# ANNUAL REPORT 1993

## United States Nuclear Regulatory Commission

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UNITED STATES NUCLEAR REGULATORY COMMISSION

August 25, 1994

The President
The White House
Washington, DC 20500

Dear Mr. President:

This Annual Report for 1993 of the United States Nuclear Regulatory Commission is forwarded for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974.

The report is devoted mainly to coverage of events and activities occurring in fiscal year 1993, with additional treatment of events after that period where circumstances warranted.

Respectfully,

Ivan Selin

Ivan Selin
Chairman

# ANNUAL REPORT 1993

## United States Nuclear Regulatory Commission

**PREVIOUS REPORTS IN THIS SERIES**

*1975 NRC Annual Report*, published April 1976

*1976 NRC Annual Report*, published April 1977

NUREG-0400, *1977 NRC Annual Report*, published April 1978

NUREG-0516, *1978 NRC Annual Report*, published February 1979

NUREG-0690, *1979 NRC Annual Report*, published March 1980

NUREG-0774, *1980 NRC Annual Report*, published June 1981

NUREG-0920, *1981 NRC Annual Report*, published June 1982

NUREG-0998, *1982 NRC Annual Report*, published June 1983

NUREG-1090, *1983 NRC Annual Report*, published June 1984

NUREG-1145, Vol. 1, *1984 NRC Annual Report*, published June 1985

NUREG-1145, Vol. 2, *1985 NRC Annual Report*, published June 1986

NUREG-1145, Vol. 3, *1986 NRC Annual Report*, published June 1987

NUREG-1145, Vol. 4, *1987 NRC Annual Report*, published July 1988

NUREG-1145, Vol. 5, *1988 NRC Annual Report*, published July 1989

NUREG-1145, Vol. 6, *1989 NRC Annual Report*, published July 1990

NUREG-1145, Vol. 7, *1990 NRC Annual Report*, published July 1991

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# NRC Annual Report
# Statutory Reporting Requirements

## ENERGY REORGANIZATION ACT OF 1974, AS AMENDED

Section 307(c) directs the Commission to include in its Annual Report statements and descriptions concerning:

"...the short-range and long-range goals, priorities, and plans of the Commission as they are related to the benefits, costs, and risks of nuclear power." (See Chapters 1, 2, 3, 4, 6, 9 and 11.)

"...the Commission's activities and findings in the following areas—

"(1) insuring the safe design of nuclear power plants and other licensed facilities...." (For reactor design, see Chapters 2 and 9; for materials facilities, devices, and transportation packaging, see Chapters 4 and 5; for waste disposal facilities, see Chapters 6 and 9.)

"(2) investigating abnormal occurrences and defects in nuclear power plants and other licensed facilities...." (See Chapters 2, 3 and 4.)

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle...." (See Chapters 5, 8 and 9.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing contingency plans for dealing with such incidents...." (See Chapters 5 and 9.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities...." (See Chapters 6 and 9.)

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities...." (See Chapters 2, 4 and 6.)

Section 205 requires development of "a long term plan for projects for the development of new or improved safety systems for nuclear power plants" and an annual updating of that plan. (See Chapter 9.)

Section 209 requires the Commission to include in each Annual Report a chapter describing the status of the NRC's domestic safeguards program. (See Chapter 5.)

Section 210 requires the Commission to submit "a plan providing for the specification and analysis of unresolved safety issues relating to nuclear reactors," and to include progress reports in the Annual Report thereafter concerning corrective actions. (See Chapter 9.)

## NUCLEAR NONPROLIFERACTION ACT OF 1978

Section 602 requires annual reports by the Commission and the Department of Energy to "include views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibilities of those agencies...." (See Chapter 8.)

## ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 170(i) directs the Commission to report annually on indemnity action implementing the Price-Anderson Act which provides a system to pay public indemnity claims in the event of a nuclear accident. (See Chapter 2.)

## PUBLIC LAW 96-295

Section 303 directs the Commission to report annually a statement of—

(1) the direct and indirect costs to the Commission for the issuance of any license or permit and for the inspection of any facility; and (2) the fees paid to the Commission for the issuance of any license and for the inspection of any facility." (See Chapter 1.)

# 1993 Highlights/Licensing and Inspection Summary

Chapter 1

This is the 19th annual report of the U.S. Nuclear Regulatory Commission (NRC), covering events and activities occurring during fiscal year 1993 (October 1, 1992 through September 30, 1993), with some treatment of noteworthy events after the end of the fiscal year.

The NRC was created by enactment in the Congress of the Energy Reorganization Act of 1974. It is an independent agency of the Federal Government. The five NRC Commissioners are nominated by the President and confirmed by the United States Senate. The Chairman of the Commission is appointed by the President from among the Commissioners confirmed by the Senate.

The mission of the NRC is to assure that civilian uses of nuclear materials in the United States—in the operation of nuclear power plants and fuel cycle plants, and in medical, industrial and research applications—are carried out with adequate protection of public health and safety, of the environment, and of national security. The agency also has a role in combating the proliferation of nuclear materials world-wide. The NRC accomplishes its purposes by the licensing and regulatory oversight of nuclear reactor operations and other activities involving the possession and use of nuclear materials and wastes; by the safeguarding of nuclear materials and facilities form theft and/or sabotage; by the issuance of rules and standards; and by inspection and enforcement actions.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1993 within the NRC or involving the NRC. The report is issued in compliance with Section 307(c) of the Energy Reorganization Act of 1974, which requires that an annual report be submitted to the President for transmittal to the Congress. This chapter takes note of significant changes in the makeup of the Commission and in agency structure, and provides a summary of licensing and inspection activity treated in detail in subsequent chapters of the report, as well as describing the status of agency consolidation.

## Changes in the Commission And in NRC Structure

The term of Commissioner James R. Curtiss ended June 30, 1993, and, as of the end of calendar year 1993, the vacancy on the Commission had not been filled. (See listing in Appendix 1 of NRC Commissioners and senior staff.)

The Director of the NRC's Office of Nuclear Reactor Regulation (NRR), Thomas E. Murley, announced his intention to retire early in 1994. William T. Russell, a former Regional Administrator at NRC's Region I (Philadelphia), was named to succeed Dr. Murley, who had served as NRR Director since 1987.

A new Deputy Executive Director for Operations for Nuclear Reactor Regulation, Regional Operations and Research, was appointed toward the end of 1993. James L. Milhoan was appointed to the post, succeeding James H. Sniezek, who announced his intention to retire early in 1994. Mr. Milhoan was previously Regional Administrator in NRC's Region IV (Dallas). Leonard Joe Callan succeeded Mr. Milhoan as Regional Administrator at Region IV.

Near the end of the report period, the Commission decided to reduce staff size and the scope of activity at Region V (San Francisco), and to designate the installation a Field Office, consolidated with activities of Region IV (Dallas). During fiscal year 1993, Region V Administrator, John B. Martin, was assigned Regional Administrator in Region III (Chicago), succeeding A. Bert Davis, who had retired. Bobby H. Faulkenberry was named Regional Administrator at Region V prior to its merger with Region IV; Mr. Faulkenberry is retiring in early 1994.

## Power Reactor Regulation

**Power Reactor Licensing Actions.** An operating license was issued for the Comanche Peak Unit 2 (Tex.) nuclear power plant, during the report period. The low power license was issued on February 2, 1993. After the licensee completed fuel loading and low power testing, the Commission met on March 16, 1993, to consider issuing a full power license. The Commission later granted the full power license, which was issued on April 6, 1993. The unit achieved commercial operation on August 3, 1993.

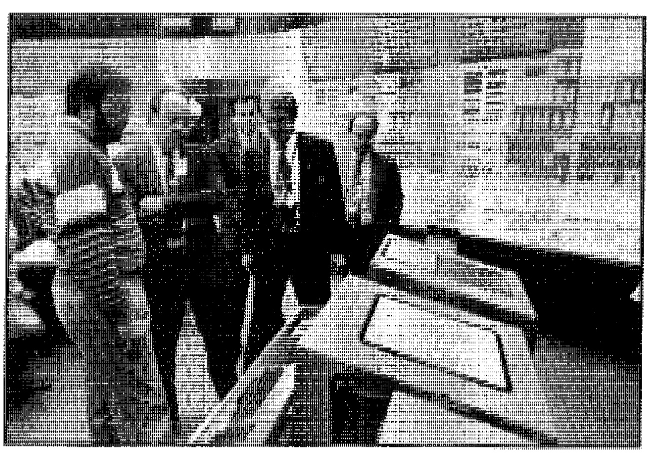
**Licensing Actions for Operating Power Reactors.** Either routine activity or unexpected events at a nuclear facility can result in a need for the NRC to take licensing actions. Routine matters occurring after license issuance include license amendment requests, possibly involving public hearings; requests for exemption from regulations; new regulations requiring "backfit" modifications to operating reactors; or orders for modification of a license. During fiscal year 1993, NRR completed about 1,400 licensing actions. About 97 percent of these actions were directed at specific plants and licensees. The balance were multi-plant actions deriving from the imposition of NRC requirements. The total inventory of licensing actions inventory has increased from about 1,145 to 1,187 licensing actions under review. (See Chapter 2.)

**Implementation Status of Safety Issues.** The NRC publishes a document annually giving the status of the implementation and verification of actions involving major safety issues. The 1993 annual report, published in December 1993, includes the status, as of September 30, 1993, of implementation and verification of all safety-issue actions affecting multiple facilities, that is, the TMI Action Plan Requirements, Unresolved Safety Issues (USIs), Generic Safety Issues (GSIs), and all other multi-plant actions (MPA). The 1993 annual report, states that more than 99 percent of the TMI Action Plan items have been implemented at the 109 licensed plants, about 90 percent of the USI items have been implemented, about 94 percent of the GSI items have been implemented, and about 87 percent of the other MPA items have been implemented.

**Renewal of Operating Licenses.** The first operating license of a current active plant will expire in the year 2004, and the operating licenses of nearly 20 percent of these plants will expire by the end of the year 2010. Because some of the licensees for these plants may soon be submitting applications to renew their operating licenses, the NRC has placed a high priority on defining the requirements that must be met before a renewal can be granted, and establishing the regulations to process such applications. The NRC published a final rule in December 1991 requiring a utility to perform a systematic review of systems, structures and components in a plant for which a license renewal is sought, in order to evaluate effects of age-related degradation and to determine whether any actions are needed to ensure continued plant safety during a period of extended operation. During 1993, at the request of the Commission, the NRC staff reviewed the practical aspects of implementing the license renewal rule, held a public workshop on the subject, and solicited additional comments from the industry and the public. The staff has proposed revisions to the existing rule under consideration.

**Improving the Licensing Process.** The Commission strongly encouraged the nuclear industry to standardize the next generation of reactor designs and to identify sites for nuclear power plants early in the licensing process. The NRC plans to realize the benefits of standardization with the new licensing process described in 10 CFR Part 52, including provisions for Design Certification, Early Site Permits, and Combined Licenses. In November 1993, the NRC issued an advanced notice of proposed rulemaking, requesting comments on a draft proposed standard design certification rule for evolutionary light-water reactor designs. The NRC is also developing the inspections, tests, analyses, and acceptance criteria (ITAAC) necessary to verify that a facility which has referenced a certified design has been constructed and will be operated in conformity with the license and the Commission's rules and regulations.

**Power Plant Maintenance.** During fiscal year 1993, the NRC and the industry developed parallel implementing guidance documents with the provisional intention that, when it found the guidelines of the Nuclear Management and Resources Council (NUMARC) guidelines acceptable, the NRC would endorse them in a regulatory guide. The NRC's regulatory guidance for implementing the maintenance guidelines was issued in June 1993. Having



**NRC Chairman Ivan Selin continued his on-site visitations to nuclear power plants at home and abroad during fiscal year 1993. Above the Chairman tours the control room at the San Onofre (Cal.) facility and speaks with operators there.**

**San Onofre, which comprises two pressurized water reactors, is located on the Pacific coast near San Clemente, Cal.**

held discussions of the guidelines in public meetings with NUMARC, and following further revision of the NUMARC guidelines, the NRC undertook a trial verification and validation program, under way at the close of the report period.

**Special Reactor Plant Inspections.** During 1993, the NRC headquarters and regional staffs continued to perform special team inspections, involving 4-to-10 inspectors and requiring 1-to-3 weeks of on-site inspection. The objective of these special inspections was to determine whether, when called upon to do so in an emergency, the nuclear plant's systems and personnel would perform their safety functions in the manner set forth in the facility's safety analysis report.

The staff performed Electrical Distribution System Functional Inspections at plants on 67 sites and plans to complete the program at the remaining two sites by December 1993.

The staff continued developing two new types of team inspections—Service Water System Operational Performance Inspection (SWSOPI) and Shutdown Risk and Outage Management (SROM). The last of five pilot SWSOPIs and three SROM pilot inspections were completed in testing and further developing the methodology. The NRC plans to conduct SWSOPIs at sites licensed before 1979 and also at sites having problems with service water systems or more general problems with maintenance, engineering, or technical support. At the end of the fiscal year, six SWSOPIs had been completed in addition to the pilot inspections.

An inspection procedure titled "Licensee Self-Assessments Related to Area-of Emphasis Inspections" (IP 40501) was issued to allow for a reduced NRC inspection activity at facilities which demonstrate good performance. Under this pilot effort, the NRC would evaluate a licensee's self-assessment effort as an alternative to a full scope NRC area-of-emphasis inspection.

**Thermo-Lag Fire Barrier Systems.** Following extensive investigation of a fire at the Browns Ferry (Ala.) nuclear power plant in 1975, the Commission, in 1981, issued a fire protection rule (10 CFR 50.48) which licensees could satisfy by, among other acceptable modes, installing a fire barrier. Beginning in 1981, licensees began requesting and receiving approval for the use of a substance called Thermo-Lag 330-1, with the result that, as of fiscal year 1993, Thermo-Lag fire barriers were installed in a majority of operating plants. Some licensees have also used Thermo-Lag to construct walls, ceilings and vaults.

By 1991, the NRC had received information which raised questions as to the adequacy of Thermo-Lag as an effective fire barrier. A Special Review Team, in its final report, issued April 1992, concluded that the fire-resistance ratings and "ampacity derating" factors (lowering the current-carrying capacity of cables, taking into account the insulating effects of the fire barrier) for Thermo-Lag were indeterminate, and that some evaluations of test results, as well as some procedures for installing Thermo-Lag, had been inadequate. Qualification fire tests of cable tray and conduit barriers conducted by the nuclear industry, and small-scale panel tests performed for the NRC staff, also demonstrated that certain Thermo-Lag fire barrier configurations may not provide the level of fire-resistant protection needed to satisfy the NRC's requirements. The staff has developed an action plan to ensure that concerns raised through the staff's review of the Thermo-Lag issue, including the adequacy of other fire barriers and the appropriateness of aspects of the NRC fire protection program, are tracked, evaluated and resolved. The staff issued seven information notices to the industry (including two on fire barriers other than Thermo-Lag), a Generic Letter, and a bulletin describing test criteria; reviewed various industry full-scale test programs; and conducted toxicity and combustibility tests. For the short term, licensees have addressed the fire endurance problem by implementing compensatory measures—such as fire watches, where the Thermo-Lag is installed. Long term actions may range from barrier upgrades and repairs to complete replacement of some barriers. Additional plant specific analyses may also be required to resolve the ampacity derating problem. Regulatory action and coordination with the industry will continue until the technical and programmatic issues in the staff's action plan have been resolved. (See discussion under "Safety Reviews," in Chapter 2.)

## Nuclear Materials Regulation

Nuclear materials regulation during fiscal year 1993 comprised:

- Over 5,000 licensing actions on applications for new byproduct materials licenses, amendments, and renewals of existing licenses and reviews of sealed sources and devices.
- Approximately 2,400 materials licensee inspections.
- Over 100 fuel storage and transportation package reviews and 15 route approvals for transporting special nuclear material and spent fuel.
- 11 inspections of supplier quality assurance (QA) programs.

**Materials Licensing and Inspection.** The NRC currently administers approximately 6,850 licenses for the possession and use of nuclear materials in medical and industrial applications. This represents a reduction of about 350 licenses in the past year. The 29 Agreement States administer about 15,000 licenses. NRC regional staff completed approximately 2,400 inspections of materials facilities in fiscal year 1993. The NRC Regional Offices administer almost all materials licensees, with the exception of exempt distribution licenses, sealed source and device design reviews, and licenses for companies that extract other metals from ores and slags containing uranium and thorium. These licenses are handled at NRC Headquarters.

The NRC completed 5,043 licensing actions during the fiscal year. Of this total, 366 were new licenses, 3,217 were amendments, 1,088 were license renewals, and 372 were sealed source and device reviews. (See Chapter 4.)

**Fuel Cycle Licensing Activities.** By the end of fiscal year 1993, the NRC had completed 120 fuel cycle licensing actions. These included 43 uranium fuel fabrication inspections; nine uranium hexafluoride production inspections; seven critical mass materials inspections; four fuel research and development and pilot plant inspections; four other source materials (metals extraction) inspections; three fuel facility decommissioning inspections; two fresh fuel storage inspections; 35 material control and accounting inspections; and 13 physical security inspections.

**Uranium Enrichment.** In November 1990, the President signed into law the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990 (Public Law 101-575). The law amended the Atomic Energy Act to establish new requirements for regulation of commercial uranium enrichment facilities.

In January 1991, the Louisiana Energy Services, L.P., submitted an application for a license to construct and operate a gas centrifuge uranium enrichment plant, to be known as the Claiborne Enrichment Center. It is to be located in Claiborne Parish near Homer, La., and to have a capacity of 1.5 million kilograms of "separative work units-per-year," about 15 percent of the annual requirement of United States nuclear utilities for enrichment services.

In July 1991, a "scoping meeting" was held in Homer, La., as part of the process leading to preparation of the required environmental studies. During 1993, staff continued review of the license application and preparation of the draft Environmental Impact Statement (EIS) and Safety Evaluation Report (SER). The draft and final EIS documents will be published in late 1993 and 1994, respectively. The SER will be published in early 1994. The required hearings on technical and environmental issues will begin following publication of the final SER and EIS documents, respectively.

**Fuel Cycle Safety Inspection.** As part of the February 7, 1993 reorganization of fuel cycle activities within the Office of Nuclear Materials Safety and Safeguards, several fuel cycle facility inspection activities have been consolidated in Headquarters. During fiscal year 1993, headquarters staff provided technical expertise to address difficult design, integration and adequacy concerns in the areas of criticality and chemical safety.

**Region-Based Inspection Activities.** The five Regional Offices conducted more than 100 safety inspections at 15 operating and decommissioning fuel cycle facilities during fiscal year 1993. The inspections included resident inspector activities at two of these fuel cycle facilities.

**Fuel Cycle Safeguards Licensing.** There were 13 active, licensed nuclear fuel cycle facilities subject to NRC comprehensive safeguards requirements during fiscal year 1993. Of these, eight were major fuel fabrication facilities. Two of the 13 facilities contain significant quantities of high-enriched uranium (HEU), requiring extensive physical security and MC&A measures. One of these two facilities—NFS, of Erwin, Tenn.—essentially phased out its naval reactors program work during calendar year 1993. An agreement with the Russian Federation, involving the conversion of HEU from the former Soviet Union nuclear weapons program into light water reactor fuel, did not lead to any further measures taken during 1993.

**Fuel Cycle Safeguards Inspection.** Headquarters staff conducted 15 comprehensive MC&A inspections, while the regional and resident inspectors continued to perform inspections for physical security at major fuel fabrication facilities. Approximately 17 physical security inspections were performed by region-based inspectors. Newly implemented physical security improvements were subject to thorough inspections at the two facilities cited above as possessing significant quantities of HEU. Performance-based inspection procedures were followed by both MC&A and physical security inspectors.

**Reactor Safeguards.** Within the five NRC Regional Offices, a total of 185 safeguards inspections were conducted at licensed nuclear power reactors subject to NRC safeguards requirements. Approximately 227 revisions to licensee security, contingency, and guard training plans were reviewed and approved by both regional and headquarters staff.

**Operational Safeguards Response Evaluations at Power Reactors.** After completion of the Regulatory Effectiveness Review Program in May 1991, the NRC staff initiated an Operational Safeguards Response Evaluation (OSRE) program at power reactors. An OSRE is an effectiveness review conducted by an interdisciplinary team consisting of a nuclear engineer and physical security specialists, assisted by U.S. Army Special Forces personnel. The team evaluates a licensee's contingency response capabilities by focusing on the interactions between operations and security personnel in establishing priorities for the protection of safety equipment, and by scrutinizing and testing the defensive strategies used. OSRE teams also conduct safety/safeguards interface reviews, to ensure that safeguards measures do not adversely affect the safe operation of the plant. Seventeen OSREs were conducted during fiscal year 1993, resulting in a combined total of 15 significant improvements at seven power reactor sites.

Sixteen inspections of licensee access authorization programs have been conducted under a temporary inspection program (TI 2515/116) for the purpose of assessing initial implementation of selected programs to determine whether they meet regulatory requirements and to identify program strengths and weaknesses. The results of these inspections are being evaluated to determine if changes to the program requirements are needed and if modifications are needed in the scope and depth of the inspection program.

**Non-Power Reactors.** The NRC conducted 34 safeguards inspections of Non-Power Reactors (NPRs) during fiscal year 1993. Efforts are continuing toward converting 25 NPRs from the use of high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel. NRC regulations governing this project continue to be predicated on (1) the availability of Department of Energy (DOE) funding, (2) the availability of a suitable replacement fuel, and (3) whether a reactor has a "unique purpose" requiring the use of HEU. The status of the conversion program at the end of the fiscal year is as follows: one license has been terminated; two licensees have been issued decommissioning orders; one licensee is planning to decommission its reactor, and seven reactors have been converted from the use of HEU to LEU fuel. There are 14 reactors that are still operating with HEU. Of these 14 reactors, nine have been funded by the DOE for evaluating the operational effects of the conversion and the writing of an Safety Analysis Report. Also, there are two "unique purpose" applications being reviewed by the Commission; in the one case, there is no suitable replacement fuel for the reactor, and the other involves two commercial reactor licensees not scheduled to receive DOE funding.

## Transportation Safeguards

**Spent Fuel Shipments.** Safeguards requirements were applied to 29 shipments of irradiated spent reactor fuel made over approved routes during fiscal year 1993, including nine rail shipments to the spent fuel pool at the Harris (N.C.) nuclear power plant, which is configured to store a large number of spent fuel assemblies. These shipments, planned by Carolina Power and Light, will, over a five-year period, transfer approximately 1,170 fuel assemblies from other reactors to the Harris pool for storage. One of the shipments was an export.

**Strategic Special Nuclear Material Shipments.** Four domestic shipments of less than five but more than one kilogram of HEU were completed during fiscal year 1993. Two export shipments of five or more kilograms were also made during fiscal year 1993.

**Tracking International Shipments of SNM.** NRC regulations require licensees to notify the NRC of international shipments of Special Nuclear Material (SNM) and natural uranium. During fiscal year 1993, the NRC received about 200 such notifications. When appropriate, these were forwarded to the Department of Transportation, for notification of international authorities.

The NRC continues to contribute to the total U.S. support of the International Atomic Energy Agency (IAEA) safeguards through interagency efforts involving the DOE, the Arms Control and Disarmament Agency, the Department of State, and the NRC. These interagency activities serve to coordinate U.S. Government technical safeguards support to the IAEA.

The NRC continues to provide support to the interagency Comprehensive Threat Reduction Program. This initiative, originally called the Safe and Secure Dismantlement program, is intended to coordinate support to the republics of the former Soviet Union in the dismantling of their nuclear arsenals and thus helping to stem proliferation of weapons of mass destruction. The NRC's role is to supply assistance to these republics in setting up national regulatory systems for material control and accounting (MC&A) and physical protection, as well as to assist individual facilities in developing and evaluating site-specific MC&A and physical protection plans.

Russia signed the MC&A implementing agreement in September 1993. Kazakhstan has agreed to the text of the MC&A implementing agreement and is expected to sign in January 1994. Ukraine is still waiting for parliamentary approval of several of its agreements, including MC&A. Belarus, which has no reactors or fuel facilities, has also requested U.S. assistance in setting up a national regulatory program, but discussions in this area have been limited.

In February 1993, the United States and the Russian Federation reached agreement on the disposition of HEU recovered from decommissioned Russian nuclear warheads. The bilateral agreement allows the United States to purchase approximately 500 metric tons of HEU extracted from dismantled nuclear weapons and reduced by blending, in Russia, down to low-enriched form. The material will be fabricated into nuclear fuel in the United States, by NRC licensees, for use in light water reactors. The NRC's role is to ensure that "transparency" measures in U.S. facilities are practical, and, in this context, the NRC solicited comments from fuel fabricators and ensured that their concerns were considered in the agreements. The United States and Russia are negotiating final details related to transparency, with the intent of starting the blending process in early 1994.

In August 1993, in response to an invitation by the government of the People's Republic of China (PRC), a senior safeguards specialist presented seminars on the NRC's safeguards programs related to the commercial nuclear industry in the United States. The visit resulted in a better understanding on the part of the Chinese of the regulations constituting the U.S. safeguards system. The PRC is trying to improve regulation of its nuclear industry in this and other areas.

**Other Licensing and Inspection Activities.** In the fall of 1989, the NRC received an application from Envirocare of Utah, Inc., for a license to dispose of commercial uranium and thorium mill tailings and other 11e.(2) byproduct material at its existing radioactive disposal facility in Clive, Utah. Early in 1991, a *Federal Register* notice was published announcing receipt of the application, describing the regulatory requirements to be applied in the licensing review, and giving notice of the opportunity for a public hearing. The safety portion of the licensing review concluded with the issuance of the Final Safety Evaluation Report (SER) in June 1993, and an SER supplement in September 1993. The environmental portion of the licensing review was completed with the issuance of the Final Environmental Impact Statement in September 1993. The license for Envirocare of Utah, Inc., was issued on November 19, 1993.

In fiscal year 1993, the Denver field office staff performed 36 inspections of uranium recovery facilities and issued the following: one new license for a commercial in-situ solution mining operation, 81 new licenses for a commercial laboratory, three license renewals, 84 license amendments, and five mill tailings reclamation plan amendments. In addition, 121 environmental and radiological monitoring report reviews were completed and pre-licensing guidance was provided to two potential license applicants.

## OFFICE OF POLICY PLANNING

The Director of the Office of Policy Planning (OPP) serves as Chair of the Strategic Planning Committee for NRC. The committee updated the NRC's strategic plan during the report period, leading to the development of the Five-Year Plan for fiscal years 1994–1998. Other major activities of the OPP during fiscal year 1993 included the completion of six policy studies, each incorporated in a report to the Commission. Issues treated in these policy assessments were:

- The effectiveness and health of the reactor inspection program.
- The current licensing basis for operating plants.
- The utility decision-making perspective for the renewal of power reactor operating licenses.
- The cost of low-level radioactive waste disposal facilities.
- The separation of civilian and military nuclear programs.
- The protection of public health and safety in the medical uses of ionizing radiation.

The approach in each of the assessments included actively seeking the views of industry, public interest groups, and other affected parties, as well as of appropriate NRC staff, so that the issues are illuminated and examined from a number of relevant perspectives. Each of the reports sets out specific findings and recommendations for agency action. Highlights of some of these findings are discussed below.

The reactor inspection program was found to be effective in terms of its overall positive impact on the safety of operating reactors. The problem plant list is highly effective in meeting the NRC's objectives, but the committee felt that the process could be improved if the NRC were to inform a utility's chief executive officer when performance at the utility's plant was tending toward inclusion on the problem plant list. The SALP program was found to be effective and no further changes in it were recommended. Escalated enforcement was also found to be effective, but the safety benefit does not appear commensurate with resources expended. These and other findings and recommendations are set forth in OPP–92–01 dated November 16, 1992.

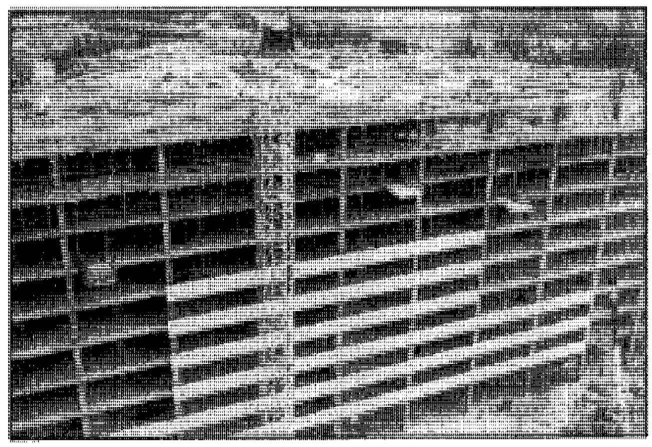
On the issue of current licensing basis (CLB), OPP–92–02, dated November 30, 1992, notes that there is a need to define CLB in order to provide a clear understanding of the bases upon which reactors are licensed, as distinct from the bases upon which operational performance is measured. The report recommends a definition of CLB that represents the minimum safety envelop within which operations must take place and should not be changed without prior NRC review and approval. This definition would apply to both the current license and a renewed license, although the basis for renewal would necessarily be more expansive than CLB alone.

The report on the utility decision-making perspective on reactor license renewal confirms that the environment in which decisions are made for license renewal has become more complex in recent years. The uppermost consideration regarding NRC activities is the need for clarity and predictability in the license renewal rule and related documents, and in the NRC's case-by-case implementation of the rule. Even with this proviso, however, the report (OPP–93–01, dated February 19, 1993) notes that the determinative issue is that of the economics of future operation of the nuclear plant. Other important matters involving the utility's decision-making processes are the disposal of highand low-level radioactive waste, the diversity of fuel supplies, and public sentiment regarding future operation.

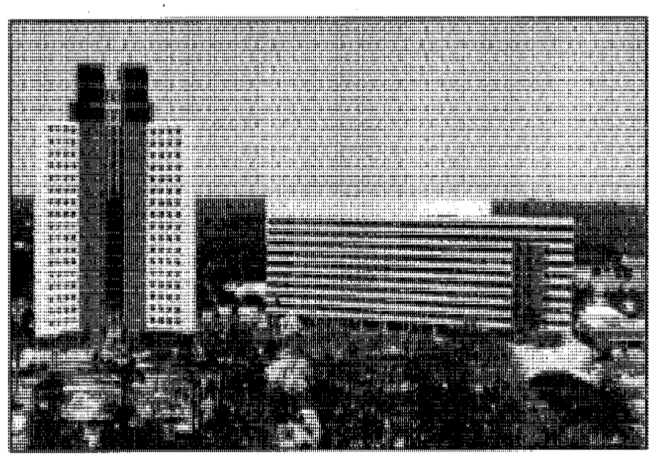
## Table 1. License and Annual Fee Collections – FY 1993

| *Fees* | *Facilities Program* | *Materials Program* | *Total* |
|---|---|---|---|
| 10 CFR Part 170 | $93.7 million | $11.9 million | $105.6 million |
| 10 CFR Part 171 | $351.4 million | $50.1 million | $401.5 million |
| TOTAL FEES | $445.1 million | $62.0 million | $507.1 million |

The costs to all sectors having responsibility for the regulation, administration, and facility development for low-level waste management are compiled and laid out in OPP-93-02 dated April 30, 1993, to provide the Commission with an informed perspective on principal cost-driving factors. Although the cost data cannot be taken as precise because of the diverse methods for cost-accounting employed by States and State-compacts, the total to date is estimated to exceed $320 million. Cost information for several foreign programs which have resulted in the operation of low-level radioactive waste disposal facilities is incorporated to provide perspective relative to development and operating costs for U.S. facilities.



**At the start of fiscal year 1993, installation of the exterior, concrete pre-cast panels and windows was under way at the new NRC office building, Two White Flint North (TWFN), as shown above. By the end of the fiscal year, the exterior of TWFN was complete, and the facility was near readiness for occupancy in calendar year 1994. NRC Headquarters, with One White Flint North on the left and TWFN on the right, is pictured below.**



A focus on national policy regarding the separation of civilian and military nuclear programs arose from the possibility of converting Russian military nuclear materials to U.S. civilian uses. OPP-93-02 confirms that there has been a longstanding national policy of separation of military and civilian nuclear programs, but that it has often been only one of a number of considerations in the framing of major policy decisions. While military material and technology have, in the past, been converted to civilian uses, no case was found where civilian materials were converted to military uses. This determination provides the fundamental policy linchpin for dealing with this issue.

The report on medical uses of radiation was prepared in response to a direct request from Senator John Glenn (D.-Ohio). An NRC task force, with assistance from the Food and Drug Administration (FDA), reviewed options for the regulation at the Federal and State level, in light of certain highly publicized instances of harm to patients from medical procedures involving radiation. Medical uses of radiation are regulated directly by the NRC, or their Agreement States, and by the Food and Drug Administration; some uses are regulated directly at the State level. The task force determined that sufficient data are not available to assess the level of protection afforded for all sources of radiation by the current regulatory framework; however, it is not clear that the current regulatory framework for all sources of radiation does not adequately protect the public health and safety. OPP-93-04 concludes that until more definitive data are collected concerning the magnitude of the health problem, the current regulatory framework should be kept in place.

## NRC LICENSE AND ANNUAL FEES

The Omnibus Budget Reconciliation Act of 1990 (Public Law 101-508) requires that, in fiscal year 1993, the NRC collect license fees (under 10 CFR Part 170) and annual fees (under 10 CFR Part 171) that approximate 100 percent of the agency's budget authority, less the amount appropriated to the NRC from the Nuclear Waste Fund. For fiscal year 1993, a total of $540 million was appropriated to the NRC (Public Law 102-104), of which $21.1 million was derived from the Nuclear Waste Fund. Of the remaining $518,900,000, approximately 98 percent, or $507,072,406, was collected through license fees and annual charges. Therefore, the net amount appropriated to the NRC in fiscal year 1993, including appropriations from the Nuclear Waste Fund, was $32,927,594. The Table 1 shows the amounts collected through license and annual fees in fiscal year 1993. A detailed account of NRC financial management, with an audited financial report, is given in the *NRC Financial Statement for FY 1993* (NUREG-1470, Vol. 3).

## NRC CONSOLIDATION NEAR COMPLETION

At the start of fiscal year 1993, the installation of the exterior concrete pre-cast panels and windows had just commenced at the new NRC Headquarters building in Rockville, Md., called Two White Flint North (TWFN). By mid-summer, the base-building construction and landscaping were substantially completed and the interior construction had begun.

During the period, the General Services Administration entered negotiations with the developer to lease the plaza level and garage for the NRC. A full-service cafeteria, credit union, fitness center, and employee store are planned for the plaza level. Occupancy of TWFN for more than 1,300 NRC staff was scheduled to commence in late spring of 1994.

# Nuclear Reactor Regulation

Chapter 2

The Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC) is responsible for developing and issuing regulations for the safe operation of the nation's operating nuclear power and research reactors and for assessing applications to construct and operate new reactors and issuing permits and licenses to do so. The operating and proposed new reactors include both nuclear power reactors operated by electric utilities and non-power reactors, such as those operated by various universities. The NRC does not regulate reactors operated by the Department of Energy (DOE) for furnishing fissionable materials for use in nuclear weapons. More specific NRR responsibilities include the approval and oversight of reactor siting, design, construction, operation, maintenance, and decommissioning. NRR's review responsibilities encompass the safety, safeguards, environmental and antitrust considerations related to reactor facilities. NRR also directs and oversees the NRC Regional Offices in their conduct of reactor licensing and inspection activity.

The licensing activity of NRR begins with the extensive review of applications for construction permits and operating licenses for new reactors, and the complex procedures—including inspections from the outset of plant construction and throughout a facility's operating lifetime—leading to issuance of permits or licenses, and licensing actions taken thereafter. In recent years, a steady increase in the number of licensed operating reactors and a decrease in the number of plants still under construction have brought about a substantial shift in NRC activity. NRC staff focuses on the safety regulation of the 109 nuclear power plants licensed for operation in the United States. (See Appendix 7 for listing of and data on all NRC-licensed power plants.) At the same time, the NRC is increasing attention to the development of criteria and procedures for conducting safety reviews of the advanced reactor designs proposed for nuclear plants of the future.

## STATUS OF LICENSING

### Reactor Engineer Intern Program

The Reactor Engineer Intern Program was established in 1988 to train new personnel in anticipation of the agency's future work force requirements. The program seeks out recent engineering graduates, recruited primarily from colleges and universities with reputations for strong engineering programs. Through individually tailored assignments at Headquarters, Regional Offices, and plant sites—coupled with extensive formal training in nuclear reactor technology—Reactor Engineer Interns are exposed to a wide range of activities of the NRC so that they may acquire a broad grasp of the various concerns, roles and tasks of the agency. Upon completing the rigorous two-year program, interns are given permanent technical professional assignments based on their educational background, personal and career preferences, and the needs of the agency.

In June 1993, at a joint ceremony recognizing the first graduates of intern programs established by NRR, by the Office of Nuclear Regulatory Research, and by the Office of Nuclear Materials Safety and Safeguards, Headquarters and Regional Offices honored 35 graduating Reactor Engineer Interns, the largest class of graduates since the inception of the program. Since 1988, a total of 53 interns have completed the Reactor Engineer Intern Program and have assumed permanent positions at Headquarters and in the Regions. Currently, 32 headquarters-based interns are pursuing the requirements of the Reactor Engineer Intern Program.

### License Applications, Issuances and Decommissioning

On June 27, 1988, the NRC staff published a final rule amending requirements for the decommissioning of nuclear power plants. Decommissioning, as defined in that rule, means the removal of a nuclear power generating facility safely from service, the reduction of residual radioactivity to a level that permits release of the property for unrestricted use, and termination of the license. The rule is written with the assumption that a licensee would not cease operations before completing the full term of a facility operating license.

## LICENSING THE NUCLEAR POWER PLANT

The nuclear power plant licensing process begins when a utility files an application by a utility for a construction permit with the NRC. The application usually follows considerable consultation between the utility and the NRC staff and comprises many volumes of data, covering both safety and environmental aspects of the intended operation, in accord with NRC requirements and guidance. The NRC staff completes the second phase by reviewing various safety, environmental, safeguards (from theft or sabotage), and antitrust issues. Thereafter, as required by law, the independent Advisory Committee on Reactor Safeguards (ACRS) assesses the proposed project and the results of the earlier reviews and makes its recommendations. The fourth phase is a mandatory public hearing on the matter conducted by a three-member Atomic Safety and Licensing Board (ASLB) which makes an initial decision as to whether a construction permit should be granted. This decision is subject to appeal by any person or group with standing in the proceeding to the Commissioners for a final NRC decision. Appeal beyond the NRC decision is available by recourse to the Federal courts.

When the NRC staff accepts ("dockets") the initial application of a utility, the staff publishes a notice of the fact in the *Federal Register*, and furnishes copies of the application to the appropriate State and local authorities and to a local public document room established by the NRC near the proposed plant site, and to the NRC public document room in Washington, D.C. At the same time, the NRC publishes a notice of a public hearing in the *Federal Register* and in local newspapers giving 30 days for members of the public to petition to intervene in the proceeding. Such petitions are entertained and adjudicated by the ASLB appointed to the case, with rights of appeal by the petitioner to the Commission.

With guidance of the standard format (Regulatory Guide 1.70), the applicant for a construction permit describes the proposed nuclear plant design in a preliminary safety analysis report. Upon finding this report sufficiently complete to warrant review, the NRC staff dockets the application and begins the safety, environmental, safeguards, and antitrust reviews in parallel. Even before receiving a safety report, NRC staff will conduct a substantive review and inspection of the design and procurement activities in the applicant's quality assurance program. The safety review is performed in accordance with the Standard Review Plan for Light-Water-Cooled Reactors, initially published in 1975 and periodically revised since then. The plan sets forth the acceptance criteria used in evaluating the various systems, components, and structures related to safety and in appraising the suitability of the proposed site; it also describes the procedures to be used in performing the safety review.

The NRC staff examines the applicant's PSAR to determine whether the plant design is safe and consistent with NRC rules and regulations, whether valid methods of calculation were used, and whether the applicant has conducted its analysis and evaluation in sufficient depth and breadth to ensure adequate safety. Upon verifying that the applicant's preliminary report meets the acceptance criteria of the Standard Review Plan, the staff prepares a Safety Evaluation Report describing the expected effect of the construction and operation of the proposed facility on public health and safety.

Following publication of the Safety Evaluation Report, the ACRS completes its assessment and meets with the staff and the applicant. The ACRS then prepares a report under letter to the Chairman of the NRC, presenting the results of its independent evaluation and its recommendations as to whether a construction permit should be issued. The staff issues a supplement to the Safety Evaluation Report which incorporates any changes or actions adopted as a result of ACRS recommendations. A public hearing can then ben held, generally in a community near the proposed facility site, on the safety aspects of the licensing decision.

Where appropriate, the NRC may grant a Limited Work Authorization to an applicant in advance of a final decision on the construction permit, in order to allow work to begin at the site; such a step can save months in construction time. This authorization will not be given until the NRC staff has completed its reviews of environmental impact and site suitability and the ASLB has conducted a hearing on environmental impact and site suitability and has reached a favorable finding. To realize the desired saving in construction time, the applicant must submit the environmental portion of the application early in the process.

The environmental review begins with an assessment of the acceptability of the applicant's Environmental Report. If that report is judged sufficiently completed to warrant review, the staff dockets the report and begins an analysis of the consequences to the environment from the construction and operation of the proposed facility. Upon completion of the analysis, a Draft Environmental Statement is published and distributed with specific requests for evaluation and comment by Federal, State and local agencies, other interested parties, and members of the general public. Comments received are taken into account in the preparation of a Final Environmental Statement. Both the draft and the final statements are made available to the public at the time of their publication. During this same period, the NRC staff conducts analyses and prepares a report on the site suitability concerns of the proposed licensing action. After these efforts, a public hearing, presided over by the appointed ASLB, may be held on the environmental and site suitability issues related to the proposed licensing action (or a single hearing on both safety and environmental matters may be held).

The antitrust reviews of license applications are carried out by the NRC and the U.S. Attorney General before or during other licensing reviews. If an antitrust hearing is required, it is held separately from hearings on safety and the environment.

# Table 1. Power Reactor Licensing by Category—FY 1993

| | |
|---|---|
| Low-Power Operating License issued | 1 |
| Full-Power Operating Licenses issued | 1 |
| Operating License applications under review | 7 |

Since 1988, several licensees announced decisions to permanently cease power operations at nuclear power generating facilities before the expiration of their operating licenses. The reasons for these decisions were related to political, technical or economic problems. These facilities have entered prematurely into the decommissioning process. Six such facilities have been shut down prematurely since 1988: the Fort St. Vrain (Colo.) nuclear power plant, the Shoreham (N.Y.) nuclear power plant, the Rancho Seco (Cal.) nuclear power plant, the Yankee-Rowe (Mass.) nuclear power plant, the San Onofre Unit 1 (Cal.) nuclear power plant, and the Trojan (Ore.) nuclear power plant. Three Mile Island Unit 2 (Pa.) ceased operation following the March 28, 1979 accident.

On November 23, 1992, the NRC issued an order approving the Fort St. Vrain decommissioning plan. The decommissioning plan for Shoreham was approved in June 1992 and decommissioning and dismantling activities are taking place there as well. The Rancho Seco decommissioning plan has not been approved because the Commission has remanded three issues to the Atomic Safety Licensing Board for consideration (loss of off-site power, decommissioning funding plan, and a decommissioning environmental assessment, raised by the Environmental and Resources Conservation Organization).

During fiscal year 1992, Yankee Atomic Electric Company, Southern California Edison Company, and Portland General Electric Company announced their decisions to prematurely shut down and decommission the Yankee-Rowe, San Onofre Unit 1, and Trojan facilities, respectively. In January 1993, the Portland General Electric Company announced its decision to permanently cease operations at the Trojan plant, further accelerating the Trojan shutdown schedule from its previously proposed 1996 shutdown date. Each of the three facilities is now permanently shut down and has been issued a possession-only license. The NRC has issued various license amendments to reduce requirements, consistent with the reduced potential safety hazards of these facilities.

In January 1993, the Commission issued guidance regarding activities which may be permitted before a decommissioning plan is approved. Licensees of plants which have possession-only licenses or permanent shutdown orders will be allowed to undertake any decommissioning activity that does not (1) foreclose the release of the site for possible unrestricted use, (2) significantly increase decommissioning costs, (3) cause any significant environmental impact not previously reviewed, or (4) violate the terms of the existing license. Licensees may be permitted to use their decommissioning funds for such activities, notwithstanding the fact that their decommissioning plans have not yet been approved by the NRC. In accordance with this guidance, Yankee Atomic Electric Company is conducting early removal of the four steam generators, the pressurizer, and reactor vessel internals from the Yankee-Rowe plant. In a July 15, 1993 letter to Yankee-Rowe management, the staff stated that it has no objection to these activities. By October 28, 1993, all four steam generators at Yankee-Rowe had been removed from containment. On October 16, 1993, the licensee began shipping these components for burial at the low-level waste disposal facility at Barnwell, S.C., and completed shipments by December 14, 1993.

An operating license was issued for Comanche Peak Unit 2 (Tex.) during fiscal year 1993. The low power license was issued on February 2, 1993. After the licensee completed fuel loading and low power testing, the Commission met on March 16, 1993, to consider issuing a full power license. The Commission subsequently granted the full power license, which was issued on April 6, 1993. The unit achieved commercial operation on August 3, 1993.

The licensee for this plant is the Texas Utilities Electric Company (TU). Comanche Peak is a two-unit 2,300-megawatt nuclear power plant located in central Texas, about 70 miles southwest of Dallas. Comanche Peak Unit 1 has been in operation since 1990.

The construction permits for Comanche Peak Units 1 and 2 were issued in 1974. In 1978, extensive design and construction deficiencies were identified. Resolution of these problems and the need to address new safety requirements imposed by the NRC following the Three Mile Island accident were the principal reasons for the delays in licensing the plants until the 1990s. TU suspended Unit 2 construction activities until July 1989, in order to concentrate on completing Comanche Peak Unit 1. Comanche Peak Unit 2 management and engineering activities resumed in July 1989 and June 1990, respectively; and construction activities resumed in January 1991.

The NRC has conducted extensive inspections at Comanche Peak to ensure that the facility was constructed properly and can be operated safely. Since the resumption of construction on Comanche Peak Unit 2 in January 1991, the NRC staff has expended about 12,000 hours of direct inspection effort at Unit 2, including four major team inspections: a configuration management inspection, a design attribute verification inspection, a fire protection team inspection, and an operational readiness assessment team inspection. Initial full power operation of Unit 2 has been satisfactory.

**Licensing Actions for Operating Power Reactors.** Either routine activity or unexpected events at a nuclear facility can result in a need for the NRC to take licensing action. Routine post-licensing activities affecting reactor operations include such matters as license amendment requests, possibly involving public hearings; requests for exemption from regulations; new regulations requiring "backfit" modifications to operating reactors; or orders for modification of a license. During fiscal year 1993, the Office of Nuclear Reactor Regulation (NRR) completed about 1,400 licensing actions. About 97 percent of these actions pertained to specific plants and licensees. The balance were multi-plant actions deriving from the imposition of NRC requirements. The total licensing action inventory has increased from about 1,145 to 1,187 licensing actions under review.

## Special Cases

**FitzPatrick.** The FitzPatrick (N.Y.) nuclear power plant is a boiling water reactor owned and operated by the New York Power Authority (NYPA). In February 1992, the FitzPatrick plant was placed on the NRC's list of plants requiring close monitoring because of regulatory concerns about declining performance. The plant remained in an extended refueling outage throughout 1992 to resolve numerous design and engineering deficiencies, most notably those in the fire protection and 10 CFR Part 50, Appendix R programs. The licensee agreed not to restart the plant until the NRC was satisfied with the plant's readiness for power operation.

In October 1992, the NRC conducted a Restart Readiness Team Inspection to obtain an independent, in-depth evaluation of the readiness of plant management, programs, equipment and staff to safely restart and operate the FitzPatrick plant. The team noted significant improvement in performance in each of the areas reviewed. The NRC subsequently determined that sufficient progress had been made to support safe plant operation and advised the licensee of this determination on December 29, 1992.

The licensee restarted the FitzPatrick plant on January 2, 1993, and achieved full power on January 30, 1993. All aspects of the restart were well controlled and management oversight of restart activities was excellent. During the next 10 months, the plant experienced one manual shutdown, three automatic reactor shutdowns, and two forced shutdowns.

In the Systematic Assessment of Licensee Performance (SALP) report for the period of April 19, 1992, through April 17, 1993, the NRC staff assigned the rating "good" to performance in the Operations, Maintenance/Surveillance, Radiological Controls, and Safety Assessment/Quality Verification functional areas. Although improvements were noted in the Engineering/Technical Support functional area, performance in this area was again assessed as being only "adequate." However, "superior" performance was demonstrated in the areas of Security and Emergency Preparedness.

In October 1993, the NRC conducted an Operational Safety Team Inspection at the plant site and corporate offices to assess the quality of management programs, self-assessment programs, corrective action programs, and engineering and technical support. The team found that corporate and plant management practices were effective in ensuring safe plant operation. Self-assessment programs gave plant and corporate managers accurate assessments of plant performance. However, the team noted that additional attention was needed to ensure that deficiencies were closed in a timely manner. The processes for identification, assessment and resolution of utility- and industry-identified problems were generally effective. But the root cause evaluation program was not consistently performed and some corrective actions were not deemed thorough and timely. The licensee made significant management changes and planned similar changes for the engineering and technical support organizations. Although finding improved corporate engineering interfaces with the plant and technical support for maintenance and operations, the team noted that the licensee should continue giving attention to successfully managing the significant changes in the engineering and technical support areas.

The NRC observed improved performance at the FitzPatrick plant during the report period and has created the NRC's NYPA Assessment Panel, a group of NRC regional and headquarters staff, to continue the assessment of the licensee's improvement efforts and overall plant performance.

**Brunswick Steam Electric Plant.** The Brunswick Steam Electric Plant (BSEP) Units 1 and 2 (N.C.), owned and operated by the Carolina Power & Light Company (the licensee), includes two General Electric 849-megawatt (electric) boiling-water reactors. In July 1992, the BSEP was added to the list of facilities which, while still authorized to operate by the NRC, warrant increased NRC oversight because of concerns about the condition of the plant and evidence of declining personnel performance.

The licensee voluntarily shut down both BSEP units on April 21, 1992, after declaring the emergency diesel generators (EDGs) inoperable because of concerns over the seismic response adequacy of the interior walls of the EDG building. After shutting down the plant, the licensee found other civil and structural concerns, such as improper evaluation for safety-related classification of some masonry walls, numerous original installation deficiencies with miscellaneous structural steel supporting safety-related equipment, and significant corrosion of instrument racks and service water system piping and supports. The licensee evaluated the extent of the material deficiencies and prepared a list of the corrective maintenance activities that would be completed before restarting of either unit. The licensee also committed to significant corporate and BSEP improvement efforts to resolve these deficiencies and raise the overall performance of its nuclear facilities.

To determine the root causes and correct the previous decline in performance, the licensee instituted various initiatives and improved the material condition of the facility. On November 30, 1992, the licensee issued Corporate Improvement Initiatives (CII) to address problems in organization and management effectiveness, nuclear safety oversight, basic work control systems, and plant material condition. The scope of the CII was captured in the Brunswick Nuclear Plant Three-Year Plan (1993–1995) of December 15, 1992, which seeks to improve (1) performance to "top-quartile" level by 1996 in the areas of safety, operations, and cost performance, (2) employee satisfaction, and (3) schedule and commitment achievement. The licensee has successfully met several of the objectives listed in the Three Year Plan.

After the year-long shutdown of BSEP Unit 2, the licensee restarted the unit on April 29, 1993, and resumed normal operation on June 3, 1993. The restart of the unit proceeded without significant difficulty, and the unit continued to operate well at the close of the report period. The licensee considered the remaining core life and the extent of the corrective actions needed before restart and elected to enter Unit 1 Refueling Outage No. 8 on April 8, 1993. The licensee had projected a fall 1993 restart of this unit. However, after the discovery of cracks in the core shroud that required a repair modification, the restart schedule was delayed until January 1994. The licensee continues to improve the material condition of the facility and is replacing the conventional and nuclear service water pumps in sequence.

The NRC continues overseeing the Unit 1 outage activities and the operation of Unit 2. As during the restart of BSEP Unit 2, the NRC will perform augmented inspection during the restart of Unit 1. The NRC also will continue to monitor the completion of the licensee's improvement initiatives to verify that the stated goals and objectives are achieved. The NRC will continue the periodic public meetings with the licensee management to assess the status of the restart efforts and the accomplishment of longer range activities.

**Commonwealth Edison Company.** The Commonwealth Edison Company (CECo) is the owner and operator of 12 nuclear power plants at six sites in the state of Illinois. The sites are Braidwood, Byron, Dresden, LaSalle, Quad Cities, and Zion, and they range in time of operation from 23 years for Dresden to six years for Braidwood. Each site houses two operating reactors, giving the utility a total nuclear generating capacity of 11,500 megawatts-electric.

Cyclical performance of CECo plants has concerned the Commission and NRC staff for some time. In 1991, regulatory concerns with declining performance at both Dresden and Zion prompted the NRC to add these plants to the list of operating plants that warrant increased NRC attention. During 1992, the NRC staff determined the following probable root causes for the utility's difficulties: (1) continued hardware and programmatic problems resulted from insufficient management attention and resources committed to operating sites during the construction of new facilities; (2) limited effectiveness of corporate level oversight of nuclear operations brought about disparities in the quality of operations at the various sites; (3) the licensee was slow to recognize situations requiring increased management attention and to ensure permanent correction of problems, as they came to light; (4) weak engineering support to the operating units brought on equipment operability concerns that were not being addressed promptly, as well as modifications that were not being properly implemented; and (5) the utility had not substantially benefitted from experiences of other utilities or from experience at its own sites.

CECo continued to implement its Integrated Management Action Plan to improve organizational and management effectiveness, business planning, and management of issues. In 1993, the licensee reorganized the corporate office and management structure at each site to establish a standard organization for each. The licensee transferred corporate engineers to the site and created a Site Vice President position accountable for developing and implementing the technical and business plans for the site. Quality assurance organizations at each site were also reorganized to provide additional staffing and improve communication with the site organizations.

The NRC monitored and evaluated operations at CECo plants under the Systematic Assessment of Licensee Performance (SALP) program and found that activities at the Byron plant exhibited generally excellent performance and that Braidwood demonstrated good performance. Performance at Dresden and Zion continued to improve. While performance at Quad Cities and LaSalle was acceptable, there were indications that it was declining.

As a result of its concern with the performance at the Quad Cities (Ill.) nuclear power plant, the NRC conducted a diagnostic evaluation team (DET) inspection in September 1993. The DET evaluated operations and training, maintenance and testing, engineering and technical support, and management and organization at the

Quad Cities units. The team found performance deficiencies and found that weaknesses in management had contributed to these deficiencies. Specifically, the team found that (1) managers accepted equipment problems without aggressively pursuing corrective actions; (2) operations managers rarely formally evaluated operability of degraded equipment; (3) engineering assessments of degraded plant hardware were not rigorous; (4) the work control process was ineffective and inefficient; (5) the effects of vibration on several plant systems had not been evaluated; (6) the large number of uncorrected component problems resulted in the degradation of safety systems; (7) significant leadership weaknesses hindered site and corporate management; and (8) previous initiatives and self-assessments to improve performance had not been successful. Although senior site managers had been aware, for some time, of many of the problems identified by DET, they had not corrected underlying root causes and improved performance. The licensee began efforts to improve the performance at Quad Cities and to develop a long term response to the DET findings. The NRC is monitoring the licensee's performance and its corrective actions.

During the past year, the NRC has closely monitored the corrective action program at the Dresden site. Close surveillance was maintained through increased inspections by the resident and regional inspectors and by the Dresden Oversight Team (DOT). The DOT consisted of personnel from Headquarters and Region III management and staff who visited Dresden quarterly to evaluate licensee performance and the status of the improvement programs. The licensee was slow at implementing improvement programs but showed continuous improvement. During 1993, both units completed extensive outages to correct several longstanding material problems that reduced equipment reliability for several years. Following these outages, both units experienced improved operations without forced outages through the end of the report period. The NRC determined that continued close monitoring is warranted until the licensee demonstrates further sustained improved performance at Dresden.

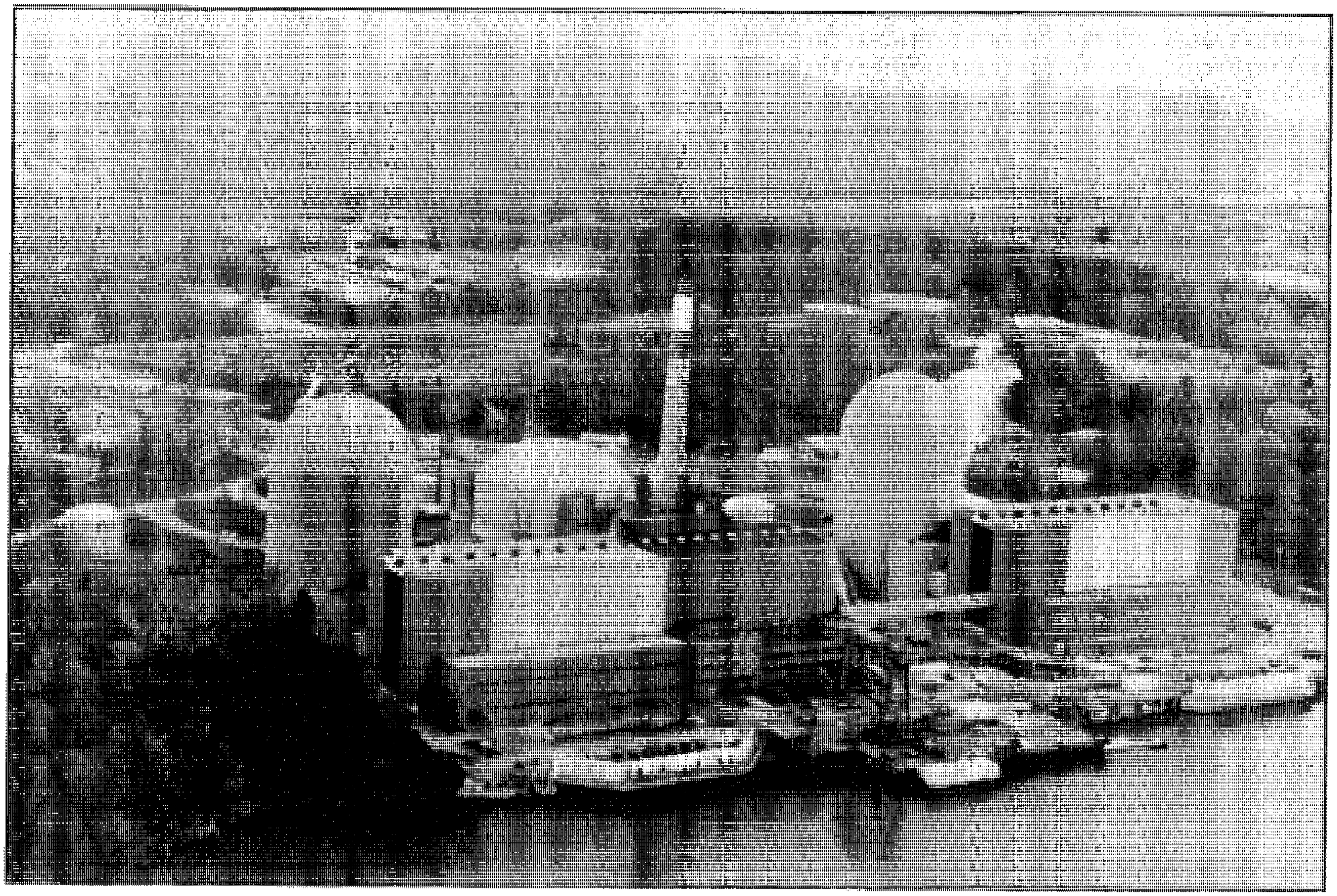
**Indian Point Unit 3.** The Indian Point Unit 3 (N.Y.) nuclear power plant, owned and operated by the New York Power Authority (NYPA) is a Westinghouse four-loop, 965 megawatt (electric) pressurized-water reactor located 24 miles north of New York City. In June 1993, the plant was added to the list of facilities which, while still authorized to operate by the NRC, warrant increased NRC headquarters and regional oversight because of declining performance.

The NRC conducted a DET inspection at the FitzPatrick plant (also owned and operated by the licensee) in the fall of 1991. Following this inspection (see above), several concerns emerging at Indian Point Unit 3 suggested problems similar to those found at FitzPatrick. The concerns grew out of a poor corrective action program that often resulted in untimely or ineffective corrective actions.

The NRC staff's SALP report for the period ending September 1992 confirmed an overall decline in performance. The licensee continued to display superior performance in the radiological controls functional area. However, the team noted weaknesses in the operations, maintenance/surveillance, emergency preparedness, engineering/technical support, and safety assessment/quality verification functional areas. The most significant weaknesses were in the engineering/technical support functional area. In general, the overall weak performance resulted from inadequate management oversight. Specifically, the licensee was not effective in implementing corrective actions for both long-standing and newly emerging issues. The weak performance was also evidenced by the escalated enforcement record of Indian Point Unit 3. Since May 1992, the NRC has issued the licensee eight Severity Level III violations, with individual Civil Penalties totaling $762,500. In January 1993, the licensee submitted a performance improvement plan to the NRC. The plan addresses the licensee's self-assessment efforts and the performance issues noted in the SALP report.

On February 26, 1993, licensee management shut down Indian Point Unit 3 to correct deficiencies associated with the anticipated transient without scram (ATWS) mitigation system actuation circuitry system (AMSAC) and with programmatic weaknesses in the surveillance testing program. The growing number of issues facing the plant prompted licensee management to keep the plant shut down while effecting plant-wide programmatic improvements.

In May 1993, the NRC conducted a Special Team Inspection at Indian Point Unit 3 and confirmed significant fundamental weaknesses at the plant. The team determined that the root causes for the declining performance were weak managerial processes, controls and skills. The team also identified two contributing causes. First, the licensee failed to identify and resolve underlying root causes for problems identified by the Quality Assurance organization. Second, the licensee did not have an effective self-assessment process, because the function was fragmented and selectively applied, and the on-site and off-site oversight committees were narrowly focused. In June 1993, the NRC issued a Confirmatory Action Letter (CAL) that documented restart commitments made by the licensee.



**The Indian Point Unit 3 (N.Y.) nuclear power plant was shut down by licensee management in May 1993 to correct certain deficiencies associated with a circuitry system and to deal with other concerns. The NRC was closely monitoring restart plans at the close of the report period. The Indian Point facility is located about 40 miles north of New York City, on the Hudson River.**

During 1993, the New York State Assembly (NYSA) held several committee hearings to discuss issues related to the NYPA. On May 6, 1993, the NYSA Environmental Conservation Committee and the Standing Committee on Energy held a joint informational hearing in the town of Cortland Manor to gather information on the performance decline at Indian Point Unit 3. Most recently, the NYSA Standing Committee on Corporations, Authorities, and Commissions and the Standing Committee on Energy scheduled two joint hearings to solicit public comment on methods of improving the oversight and accountability of the NYPA. The first of these hearings was held on October 15, 1993, in New York City and the second was held on October 25, 1993, in Albany. At the May 6 and October 25 hearings, the NRC described the conditions that led to Indian Point Unit 3 and FitzPatrick being placed on the NRC's list of facilities to be closely monitored and the steps the licensee had taken to improve the management of the facilities.

The NRC is closely monitoring the licensee's performance in restarting from the current outage at Indian Point 3. In July 1993, the staff issued the Indian Point Unit Restart Action Plan, a list of 32 plant-specific restart issues. After the licensee certifies that Indian point Unit 3 is ready in all respects to restart, the NRC will conduct a Readiness Assessment Team Inspection before determining whether or not to allow restart of the facility.

**South Texas Project.** On February 12, 1993, the Executive Director for Operations sent a Diagnostic Evaluation Team to the South Texas Project (STP), a two-unit Westinghouse pressurized-water reactor facility, located in Matagorda County, Tex. The licensees for the facility are Houston Lighting and Power Company, City Public Service Board of San Antonio, Central Power and Light Company, and the City of Austin, Tex. The operator of the South Texas Project is Houston Lighting and Power Company. Each unit is rated at 3,800 megawatts (thermal).

The NRC performed a DET inspection to evaluate continuing operational events and problems, after a decline in licensee performance during the previous two SALP periods. The NRC concluded that the performance problems stemmed from three broad causes: managerial and organizational performance, human performance, and material condition and housekeeping. Historically, hardware problems, some of which were repetitive, resulted in numerous plant trips, transients, engineering safety feature (ESF) actuations, and forced outages. Most of these system and component problems were limited to "balance-of-plant" equipment, but long-standing, safety-related hardware problems had not been fully resolved. Personnel errors resulted in reactor trips, ESF actuations, and violations of the STP Technical Specifications. Other problems and concerns pertaining to organizational performance were also noted. Over the past two years, low-level licensee and constractor personnel had committed several willful violations. There were instances of internal and external failures of communication. The NRC staff believed that the causes of the performance problems had not been fully identified.

At the time that the NRC determined that a DET was necessary, both units were shut down because of continuing problems with the auxiliary feedwater system. The licensee was operating under a confirmatory action letter and under special review by an oversight committee of NRC Region IV and Office of Nuclear Reactor Regulation staff members. This committee later undertook the responsibilities of a restart panel, in accordance with NRC procedures. During this time, the NRC was considering significant enforcement action on many broad managerial and technical issues.

Evaluation of facility operation prompted the NRC staff to place the South Texas Project on the "plant watch" list, in June 1993. The licensees addressed issues raised by the DET by issuing two supplements to the confirmatory action letter. In response to public interest, the NRC staff initiated routine public meetings with the licensee to review outstanding issues.

On August 28, 1993, the licensee submitted an operational readiness plan to address the short-term activities—specified in the confirmatory action letter, its supplements, inspection reports and the DET report—required for the restart of the units. On October 15, 1993, the licensee submitted a business plan to address the need for long term facility improvements cited in the DET report.

The NRC conducted numerous inspections in 1993 and more on-site inspections, including one by an operational readiness assessment team, to assess the operational readiness of the plant. The licensee anticipated a return to service in January 1994 for Unit 1, and in April 1994, for Unit 2.

**Diablo Canyon.** The Diablo Canyon (Cal.) nuclear power plant consists of two pressurized-water reactors, owned and operated by Pacific Gas and Electric Company, the licensee. In a letter of July 9, 1992, the licensee requested an amendment to the Unit 1 and 2 operating licenses to recapture each unit's construction period, which would, if granted, result in more than 13 and 14 years of additional operation for Units 1 and 2, respectively. The amendment request was published in the *Federal Register*, and a local group, the San Luis Obispo Mothers for Peace (SLOMFP), requested a hearing. A prehearing conference was held before the Atomic Safety and Licensing Board (ASLB) and two contentions were admitted into the proceeding. One contention concerned the effectiveness of the licensee's maintenance and surveillance program. The other concerned the adequacy of interim corrective measures implemented in lieu of Thermo-Lag fire barrier material. (See above.) The counsel for the intervenor, SLOMFP, presented numerous licenseeinitiated non-conformance reports, licensee event reports and NRC inspection reports as evidence. NRC Headquarters and Region V staff prepared and presented testimony at the hearings, held in August 1993. The ASLB's decision is expected in 1994.

**Palo Verde Unit 2.** On March 14, 1993, while at approximately 99 percent power, Palo Verde Unit 2 (Ariz.) experienced a steam generator tube rupture in one of the unit's two steam generators. The tube rupture resulted in an initial primary-to-secondary system leakage of approximately 240 gallons-per-minute. The NRC sent an Augmented Inspection Team (AIT) to investigate the event. As a result of the team's review of the licensee's response to the event, the NRC issued an information notice to all pressurized-water reactor licensees detailing emergency operating procedure weaknesses.

The Palo Verde plant (three units) is the only Combustion Engineering System 80 design in the United States. The System 80 steam generators are recirculating U-tube steam generators which contain approximately 11,000 high-temperature, mill-annealed alloy 600 tubes. The rupture can be characterized as an "axial fishmouth crack" in the freespan area, with a total crack length of approximately eight inches. This tube rupture was noteworthy for two reasons: (1) the tube had been inspected at the previous refueling outage (15 months earlier) with no indications of cracking present; and (2) inspections following the tube rupture revealed many crack indications in both steam generators, primarily located along an arc in the upper portion of the outer periphery of the tube bundle.

The NRC staff extensively studied the results from the two latest steam generator tube inspections performed at the Palo Verde site. The staff issued a safety evaluation authorizing interim operation of Unit 2 until a mid-cycle steam generator tube inspection is conducted. Confirmatory Action Letters were issued to hold the licensee to commitments regarding inspection methodology and restart decisions. The NRC continues to carefully evaluate the root-cause determinations and corrective actions taken by the licensee in its assessment of potential generic applicability to the other Palo Verde units, as well as other plants.

## TVA Projects

In September 1985, the NRC staff issued a letter to the Chairman of the Board of Directors of the Tennessee Valley Authority (TVA), discussing significant continuing weaknesses in TVA performance and stating that management of the TVA nuclear program was ineffective. By that time, TVA had already placed the Browns Ferry (Ala.) and Sequoyah (Tenn.) nuclear plants in a cold shutdown status and had made commitments to the NRC to implement comprehensive corrective actions. TVA had confirmed that these plants would not be restarted without NRC concurrence. The number and complexity of relevant issues were not limited to the operating reactors, since questionable construction practices had also been identified at the TVA's Watts Bar (Tenn.) project.

**Browns Ferry.** Unit 2 was shut down in September of 1984 for a planned refueling outage. Units 1 and 3 were shutdown in early 1985, because of equipment problems and operational incidents. In March of 1985, TVA volunteered to maintain all three units in a shutdown condition until corrective actions could be effected to resolve serious NRC concerns regarding TVA's ability to safely operate and manage the Browns Ferry facility.

Having been shut down for nearly seven years, Browns Ferry Unit 2 was restarted on May 24, 1991, following extensive NRC review and inspection of TVA's corrective action programs. TVA focused its efforts at Browns Ferry exclusively on Unit 2 to develop and implement necessary corrective actions; restoration of Unit 3 and then Unit 1 were to follow. In August of 1991, Unit 2 returned to normal full power commercial operation, having successfully completed a Power Ascension Test program. In a letter of June 30, 1992, the NRC notified TVA that Unit 2 had demonstrated excellent plant performance and would, therefore, be removed from the list of plants warranting close NRC monitoring. However, the NRC informed TVA that Units 1 and 3 would continue to remain in the close monitoring category and would require explicit NRC authorization to be operated.

On January 29, 1993, Browns Ferry Unit 2 was shut down for its first refueling outage following restart from the extended recovery outage. This outage was the milestone for completing numerous post-restart commitments. These large plant modifications included installation of the hardened wetwell vent, completion of the control room design upgrade, and installation of a new plant process computer including full-function safety parameter display systems. The plant was restarted on schedule in late May 1993, and has operated well.

TVA scheduled to restart Browns Ferry Unit 3 in August 1995 and is applying the same corrective action plans and criteria used to effect the Unit 2 restart. The NRC staff will review any changes proposed by TVA.

**Sequoyah.** Sequoyah Units 1 and 2 were voluntarily shut down in 1985 to address environmental qualification issues, performance weaknesses, and management problems. Both units were restarted in 1988. From October 1986 to May 1989, Sequoyah was on the NRC's list of plants requiring close monitoring because of regulatory concerns about declining performance.

Performance improved into 1991, but then slowly declined. In 1992, an increase in the number of plant events, and escalated enforcement actions caused by poor procedure adherence, lack of attention to detail, and configuration control problems, caused increased NRC staff concerns. These concerns related to a lack of leadership and an inability to effectively communicate expectations within the organization, especially in operations, maintenance and engineering.

These concerns were heightened further by three dual-unit events that occurred in 1992 and 1993. The first was an inadvertent water injection into the control air system, the second was a simultaneous trip of both units during breaker testing in the switchyard, and the third was an unanticipated steam leak in the secondary system. The water injection event was caused by failure to adequately maintain air system components, and caused one unit to trip and the other unit to automatically reduce power. The breaker-testing event caused both units to trip.

When Unit 2 tripped in 1993 because of a steam leak in the secondary system, the licensee found a significant deficiency in the process to monitor and predict weakening of steam line piping caused by steam impingement on the inside surface of the piping (the socalled erosion/corrosion program). Since these program weaknesses was evident in both units, TVA voluntarily shut down Unit 1, and management agreed that neither unit would be restarted until various issues were addressed and the NRC determined that the licensee identified and corrected the root cause of the problems. TVA determined that the root cause was the failure of management to clearly assign responsibilities and provide appropriate oversight and direction for monitoring and maintaining the balance-of-plant piping.

While the plant was shut down, TVA performed evaluations that revealed problems in hardware and other areas. The problems were grouped into six focus areas: Balance of Plant; Operations; Backlogs; Programs; People, Organization and Culture; Corporate/Site Interface.

The licensee identified ineffective resource management and ineffective personnel and management performance as the underlying causes of the problems in these areas. The licensee adopted a comprehensive performance improvement plan, including a Restart Plan and a Post-Restart Site Improvement Plan. As a result, many site technical programs have been restructured and reorganized to more clearly assign responsibilities, and management focused on creating an atmosphere conducive to improved performance.

The NRC established a Restart Panel to monitor activities at the plant. The NRC also conducted an Operational Readiness Assessment Team inspection to confirm the overall effectiveness of plant programs to correct the deficiencies and to conduct power operation. In general, the results of this inspection were favorable. TVA management and the NRC will continue to closely monitor the effectiveness of these changes.

On October 18, 1993, the NRC approved TVA's statement that the plant had completed all items necessary for restart of Unit 2. TVA restarted Unit 2 and scheduled the restart of Unit 1 for early 1994. The changes will likely yield significant improvements.

**Watts Bar.** Having restarted Sequoyah and Browns Ferry Unit 2, TVA stepped up its activities on Watts Bar Unit 1 and established a fuel-loading date for late summer or early fall of 1994.

Although construction of Unit 1 was complete in 1985, extensive corrective programs were required to resolve deficiencies from allegations, employee concerns, inspections and audits. The staff reviewed and approved all 28 corrective action programs. Details of the staff's review may be found in the latest supplement to the Watts Bar Safety Evaluation Report (NUREG–0847). TVA must implement all corrective actions programs before the NRC will issue an operating license.

The NRC staff is implementing an extensive inspection program at Watts Bar to ensure that the plant has been built in accordance with applicable NRC requirements.

**Bellefonte.** In July 1988, TVA informed the NRC that the TVA Board of Directors had decided to defer construction of Bellefonte Units 1 and 2 (Ala.) because of lower-than-expected load forecast for the near future, cost-cutting efforts to improve the TVA's financial position, and the TVA's effort to hold electric rates constant for a specific period of time. TVA continued activities at the plant during the deferral period, and the NRC staff continues performing periodic inspections at the site.

On November 8, 1990, TVA met with the NRC staff and presented a plan to resume construction of the Bellefonte plant. At that time, Unit 1 was 80 percent complete, and Unit 2 was 45 percent complete. TVA evaluated three options for completing Bellefonte: (1) completing the plant as a nuclear facility, (2) converting the plant to a combined-cycle gas facility, or (3) converting the plant to a pulverized coal facility. Following this evaluation, TVA decided to complete the two Bellefonte units as nuclear units.

In a letter of December 4, 1990, TVA informed the NRC staff that it would submit technical position papers for NRC review. The technical areas were those in which differences between the expectations of TVA and those of the NRC, in technical approach or criteria, could significantly affect the schedule and the scope of work necessary to complete and license the two Bellefonte units. TVA requested that the NRC staff review the position papers and provide docketed agreements or comments on each of TVA's positions. TVA formally submitted 14 position papers and met with the NRC staff to discuss these papers.

The NRC staff reviewed the TVA position papers and issued responses clearly defining its position and providing comments where agreement could not be reached. TVA has incorporated the agreements into the FSAR (Amendment 30, dated December 20, 1991). NRC staff is also defining the inspection activity that will be needed when TVA resumes construction of Bellefonte.

On March 23, 1993, TVA notified the NRC that it plans to complete Bellefonte Units 1 and 2. TVA's plans call for loading fuel in Unit 1 by 1998 and in Unit 2 by 2002. Following receipt of TVA's letter, the staff prepared an inspection plan and conducted a special reactivation inspection. The staff concluded that TVA's knowledge of Bellefonte structures, systems, and components was adequate for reactivation of Bellefonte.

## GE Generic BWR Power Uprate Program

The NRC staff continued to work closely with both General Electric and industry representatives to ensure the safe, efficient implementation of the generic power uprate program. In September of 1992, the staff issued the first license amendment allowed under the boiling-water reactor (BWR) power uprate program to Enrico Fermi Unit 2 (Mich.). During fiscal year 1992, the staff subsequently received applications for power uprates for the Susquehanna (Pa.) and FitzPatrick (N.Y.) facilities. At the close of fiscal year 1993, the staff was completing its review of these submittals. The staff is also reviewing submittals for the Peach Bottom (Pa.), Washington Nuclear Unit 2 and Nine Mile Point Unit 2 (N.Y.) facilities.

To date, 20 BWR units, representing almost 1,000 megawatts of new generating capacity, have expressed interest in requesting license amendments under the generic uprate program. This cumulative increase in generating capacity would be equivalent to the capacity of a new nuclear power facility, and would be achieved at a small fraction of the cost of new construction. Industry estimates indicate that a 5 percent increase in power for a 1,000-megawatt plant could save a utility as much as $120 million to $180 million over a 25-year period, by displacing other generating sources, such as coal or oil.

## PLANT LICENSE RENEWAL

The U. S. Department of Energy has projected an increase in national demand for electricity of 100,000 megawatts in the next decade. In light of the anticipated demand, the electric utility industry has urged the NRC to expedite preparations for license renewal applications. According to the industry, if the current operating license for a plant is not renewed, the licensee will need a lead time of 10-to-12 years before the license expires to plan for replacement power alternatives and capital acquisition.

The prospect of renewing operating licenses for nuclear power plants has long been a top priority for the NRC and the nuclear industry. Within the next 20 years, many commercial nuclear power plants will have reached the standard 40-year term of their operating licenses, a figure adopted by Congress in the Atomic Energy Act of 1954, as amended. The Act permits the NRC staff to renew operating licenses but does not set forth a process to be followed; for that reason, the NRC is giving substantial attention to defining the process for review of licensee renewal applications.

To help meet the electrical energy needs of the nation in the early 21st century, some utilities are now carefully examining steps to extend the life of their nuclear power plants beyond 40 years. The NRC's parallel activities include rulemaking proceedings, regulatory guidance development, and "lead-plant" reviews.

### Rulemaking

The NRC published the proposed license renewal rule (10 CFR Part 54), in the *Federal Register*, July 17, 1990; the final rule was published in December 1991. The basic premise of the final rule is that, for age-related degradation unique to the period of extended operation, the regulatory process ensures that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety. The final rule also states that each plant's current licensing basis must be maintained during the renewal term, in part through a program to manage age-related degradation for systems, structures and components that are important to license renewal.

Since publishing the final rule, the staff has been conducting various activities for implementing the license renewal rule. These actions have included revision of the regulatory guide and Standard Review Plan (SRP), interaction with the lead plant licensees, and reviews of industry technical reports sponsored by NUMARC. Over the past year, the NRC found a number of significant policy issues.

On December 7, 8, and 18, 1992, the staff briefed the Commission on the status of the various license renewal activities and on the staff's plans to resolve key license renewal issues. The staff informed the Commission that a senior management review group would address these issues. The staff also stated that it would interact with NUMARC in public meetings to obtain the industry's views. In a staff requirements memorandum (SRM) of December 21, 1992, the Commission endorsed the staff's senior management review, identified issues for consideration by the group, and directed the staff to submit its recommendations to the Commission.

The staff submitted its recommendations for implementing the final rule to the Commission, in SECY-93-049, dated March 1, 1993, and SECY-93-113, dated April 30, 1993. In the SRM of June 28, 1993, the Commission directed the staff to hold a public workshop to evaluate alternative approaches to the better use of existing licensee activities and programs as a basis for concluding that aging will be acceptably addressed. The public workshop was held on September 30, 1993, and attended by over 170 representatives from the nuclear industry, engineering and consulting firms, Federal and State governments, and public interest groups. DOE, NUMARC, and Yankee Atomic Electric Company (YAEC) made formal presentations at the workshop. The use of existing licensee programs in the license renewal process was a central theme of each presentation, and DOE, NUMARC and YAEC affirmed that a rule change is necessary. The staff was evaluating the results of the workshop and preparing recommendations for the Commission, at the close of the report period.

The NRC is also putting forth environmental initiatives relevant to license renewal, in the context of National Environmental Policy Act (NEPA) requirements. The NRC has proposed amendments to the "Environmental Protection Regulations For Domestic Licensing and Related Regulatory Functions" (10 CFR Part 51), and a generic environmental impact statement (GEIS), in support of the proposed amendment. This rulemaking is based on the belief that certain environmental issues should be treated generically, rather than in each plant-specific licensing review. The NRC published the proposed amendments and a draft GEIS for public comment in September 1991, and held a public workshop on them in November 1991. The NRC received numerous public comments on the amendments and is completing its response to these comments and revising the documents as appropriate. Public comments have also raised concerns regarding NRC policy for treatment of environmental issues. The staff plans to conduct workshops with the commenters to resolve these policy issues.

### Regulatory Guidance Development

To aid in implementing the license renewal rule, 10 CFR Part 54, the NRC has developed a draft regulatory guide and a draft Standard Review Plan for License Renewal (SRP–LR), in parallel with (1) the renewal rulemaking, (2) reviews of industry technical reports, and (3) lead plant and owners groups activities. The draft regulatory guide and the draft SRP–LR were published for comment in December 1990. The staff plans to publish a revised draft regulatory guide and SRP–LR for public comment, after resolving policy issues and determining the need for further license renewal rulemaking.

The NRC published for public comment a draft regulatory guide and a draft Environmental Standard Review Plan for license renewal (ESRP–LR), in September 1991. The staff expects to complete the final regulatory guide and ESRP-LR about six months after issuing the final Part 51 rule and the GEIS.

### Industry Technical Report Reviews

NUMARC prepared 11 industry reports and requested NRC review and approval of them, so that each can be referenced in a license renewal application, thus obviating any need for an entirely plant-specific evaluation. The industry reports addressed aging for PWRs and BWRs on the reactor vessel and its internals, the reactor coolant system, the containment, and Class I structures and cables in harsh environment. A screening methodology report was also provided.

The NRC has completed the first review of all 11 industry reports. The staff gave NUMARC comments on the initial versions of the proposed reports and met with NUMARC on each report to clarify its review findings. In response to NRC comments, NUMARC revised and resubmitted 10 of the reports addressing aging of specific structures and systems. However, in order to make best use of the technical information and agreements from the NUMARC program, the Commission, in a SRM of June 28, 1993, approved the staff's recommendations that instead of writing an SER for each industry report, the staff will incorporate appropriate technical information from the industry reports into the draft SRP for license renewal. This approach is expected to result in a single document that will include industry report insights and also establish the staff's review acceptance criteria.

### Lead Plant Reviews

The Yankee-Rowe (Mass.) and Monticello (Minn.) nuclear power plants were initially identified as the industry "lead plants" in activating the license renewal procedure. Yankee Atomic Electric Company decided to permanently shut down the Yankee Rowe plant on October 1, 1991. On November 3, 1992, Northern States Power (NSP) decided to delay indefinitely plans to submit a license renewal application for Monticello.

In October 1992, the Babcock & Wilcox (B&W) Owner's Group announced that they had begun a generic license renewal program for the seven currently operating B&W facilities. The program will attempt to resolve generically (for these B&W facilities) as many technical and process issues as possible. The staff received and reviewed screening methodology reports in 1993. A major goal of this program is to have a plant-specific application, which would incorporate generic technical reports, prepared and submitted to the NRC in 1998.

In March 1993, Baltimore Gas and Electric Company, the owner of the Calvert Cliffs nuclear plant, also submitted a screening methodology which the staff has reviewed. The staff also received information on the associated facility procedures and results demonstrating application of the screening methodology.

The staff is preparing SERs on both of these license renewal screening methodology submittals.

Other owner groups, representing Westinghouse and General Electric plants, have indicated that they also will become actively involved with license renewal.

## IMPROVING THE LICENSING PROCESS

### Standardization of Reactor Design

The Commission strongly endorses regulatory policies that encourage industry to pursue standardization of next-generation reactor designs. Standard designs are expected to benefit public health and safety by (1) concentrating industry resources on common approaches to design problems that have wide application, (2) stimulating adoption of sound construction practices and quality assurance, (3) fostering constantly improving maintenance and operating procedures, (4) and permitting a more efficient and effective licensing and inspection process. In this regard, the NRC plans to achieve the benefits of standardization with the Design Certification process which, along with Early Site Permits and Combined Licenses, constitute the major provisions of the new licensing process in 10 CFR Part 52. In November 1993, the NRC issued an advanced notice of proposed rulemaking that requested comments on a draft-proposed standard design certification rule. The NRC is also developing the inspections, tests, analyses, and acceptance criteria (ITAAC) to verify that a facility, which has referenced a certified design, has been constructed and will be operated in conformity with the license and the Commission's rules and regulations.

## Next-Generation Reactor Designs

The staff is reviewing four applications for design certification under Subpart B of 10 CFR Part 52. Two of the applications are for evolutionary light-water reactor designs (ABWR and System 80+) and the other two are passive light-water reactor designs (AP600 and SBWR). The following discussion describes the status of each of these reviews, including the Electric Power Research Institute (EPRI) Advanced Light-Water Reactor Program.

**ABWR.** GE Nuclear Energy (GE), in cooperation with its international technical associates, is developing an advanced boiling water reactor (ABWR). The staff is nearing completion of the ABWR review and expects to issue the final safety evaluation report (FSER) in 1994.

**System 80+.** From March 1989 to March 1991, Combustion Engineering (CE), a manufacturer of pressurized-water reactors, submitted documentation to the NRC in support of an application for final design approval and design certification of its System 80+ nuclear power plant design. The NRC staff issued the draft safety evaluation report (DSER) in September 1992. The staff and CE have resolved most of the issues in the DSER, and the staff plans to issue an FSER in 1994.

**AP600.** Westinghouse Electric Corporation submitted an application for final design approval and design certification of its AP600 design in June 1992. The AP600 is a 600 megawatt-electric pressurized-water reactor plant incorporating passive safety systems and features into its design. In support of the passive design, Westinghouse has established a test program for the AP600 which includes separate-effects (SE) experiments on the passive approach and two integral systems test (IST) programs (see "Testing for Passive Designs," later in this chapter). During its review, the staff issued approximately 1,200 requests for additional information to support its evaluation of the application. Westinghouse has responded to most of the questions raised by the staff. The staff is continuing its review, and expects to issue a DSER discussing its findings in 1994.

**SBWR.** GE Nuclear Energy submitted an application for final design approval and design certification of a simplified boiling-water reactor (SBWR) design on August 27, 1992, and furnished supplements to it on February 25, February 28, and May 7, 1993. The SBWR is a 600 megawatt-electric advanced reactor design that employs passive features, such as gravity flow and natural convection, to perform essential safety functions. The staff docketed GE's application for design certification in May 1993. The staff is continuing its review, and expects to issue a DSER discussing its findings in 1994.

**EPRI Advanced Light-Water Reactor Program.** EPRI prepared a compendium of technical requirements for advanced light water reactors, referred to as the ALWR Utility Requirements Document (URD). These requirements are intended to apply to the design of future evolutionary and passive ALWR power plants. Volume I of the URD, "ALWR Policy and Summary of Top-Tier Requirements," is a management-level synopsis of the URD, covering design objectives and philosophy, the overall physical configuration and features of future commercial nuclear power plant design, and the steps needed to apply the proposed ALWR design criteria to a functioning power plant. Volume II contains the utility design requirements for an evolutionary nuclear power plant (with a power rating of approximately 1,350 megawatts-electric). Volume III contains the utility design requirements for nuclear power plants (of approximately 600 megawatts-electric) in which passive safety features and systems will be used for the ultimate safety protection of the plant. The URD also proposes to resolve certain unresolved safety issues and generic safety issues and delineates ways of complying with 10 CFR Part 52.

The NRC staff issued the FSER on Volumes 1 and 2 (NUREG-1242) of the EPRI ALWR URD in August 1992. The staff issued the FSER on Volume III for ACRS and Commission review in August 1993 and scheduled to publish it as Volume 3 of NUREG-1242 in 1994.

## Pre-application Reviews

The staff is conducting pre-application reviews of four advanced reactor designs (MHTGR, PRISM, CANDU 3, and PIUS) in response to the Commission's "Statement of Policy for the Regulation of Advanced Nuclear Power Plants," which calls for early Commission review and interaction with potential applicants for the licensing of advanced designs. The staff performed a preliminary assessment of these designs and set out, in SECY-93-092, April 8, 1993, the major policy and technical issues pertaining to these four advanced reactor designs, in the context of current regulatory requirements. The Commission's decisions on these issues gave the pre-applicants important information for their design development and provided the NRC staff with the necessary guidance regarding the pre-application reviews. The following discussion deals with each of these reviews.

**PRISM.** The Department of Energy (DOE) submitted the Power Reactor Innovative Small Module (PRISM) design concept to the NRC for a pre-application review, under provisions of the NRC Statement of Policy for the Regulation of Advanced Nuclear Power Plants. PRISM is a liquid-sodium-cooled reactor with a ternary metal-alloy-fueled core. The proposed PRISM plant design would integrate nine reactor modules, producing 471 megawatts (thermal) each, with three steam turbine generator sets to produce a total plant output of 1,395 megawatts (electric). Plant design and performance will be highly automated, with little reliance on operators for response to most off-normal events and provision for the passive response of systems to transient events, so that power excursions will be kept small and promptly shut down and decay heat removal will be assured with high reliability.

The NRC issued a draft pre-application safety evaluation report (PSER) on PRISM in November 1989. In 1990, DOE submitted two new amendments to its Preliminary Safety Information Document (PSID), in response to open issues in the draft PSER. The staff reviewed the two new amendments and the PSID in completing the final PSER. The staff reviewed a conceptual design, but did not approve the design, upon issuing the PSER. Instead, the staff took note of certain key safety issues, gave DOE guidance on applicable licensing criteria, and assessed the adequacy of the pre-applicant's research and development programs.

On October 19, 1993, the Commission released the draft final PSER to the Advisory Committee on Reactor Safeguards (ACRS) and to the public. On November 4, 1993, the ACRS reviewed the PSER and concluded that no obvious impediments to licensing the PRISM design had been identified. The final PSER was scheduled for publication in December 1993.

**CANDU 3.** In a letter of May 25, 1989, Atomic Energy of Canada, Limited (AECL), Technologies informed the NRC of its intent to seek design certification of the CANDU 3 power plant design, under provisions of 10 CFR Part 52.

The CANDU 3 design is a pressurized-water reactor, rated at 450 megawatts (electric), consisting of 232 reactor fuel channels, two steam generators, four electrically driven heat transport pumps, four reactor inlet headers, and two reactor outlet headers. The design employs natural uranium fuel, heavy-water moderator and reactor coolant, computercontrolled operation, and refueling-without-shutdown. Major technical issues to be resolved include those involving reactivity feedback and control, reactor shutdown reliability, and online refueling.

NRR staff met repeatedly with AECL Technologies and with the Atomic Energy Control Board, the Canadian regulatory body, to explore various aspects of the CANDU 3 design. AECL Technologies expects to file a design certification application in 1994.

**PIUS.** In October 1989, ABB Atom (Sweden) asked that the NRC review the PSID related to the Process Inherent Ultimate Safety (PIUS) reactor design, under provisions of the Advanced Reactor Policy Statement, to determine whether the design could be licensed. ABB/Combustion Engineering, the U.S. representative for ABB Atom, presented the PIUS design to the NRC for pre-application review.

PIUS is an advanced pressurized-water reactor (PWR) design which exploits physical phenomena to accomplish control and safety functions in a nuclear power plant that are usually performed by mechanical means. The PIUS design consists of a reactor module (containing the core) submerged in a large pool of highly borated water, intended both for core cooling and for effective reactor shutdown. The reactor module will be open at the bottom and again at the high point of the "hot leg." At these two openings, density locks will prevent mixing of the coolant and pool water, under normal operating conditions. The density locks will not include a physical flow barrier in the density locks, but the difference in density between the reactor water and the cooler borated pool water will maintain a stationary interface. During certain transient conditions, the density difference would be overcome and the borated water would flow into the core and shut down the reactor.

In May 1993 the Commission directed the staff to document its evaluation of the preapplication review of the PIUS design, to date, and to end all other activities until a design certification application is submitted by ABB/Combustion Engineering.

**MHTGR.** DOE submitted the Modular High Temperature Gas-Cooled Reactor (MHTGR) design to the NRC, a concept which features a helium-cooled, graphite-moderated 350-megawatt (thermal) standard reactor module. One objective of the design is to meet the Protective Action Guidelines of the Environmental Protection Agency at the exclusion area boundary during an accident, with a minimal reliance on active systems and without reliance on operator actions. A high reliance will be placed on the containment strength and on the reliability of the individual fuel particles, which will be coated microspheres embedded in a graphite fuel block (identical in shape to those formerly used in the Fort St. Vrain (Colo.) reactor). Other key features of the design are its passive reactor shutdown characteristics and a passive decay heat removal system. The MHTGR design may not require the conventional light-water reactor low-leakage containment building. The staff issued a draft pre-application safety evaluation report in March 1989.

## Early Site Permits

On April 18, 1989, the Commission issued, in 10 CFR Part 52, the regulatory framework for obtaining early resolution of site-related issues. In 1993, the NRC began an upgrading of its capabilities for managing and conducting environmental and site-licensing reviews, and for accessing and analyzing requisite geographical and land use information. The NRC continues to monitor the progress of the Department of Energy demonstration program, looking to identify an initial applicant for an early site permit.

## Standard Review Plan Update And Development Program

The NRC established the Standard Review Plan Update and Development Program (SRP–UDP) in fiscal year 1991 to update the Standard Review Plan (SRP) for use in reviewing future reactor design applications. The revised SRP will incorporate changes made in the regulation of the nuclear power industry since the 1981 revision of the SRP and will include guidance for the staff in reviewing unique technology, or the unique application of existing technology, in future reactor designs.

In fiscal year 1992, the staff established a modification data base of SRP information and will keep the data up-to-date. A text retrieval system that provides the capability to conduct a full text search of generic regulatory documents was installed on the NRC's local area network. Procedures were also developed for revising and developing SRP sections and published as NUREG–1447.

In fiscal year 1993, the staff reviewed generic regulatory documents to identify information applicable to the SRP. The information was sorted by SRP section and loaded into the modification data base. A list of codes and standards cited in regulatory documents was completed and published as NUREG/CR–5973. The current SRP combined with the work completed to date in the SRP–UDP, as described above, form an interim SRP the staff will use to review future reactor designs. This information allows staff reviewers to identify those changes to the current SRP that will be analyzed and incorporated in the revised SRP. The reviewer will use these data with the SRP text retrieval system of generic regulatory documents in conducting design certification reviews.

With the completion of the interim SRP, the primary objectives in fiscal year 1993 were to maintain current previously collected data, and to continue to enhance the usefulness of the interim SRP. The staff reviewed newly issued generic regulatory documents to find new SRP information, and reviewed new versions of codes and standards to update the listing of code and standard citations. The staff continued to improve the SRP by (1) searching for SRP information in the SERs for evolutionary advanced light water reactor designs, (2) preparing required or recommended changes to the SRP by analyzing SRP information from generic regulatory documents and design certification SERs, and (3) comparing the latest codes and standards to the codes and standards referenced in the existing SRP and regulatory guides for selected codes and standards issued by the Institute of Electrical and Electronics Engineers (IEEE) and the American Society of Mechanical Engineers (ASME).

In fiscal year 1994, the staff will continue improving the SRP by preparing required or recommended changes for all sections and comparing of selected codes and standards for the remaining code groups.

## Technical Specifications Improvements

In July 1993, the NRC issued a final policy statement on technical specifications improvements for nuclear power reactors. Under NRC regulations, technical specifications are incorporated into the utility's operating license for a nuclear reactor facility. The technical specifications specify the safety limits, limiting conditions for operation, surveillance requirements, design features, and administrative controls necessary to ensure the safe operation of the facility. The final policy statement encourages licensees to undertake a voluntary program to update their technical specifications so as to make them consistent with the improved vendor-specific standard technical specifications (STS) issued by the NRC in September 1992. The improved specifications were based on the criteria in the interim policy statement, published in February 1987. This statement provided for the improvement of the STS and set up a parallel program for interim improvements to plantspecific technical specifications, prior to converting to the improved STS. The staff prepared the improved STS in order to raise the level of safety in nuclear power plants by making technical specifications clearer and easier to use, and to focus them more sharply on safety concerns.

In September 1992, the staff completed the improved STS for each of the nuclear steam supply system (NSSS) vendors—Babcock & Wilcox (NUREG–1430), Westinghouse (NUREG–1431), Combustion Engineering (NUREG–1432), and General Electric (NUREG–1433 for the BWR/4 model and NUREG–1434 for the BWR/6 model). The improved STS embody the following improvements: (1) technical specifications are presented using a tabular format, based on human factors principles, rather than a narrative format; (2) guidance is provided on the use and applicability of the STS; (3) operational requirements that do not meet the criteria for inclusion in the STS are placed in licenseecontrolled documents; (4) the bases for technical specifications more explicitly delineate the relationship between operational requirements and safety analyses; and (5) there is greater consistency between the reactor-plant technical specifications and those for the NSSS vendor designs. The plants within the various owners groups that have volunteered for "lead plant" conversion to the improved STS include Crystal River Unit 3 (Babcock & Wilcox owners group), Zion Units 1 and 2 (Westinghouse owners group), San Onofre Units 2 and 3 (Combustion Engineering owners group), and Hatch Unit 2 (General Electric BWR/4 owners group). All BWR/6 plants of the General Electric owners group have chosen to convert to the improved STS at the same time. Peach Bottom Units 2 and 3, Browns Ferry Unit 2, and Vogtle also anticipate complete conversions to the improved STS in 1994 and 1995.

Plants adopting the improved STS will relocate certain requirements to licenseecontrolled documents, such as the final safety analysis report. The Commission has, therefore, directed the staff to ensure that licensees have adequate programs in place to exercise effective control over such relocated requirements. NUMARC coordinated an industry effort to develop guidelines for internally evaluating plant design changes or changes to procedures, in accordance with 10 CFR 50.59. The guidelines were published in June 1989, as NSAC-125, "Guidelines for 10 CFR 50.59 Safety Evaluation," and are in use by a number of licensees. In December 1992, the staff issued improved inspection guidance for appraising licensee safety evaluation programs.

The NRC is also continuing too improve existing technical specifications by preparing and promulgating "line-item" generic STS improvements. Licensees may request license amendments to selectively adopt these STS improvements as "line-item" improvements to their existing technical specifications. Examples of line-item improvements include extending surveillance intervals and outage times, as for the reactor protection system and engineered safety features actuation system instrumentation or for relocating to licensee-controlled documents the technical specifications that address such administrative matters as component lists, organization charts, or the reactor vessel material specimen withdrawal schedule.

The NRC is evaluating means to improve technical specifications by applying risk insights in preparing technical specification requirements for low-power and shutdown operations, for advanced reactor design, and for one operating reactor (South Texas).

The NRC staff also encourages and monitors industry initiatives to apply risk insights to regulatory requirements and to develop risk-based technical specifications.

## IMPROVING NRC ANALYTICAL CAPABILITY

The engineering community has used computer codes to analyze the performance of engineered structures and systems for many years. Computer codes allow structures and systems to be modeled, and their design capabilities to be determined, without subjecting the actual facility to the conditions under scrutiny. Models are the only practical means available to examine the response of a facility, because actual tests of reactor accidents are either impractical or physically impossible.

In 1993, the staff continued to expand its use of advanced computer codes for safety analyses of operating reactor events and predictions of advanced reactor behavior. In NRR, the Analytical Support Group (SASG) continued to build its analytical capacity in several areas. SASG performed sensitivity calculations of severe accident scenarios in several advanced reactor designs and began to do the thermal-hydraulic calculations of both the new designs and the test facilities that will be used to support the licensing process. SASG evaluated the capabilities of degraded emergency core cooling systems in operating reactors and studied steam generator tube rupture events in pressurized-water reactors.

The Office of Nuclear Regulatory Research increased the number of staff members who are using high-performance workstations to run the advanced computer codes and added new members with broad backgrounds in code development and performance. RES members performed analyses of advanced light water reactor designs, and also began to investigate the behavior of the CANDU 3 heavy water reactor using computer codes provided by Atomic Energy of Canada, Ltd. Research is devoting particular attention to assessing codes for the passive advanced light-water reactors and is analyzing operating and advanced passive plants, as part of accident management and probabilistic risk assessment studies.

The Office for Analysis and Evaluation of Operational Data (AEOD) is continuing to upgrade the NRC simulators and develop and use the Nuclear Engineering workstation and the RELAP5 desktop analyzer. In coordination with NRR and RES, AEOD is also using and developing the Reactor Safety Assessment System, which is used by the reactor safety teams in the NRC operations center and by regional teams to monitor the status of critical safety functions (CSFs) during reactor transients and the availability of success paths needed to maintain or restore the CSFs.

The Division of High-Level Waste Management in the NRC's Office of Nuclear Materials Safety and Safeguards is continuing a pilot program using high-performance computer workstations integrated with staff personal computers and special peripheral equipment. This program supports high-resolution three-dimensional visualization technology, geosciences information systems, and complex mathematical natural systems modeling and engineering design, for computer-aided studies and reviews of radioactive waste sites and facilities.

### Testing for Passive Designs

The requirements for certification of advanced reactor designs, under 10 CFR 52.47(b)(2), include demonstration that the reliability of each safety feature of the design has been confirmed through analysis, testing, experience, or a combination thereof, and that sufficient data exist on the safety features to confirm the accuracy of the analytical tools used in safety analyses. Both the AP600 and the SBWR designs rely on passive systems for reactor safety. Accordingly, the vendors for both designs have developed testing programs to provide data to satisfy the requirements of 10 CFR 52.47(b)(2). The NRC is monitoring the vendors' test programs by the procedure described in SECY-91-273 and is reviewing these test programs to determine if they will yield the necessary data. The staff will also examine the experimental data, when available, to ensure that the codes were adequate.

Westinghouse's test program for the AP600 includes separate-effects (SE) experiments on several of the key systems and components involved in the passive safety approach. These tests will examine the performance of the passive residual heat removal (PRHR) system, the core makeup tanks (CMTs), the automatic depressurization system (ADS), and the passive containment cooling system (PCCS). Two integral systems test (IST) programs are also planned. A low-pressure IST facility is being constructed at Oregon State University to study the behavior and interactions of the safety and important non-safety systems at low pressures corresponding to the later stages of several accident sequences. A high-pressure, full-height IST facility is also under construction at the Societa Informazione Esperienze Termoidrauliche (SIET) laboratories in Piacenza, Italy, to examine the behavior of the passive safety systems during the high- pressure phase of accidents. Testing in both integral facilities is due to be concluded in 1994. The staff continues to evaluate these programs.

GE Nuclear Energy designed a broad testing program to support its SBWR design. GE completed much of the testing, which it planned to submit as part of the SBWR application including SE experiments on the unique squib-type, explosive-actuated depressurization valves used in the SBWR ADS, and SE heat transfer tests related to the operation of the SBWR PCCS. Inservice testing (IST) programs have also been carried out at the Toshiba facility in Japan and at GE's San Jose site to study the behavior of the PCCS and the gravity-driven cooling system (GDCS), respectively. Further SE tests are planned at SIET in a new facility, PANTHERS, and at a test facility, PANDA, which is under construction at the Paul Sherrer Institute (PSI) in Wuerenlingen, Switzerland. The staff has identified several other tests which must be included in the GE test program. GE must also address certain questions regarding tests conducted before the staff's review.

The NRC will conduct confirmatory research for both the AP600 and SBWR designs. The research will provide valuable data to aid in validating the NRC's analytical codes used to audit the vendors' calculations and will provide experimental knowledge to improve the staff's understanding of the unique behavior of the passive ALWR's safety systems. (The need and planning for confirmatory research are discussed in SECY-92-037 and SECY-92-219 for the AP600, and in SECY-92-211 for the SBWR.) The NRC is helping to plan the confirmatory research programs for the passive ALWRs, including the modified ROSA-V/LSTF loop in Japan and a small-scale integral systems SBWR loop at Purdue University.

### Design Bases Reconstitution

The NRC staff proposed a Generic Letter on design reconstitution requesting that all licensees describe (1) the programs that are in place to ensure that design information is correct, accessible and current, and (2) the schedule for program implementation and completion. Those licensees not implementing a design reconstitution program would be requested to provide their rationale for not doing so.

The staff reviewed public comments (primarily from licensees) on the proposed Generic Letter and recommended that the Commission discontinue consideration of the proposed Generic Letter. It was decided that the staff already obtains sufficient insights as to licensee design bases reconstitution programs through its design-related inspections.

## INSPECTION PROGRAMS

NRR is responsible for developing, maintaining and assessing the effectiveness of the reactor inspection program, which encompasses all applicant and licensee activity carried out in connection with the construction and operation of nuclear facilities. Most of the inspection effort is dedicated to operations at the 109 plants where operating licenses are in effect (as of September 30, 1993), with added coverage of the seven facilities with construction permits. Responsibility for developing, maintaining, and assessing the effectiveness of the reactor inspection program is shared among the NRR staff.

NRR continued to improve the operating reactor program throughout fiscal year 1993 on the basis of its field experience in implementing the current program. The objectives of the inspection program are (1) to ensure that an adequate level of inspection is conducted at every plant, (2) to integrate headquarters and regional inspection programs, (3) to provide more flexibility for Regional Administrators to allocate resources on the basis of plant performance, and (4) to explicitly allocate resources in response to safety issues and regulatory concerns. The inspection staff seeks to obtain sufficient information through direct observation and verification of licensee activity to ascertain whether the facility is being operated safely, whether the licensee's management-control program is effective, and whether regulatory requirements are being satisfied. The inspection staff also gathers information for SALP evaluations (see "Performance Evaluation," below). In the "initiatives" phase of the inspection program, Regional Offices may redirect certain of their inspection resources from plants exhibiting a high level of performance to those showing a lower level of performance.

A basic element in the NRC reactor regulation program is the inspection of licensed reactor facilities to assure reactor safety by confirming that the operations comply with the provisions of the license, and to look for other conditions that have safety implications serious enough to warrant corrective action. The five NRC Regional Offices conduct most of the NRC inspection programs, while the NRC Headquarters directly conducts only a limited number. NRR is responsible for developing inspection policies and procedures and for monitoring and assessing the effectiveness and uniformity of the programs carried out by the NRC Headquarters and Regional Offices. Regional Administrators report to the NRC Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Resources.

The NRC conducts a program of regular inspections for reactor, fuel cycle facility, and materials licensees. The NRC is also committed to dealing aggressively with unsafe or potentially unsafe events or conditions occurring at individual plant sites, or other facilities involving licensed operations, through "reactive" inspections. The NRC conducts reactive inspections to determine the root cause of such an event or condition; evaluate the licensee management's response to it, including action to prevent recurrence; and decide whether a similar problem could occur at other facilities.

## Reactor Inspection Program

The operating reactor inspection program is implemented by headquarters and regionbased inspectors. Headquarters inspectors conduct, or support the Regional Office in conducting, inspections under the Team Inspection Program. The Regional Offices conduct most of the required program inspections by both region-based and resident inspectors. Most region-based inspectors are specialists, and resident inspectors are generalists. The resident inspectors provide the major on-site NRC presence for direct observation and verification of licensee activity. This effort includes in-depth inspections of control room operations; maintenance and surveillance testing carried out by the licensee; periodic "walk-down" inspections to verify the correctness of system lineups for those nuclear systems important to safe operation; and frequent plant tours to assess radiation control, physical security, equipment condition, and housekeeping. The resident inspector is the primary on-site evaluator in the NRC inspection effort with respect to licensee event reports, events and incidents, and other general inspections of licensee activity. The resident inspector is also the NRC contact with local officials, the press and the public. Region-based inspectors perform technically detailed inspections in such areas as engineering, system modifications, inservice inspection, fire protection, physics testing, radiation protection, physical security and safeguards, maintenance, and licensee management systems.

The inspection program allows headquarters and regional inspections to focus on those plant operations that contribute most to ensuring reactor safety and on the identification of safety problems. The NRC continued to improve the program in fiscal year 1993, based on knowledge gained from experience with the current program.

The inspection program comprises the following three elements:

**(1) Core Inspections.** The regular inspections conducted at every plant. They provide a balanced look at a cross-section of plant activities considered important to maintaining safety.

**(2) Area-of-Emphasis Inspections.** This program element consists of two parts:

*(a) Generic Area Team Inspections* addressing a subject area in which an emerging safety concern was found, or in which increased attention is needed because of a history of long-standing or recurring problems. Inspections of this kind are scheduled for all sites. In fiscal year 1993, the NRC continued conducting generic area team inspections of electrical distribution systems. This effort is expected to be completed during fiscal year 1994. The NRC also began conducting generic area team inspections in 1993 to evaluate the operational performance of service water systems. The staff will continue with these inspections during fiscal year 1994.

*Safety Issues Inspections* are one-time inspections to address a specific safety concern. The staff institutes these inspections by a temporary instruction (TI). A TI may be issued to ensure inspection follow-up of safety issues addressed in a bulletin or Generic Letter, or any other specific safety issue that calls for a one-time confirmatory inspection. During fiscal year 1993, the staff conducted six TIs for such issues as licensed operator re-qualification, BWR water level instrumentation errors, access authorization, and station blackout.

**(3) Initiative Inspections.** Inspections instituted by the Regional Administrator to follow up on problems identified in licensee performance during other inspections and to address areas where the greatest safety benefit can be obtained. This category also includes those reactive inspections that are conducted unannounced, at the discretion of the Regional Administrator, in response to various plant events or conditions of concern.

The Regional Offices also implement the construction inspection program (CIP) to confirm that the requirements of construction permits for nuclear plants are being met, and that the plants are being built in accordance with their approved designs and applicable codes and standards.

In 1993, the staff conducted construction inspection activities at two facilities, one of which received its operating license during the year. Another licensee notified the staff that it intended to resume construction activities on a deferred construction project. The staff conducted a team inspection to assess the licensee's readiness to resume construction and created an automated data base to record the construction inspections performed at the plant before the moratorium on construction.

The staff is developing a new CIP to guide the conduct of inspections at construction sites for advanced reactors licensed under 10 CFR Part 52. The new program will be structured to ensure that inspections are systematically planned, performed and documented, and to ensure that the "Inspections, Tests, Analyses, and Acceptance Criteria" required by 10 CFR Part 52 are satisfied. In 1993, the staff defined the inspection concepts to be employed and established a prototype of the computerized data base to be used to document and track inspection results for new construction projects.

## Assessment of the Operating Reactor Inspection Program

In fiscal year 1993, the staff assessed the effectiveness and implementation of the operating reactor inspection program, using a process developed in November 1992 for performing quantitative and qualitative analysis of the inspection program. The objective of the assessment process was to ascertain whether the inspection program (1) was effective in achieving its regulatory objectives, (2) covered the appropriate areas, (3) devoted resources at a level consistent with licensee performance, and (4) was implemented consistently across the nation. NRR evaluated the effectiveness and execution of the inspection program in each Region; the results of these assessments were conflated to arrive at overall conclusions and recommended improvements.

The overall results of the assessment indicated that the program was generally effective in meeting its regulatory objectives, was examining the appropriate areas, and did result in the identification of significant issues and adverse trends in licensee performance. The staff set out to improve the effectiveness of the program by (1) revising the Inspection Program to better allocate inspection resources based on licensee performance; (2) gradually reducing overall inspection effort by means of improved allocation of inspection resources; (3) improving the guidance and implementation of NRC management oversight activities; and (4) sharpening the inspection focus on licensee management programs, such as root cause analysis, corrective action, and self-assessment. The staff will also seek to improve the program by making changes that reflect revisions to the SALP process, issuing new procedures for maintenance and engineering, and incorporating other lessons learned during the evaluation.

## Special Team Inspections

During fiscal year 1993, NRC headquarters and regional staffs continued to perform special team inspections. A special team inspection usually involves a team of 4-to-10, with several engineering disciplines represented, and requires 1-to-3 weeks of on-site inspection. The team examines in detail various aspects of selected systems and components that are critical to safe shutdown of a plant or that are required to maintain the plant in a safe condition after shutdown. The team may inspect design, installation, testing, maintenance and operation of the systems selected. The overall objective of such inspections is to determine whether, when called on to do so in an emergency, plant systems and personnel will perform their safety functions as set forth in the Safety Analysis Report.

Headquarters develops the method for each new type of team inspection, tests it during a limited number of pilot inspections, and incorporates the developed inspection methodology into the NRC Inspection Manual. Responsibility for most of the special team inspections is assigned to the Regional Offices. Headquarters may lead a team inspection in some circumstances. Examples of special team inspections during 1993 were the Integrated Design Inspection at the Watts Bar (Tenn.) facility, the Plant Design Change inspection at Browns Ferry (Ala.), and the Augmented Inspection Team inspection at LaSalle (Ill.) after the loss of the station auxiliary transformer. Headquarters led Operational Readiness Assessment Team (ORAT) inspections at Comanche Peak Unit 2 (Tex.), Brunswick Unit 2 (N.C.), and the Sequoyah (Tenn.) nuclear plants. An ORAT is an independent review of licensee readiness to begin initial plant operation or to resume plant operation after an extended outage.

Some types of team inspections are performed "as needed" at particular plants, while others are designated "area-of-emphasis" inspections and are performed at a designated population of plants. Established types of special team inspections cover the following areas: emergency operations, maintenance, ability of systems to perform safety functions as designed, motor-operated valves, modification of safety systems during reactor outages, operational safety, operational readiness, and plant designs.

**Electrical Distribution System Functional Inspections.** A special team inspection—the Electrical Distribution System Functional Inspection (EDSFI)—was developed in 1990. After testing the program at six plants in 1990 and evaluating the results of those pilot inspections, NRC staff decided, in early 1991, that EDSFIs should be considered "area-of-emphasis" inspections and should be performed at all plants in the country (but with reduced scope wherever an in-depth NRC inspection had recently been performed, in the same program area). By end of fiscal year 1993, the inspection had been completed for the plants at 67 of a total 69 sites. The NRC planned to complete the remaining two EDSFIs by December 1993.

## INSPECTING THE NUCLEAR POWER PLANT

The primary safety consideration in the operation of any nuclear reactor is the control and containment of radioactive material, under both normal and accident conditions. Numerous controls and barriers are installed in reactor plants to protect workers and the public from the effects of radiation.

Both the industry and the NRC have roles in providing these protections and in ensuring that they are maintained. The NRC establishes regulations, and guides for the construction and operation of nuclear reactors. Organizations licensed by the NRC must abide by these regulations and are directly responsible for designing, constructing, testing, and operating their facilities in a safe manner. The NRC, through its licensing and inspection programs, provides assurance that its licensees are meeting their responsibilities.

The NRC inspection program is designed, through selective examinations, to ensure that the licensee is meeting its responsibility. The NRC inspection program is audit-oriented to verify, through scrutiny of carefully selected samples, that relevant activities are being properly conducted and equipment properly maintained so as to ensure safe operations. The staff determines the items to sample, sample size, and the frequencies of inspection based on the importance of the activity or system to overall safety and on available resources. The inspection program is preventive in nature and is intended to anticipate and preclude significant events and problems by identifying underlying safety problems. The inspection process monitors the licensee's activity and gives feedback to licensee's management for appropriate corrective action. However, the NRC inspection program does not supplant the licensee's programs or attenuate its responsibilities. The inspection program seeks to independently verify the effectiveness of the licensee's implementation of its programs, to ensure that operations are being carried out safely and in accordance with applicable NRC requirements. Inspections are performed on power reactors under construction, in test conditions, and in operation. The inspections are conducted primarily by region-based and resident inspectors. Resident inspectors are stationed at each reactor under construction and in operation. Region-based inspectors operate out of the five Regional Offices located in or near Philadelphia, Atlanta, Chicago, Dallas, and San Francisco. These programs are supplemented by inspections conducted by special teams made up of personnel from both NRC Headquarters and Regional Offices.

Before construction, the inspection program concentrates on the applicant's establishment and implementation of a quality assurance program. Inspections cover quality assurance activity related to design, procurement, and planning for fabrication and construction of the facility.

During construction, samples taken across the spectrum of licensee activity are examined to confirm that the requirements of the construction permit issued by the NRC are being followed and that the plant is being built according to the approved design and applicable codes and standards. Construction inspectors look for qualified personnel, quality material, conformance to approved design, and a well formulated and implemented quality assurance program. As construction nears completion, pre-operational testing begins, in order to demonstrate the operational readiness of the plant and its staff. Inspections during this phase seek to determine whether the licensee has developed adequate test plans - both to verify that tests are consistent with NRC requirements, and to ascertain whether the plant and its staff are thoroughly prepared for safe operation. Inspections during the pre-operational phase involve (1) reviewing overall test procedures, (2) examining selected test procedures for technical adequacy, and (3) witnessing and assessing selected tests to verify that test objectives have been met and to confirm the consistency of planned and actual tests. Inspectors also review the qualifications of operating personnel and verify that operating procedures and quality assurance plans are properly developed and implemented.

About six months before the operating license is issued, the licensee begins a startup phase to prepare for fuel loading and "power ascension." After issuance of the operating license, fuel is loaded into the reactor and the startup test program begins. As in pre-operational testing, NRC inspection emphasis is given to test procedures and results. The licensee's management system for startup testing is appraised, test procedures are analyzed, tests are witnessed, and licensee evaluations of test results are reviewed. Thereafter, the NRC continues its inspection program for the rest of the operating life of the plant.

The staff is developing a new construction inspection program for reactors to be built under combined construction and operating licenses issued under 10 CFR Part 52. The new inspection program will continue to verify the safety aspects of a plant's construction and testing, as described above for the current program, and will allow for more systematic inspection planning and documentation of inspection results. The new construction inspection program will be structured to ensure verification of satisfactory completion by licensees of the inspections, tests, analyses and acceptance criteria (ITAAC) included in a combined license and required by 10 CFR Part 52.

The NRC verifies that the licensee is operating safely through selective inspections. An on-site resident inspector provides a continual inspection and regulatory presence, as well as a direct contact between NRC management and the licensee. The activity of the resident inspector is supplemented by the work of engineers and specialists from the Regional Office who perform inspections in a wide variety of engineering and scientific disciplines, ranging from civil and structural engineering to health physics and reactor core physics.

The NRC Inspection Manual defines the frequency, scope and depth of the inspection program for operating reactor plants. Detailed inspection procedures provide instructions and guidance for NRC inspectors. The program consists of three major elements: core inspections—the minimum required at all plants; "area-ofemphasis" inspections—special inspections to focus on a specific issue; and regional initiative inspections—those required to resolve safety issues brought to light by other inspections or from plant operational experience. The program is structured to ensure that the resources available for inspection are used efficiently and effectively, with particular attention accorded those plants where past performance indicates the need to improve the levels of protection and safety-consciousness.

The inspection program is designed to ensure that nuclear power plants are constructed and operated safely and in compliance with regulatory requirements. The NRC considers the results of the inspection program when making its overall evaluation of licensee performance for the SALP program. When a safety problem or failure to comply with requirements is discovered, the NRC requires prompt corrective action by the licensee, confirmed, if necessary, by appropriate enforcement action.

The NRC periodically assesses the inspection program to evaluate its effectiveness in achieving its regulatory objectives.

The NRC has established an electronic data base of EDSFI findings which will allow them to be tracked and trended by plant, component and technical issue. The staff made the data base and associated software available to the industry, as NUREG 1473. EDSFI inspection results indicated a need for better licensee engineering and technical support, better licensee self-assessment programs, more detailed understanding of the design bases for the plant, and greater availability of design documents to the engineering staff. In parallel with NRC's attention to electrical distribution systems, licensees are conducting their own electrical inspections, are devoting more effort to evaluating the design basis for their electrical distribution systems, and are improving the functional capability of these systems.

**New Initiatives.** In 1991, the staff began preparing for two new types of team inspections in areas of concern to the NRC: Service Water System Operational Performance Inspection (SWSOPI) and Shutdown Risk and Outage Management (SROM) inspection. The staff conducted pilot inspections of both types in each Region, to test the methodology and scope of each. The NRC plans to proceed with the SWSOPI, as an "area-of-emphasis" inspection, at sites licensed before 1979 and at other sites having service water system problems, or more general maintenance, engineering or technical support problems. At the end of the fiscal year, six SWSOPIs had been completed, besides the pilot inspections. The staff has not planned additional SROM pilot inspections.

The staff issued an inspection procedure, "Licensee Self-Assessments Related to Area-of-Emphasis Inspections" (IP 40501), to allow reduced NRC inspection at those facilities which demonstrate good performance over time. Under the pilot effort, the NRC would evaluate a licensee's self-assessment effort as an alternative to a full scope NRC area-of-emphasis inspection. The NRC would sample areas covered by a licensee's self-assessment and significant areas not covered. The goal of this approach is to more effectively apportion NRC inspection resources and to reduce the impact on licensee operations of NRC inspection activities.

## Inspection of Emergency Operating Procedures

During fiscal year 1993, the staff continued to perform routine inspections of emergency operating procedures (EOPs), to assess the usefulness of the EOPs by evaluating their technical accuracy and by considering human factors. Findings from recent EOP inspections are addressed in a supplement to NUREG–1358, published in October 1992. Among the identified concerns are: inadequate documentation of deviations from the NRC approved generic technical guidelines, inadequate verification and validation of the EOPs and EOP support procedures, inadequately defined or implemented EOP usage guidance, and inadequate control of the EOP revision process. Although licensees have improved their implementation of EOP programs, the staff continues to find these kinds of deficiencies. The staff will continue inspection of EOPs to verify continued improvement.

## Vendor Inspection Program

During fiscal year 1993, the vendor inspection program remained a "reactive" program structured to respond to vendor and licensee reports of deviations and defects in the parts, components, materials and services provided by vendors to nuclear power plants. The staff devised the tasks and set the priorities by which to identify and deal with issues according to their safety significance and generic applicability.

Vendor inspections addressed reports from industrial organizations and allegations from members of the public concerning parts, components and materials that may be defective or misrepresented. Licensees and vendors are required to report problems and defects in safety-related equipment, materials and services to the NRC, under provisions of 10 CFR Part 21, as appropriate. In fiscal year 1993, the Vendor Inspection Branch of NRR assumed responsibility for screening, tracking and ensuring the proper disposition of these notifications. The NRC staff gauged the validity, extent and safety significance of each reported and alleged deficiency and ensured that licensees were apprised of any potential problems, so that appropriate action could be taken to prevent the use of defective components in nuclear plant safety systems. The NRC vendor inspection staff also frequently corresponded with vendors and licensees, by both written and oral communication, to clarify the NRC's position on specific interpretations and applications of 10 CFR Part 21, and on other Federal regulations.

In fiscal year 1993, the NRC vendor inspection staff conducted 21 inspections of vendors who manufacture or supply printed circuit assemblies, nuclear fuel assemblies, motor control centers, transmitters, fire barrier material, air flow meters, circuit breakers, water level instrumentation, motor-operated valves, air-operated valves, spare circuit breaker parts, piping supports and snubbers, and commercial grade dedication and equipment qualification services. The vendor inspection staff also conducted inspections to review the General Electric Advanced Boiling Water Reactor quality assurance program and to give technical support to the NRC Office of Investigations. The vendor inspection staff also assisted the Office of Investigations and various U.S. attorneys in criminal cases.

The Vendor Inspection Program includes inspection of foreign vendors who supply components for use in U.S. nuclear power plants. In this phase of the program, the NRC inspected circuit breakers being manufactured and tested in Germany by Klockner-Moeller GmbH and pipe support component procurement and dedication activities of Lisega GmbH for domestic nuclear plants.

As a result of inspection findings and other information in the vendor program area, the NRC issued 15 Information Notices during the report period informing the nuclear industry of potential problems. The Information Notices dealt with questionable selection of relays, tin smearing on diesel engines, deficiency of electrical cables, overstressing of motor-operated valves, accuracy of motor-operated valve diagnostic equipment, failure of under voltage trip on reactor trip breaker, tripping of molded-case circuit breakers, deficiencies in switchboard meters, degradation of electrical penetration assembly, deficiency of cables, installation of eyebolts with indeterminate properties, key failure in motor-operated valves, testing and preventive maintenance of molded-case circuit breakers, criminal prosecution of nuclear suppliers, and spurious tripping of low voltage power circuit breakers. The NRC also issued a bulletin supplement regarding loss of fill oil in transmitters and a Generic Letter dealing with deficient fire barrier material.

The staff continued to supply information to and participate in the Federal Interagency Working Group on Problem Parts and Suppliers, an activity that NRC helped to sponsor in 1988 and 1989. The Working Group is developing an interagency data base for the interchange of information regarding counterfeit and misrepresented parts.

In April of 1993, the NRC staff held a public workshop to discuss with nuclear industry representatives the procurement and dedication of commercial grade items used in safety-related applications and to solicit comments on the NRC's draft inspection guidance for those activities. The staff revised an inspection procedure, "Commercial Grade Procurement Inspection" (IP 38703), for release in November 1993. The procedure will call for more performance-based and results-oriented inspections for the dedication of commercial grade items, taking into account the significant feedback received from pilot inspections, licensee and industry meetings, and the public workshop.

## PERFORMANCE EVALUATION

The performance evaluation process is intended to enhance the NRC's ability to evaluate the effectiveness of licensee performance at nuclear power plants. It involves the integration of information from a variety of the NRC's continuing activities—such as the SALP program, enforcement actions, performance indicator tracking, trend analysis, event evaluation, operator examinations, and inspection findings. The process culminates in a semiannual meeting of NRC senior managers for discussions and appraisals of operating plant performance. On that occasion, the NRC managers agree upon the plants of greatest concern to the agency and plan a coordinated course of action, including recommendations for special inspections and intensified management attention. The staff presents the results of each such meeting to the Commission and informs each identified licensee of NRC senior management's characterization of its overall performance.

### Systematic Assessment of Licensee Performance

The SALP program is a principal and regular method for judging licensee performance. Under the program, the performance of each licensee with a nuclear power facility in operation or under construction in the United States is evaluated through the periodic, comprehensive examination of available data-including inspection findings, special results review, and similar licensing and inspection-related information.

The SALP program is designed to arrive at an overall assessment of how well licensee management at a given plant is directing and guiding operations, and providing needed resources for the requisite assurance of plant safety. The purpose of the SALP review is to focus both NRC and licensee attention on, and to direct resources to, those areas that can most closely affect nuclear safety and that need improvement.

The SALP includes a review of the previous year's reported events, inspection findings, enforcement history, and licensing issues. Also important are evaluations by resident and region-based inspectors, licensing project managers, and senior managers, all of whom are familiar with the facility's performance. New data are not necessarily generated in the conduct of a SALP assessment, which consists of performance evaluations in specific functional areas.

The Commission recently approved certain changes to the SALP program, including a reduction in the number of functional areas to be scrutinized from seven to four, restriction of the SALP Board membership to Senior Executive Service (SES) members only, a focus on the assessment of the most significant issues in each functional area, emphasis on recent (within the preceding six months) licensee performance when determining the SALP category ratings, and reduction of the length of the report to promote clearer communications with the licensee and the public. The staff effected the program changes for assessment periods ending after July 19, 1993.

## Human-Systems Interface

During fiscal year 1993, the staff continued its review of the human-systems interfaces of advanced reactor designs. Considerable staff resources have been devoted to reviewing the human factors engineering aspects of the Advanced Light Water Reactor (ALWR) Requirements Document of the Electric Power Research Institute (EPRI), and the designs of the General Electric (GE), Advanced Boiling Water Reactor (ABWR) and Simplified Boiling Water Reactor (SWBR), the ABB-Combustion Engineering (CE) System 80+, and the Westinghouse AP600.

Human factors constitutes one of the crucial areas affected by proposed advanced reactor designs, mainly because of the significantly different control rooms being proposed. New control room designs incorporate compact workstations with computerized display and control functions, as well as some conventional hardwired controls. The staff developed a Human Factors Engineering (HFE) program review model and acceptance criteria for reviewing the advanced control room design process proposed by the advanced reactor applicants. The model consists of eight elements, each of which includes general design acceptance criteria derived from accepted HFE practices. It is expected that the model will be published as a NUREG series report during fiscal year 1994

During fiscal year 1993, the staff prepared final safety evaluations for the humansystems interface portion of the EPRI ALWR Requirements for Passive Plant Designs and the GE ABWR advanced reactor design, and a draft safety evaluation for the human-system interface portion of the CE System 80+ design. The staff also developed two rounds of questions on the Westinghouse AP600 and GE SBWR human-systems interfaces, and continued exchanges with foreign utilities, researchers and regulatory organizations to examine their efforts to design and evaluate advanced control room designs.

The staff increased its efforts to conduct follow-up investigations of events involving human performance issues. The staff participated in two AITs and seven special inspections to help determine the root causes of such events and to identify and analyze those conditions which contribute to human errors. Investigations are conducted using a protocol developed by the NRC specifically for human-performance-related issues, considering the design of human-system interfaces, plant procedures, training, communications and the effects of supervision, management and organization on human performance.

## Training

During fiscal year 1993, the staff completed framing a performance-based training rule to meet the requirements of Section 306 of the Nuclear Waste Policy Act of 1982, as required by the U.S. Court of Appeals for the District of Columbia Circuit, in its April 17, 1990 decision. The final rule was published in the *Federal Register* on April 26, 1993. Full implementation of the training rule was required by November 22, 1993. The rule is not expected to have more than minimal impact on current industry training initiatives.

The staff conducted inspections of training programs when conditions at a particular licensee site warranted staff evaluation. During the report period, the staff conducted 10 training inspections at eight sites. Also during fiscal year 1993, the staff continued to evaluate implementation of the Institute of Nuclear Power Operations (INPO) accreditation program, in order to ensure that the industry's voluntary efforts are maintaining effective training programs for nuclear power plant workers. NRC personnel are present as observers during utility presentations to the National Nuclear Accrediting Board. NRC staff members also attend some INPO accreditation team visits as observers.

The staff has concluded that the industry continues to make progress in bringing about improvements in training, even though deficiencies continue to be found requiring corrective actions. The Commission continues to endorse the industry accreditation program as an effective means of ensuring proper nuclear plant personnel training.

# QUALITY ASSURANCE

Reviews of quality assurance programs for the design phase of Advanced Reactors continue to be an area of considerable activity. During fiscal year 1993, the NRC staff continued the review on the Quality Assurance portions of the Standard Safety Analysis Reports (SSARs) for the GE-ABWR, CE System 80+, and the Westinghouse AP600 advanced reactors. Two implementation inspections were performed for the GE-ABWR and SBWR design with respect to quality assurance controls for design, testing and analysis.

Reviews of Quality assurance program elements associated with material traceability and vendor supplied information were performed for the Watts Bar (Tenn.) nuclear power plant. The effectiveness of a licensee initiative for ensuring that commitments have been properly implemented was examined during the course of a site inspection.

Reviews were performed both of topical QA report revisions submitted by vendors and of some QA program revisions submitted by licensees.

The NRC staff held a quality assurance "counterparts meeting" to discuss issues related to QA inspection and QA program review topics, and periodic inter-office meetings have been held to coordinate staff activities on QA issues. The staff has also interacted with the "Appendix B Working Group" of the Nuclear Management and Resources Council (NUMARC) toward the development of a gradual approach to QA implementation.

## Maintenance

On July 10, 1991, the Commission published a new maintenance rule, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (10 CFR 50.65), in the *Federal Register* (56 FR 31306). The rule will require commercial nuclear power plant licensees to monitor the effectiveness of maintenance activities for safety-significant plant equipment, in order to minimize the likelihood of failure and of events caused by the lack of effective maintenance. This rule takes effect on July 10, 1996.

During the fiscal year 1993, NRC's maintenance efforts were primarily dedicated to issuing a regulatory guide for the implementation of the maintenance rule, amending provision (a)(3) of the maintenance rule, and preparing inspection guidance for verifying implementation of the rule.

The NRC regulatory guidance development effort was headed by a steering group of senior managers from the Office of the Executive Director for Operations, NRR and the Office of Nuclear Regulatory Research. In conjunction with NRC's guidance development, the NRC steering group conducted numerous public meetings to discuss the development of NUMARC's implementing guidance for the maintenance rule. Following a series of public meetings with NUMARC, the NRC issued NRC Regulatory Guide 1.160, dated June 1993, which endorsed NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated May 1993, as an acceptable method for implementing the maintenance rule.

Provision (a)(3) of the original rule would have required that licensees evaluate performance and condition monitoring activities for structures, systems and components at least annually. In response to comments from the industry, the Commission changed the required frequency of the evaluation from annually to at least every refueling cycle, provided the interval between evaluations does not exceed 24 months. The Commission agreed with the industry that evaluation of data collected over the period of a refueling cycle will provide a substantially better basis for detecting problems in degraded performance of structures, systems, and components, as well as weaknesses in maintenance practices. It would also allow licensees to consider and integrate data available only during refueling outages with the data available during plant operations.

The NRC staff has begun preparation of a draft inspection procedure that will be used to review licensee's implementation of the maintenance rule. A workshop will be held in early 1994 to provide the public an opportunity to comment on the inspection procedure. In order to give licensees early feedback on their implementation of the maintenance rule, and to validate the adequacy of the inspection procedure, the NRC will perform a series of pilot inspections in 1994 and 1995. Because these inspections will be performed before the effective date of the rule, July 10, 1996, they will be performed at volunteer plants only, and the results of the inspections will be provided to licensees for information only and will not be the basis for enforcement action. The results of these inspections will be used by the NRC staff to revise the inspection procedure, as necessary. A second workshop will then be held to provide the public an opportunity to comment on the revisions before the inspection procedure is issued in final form for use by NRC regional inspectors, and before the effective date of the maintenance rule.

# OPERATOR LICENSING

The NRC continues to administer initial and requalification examinations to applicants for and holders of reactor operator (RO) and senior reactor operator (SRO) licenses at power and non-power reactor facilities. Both the initial and requalification examination procedures consist of a written examination and an operating test that includes a plant walk-through and a dynamic performance demonstration on a simulation facility. The responsibility for administering the examinations at power reactors rests with the five NRC Regional Offices, while the Operator Licensing Branch at NRC's Headquarters has responsibility for managing the program and administering the examinations at non-power facilities.

During fiscal year 1993, the NRC administered 636 licensing examinations to RO and SRO applicants at power and non-power reactor facilities and 410 Generic Fundamentals Examinations (GFEs) to prospective license applicants at power reactor facilities. The GFE tests prospective licensed operators on their understanding of the theoretical knowledge required in the operation of a nuclear power plant and must be passed before the applicant can take the site-specific written examination.

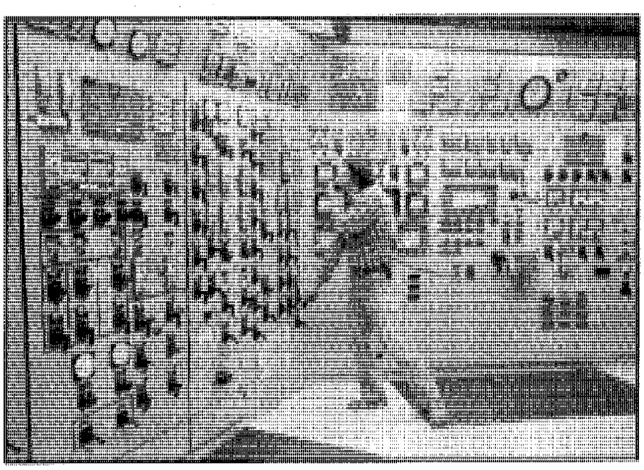
The NRC also evaluated a total of 921 licensed operators for purposes of requalification. The NRC conducted requalification program evaluations at 59 power reactor facilities and administered requalification examinations at 21 non-power reactor facilities. Twelve of the power reactor requalification programs were evaluated using the newly developed inspection procedure that is described below. The NRC requalification evaluation program assures the continued competence of individual licensed operators and also evaluates the quality of the facility licensees' requalification programs.

On August 14, 1991, the NRC amended 10 CFR Part 55 to make the facility licensee's fitness-for-duty requirements a condition of each operator's license. Through September 30, 1993, the NRC received 27 reports of licensed individuals' exceeding their facility licensee's cutoff levels for drugs or alcohol.

The NRC received two plant-referenced simulator certifications during fiscal year 1993. As of September 1993, every facility licensee has either certified a plant-referenced simulator or obtained NRC approval of a simulation facility.

The NRC is continuing its efforts to improve the operator licensing program. The NRC staff has implemented or is considering a number of initiatives that will enhance the initial licensing and requalification examination processes. The following improvements were either accomplished during fiscal year 1993 or are in progress:

(1) The staff formally promulgated the crew-based dynamic simulator grading procedures. Revision 7 of NUREG–1021, "Operator Licensing Examiner Standards," was published in January 1993 and has been used for all requalification examinations conducted since August 1993. The staff believes that the revised grading procedures encourage better teamwork, communications, and command and control among the control room operators, thereby providing a more accurate measure of the operators' abilities.



**The NRC continues to administer initial and requalification exams to applicants for or holders of reactor operator and senior reactor operator licenses at reactor facilities. During fiscal year 1993, the NRC administered 636 licensing exams and 410 Generic Fundamentals exams to prospective license applicants. Above is a reactor operator on the job at the Donald C. Cook (Mich.) nuclear power plant.**

(2) In December 1992, the staff issued a temporary instruction (TI) for use during a number of trial inspections to assess the viability of a new requalification oversight program that would replace most NRC-conducted requalification examinations. In May 1993, the NRC published a *Federal Register* notice proposing to delete the 10 CFR Part 55 requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's six-year license. The proposed rule change and the associated inspection program are described below.

(3) The staff is reviewing and updating the "Knowledge and Abilities Catalog for Nuclear Power Plant Operators: Pressurized [Boiling] Water Reactors" (NUREG–1122), which was originally published in 1985, in order to incorporate evolutionary changes in licensed operators' tasks and in the operator licensing program.

(4) Concerns regarding regional variations in the NRC's operator licensing examinations prompted the staff to competitively select an independent contractor to study examination quality and consistency and to recommend possible solutions to any problems that might be identified. The study disclosed that the individual differences among examiners are the most important determinant of variations in the development and administration of operator licensing examinations, and that the current regional structure of the operator licensing program is not a key contributor to examination inconsistency. The contractor concluded that both the requalification and initial examinations are sufficiently consistent to ensure that appropriate licensing decisions are being made.

## Operator Licensing Requalification Changes

As noted above, the NRC published a *Federal Register* notice in May 1993 proposing to delete the 10 CFR Part 55 requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's six-year license. Eliminating that requirement will enable the NRC to more efficiently accomplish its regulatory task of actively overseeing the requalification program at each facility, because it will allow NRR to allocate resources based upon the program's performance, rather than on the number of individuals licensed to operate the facility. Under the revised oversight program, the

NRC will either inspect the facility licensee's requalification program in accordance with a newly developed procedure or conduct "for cause" requalification examinations in accordance with existing examination procedures. The NRC does not plan to conduct its own periodic requalification examinations under the revised oversight program. The staff anticipates that the final rule will be published in fiscal year 1994.

In December 1992, the staff issued a temporary instruction (TI) to be used in a number of trial inspections for the purpose of appraising the viability of the new requalification inspection process. The staff believes that the guidance in the TI enabled the inspectors to conduct adequate assessments of the facility licensees' operator requalification programs. The proposed oversight program may further improve facility requalification programs, because the trial inspections performed in accordance with the TI have brought several issues to light that went undetected during previous NRC-conducted requalification examinations. The staff is evaluating lessons learned from the trial inspection program and is developing a final requalification inspection procedure. The staff intends to conduct an inspection at each facility during each SALP cycle. Those facilities that are good performers could go up to two years between inspections, while the weaker performers could be inspected annually.

## EMERGENCY PREPAREDNESS

The staff continued to assess emergency preparedness (EP) at nuclear power facilities through on-site inspections and observation of the annual exercises conducted at the more than 70 nuclear power reactor sites throughout the United States. The staff has also reviewed changes in licensee emergency plans and in implementing procedures to verify compliance with current regulations. The overall quality of the emergency preparedness programs at these facilities remained high for fiscal year 1993. Oversight of research and test reactors involved selected on-site inspections and staff review of changes to emergency plans submitted by the licensees. In addition, the staff worked closely with the Federal Emergency Management Agency (FEMA) to address issues related to off-site emergency preparedness.

In June of 1993, the NRC and FEMA signed a new Memorandum of Understanding (MOU) related to radiological emergency planning and preparedness. The new MOU replaced one that had been in effect since 1985. Principal changes in the MOU address (1) withdrawal of "reasonable assurance" findings, (2) recovery from disasters affecting off-site emergency preparedness, and (3) review of applications for early site permits under 10 CFR Part 52. The new MOU with FEMA will help facilitate continued improvement in joint agency efforts in the area of emergency planning and preparedness for licensed nuclear power reactors.

The staff also worked closely with FEMA in fiscal year 1993 on (1) the development of emergency planning guidance to assist applicants filing for early site permits under 10 CFR Part 52, (2) the development of guidance on recommended protective actions to protect the public in the event of a severe reactor accident, (3) the development of policy on implementation of the Environmental Protective Agency's revised Protective Action Guides and, (4) the review and response to petitions concerning off-site emergency preparedness.

The sustained heavy rains and flooding in the midwestern States during the summer of 1993 had a significant impact on emergency planning at several reactor sites. One licensee, the Commonwealth Edison Company, delayed restart of its Quad Cities (Ill.) nuclear power plant, because of concerns over flooded evacuation routes in Iowa. State and local officials established several alternative evacuation routes for the Callaway nuclear power plant in Missouri, where river conditions and high water caused partial evacuation in several areas. Another licensee, Nebraska Public Power District, initiated precautionary sandbagging around the Cooper (Neb.) nuclear power plant and issued a Notification of Unusual Event, because of rising river levels. This licensee also executed a precautionary shutdown of the Cooper plant. By being well informed of current and projected conditions, licensees and State and local authorities were able to effectively cope with the effects of the flooding. The value of established emergency plans, well trained responders, and effective communications was clearly demonstrated during the 1993 Midwest floods.

As a follow-up to lessons learned from Hurricane Andrew in fiscal year 1992, the NRC staff evaluated the adequacy of current NRC regulations and guidance on licensee off-site communications capabilities and reviewed actions taken by licensees to improve their ability to maintain communications during hurricanes.

During fiscal year 1993, the NRC staff worked with NUMARC on the implementation of NUMARC/NESP-007, "Methodology for Development of Emergency Action Levels." NUMARC/NESP-007 was endorsed by the NRC in Revision 3 to Regulatory Guide 1.101 as an acceptable alternative for meeting NRC requirements. The NRC coordinated with NUMARC in developing questions and answers on NESP-007 that help clarify its intent. The staff has also reviewed 12 separate emergency classification schemes from nuclear power plant licensees who have implemented the new guidance. Many more licensees are expected to revise their emergency classification schemes, based on NUMARC/NESP-007, in fiscal year 1994.

In fiscal year 1993, the NRC staff reviewed a number of emergency preparedness issues, ranging from advanced reactors to the shutdown of the Trojan (Ore.) nuclear power plant. The staff also addressed emergency planning aspects of actual events occurring at operating plants during the year, including a steam generator tube rupture at Palo Verde (Ariz.) and an unauthorized intrusion at Three Mile Island (Pa.).

The Virginia Electric and Power Company (VEPCO) has submitted a proposed change to NRC's emergency preparedness requirements through a petition for rulemaking seeking to change the frequency for the conduct of licensee emergency exercises from annual to biennial. A notice of the filing of the petition was published in the *Federal Register*, on March 4, 1993 (41 FR 12341). A total of 32 comment letters were subsequently received from utilities, States, NUMARC, FEMA and environmental and citizens groups. The matter was under study at the close of the report period.

## SAFETY REVIEWS

### Applications of Probabilistic Risk Assessment

In fiscal year 1993, the application of Probabilistic Risk Assessment (PRA) methods and insights to regulatory activity continued. As in recent years, PRA applications were made in both traditional PRA-relevant activities and in newer areas. The traditional applications include PRA reviews, setting of priorities, evaluating regulatory issues and plant-specific licensing issues, and judging the risk significance of changes in the technical specifications. Newer uses are related to advanced reactors, inspection guidance, human performance, accident management, shutdown risk, and operating plants performance.

The NRC staff has completed reviews of the PRAs for the General Electric ABWR and Combustion Engineering System 80+ designs, which were used during the staff's Design Certification review. The staff has also continued preliminary PRA reviews for the Westinghouse AP600 and the General Electric SBWR advanced passive design. The presence of passive safety systems in these designs poses unique technical challenges in the PRA review process.

The Individual Plant Examinations (IPEs) continued with approximately 63 of the IPEs completed and submitted. The staff has completed reviews of 12 submittals with 21 currently under review, including the Watts Bar (Tenn.) IPE. (The last named submittal is being employed by the staff to assess severe accident strengths and weaknesses, as part of the Severe Accident Mitigation Design Alternatives (SAMDA) process, leading to the completion of the Operating License review.) These IPE submittals deal with accident sequences initiated by internal events or internal flooding. Utilities are currently performing IPEs for seismic events, fires, external floods, high winds, and nearby industrial accidents. These IPEs are expected to be submitted within the next two years.

The application of PRA results and insights to operating reactor activities continues to prove its worth. PRA-based information is used to assess the significance of plant events and to evaluate proposed licensing actions. Currently, the staff has a program under way to enhance the application of PRA methods throughout the agency and to ensure that its use is consistent and appropriate. Four NRC offices are participating in the program which will result in a broadly based PRA implementation plan, expected to be completed in calendar year 1994.

### Reactor Vessel Materials

Reactor pressure vessel integrity is essential to assuring reactor safety. During operation, a reactor vessel is subject to neutron irradiation and, as a result, the fracture resistance of its materials is reduced. The decrease in fracture resistance is measured by an increase in the brittle-to-ductile transition temperature and a reduction in the Charpy upper-shelf energy. In Section 50.60(a) of Title 10 of the Code of Federal Regulations (10 CFR 50.60(a)), the NRC requires that licensees for all light water nuclear power plants meet fracture toughness requirements and have a material surveillance program for the reactor vessel materials that are subject to neutron irradiation. These requirements are set forth in Appendices G and H to 10 CFR Part 50. Appendix G requires that reactor vessels have a minimum value of 50 ft.-lb. Charpy upper-shelf energy or that the licensee demonstrate, by performing an "equivalent margins" analysis, that safety margins against failure equivalent to those required by Appendix G of the ASME Code are maintained.

In July 1985, the NRC promulgated "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events" (10 CFR 50.61). This rule established screening criteria to determine whether a reactor vessel has adequate fracture toughness to withstand pressurized thermal shock (PTS) events. For each material in the reactor vessel beltline region of the reactor vessel (i.e., areas directly surrounding the effective height of the active core and adjacent regions that are predicted to experience significant neutron irradiation embrittlement), there is a reference temperature ($RT_{PTS}$) value, calculated by means of methodology contained in the rule. The $RT_{PTS}$ value is an indication of the fracture resistance of the material. As the $RT_{PTS}$ value increases, the fracture resistance decreases.

Analyses performed by the NRC staff indicate that the risk from PTS events for reactor vessels with $RT_{PTS}$ values below the screening criteria is acceptable. The rule requires that licensees implement flux reduction programs, as reasonably practicable, to avoid exceeding the PTS screening criteria. For reactor vessels that are predicted to exceed the PTS screening criteria, the rule permits licensees to submit safety analyses that demonstrate what, if any, modifications to equipment, systems and operations are necessary to prevent potential failure of the reactor vessel as a result of postulated PTS events. After considering the licensee's analysis, the Commission may, on a case-by-case basis, approve operation of the facility at values of $RT_{PTS}$ in excess of the screening criteria.

The NRC issued a Generic Letter on March 6, 1992, to obtain information needed to judge compliance with requirements and with commitments regarding reactor vessel integrity, in light of certain concerns raised in the staff's review of reactor vessel integrity for the Yankee-Rowe (Mass.) nuclear power plant.

In response to the Generic Letter, all licensees have indicated that they are in compliance with Appendices G and H and have provided information to confirm that compliance. The NRC staff is reviewing the information to certify that the licensees are in compliance with the regulations and has prepared a preliminary assessment of the licensee responses (SECY-93-048).

All licensees have responded that, based on plant-specific data and evaluations, their reactor vessels meet the 50 ft.-lb. upper-shelf energy criteria in Appendix G, 10 CFR 50. However, 15 plants would currently have calculated that—when upper-shelf energies are calculated using the NRC generic criteria—the upper-shelf energies are less than 50 ft.-lb., and three others would have upper-shelf energies less than 50 ft.-lb. before the end of their operating licenses. However, generic analyses sponsored by the NRC Office of Research and performed by the Oak Ridge National Laboratory, reported in NUREG/CR-6023, July 1993, indicate that reactor vessels can be safely employed, even though their Charpy upper-shelf energies may be less than 50 ft.-lb. Individual licensees and owners' groups have also performed analyses to demonstrate that their reactor vessels with upper-shelf energies less than 50 ft.-lb. can function safely.

The NRC staff is synthesizing information received in response to the Generic Letter, in order to create a data base that will contain all the data needed to confirm that each reactor vessel is in compliance with Appendix G to 10 CFR Part 50 and the PTS rule, 10 CFR 50.61, throughout the plant's life.

## Performance of Motor-Operated Valves

The NRC staff is continuing efforts to improve the performance of motoroperated valves (MOVs) in nuclear power plants. Although improvements in MOV performance are being observed, MOV problems continue to occur or are otherwise being identified. The problems include inadequate MOV design and incorrect torque, torque bypass, and limit switch settings that have led, or could lead, to failures of MOVs to perform their intended functions.

The NRC staff issued Generic Letter 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and Surveillance," in light of problems with the performance of MOVs in nuclear power plants. In Generic Letter 89-10, the staff requested that licensees confirm the capability of MOVs in safety-related systems by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The staff requested that licensees complete the Generic Letter 89-10 program within approximately three refueling outages or five years from the issuance of the Generic Letter.

Supplement 1 to Generic Letter 89-10 was issued on June 13, 1990, to provide detailed information on the results of public workshops held to discuss the Generic Letter. On August 3, 1990, the staff issued Supplement 2 to allow licensees more time to review and to incorporate the information provided in Supplement 1 into their programs, in response to the Generic Letter. Based on the results of NRC-sponsored MOV tests, the staff issued Supplement 3 on October 25, 1990, requesting that licensees of boiling water reactor (BWR) nuclear plants take action in advance of the Generic Letter 89-10 schedule to resolve concerns about the capability of the MOVs used for containment isolation in the steam supply line of the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems, in the supply line of the Reactor Water Cleanup system, and in other systems directly connected to the reactor vessel. On February 12, 1992, the staff issued Supplement 4 to the Generic Letter, lifting the recommendation that BWR licensees address inadvertent MOV operation as part of their Generic Letter 89-10 programs, on the basis of a staff study of core melt probability. On June 28, 1993, the staff issued Supplement 5, requesting that licensees address the increased inaccuracy of MOV diagnostic equipment that had been revealed from testing and plant experience.

On February 25, 1993, the NRC staff held a public workshop to discuss Generic Letter 89-10 and to answer questions from the public on the inspections of licensee programs developed in response to the Generic Letter. On July 22, 1993, the staff issued proposed Supplement 6 for public comment to further clarify staff positions on the schedule for completing the MOV testing to verify design-basis capability, as recommended in the Generic Letter, and for grouping of MOVs to establish valve setup conditions. The staff also discusses the safety significance of the potential for pressure locking and thermal binding of gate valves and responds to general public questions in an enclosure to the proposed supplement. The staff has reviewed the public comments and revised the proposed supplement where appropriate.

On June 14, 1993, the NRC staff issued Revision 1 to Temporary Instruction 2515/109, updating guidance for regional inspections of the programs being developed by nuclear power plant licensees in response to the Generic Letter. The staff has performed inspections to review the utilities' development of MOV programs at each nuclear power plant. In 1993, the staff initiated inspections of the implementation of Generic Letter 89–10 programs and performed implementation inspections at many nuclear power plants.

The staff is closely monitoring the industry's efforts toward resolving concerns about the performance of MOVs at nuclear power plants. Although improvements in MOV performance were observed in 1993, it is apparent from nuclear plant operating events, Generic Letter 89–10 program development and implementation, industry research, and NRC inspections that nuclear power plant licensees will need to continue to apply resources to improving MOV performance. The staff will initiate regulatory action, where necessary and appropriate, to provide assurance that the health and safety of the public are protected, based on its review of the industry's efforts to improve the performance of MOVs in nuclear power plants.

## Evaluation of Shutdown And Low-Power Risk Issues

As discussed in the *1991 NRC Annual Report* and the *1992 NRC Annual Report*, an evaluation of shutdown and low-power issues was initiated following the NRC staff investigation of the loss during shutdown of all vital a.c. power, on March 20, 1990, at the Vogtle (Ga.) nuclear power plant. The evaluation sought a broad assessment of risk during shutdown, refueling and startup, addressing issues raised by the Vogtle event and by a number of other shutdown-related issues identified by foreign regulatory organizations, as well as by the NRC, and also treating new issues uncovered in the evaluation process.

In February 1992, the staff issued a report entitled "Shutdown and Low-Power Operations at Nuclear Power Plants in the United States" (NUREG–1449), as a draft report for comment by the public. NUREG–1449 documents the staff's technical findings deriving from the evaluation of shutdown and low-power operations. The comment period on NUREG–1449 ended on April 30, 1992, and a large number of comments were received from utilities and industry organizations. The comments have been addressed in the final report (NUREG–1449), issued in September 1993.

Over the past year, the staff has conducted a formal regulatory analysis of potential requirements in the area of shutdown and low-power operations. The results of the draft regulatory analysis support the staff's preliminary findings, in NUREG–1449, that public health and safety have been adequately protected while plants have been in a shutdown condition, but that safety levels could be substantially improved and that such improvement is warranted. The staff has identified the following areas for potential improvements in shutdown operations: (1) outage planning and control; (2) fire protection; (3) technical specifications; and (4) instrumentation.

The preliminary findings of the staff's draft regulatory analysis are documented in "Regulatory Approach to Shutdown and Low-Power Operations" SECY–93–190, issued in July 1993.

On July 20, 1993, the staff briefed the Commission on the status of the shutdown risk program and the results of the draft regulatory analysis, and recommended rulemaking for the purpose of implementing cost-justified safety improvements in the area of shutdown and low-power operations. The staff is currently drafting a proposed rule for consideration by the Commission in the mid-1994 time frame.

In the interim, the staff has acted in response to concerns about shutdown operations. The staff has issued Information Notices regarding those operations, the use of freeze seals, and the potential for boron dilution. The staff also issued a temporary instruction, "Reliable Decay Heat Removal During Outages" (TI) 2513/113, calling for increased inspection emphasis during outages, focusing primarily on residual heat removal capability and activities involving electrical systems. The staff has also modified NRC standards for operator license exams to (1) place more emphasis on shutdown operations and (2) review the licensee's requalification exam test outline for coverage of shutdown and low-power operations, consistent with the licensee's job task analysis and operating procedures. Finally, headquarters staff advised regional staff that current emergency plans should address the protection of plant workers in any emergency occurring during shutdown operations.

## Steam Generator Issues

The thin-walled tubing of the steam generator constitutes well over 50 percent of the reactor coolant pressure boundary. The integrity of the boundary is particularly important in minimizing the release of radioactive fission products to the environment. Steam generator tubing has, however, exhibited widespread degradation by a variety of corrosion and mechanical mechanisms.

Because of the potential consequences of the loss of steam generator tube integrity, the agency has established measures for ensuring that the integrity of the tubing is maintained. The traditional tube plugging criteria have typically been based on a minimum wall thickness requirement which assumes that the degradation involves uniform thinning of the tube wall. The assumption of uniform thinning conservatively defines the effects of all flaw types occurring in the field and is the basis for the standard 40 percent depth-based plugging limit. But the 40 percent plugging limit is very conservative for highly localized flaws, such as pits and short cracks. Since the dominant degradation mechanism currently affecting steam generator tubes is stress corrosion cracking, the 40 percent depth-based plugging limit may result in unnecessary repair of many steam generator tubes.

In August 1993, the nuclear industry proposed a generic approach—designated the "steam generator degradation specific management program"—for addressing various forms of steam generator tube degradation. Under this approach, inspection methods and repair criteria would be developed for a specific type of degradation. The program is designed to improve the technical and regulatory aspects for ensuring steam generator tube integrity. It consists of guidelines for generic steam generator tube inservice inspection, degradation-specific inspection and repair criteria, and certain plant-specific measures for monitoring the performance of steam generators (e.g. leakage detection). A degradation-specific approach to managing steam generator tube degradation has several important benefits including: (1) improving the scope and methods for inspecting steam generator tubes; (2) incentives to continue to improve inspection methods; and (3) development of plugging/repair criteria based on the most appropriate nondestructive examination parameters, thereby improving the efficacy of the criteria and eliminating unnecessary conservatism.

One plugging limit that has been proposed is a voltage-based limit for axially oriented outside diameter stress corrosion cracking (ODSCC) at the tube support plate elevations. This proposed plugging criterion was implemented on the basis of commitments to the use of enhanced inspection methods, enhanced sampling plans, and reduced primary-to-secondary leak rate limits. The voltage limit is intended to ensure adequate structural and leakage integrity of the tubing throughout the operating cycle. Currently five nuclear power plants have implemented, on an interim basis, voltagebased limits for ODSCC at the tube support plates. These interim limits are more restrictive than what the industry is currently proposing. The staff is preparing a Generic Letter for issuance in 1994 on voltage-based interim plugging criteria for ODSCC of tubes at tube support plate elevations. This Generic Letter would provide the staff position on voltage-based limits for ODSCC of tubes at tube support plate elevations, pending completion of longer term actions that the staff is considering.

Another flaw-specific plugging limit proposed to the NRC includes length-based limits for primary water stress corrosion cracking at roll transition locations. The roll transition location is the region in the tube where the diameter changes as a result of the tube's expanding into the tubesheet. As is the case for voltage amplitude-based limits, proposals for length-based limits are programmatic, involving commitments to specific inspection methods, inspection sampling plans, and reduced primary-to-secondary leak rate limits, as well as revised plugging limits. Review of this proposal is expected to be completed in 1994.

In 1993, the staff devoted considerable resources to generic steam generator tube integrity issues. One important step was the preparation of a draft document, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" (draft NUREG–1477), nearing completion at the end of the report period, in support of the Generic Letter on voltage-based criteria. The staff also began reviewing technical reports setting out the industry's proposal on degradation-specific management.

## Primary Water Stress Corrosion Cracking

Primary water stress corrosion cracking (PWSCC) of Alloy 600 was identified as an emerging issue by the NRC staff, following a 1989 leakage from an Alloy 600 pressurizer heater sleeve penetration at the Calvert Cliffs Unit 2 (Md.) nuclear power plant. The unit is a Combustion Engineering designed pressurized water reactor (PWR). Several instances of PWSCC of Alloy 600 pressurizer instrument nozzles have been reported to the NRC since 1986, in both domestic and foreign pressurized water reactors. Arkansas Nuclear One Unit 1, a Babcock & Wilcox (B&W) designed PWR, reported a leaking pressurizer instrument nozzle in 1990, after 16 years of operation. (Westinghouse PWR's do not use Alloy 600 for penetrations or nozzles in the pressurizers.)

In 1991, a leak was discovered during a hydrostatic test of an Alloy 600 control rod drive mechanism (CRDM) penetration adaptor tube at the Bugey Unit 3 (France) reactor, after 12 years of commercial operation. A visual examination of the CRDM penetration adaptor tube revealed axial cracks in the inside diameter (ID) of the CRDM penetration adaptor tube. The remaining 65 CRDM adaptor tubes at Bugey Unit 3 were examined, and axial cracks were found on the ID of two additional CRDM adaptor tubes. An examination of 24 CRDM adaptor tubes at Bugey Unit 4 revealed axial ID cracks in eight CRDM adaptor tubes. CRDM adaptor tubes have been examined at 37 nuclear power plants in France, Sweden, Switzerland, Japan and Belgium. Fifty-nine of the 1,850 penetration tubes examined have short, axial crack indications.

The staff has received safety assessments from NUMARC prepared by the Westinghouse Owners Group (WOG), Combustion Engineering Owners Group

(CEOG), and B&W Owners Group (B&WOG). They address the potential for, and consequences of, CRDM penetration tube or control element drive mechanism (CEDM) penetration tube cracking. Based on these evaluations and the inspection results at foreign plants, the NRC staff has concluded that this issue is of low safety significance, because all cracks reported to date, with the exception of one apparent fabrication defect, are short in length and axially oriented, in an extremely flaw-tolerant material; for those reasons, ejection of a CRDM continues to be deemed an unlikely event. Furthermore, the effects of wastage by borated water on a crevice area, such as between CRDM penetration and the reactor head, have been evaluated on the bases of laboratory testing and similar field experience. The results of these laboratory tests and the field data indicate that any degradation induced by boric acid corrosion would occur very slowly. Furthermore, plant operators conducting surveillance walkdowns of the vessel head, as specified in Generic Letter (GL) 88-05, would detect any leakage before unacceptable corrosion degradation could occur in the vessel head.

NUMARC submitted proposed flaw acceptance criteria to the NRC staff on July 30, 1993, setting forth certain criteria to be used in determining the nature of flaws found during CRDM/CEDM inspections. The staff accepted the criteria for axial cracks, because the criteria conform to the ASME Section XI criteria. However, based upon information submitted to date and the more serious safety consequences of circumferential flaws, the staff decided not to pre-approve criteria for circumferential flaws. Instead, circumferential flaws found during CRDM/CEDM inspections will have to be evaluated, following review by the staff, on a case-by-case basis.

On the basis of the low probability of the ejection of a CRDM and the low safety significance of CRDM leakage, the staff concluded that there was sufficient time available for the industry to implement a well-planned inspection, evaluation and repair program that would minimize personnel radiation exposures.

## Radiation Protection at Nuclear Reactors

Daily monitoring of licensee and Region reports to the NRC Operations Center alerts staff to potential problems developing in radiation safety, ranging from major repair problems involving highly radioactive components to contamination from the cleanup of small leaks of liquid and gaseous materials. These initial reports are followed up by discussions with regional NRC representatives and eventual action to be taken on any health physics problems uncovered in regional inspections. Further involvement of headquarters staff in regional and licensee problems may come about as the result of the staff participation in routine environmental and radiological inspections, as well as participation in special regional team inspections of significant licensee problems.

During fiscal year 1993, the NRC staff provided radiation protection support in licensing activities at most of the operating nuclear power reactors, as well as reviews of design criteria and conceptual designs for advanced reactors. This work was initiated for the Westinghouse AP600 and the General Electric (GE) Simplified Boiling Water Reactor (SBWR) and was continued for the GE Advanced Boiling Water Reactor (ABWR) and the Asea Brown Boveri-Combustion Engineering (ABB-CE) System 80+. The reviews of the Electric Power Research Institute (EPRI) evolutionary and passive plant designs were also completed. This task included detailed evaluations of occupational radiation protection design features, systems, equipment and drywell. Evaluations continued for the off-site consequences of design basis accidents for the ABB-CE System 80+ project. Also included in these activities were reviews of spent fuel pool re-racking plans for such plants as Beaver Valley (Pa.), Ft. Calhoun (Neb.), Millstone (Conn.), and Maine Yankee. Other licensing support actions included reviews of main steam line radiation monitors at the Hatch (Ga.), Fitzpatrick (N.Y.), and Millstone (Conn.) facilities. And reviews were performed of the proposed steam generator interim tube plugging criteria for the Catawba (S.C.), Farley (Ala.), and Palo Verde (Ariz.) plants. Licensing action support during the period included reviews of the radiation protection operating histories at the Turkey Point (Fla.), and Peach Bottom (Pa.), in support of requests for operating license extensions.

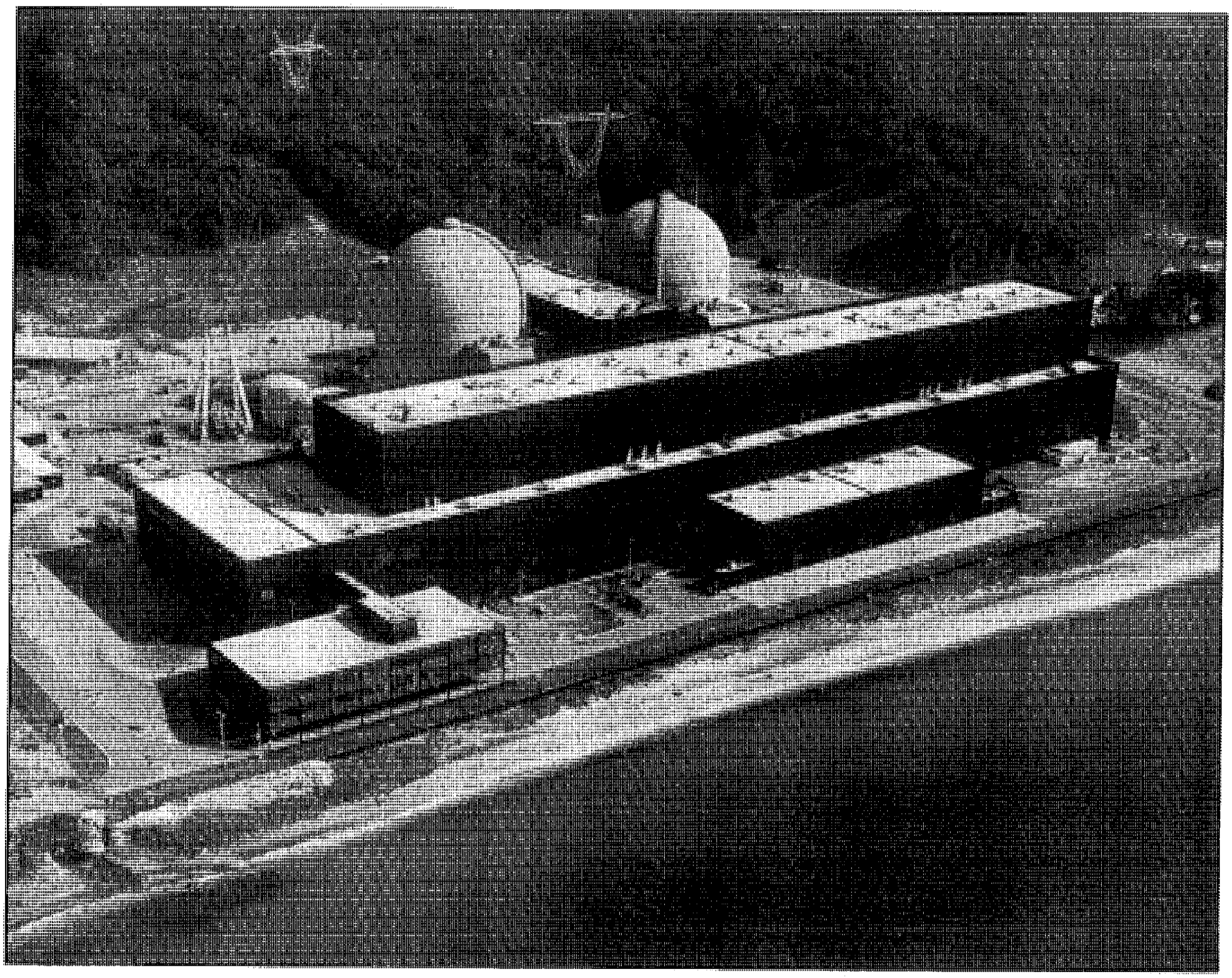
Another important staff function has been to provide radiation protection evaluation of low-level waste handling and disposal activities at power reactors. In this area, the staff has evaluated proposals from the D.C. Cook (Mich.), Pilgrim (Mass.), St. Lucie (Fla.), and Turkey Point (Fla.) plants, for the on-site disposal of wastes contaminated with very low levels of radioactivity. Besides these reviews, the staff participated in a meeting with local citizens, in conjunction with a proposed interim storage facility for low-level solid waste at the Perry (Ohio) plant. In the area of generic communications on radiation protection matters, during the report period, Information Notices were prepared and issued on such subjects as the improper control of radiography activity at nuclear power plants (various Region III plants), and the discovery of an intense radiation beam in containment at the Limerick (Pa.) facility.

NRR staff provided significant technical support to the Office of Nuclear Regulatory Research to ensure smooth implementation of the major revision of 10 CFR Part 20. The support focused on the development and preparation for public comment of 10 regulatory guides, and the issuance of six of these guides in final form, all associated with the revised 10 CFR Part 20. In order to provide regional inspectors guidance and to assist the licensees in implementing the revised rule, NRR staff also undertook a question-and-answer process (Q&A), dealing with the implementation questions of both licensees and inspectors. More than 450 Q&As were prepared; this process will be continued.

All licensed U.S. nuclear power plants are required under Federal regulations to periodically measure samples from the environment outside the boundaries of the plant site for indications of radioactivity originating in the plant. This environmental monitoring program is conducted to verify that measurable concentrations of radioactive material and levels of radiation are not higher than allowed or expected, based on a measurement of plant effluents and the analytical modeling of the environmental exposure pathways. In turn, the studies certify that the plant is in compliance with regulations and that the releases measured do not exceed the amounts defined in the Final Environmental Statements as representing very small risks to members of the public.

## Environmental Radioactivity Near Nuclear Power Plants

Extensive weekly and monthly monitoring is required for each plant by its Radiological Effluent Technical Specifications (RETS) or by effluent control procedures in licensee-controlled documents which provide the overall level of effluent management and control required by the



**NRC staff evaluated a number of proposals from reactor licensees for the on-site disposal of wastes containing very low levels of radioactivity. Among those proposing on-site low level waste disposal was the Indiana & Michigan Electric Company, licensee for the Cook (Mich.) plant shown here. The plant is on the eastern shore of Lake Michigan near the Michigan-Indiana border.**

Technical Specifications. The radiological environmental monitoring program records when, if ever, radioactive contamination above natural background is detected outside the plant boundaries. Samples come from sources that such as lake, river, and well water, for water-borne contaminants; radio-iodine and particulate dusts, for airborne contaminants; milk, fish, shellfish and vegetables, for contaminants that might be ingested as foods. Direct radiation from each of up-to-16 specific sectors of land surrounding the plant is also measured, by special radiation dosimeters that gauge the cumulative radiation dose at locations in each sector for each calendar quarter.

Results of all licensee measurements in their radiological environmental monitoring program are recorded in an annual radiological environmental report, submitted each May for the preceding calendar year. These reports for each year of operation of a power reactor are available for public inspection in Local Public Document Rooms (LPDRs; see Appendix 3 for listing).

Independent from, but supplemental to, these licensee monitoring programs are two programs conducted by the NRC. In one, the direct radiation in the sectors surrounding each plant is measured independently by NRC dosimeters at locations similar to those of the licensee. The results of these measurements for each power reactor site, from the NRC Direct Radiation Monitoring Network, are published quarterly in NRC documents, also available in the LPDRs.

In addition, NRR sponsors, through the five Regional Offices, contracts with 34 States for the States to carry out environmental monitoring. The State contracts establish policies and procedures under which the States independently monitor the environs of the NRC licensed facilities. The States collect samples or make radioactivity measurements in the environs of licensed facilities. The measurements duplicate, as closely as possible, certain parts of the licensee's environmental monitoring efforts, but they are executed independently of the licensee. Results of State monitoring are used to confirm the results of licensee monitoring programs.

## Occupational Exposure Data And Dose Reduction Studies

The NRC staff has been collating the annual occupational doses at light water reactors (LWRs) since 1969. Although the annual dose averages for both pressurized water reactors (PWRs) and boiling water reactors (BWRs) have fluctuated over the years, the overall trend between the early 1970s and 1980 was one of increasing annual dose averages. Annual dose averages peaked in the early 1980s, mainly because of the NRC-mandated plant upgrades imposed on all LWRs shortly after the 1979 accident at Three Mile Island (Pa.). Since 1983, the annual average doses for both PWRs and BWRs have been steadily declining.

In 1992, the average collective dose-per-unit for all LWRs was 266 person-rems. This is 5 percent higher than the 1991 average of 253 person-rems. The small increase is the result of an increased number of refueling outages for BWRs in 1992.

In 1992, the average collective dose-per-unit for PWRs was 219 person-rems, down 2 percent from the average dose-per-unit of 223 person-rems in 1991. The activities which most frequently contributed to PWR doses in 1992 were steam generator-related work, refueling, valve maintenance and repair, health physics surveys and inspections, and in-service inspections.

In 1992, the average collective dose-per-unit for BWRs was 360 person-rems. This is 11 percent higher than the average dose-per-unit for BWRs of 324 person-rems in 1991. The increase is due, primarily, to the 28 percent increase in the number of outage hours reported for refuelings at BWRs in 1992. Major contributors to BWR doses in 1992 included valve maintenance and replacement, in-service inspections, health physics support, drywell work, control rod drive replacement and repair, and refueling activities.

The 1992 dose compilation includes data from 73 PWRs and 37 BWRs, for a total of 110 light-water reactors. Plants which have not been in commercial operation for a full year are not included in this compilation. One PWR, Yankee-Rowe (Mass.), has been permanently shut down and has been dropped from this year's annual listing. Other plants no longer in operation and not included in the dose compilation are Dresden Unit 1 (Ill.), Fort St. Vrain (Colo.), Humboldt Bay (Cal.), Indian Point Unit 1 (N.Y.), LaCrosse (Wis.), Rancho Seco (Cal.), and Three Mile Island Unit 1 (Pa.).

The NRC has ongoing contracts with the Brookhaven National Laboratory (BNL) in the area of occupational dose reduction at LWRs. The NRC-sponsored program monitors U.S. and foreign nuclear power plant efforts to reduce occupational dose. Under the contract, BNL publishes the periodical *ALARA Notes*, which contains ALARA-related information submitted by U.S. and foreign nuclear power plants. (ALARA is an acronym for "as low as reasonably achievable," the criterion characterizing the dose-reduction objective.) As part of this contract, BNL is also involved in the compilation of an ongoing annotated bibliography of selected readings in radiation protection and ALARA. Other BNL studies for the NRC include a study of the impact of reduced dose limits, and a study of hot particle production, mitigation and dosimetry. The NRC also has an ongoing contract to evaluate the effects of hydrogen water chemistry on shutdown radiation levels at BWRs.

## Implementation Status of TMI And Other Safety Measures

The NRC publishes a document annually giving the status of the implementation and verification of licensing actions related to major safety issues. The most current report includes the status, as of September 30, 1992, of implementation and verification of all safety-issue actions affecting multiple facilities: TMI (Three Mile Island) Action Plan Requirements, Unresolved Safety Issues (USI), Generic Safety Issues (GSI), and, for the first time, all other multi-plant actions. As noted in the report published in December 1992, more than 99 percent of the TMI Action Plan items have been implemented at the 109 licensed plants; approximately 88 percent of the USI items have been implemented; approximately 90 percent of the GSI items have been implemented; and approximately 84 percent of other multi-plant action items have been implemented.

## Fill-Oil Loss in Rosemount Pressure Transmitters

On April 21, 1989, the NRC issued Information Notice 89-42, "Failure of Rosemount Models 1153 and 1154 Transmitters," to alert the industry to a series of reported failures of Models 1153 and 1154 pressure and differential pressure transmitters, manufactured by the Rosemount Inc. Rosemount investigated the cause of the failures and confirmed that the failure was of a glass-to-metal seal inside the sensor which allowed fill-oil to leak out of the sensor at a very slow rate. When this condition occurred, the transmitter performance gradually deteriorated and lead to failure. Rosemount attributed many of the failures to the use of stainless steel "O" rings and the increased stresses on the sensor module that result.

On March 9, 1990, the NRC issued Bulletin 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," in which it requested that licensees promptly identify and take appropriate corrective actions on Model 1153 Series B, Model 1153 Series D, and Model 1154 transmitters manufactured by Rosemount, that might or did have the potential for leaking fill-oil. These actions included removing certain transmitters from reactor protection and engineered safety feature actuation systems.

The staff continued to review the Rosemount transmitter loss of fill-oil issue by analyzing data gathered from (1) licensee event reports, (2) the licensee's responses to NRC Bulletin 90-01, (3) technical information provided by Rosemount, (4) site visits, (5) NUMARC report 91-02, "Summary Report of NUMARC Activities to Address Oil Loss in Rosemount Transmitters," (6) numerous meetings with representatives from the nuclear power industry, NUMARC, and Rosemount, and (7) a Brookhaven National Laboratory evaluation of Rosemount transmitter failures. Based on these data, it was determined that the failures were more dependent upon the operating pressure and the amount of time that the transmitter had been in service than upon the manufacturing lot of the transmitters. New techniques were also developed to identify failing transmitters. With this information, the staff issued Supplement 1 to Bulletin 90-01 on December 22, 1992. The supplement requested that utility licensees perform enhanced surveillance testing on the Rosemount transmitters, commensurate with their importance to safety and demonstrated failure rate, and inform the staff of the manner in which the enhanced surveillance program was being implemented. All nuclear power plants have responded. NRC staff review of these responses will continue through 1994.

On May 21, 1993, the NRC Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research, established the Rosemount Transmitter Review Group (RTRG) to ensure that the staff was aware of all the technical information necessary to effectively respond to the Rosemount transmitter failure concerns. The RTRG was tasked to perform an in-depth review and evaluation to determine whether the agency should require licensees to take action beyond that specified in Bulletin 90-01 and Supplement 1 to Bulletin 90-01. The RTRG completed its evaluation, and issued its report on October 12, 1993. The principle conclusions of the RTRG were that the scope and actions specified in NRC Bulletin 90-01, Supplement 1, are appropriate and that improvements in Rosemount Model 1153 B/D and 1154 transmitters manufactured since July 11, 1989, have significantly reduced the transmitter failure rate. The RTRG, however, also recommended that the following actions be taken: (1) issue a temporary instruction for NRC inspections of the effectiveness of licensee actions in response to Bulletin 90-01, Supplement 1, and collect data on calibration trending and failures of all Rosemount transmitters; (2) continue periodic dialogue with Rosemount to track the performance of the different models of transmitters; (3) review Nuclear Plant Reliability Data System data on Rosemount transmitters every six months for two years; (4) hold management meetings with NUMARC to discuss lessons learned from the Rosemount transmitter loss-of-fill-oil issue; (5) review EPRI Report TR-102908, dealing with Rosemount transmitter concerns; and (6) ask the NRC Office of General Counsel to provide a written legal interpretation regarding the circumstances under which organizations such as NUMARC and EPRI would be subject to the requirements of 10 CFR Part 21 and 10 CFR Part 50.9 for reporting defects and noncompliance. The above actions are to be implemented in 1994 and will provide the NRC with more information to ensure that actions taken in response to NRC Bulletin

90–01, Supplement 1 are sufficient to resolve the Rosemount transmitter concerns.

## Thermo-Lag Fire Barrier Systems

Following a fire at the Browns Ferry nuclear power plant (Ala.) in 1975, a Special Review Group (SRG) was established to identify lessons learned and to make recommendations for corrective actions. The SRG concluded that improvements in fire protection programs were needed, and, in 1981, the Commission issued 10 CFR 50.48 and Appendix R to 10 CFR Part 50 to require the added protection. The regulations were to apply to nuclear power plants licensed to operate before January 1979; three sections in Appendix R were considered important enough, however, to be made applicable to all plants. These three sections deal with the protection of safe shutdown capability, emergency lighting, and the reactor coolant pump oil collection system. Section III G.1.a, "Fire Protection of Safe Shutdown Capability," specifically addresses requirements involving the protection of safe shutdown systems. It requires that one train of systems necessary to achieve and maintain hot shutdown conditions, from either the control room or emergency control stations, shall be free from fire damage. Licensees can satisfy the requirement by separating redundant safe shutdown trains located within the same fire area outside primary containment, achieving the separation by providing one of the following: (1) a horizontal distance of at least 20 feet, with no intervening combustibles plus installed fire detectors and an automatic suppression system; (2) a three-hour rated fire barrier; or (3) a one-hour rated fire barrier, with fire detectors and automatic suppression.

In 1981, the NRC began receiving requests from licensees for the approval of Thermo-Lag 330–1, manufactured by Thermal Science, Inc., of St. Louis, Mo., as a rated fire barrier to satisfy the NRC's new fire protection requirements. The vendor manufactures Thermo-Lag in pre-shaped panels and molds of varying thicknesses providing one-hour and three-hour fire endurance. Thermo-Lag is a "sacrificial subliming" material that is consumed when it is exposed to a fire. When it is heated by a fire, the solid material sublimes, the subliming gases are decomposed by the fire, and the virgin Thermo-Lag material is replaced by a char layer. The sublimation process and the insulating effects of the resulting char layer protect the equipment located within the confines of the fire barrier from the effects of the fire. More traditional fire barriers, such as concrete block walls, provide fire endurance by maintaining structural integrity during the fire exposure and limiting heat transfer through the barrier.

Currently, Thermo-Lag fire barriers are installed in a majority of operating plants, in order to meet the requirements of 50.48 for the safe shutdown capability. Thermo-Lag is mainly used to separate redundant raceways, by surrounding one of the raceways within a Thermo-Lag enclosure. Some licensees have also used Thermo-Lag to construct walls, ceilings, and vaults.

Between 1982 and 1991, the NRC received sporadic reports of problems associated with the use of Thermo-Lag. By June 1991, the NRC had information about problems at the River Bend (La.) nuclear power plant which substantiated previous questions regarding the adequacy of Thermo-Lag as an effective fire barrier. The NRC established a Special Review Team to review the issues and make recommendations for their resolution. A final report was issued in April 1992. The team concluded that (1) the fire resistance ratings and the ampacity derating factors for the Thermo-Lag 330–1 fire barrier system were indeterminate, (2) some licensees had performed an inadequate review and evaluation of fire endurance test results and ampacity derating factors (actual cable temperatures may exceed the expected temperatures, accelerating aging of the cable insulation) to confirm the validity of the tests and their applicability to their plants, (3) some licensees had not adequately reviewed installed fire barriers to assure conformance with NRC requirements, and (4) some licensees had used inadequate or incomplete installation procedures.

In addition, subsequent qualification tests conducted by the nuclear industry and small-scale panel tests performed by the NRC at the National Institute of Standards and Technology (NIST) demonstrated that certain Thermo-Lag fire barrier configurations may not provide the level of fire resistive protection needed to satisfy the NRC's requirements. Furthermore, some Thermo-Lag barriers used by some licensees, as in walls and ceilings, have not been qualified as fire barriers by test.

The staff incorporated these and other issues into an action plan to assure that the issues are tracked, evaluated and resolved. There has been a high level of Congressional and intervenor interest in the matter. The NRC staff has responded to several petitions submitted pursuant to 10 CFR 2.206 requesting that all nuclear plants that use Thermo-Lag be shut down until the operability of Thermo-Lag barriers can be effectively demonstrated. The Commissioners testified before the Subcommittee on Oversight and Investigation of the Committee on Energy and Commerce, in the House of Representatives, in March 1993, on fire safety at nuclear power plants, particularly focusing on the Thermo-Lag issues. The staff has completed a reassessment of the NRC reactor fire protection program, and issues raised in its reviews are being addressed by the staff and tracked in an action plan. The staff is also evaluating fire barrier materials other than Thermo-Lag and has conducted small-scale tests at NIST. Besides the special review team report, the staff has issued seven Information Notices to the industry (including two on fire barriers other than Thermo-Lag), a Generic Letter, a bulletin and a bulletin supplement; developed a

Generic Letter supplement clarifying fire endurance test criteria; reviewed various industry full-scale test programs; and conducted toxicity and combustibility tests.

The staff continues to work closely with the NUMARC and with individual licensees to review and monitor industry fire tests, ampacity derating tests, and other industry initiatives. Licensees are implementing compensatory measures, such as fire watches, where Thermo-Lag is installed until long term corrective actions can be implemented. These actions will be based, in part, on the results of a test program developed by NUMARC for the nuclear industry. The program includes construction and testing of baseline and upgraded Thermo-Lag fire barriers representative of in-plant configurations. Upon completion of the testing in 1994, NUMARC will prepare an application guide for licensees to apply the test results to specific in-plant configurations and to determine whether the installed fire barriers meet NRC fire protection requirements. When installed barriers do not meet the requirements, the licensees may choose to repair, upgrade or replace the existing barriers. The NRC staff is also in the process of identifying the in-plant configurations which fall outside of the industry test program. The licensees for these plants may need to implement alternative plans such as additional testing and analyses. More plant-specific analyses may also be required to resolve the ampacity derating problem. Regulatory action and coordination with the industry will continue until the technical and programmatic issues in the staff's action plan have been resolved.

## Boiling Water Reactor Instability

Boiling Water Reactors (BWRs) may be subject to thermal-hydraulic and neutronic driven power oscillations when operating at low flow and relatively high power during, for example, reactor startup or loss-of-flow transients. The staff and the BWR Owners' Group (BWROG) have been reviewing safety issues which may arise from these oscillations. The review was prompted by an instability event on March 9, 1988, at the LaSalle (Ill.) nuclear power plant. The review has considered (1) the causes and characteristics of oscillations, (2) the replacement of current corrective actions by long term solutions, and (3) the possible effects of large oscillations on anticipated transients without scram (ATWS).

Previous staff actions have included issuance of NRC Bulletin 88–07 and Supplement 1 requesting that BWR licensees take specified interim actions to prevent significant oscillations until long term resolutions can be developed. Such interim actions have been generally effective in increasing awareness of the problem among reactor operators. The staff and its consultants and the BWROG have engaged in a coordinated effort to improve understanding of instability phenomena and the principal fuel, core design and operating parameters contributing to them. Substantial effort was required to develop computer codes and to validate these codes. Based on this improved understanding, analyses have been performed by the BWROG and the staff to develop and evaluate long term solutions to facilitate the detection and suppression of oscillations and evaluations of ATWS events.

The BWROG has proposed to resolve the instability issue by ensuring an automatic protection action (i.e., reactor scram or selective control rod insertion) to prevent power oscillations that could violate fuel safety limits. The BWROG has proposed several options for implemented the proposed measures. The two primary options involve (1) an exclusion region on the power/flow map, outside of which instability is very improbable and inside of which automatic control rod insertion occurs to exit the region; and (2) a local-power-range-monitor (LPRM) based detection and suppression system, in which signals from a core-wide distribution of small groups of LPRMs are analyzed on-line, using diverse characteristics of oscillation signals to detect instability and cause rod insertion. The staff review has found these solutions acceptable when augmented, in some cases, by procedures to monitor core power distribution or stability. The staff has issued and received public comments on a draft Generic Letter (*Federal Register*, Vol. 58, No. 138, Page 39044, July 21, 1993) defining requirements for an acceptable long term solution and strengthening interim administrative controls. The public comments are under review for final disposition and publication of the Generic Letter. A staff safety evaluation report on the BWROG topical reports describing proposed long term solution options from which utilities may choose has been issued. It concludes that, at this time, three of the proposed options are acceptable, subject to certain conditions. A fourth option was under review at the close of the report period.

The staff review has concluded that for some ATWS events with a very low probability of occurrence large oscillations are possible and could lead to melting of a small fraction of the fuel. However, containment integrity will be maintained, and the radiological consequences will remain within 10 CFR Part 100 limits. Furthermore, revisions to Emergency Operating Procedures have been proposed which would limit power oscillations during ATWS events. While these proposed procedures are generally acceptable for protection against large oscillations, a review seeking to optimize some of the procedures (e.g., pressure vessel water level) for overall best ATWS protection is continuing.

## Boiling Water Reactor Water Level Instruments

During the past year the staff has continued its appraisal of the potential for inaccurate reactor vessel water level indication in boiling water reactors (BWRs). Under certain conditions, the reference legs of the water level instrumentation can become saturated with dissolved non-condensible gases. In the event of a reactor depressurization, these gases come out of solution and could displace water from the reference leg. Loss of reference leg inventory would result in a "false high" indication of reactor vessel water level.

Temporary Instruction (TI) 2515/119 was issued on March 31, 1993, giving instructions for inspection of the compensatory actions taken by licensees in response to Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs pursuant to 10 CFR 50.54(f)," dated August 19, 1992. Inspections were completed by May 31, 1993, confirming that licensees had sensitized their operators to this phenomenon and that the operators had received adequate guidance to properly respond to such an event.

An event occurred during a normal plant cooldown on January 21, 1993 at the Washington Nuclear Unit 2 (WNP-2) plant, resulting in a 32-inch error in level indication which gradually recovered over a period of two hours. NRC Information Notice 93-27, "Level Instrumentation Inaccuracies Observed During Normal Plant Depressurization," was issued on April 8, 1993, to alert licensees to the potential for significant errors during normal depressurization.

By spring of 1993, the BWR Owners' Group (BWROG) had completed a reference leg de-gas program, intended to resolve the issue. The results of the program confirmed that significant errors in level indication could occur under certain conditions. The test results also confirmed earlier calculations which showed that significant errors would not occur until depressurization below 450 p.s.i., and that automatic safety system response would not be affected for events initiated at full reactor pressure.

Based on the results of the reference leg de-gas testing conducted by the BWROG and analysis of the WNP-2 event, the staff concluded that short term compensatory actions were necessary, besides those already taken in response to Generic Letter 92-04, for protection against potential events occurring during normal cooldown, and that hardware modifications to resolve this concern should be made promptly. On May 28, 1993, NRC Bulletin (NRCB) 93-03 was issued, requesting that each BWR licensee take compensatory action and implement hardware modifications to resolve the problem at the first cold shutdown after July 30, 1993. All affected licensees have completed the requested short term actions and have committed to effecting the hardware modifications at the next shutdown of sufficient duration.

Delays in implementation were granted to some licensees to allow time to complete necessary design and hardware procurement to support the modification. The majority of licensees have chosen to implement a continuous reference leg backfill system which provides constant flushing of the reference leg with water supplied at low flow rates from the control rod drive hydraulic system. This backfill modification has already been installed at several plants. To address some design and implementation problems identified by licensees with the backfill system, the staff issued Information Notice 93-89, "Potential Problems with BWR Level Instrumentation Backfill Modifications," on November 26, 1993. The staff will be issuing a Temporary Instruction which will provide guidance for the inspection of hardware modifications.

## Individual Plant Examination

In November of 1988, the Commission issued Generic Letter No. 88-20, requiring each licensee and construction permit holder to conduct an individual plant examination (IPE), by which to systematically search for any significant contributors to core damage risk. The Commission encouraged the use of probabilistic risk analysis (PRA) in the carrying out the examination. Specific guidance regarding IPE content for internal events (e.g., failure of various plant systems) was issued to all licensees in "Individual Plant Examination: Submittal Guidance" (NUREG-1335, August 1989). Further guidance for individual plant examinations for external events (IPEEE) —including earthquakes, fire, wind and floods—was issued in June 1991, as NUREG-1407, with Supplement 4 to Generic Letter No. 88-20. Most licensees will make two separate submittals to the NRC, one for IPE (internal events only) and one for IPEEE. The NRC's expectation is that when significant contributors to core damage risk (sometimes called "outliers") are discovered, prompt action will be taken by the licensee to modify plant design or operation to reduce the risk. Sixty-three IPE reports were submitted to the NRC through fiscal year 1993, covering 90 nuclear units. The NRC staff completed review of seven of these reports in fiscal year 1993, bringing the total reviewed to 11. In certain cases, the IPE process has indeed led to discovery of outliers, and licensees have taken appropriate corrective action to reduce core damage risk.

As part of the review process, the staff extracts from IPEs noteworthy insights for dissemination within the NRC and throughout the industry. For example, the NRC is preparing to issue an Information Notice alerting licensees of potential problems resulting from common dependencies in component cooling water (CCW) systems. CCW systems provide cooling for reactor coolant pump (RCP) seals, and in many cases they also supply cooling water to emergency core cooling system (ECCS) components. Consequently, failure of the CCW system could cause a loss of RCP seal cooling leading to an RCP seal loss-of-coolant (LOCA) accident (LOCA) and could also disable the necessary accident mitigation systems. Several licensees have identified these common dependencies as contributors to the total plant core damage frequency in their IPEs. The following illustrates the kinds of measures taken by utilities to reduce the vulnerabilities to such dependencies and indicates the benefits available from structured examinations of the IPE program.

The Turkey Point (Fla.) IPE led the licensee to modify charging pumps, so that the service water system can be aligned as an alternate cooling water supply to any of the charging pumps, with the result that a loss of the CCW system alone would not disable all of the ECCS components required for LOCA mitigation.

During development of the Robinson (S.C.) IPE the licensee identified loss of CCW as a significant accident initiator and implemented procedural revisions that allow the alignment of an alternate cooling supply to the charging pumps by connecting the firewater system to existing fittings on the charging pump couplings.

The Farley (Ala.) IPE led the licensee to implement new procedural guidance addressing the issue of loss of the CCW system. In particular, instructions for routing a temporary cooling water supply to the charging pump oil coolers via the fire protection water system were instituted to ensure operability of the charging pumps.

As a result of the Diablo Canyon (Cal.) IPE, hose connections and dedicated hoses were provided and operational procedures were changed to facilitate the use of the fire water system as an alternate means of cooling the charging pumps which supply cooling water to the RCP seals. The D. C. Cook (Mich.) IPE identified loss of the CCW system leading to an RCP seal LOCA as a dominant contributor to the plant core damage frequency, and the licensee is currently investigating changes to operational procedures to instruct the operator to open the cross-tie valve of the chemical and volume control system of one unit to the opposite unit early in the accident response, in order to provide RCP seal cooling and to prevent seal damage.

## Environmental Qualification of Electric Equipment

Under 10 CFR 50.49, issued in 1983, it is required that licensees establish a program for qualifying electrical equipment important to safety of plant operation during and following "design basis" events. Licensees were not required to re-qualify equipment qualified under earlier NRC directives. Reactor licensees developed environmental qualification (EQ) programs that were appraised by NRC staff. Qualification for many components in power plants was predicated on a plant life of 40 years, the term of the plant's operating license. As a result of its review of license renewal issues in 1992, the staff concluded that differences in EQ requirements constituted a potential generic issue to be evaluated independently of license renewal reviews.

During development of an inter-office action plan addressing the differing EQ requirements for older plants, the staff evaluated the technical adequacy of EQ requirements. In doing so, the staff reviewed recent tests of qualified cables performed by Sandia National Laboratories (SNL), under contract with the NRC. The purpose of the tests was to determine the effects of aging on cable products used in nuclear power plants. After accelerated aging, some of the environmentally qualified cables either failed or exhibited marginal insulation resistance during accident testing, indicating that qualification of some electric cables may have been "non-conservative." The SNL test results raise questions with respect to the environmental qualification and accident performance capability of certain artificially aged cables. Depending on the application, failure of these cables during or following design basis events could affect the performance of safety functions in nuclear power plants.

Independent of the SNL tests, the staff also performed a preliminary risk-scoping analysis of the potential impact of failures of environmentally qualified equipment on core damage frequency. The preliminary analysis concluded that (1) EQ failures could have significant risk impact if electric component reliability is reduced in the presence of a harsh environment, (2) the magnitude of the impact on core damage frequency is plant specific, and (3) lack of reliability data bases and limitations in current probabilistic risk assessment models combine to produce significant uncertainty in these preliminary results. In the preliminary risk-scoping assessment report, the staff made recommendations for further evaluation of the risk impact of EQ.

The staff's action plan addressing EQ issues includes a review of the current EQ program, review of licensee operating experience, additional risk assessment of the importance of EQ equipment, and research into qualification testing techniques and equipment condition monitoring. The programmatic review involves a look back at the EQ requirements and the basis for the different requirements, as well as a review of the adequacy of the requirements and their implementation. The staff is reviewing operating experience to determine whether there are significant problems with EQ in the industry and to focus research on those problems. The preliminary risk assessment will be refined and available reliability data will be analyzed. The staff will determine what further research is necessary in the areas of accelerated aging, condition monitoring techniques, and accident testing. The staff's technical analysis on EQ is expected to be complete by the end of fiscal year 1994.

## ECCS Strainer Blockage in BWRs

Unresolved Safety Issue (USI) A-43 deals with concerns for the performance of safety-related pumps in containment during an emergency. The principal concern was the potential loss of net positive suction head (NPSH) resulting from clogging of the suction strainers by fibrous debris. As part of that effort, Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," was revised to provide adequate assurance that debris from thermal insulation would not interfere with the performance of the pumps. The revision was based on engineering tests and analyses. However, based on an evaluation of low risk significance, the issue was resolved in 1985 without backfitting operating plants or plants under construction. Recent operational experience in the United States and abroad indicates that the potential for strainer clogging may be more significant than was perceived at the time USI A-43 was resolved.

Because of an event that occurred on July 28, 1992, at a Swedish BWR, Barsebck 2, the NRC staff is performing additional technical studies related to this issue. At Barsebck 2, while the reactor was operating at low power during restart, a safety valve for the reactor coolant system that discharges to the drywell opened. Coolant flowing from the discharge pipe stripped fibrous thermal insulation from piping located in the vicinity of the valve. This debris was transported to the suppression pool by the flow of water from the reactor coolant and containment spray systems. Strainers on the suction side of the containment cooling system were clogged with debris in an hour, causing pump cavitation. Although clogging had been anticipated by the Swedish regulatory authorities, it proceeded 10 times faster than was expected. On September 30, 1992, the NRC issued IN 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR."

In January of 1993, while Perry Unit 1 (Ohio) was shut down, the licensee discovered that two ECCS strainers were clogged with particulates and deformed by hydraulic forces. On April 26, 1993, in response to the condition discovered at Perry and the identification of a significant source of material at the Grand Gulf (Miss.) plant with the potential to restrict the flow through the sump-debris screen, the NRC issued Information Notice (IN) 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment."

In March 1993, two months after the strainers at Perry had been replaced and the suppression pool had been cleaned, the licensee discovered during testing that a strainer was again clogged. The material clogging the strainer consisted primarily of glass fibers from drywell ventilation roughing filters that had been inadvertently dropped into the suppression pool and of corrosion products that had been filtered from the pool by the glass fibers' adhering to the surface of the strainer. The strainers, once clogged with fibrous material, had acted as filters, progressively filtering out finer material, and developing larger pressure drops than previously anticipated. Supplement 1 to IN 93-34 issued on May 6, 1993, described the deposition of filter fibers on residual heat removal strainers, which had occurred.

The consequences of the filtering action of the fibrous material on the strainer was beyond the scope of Unresolved Safety Issue A-43, which addressed the transport of fibrous thermal insulation from the containment to the strainers during a LOCA. The Perry event showed that filtering of corrosion products, dust and other debris from the drywell, as occurred at Perry, may cause a loss of net positive suction head for the ECCS pumps when they are needed to perform their intended function.

Based on the identification of this previously unrecognized failure mode for ECCS recirculation—attributed to the synergistic effect between the filter material and debris—the NRC issued NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," on May 11, 1993. The bulletin discussed several instances of ECCS suction blockage resulting from the filtering action of fibrous material. All operating reactor licensees were requested to identify fibrous air filters, or other temporary sources of fibrous material not designed to withstand a LOCA, which are installed or stored in their primary containment, take prompt action to remove any such material, and take any immediate compensatory measures which may be required to assure the functional capability of the ECCS. Licensees were required to provide a written response stating whether the actions requested have been or will be performed, the locations and quantity of any identified material, and any immediate compensatory measures taken. Reports on the completion of the requested actions and a justification for any deviations from the requested actions were also required.

The responses to NRC Bulletin 93-02 indicate that approximately 95 percent of the licensees do not need, or have already performed, necessary corrective actions. For the remainder of the licensees whose responses did not provide sufficient information, further review will be performed by the staff. It is expected that the issue will be resolved for all facilities within one year of the date of the Bulletin.

The staff also has in place a program to systematically evaluate the larger implications of the Barsebck and Perry experience. This will include consideration of strainer area, containment housekeeping, pool cleanliness, strainer testing, and measures to cope with clogged strainers.

## BWR Core Shroud

During 1993, the NRC determined that cracking of boiling water reactor (BWR) internals represents an emerging technical issue for the agency. Of particular note in 1993 were the reports of cracking discovered in the core shrouds of the Brunswick Unit 1 (S.C.) and Peach Bottom Unit 3 (Pa.) reactors. The core shroud is a stainless steel cylinder which partitions feedwater in the reactor vessel's downcomer annulus region from cooling water flowing through the reactor core.

During the summer 1993 refueling outage at the Brunswick Unit 1 reactor, the Carolina Power & Light Company (CP&L, the licensee) performed an in-vessel visual examination of the Brunswick Unit 1 core shroud. Upon review of the results, the licensee determined that the shroud contained a significant amount of cracking. CP&L performed the examinations according to the recommendations in General Electric Corporation (GE) Rapid Information Services Letter (RICSIL) 054, "Core Support Shroud Crack Indications," issued as a summary of cracking discovered in the core shroud of an overseas boiling water reactor (BWR), in 1991.

The licensee has determined that the most severe crack was a 360° circumferential crack associated with the weld (H-3 weld) which fuses the top guide support ring to the lower shroud cylinder. The crack is located in the heat affected zone of the weld and extends about 1.7 inches or deeper into the support ring. The licensee has also determined that the other axial or circumferential cracks in the heat affected zones of welds associated with upper shroud (H-1 and H-2 welds) and shroud beltline region (H-4, H-5 and H-6 welds) are of lesser safety significance, and therefore do not require a repair or modification for operation of another fuel cycle. CP&L installed a number of mechanical clamps around the H-3 weld to ensure shroud structural integrity. The NRC staff has performed a Core Shroud Cracking Preliminary Safety Assessment.

Information Notice 93-79, "Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors," was issued to inform BWR licensees of the cracking discovered in the Brunswick Unit 1 core shroud. GE issued Safety Information Letter (SIL) 572 and its revision, Rev. 1., to inform the industry with respect to the cracking in the Brunswick Unit 1 core shroud. GE's SIL also provided GE's latest recommendations for performing core shroud inspections during BWR refueling outages.

The Boiling Water Reactor Owners Group (BWROG) has developed a Core Shroud Cracking Action Plan. The BWROG worked in conjunction with GE to develop the generic safety assessment. The Action Plan also includes plans for compiling and evaluating the data provided by BWR licensees who have performed shroud inspections during 1993 fall/winter refueling outages, and for developing generic core shroud inspection guidance and acceptance criteria. The NRC staff is closely following this industry initiative and will review the generic core shroud inspection guidance and acceptance criteria when they are finalized and submitted in late 1993.

## Operational Safety Assessment

The NRC headquarters staff participates with the regional staff in the review and follow-up of events at operating nuclear reactor facilities, identifying items of generic significance and determining whether an ordered derating or shutdown of a plant is indicated. These reviews involve evaluating events against existing safety analyses, appraising plant and operator performance during events, reviewing licensee analyses, and deciding if there is any need for corrective action.

In fiscal year 1993, the NRC assigned augmented inspection teams, part of the formal program for the assessment of major incidents, to determine the facts regarding the following operating reactor events:

- Unreported failure of all control room annunciators at Callaway Unit 1 (Mo.), in October 1992.
- Reactor trip with loss of off-site power at Oconee Unit 2 (S.C.), in October 1992.
- Unreported failure of control room annunciators at Salem Unit 2 (N.J.), in December 1992.
- Degraded shutdown cooling at Oyster Creek Unit 1 (N.J.), in January 1993.
- Scram with complications at South Texas Units 1 and 2 (Tex.), in February 1993.
- Uranium oxide powder spill at Siemens Power Corporation (Wash.), in February 1993.
- Extraction steam header rupture at Sequoyah Unit 2 (Tenn.), in March 1993.
- Steam generator tube leak and manual reactor trip at Palo Verde Unit 2 (Ariz.), in March 1993.
- Service water pipe break in main service water header resulted in flooding at Perry Unit 1 (Ohio), in March 1993.
- Single failure in rod control system at Salem Unit 2 (N.J.), in May 1993.
- Loss of off-site power while doing circuit breaker testing at Haddam Neck (Conn.), in June 1993.
- Broken fuel rod and stuck fuel assembly at Palisades (Mich.), in July 1993.
- Main steam leak through steam generator secondary drain at McGuire Unit 2 (N.C.), in August 1993.
- Fuel handling events at Vermont Yankee (Ver.), in September 1993.
- Loss of off-site power and reactor scram at LaSalle Unit 1 (Ill.), in September 1993.

Also as part of the formal program for the assessment of major incidents in fiscal year 1993, the staff assigned incident investigation teams to investigate in-depth the safety and regulatory implications of the following events:

- Loss of iridium-192 source and therapy misadministration at Indiana Regional Cancer Center (Ind.), in November 1992.

- Unauthorized forced entry into protected area at Three Mile Island Unit 1 (Pa.), in February 1993.

When generic problems are identified in the course of staff reviews of reported events and problems, a number of actions that may be taken by the NRC. If warranted, Information Notices are issued, notifying utilities of conditions or problems that could affect their plants. Utilities are expected to review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. Bulletins and Generic Letters have a similar function but may request that specific actions be taken by utilities and require written confirmation when such actions have been completed. In fiscal year 1993, the NRC began issuing a new class of generic communication, called Administrative Letters, to transmit information to the utilities that is essentially administrative in nature. In fiscal year 1993, the staff issued 99 Information Notices, including one revision and six supplements; six bulletins, including two supplements; nine Generic Letters, including two supplements; and four Administrative Letters.

## Cleanup at Three Mile Island

During fiscal year 1993, preparations continued for placing the damaged reactor at the Three Mile Island Unit 2 (TMI-2; Pa.) nuclear power plant in post-defueling monitored storage (PDMS), a passive, monitored state similar to the SAFSTOR option of decommissioning.

In August of 1988, the licensee, GPU Nuclear (GPUN), submitted a Safety Analysis Report (SAR) to document and support their proposal to amend the TMI-2 license to a "possession-only" license and to allow the facility to enter PDMS. The staff issued Final Supplement 3 to the Programmatic Environmental Impact Statement for the TMI-2 decontamination and cleanup, in August of 1989. In February 1992, the staff issued a safety evaluation regarding the PDMS license amendment and a technical evaluation report regarding PDMS. These three NRC staff documents form the basis for the staff position on the acceptability of PDMS. On April 25, 1991, the staff published a notice of opportunity for a prior hearing regarding the licensee's request to amend its license. A member of the public petitioned to intervene in the license amendment proceedings. The petitioner, the licensee, and the NRC staff reached a settlement agreement on September 25, 1992. The request to intervene was withdrawn and on October 16, 1992, the Atomic Safety and Licensing Board dismissed the proceeding.

The reactor building preparations for PDMS were completed in October 1992, and it is now in a pre-PDMS condition. The NRC staff issued a possession-only license on September 14, 1993; the expectation is that TMI-2 will enter PDMS late in the fourth quarter of calendar year 1993 or early 1994. GPUN plans to keep TMI-2 in the PDMS state until they simultaneously decommission TMI-1 and TMI-2 in 2014.

On February 1, 1993, GPUN notified the NRC staff that the current best estimate of the residual fuel in the reactor vessel was 2,040 pounds (925 kilograms), based on data from recently completed fast-neutron measurements. The measurement technique made use of an array of helium filled detectors to measure fast neutrons produced by the residual fuel. The estimate was derived from calculations made by on-site staff and an independent review by an off-site group headed by Dr. Norman Rasmussen of the Massachusetts Institute of Technology. The estimate was reviewed and endorsed by three other independent reviewers from national laboratories.

For the balance of the facility external to the reactor vessel, earlier licensee estimates based on measurements, sample analyses, and visual observations indicated that no more than 385 pounds (174.6 kilograms) of residual fuel remains. The NRC staff and consultants from Battelle Pacific Northwest Laboratories have performed independent evaluations and made independent measurements of these earlier fuel measurements in the auxiliary and reactor buildings. On July 6, 1993, the staff issued an analysis confirming earlier analyses done by the licensee which indicated that the fuel remaining in the TMI-2 reactor vessel will remain subcritical, with an adequate margin of safety, during PDMS.

Evaporation of the treated, accident-generated water began in January 1991, after a prolonged period of system testing, modification and repair. On August 12, 1993, the decontamination and evaporation of 2.23 million gallons of accident-generated water was completed.

The 10-member Advisory Panel for the Decontamination of Three Mile Island Unit 2, held its last meeting during fiscal year 1993. The Panel, composed of citizens, scientists, and State and local officials, was formed by the NRC in 1980 to provide input to the Commission on major cleanup issues. (See Appendix 2 for a listing of the members.) The principal topics discussed at these meetings included the NRC staff Safety Evaluation and technical evaluation report addressing PDMS, the status and progress of cleanup at the TMI-2 facility, and the decommissioning funding status and plans. Two meetings were held in fiscal year 1993: the first was held at NRC headquarters in Rockville, Md., while the last meeting (the 78th overall) was held in Harrisburg, Pa., on September 23, 1993. Commissioner Kenneth Rogers attended the final session to express the Commission's appreciation to the Advisory Panel for their dedication and service over the past 13 years.

## Loss of Spent Fuel Pool Cooling Function

The staff is evaluating a 10 CFR Part 21 report filed on November 27, 1993, contending that the design of a certain reactor facility failed to meet numerous regulatory requirements with respect to a postulated loss of normal cooling function in the spent fuel pool. The report provided a series of detailed technical and regulatory arguments to support the assertion.

**The Advisory Panel for the Decontamination of Three Mile Island Unit 2 held its last meeting during fiscal year 1993. The Advisory Panel had been formed by the NRC in 1980 to provide input to the Commission on major cleanup issues at the TMI site. The last meeting (the 78th overall) was held in Harrisburg, Pa., on September 23, 1993. Commissioner Kenneth Rogers attended the final session to express the Commission's appreciation to the Advisory Panel for their dedication and service over the past 13 years.**

**Panel members attending the final meeting are pictured above. They are, left-to-right, front row: Ann Trunk, Resident of Middletown, Pa.; Arthur E. Morris (Panel Chairman),Resident and former Mayor of Lancaster, Pa.; Joel Roth (Panel Vice Chairman), Resident of Harrisburg, Pa.; Elizabeth Marshall, Resident of York, Pa. In the back row, left-to-right, are: Kenneth L. Miller, Director of the Division of Health Physics and Professor of Radiology, Milton S. Hershey Medical Center, Hershey, Pa.; Thomas Smithgall, Resident of Lancaster, Pa.; Lee H. Thonus, Alternate Designated Federal Official, Non-Power Reactors and Decommissioning Projects Directorate, NRC Office of Nuclear Reactor Regulation (Region I); John Leutzelschwab, Professor of Physics, Dickinson, College, Carlisle, Pa.; Niel Wald, Professor, Department of Environmental and Occupational Health, University of Pittsburgh, Pittsburgh, Pa.; Michael T. Masnik, Designated Federal Official, Non-Power Reactors and Decommissioning Project Directorate, NRC Office of Nuclear Reactor Regulation; Frederick S. Rice, Resident of Harrisburg, Pa.; and Gordon Robinson, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.**

The postulated loss of spent fuel pool cooling is assumed to result from piping failure in the spent fuel pool cooling system or in the service water supply system to the spent fuel pool heat exchangers, caused by the hydrodynamic effects induced by a loss-of-coolant-accident (LOCA), or from a long term loss of off-site power coincident with a LOCA. The effects of a LOCA are assumed to prevent necessary access to the reactor building for restoration of any method of spent fuel pool decay heat removal. Subsequent boiling of the spent fuel pool is assumed to cause failure of equipment necessary for accident mitigation, attributable to the environmental conditions caused by spent fuel pool boiling within the reactor building. Based on these assumed failures, severe off-site consequences are postulated.

The concerns regarding the design of the spent fuel pool cooling system identified in the 10 CFR Part 21 may have generic implications. The concerns appear to be relevant to boiling water reactors (BWRs) with Mark I and Mark II containment designs, because of the location of the spent fuel pool within the reactor building. Certain concerns may also be applicable to BWRs with Mark III containments and to pressurized water reactors (PWRs), which have separate fuel-handling buildings.

To address this issue, the staff has initiated action to (1) determine the safety significance of the identified concerns, (2) determine the facilities where the concerns are applicable, (3) evaluate the adequacy of present spent fuel pool cooling system designs and (4) evaluate the adequacy of current NRC guidance for spent fuel pool cooling system design. The staff plans to complete an analysis for the plant identified in the 10 CFR Part 21 report by March 1994. The staff will use information from the plantspecific review, as well as a separate risk analysis to determine the scope and nature of generic activities. The staff plans to begin development of generic activities in January 1994.

The staff has initiated correspondence with the BWR Owners Group (BWROG) and requested their plans for addressing the issue. The staff will work with the BWROG and other applicable industry groups in developing a generic resolution of the issue.

## ANTITRUST ACTIVITIES

As required by law since December 1970, the staff has conducted pre-licensing antitrust reviews of all construction permit and operating license applications for nuclear power plants and certain commercial nuclear facilities. (See "Procedures for Meeting NRC Antitrust Responsibilities," NUREG-970, May 1985). In addition, applications to amend construction permits or operating licenses resulting from a proposed transfer of ownership interest or operating responsibility in a nuclear facility are subject to antitrust review. Over the past several years, the staff's antitrust activities have been concentrated in the areas of license amendment reviews—usually associated with proposed new owners or operators resulting from mergers or acquisitions involving licensees—and compliance proceedings initiated by requests to enforce antitrust license conditions.

During fiscal year 1993, the staff initiated or conducted the following activities associated with the NRC's antitrust review responsibility: (1) completed operating license amendment reviews associated with the change in operators of the Vogtle and Hatch nuclear power plants, both in Georgia; (2) initiated a Section 2.206 compliance proceeding pursuant to an alleged violation of antitrust license conditions attached to the St. Lucie Unit 2 (Fla.) plant; (3) conducted an operating license amendment review associated with the proposed merger between Gulf States Utilities and Entergy Corporation; and (4) resolved requests by three Perry and Davis-Besse (both in Ohio) license holders to suspend antitrust license conditions.

In early fiscal year 1993, the staff completed its review of amendment requests by Georgia Power Company to change the plant operator of both the Vogtle and Hatch plants from Georgia Power Company to a non-owner operator, Southern Nuclear Operating Company (Southern Nuclear). The staff negotiated a license condition for each plant that precludes Southern Nuclear from marketing or brokering power or energy from either the Vogtle or Hatch plants. In light of this license condition, the staff concluded that no significant anti-competitive effects would result from the change in operator. This proceeding contributed to setting Commission policy regarding the need for an antitrust review of a new non-owner operator. The Commission indicated that when a license condition was made a part of the license that precluded a non-owner operator from influencing the marketing or brokering of power or energy from the facility in question, there would be no need to conduct any additional antitrust review of the proposed change.

In late fiscal year 1993, the staff made a post-operating license "significant change" finding associated with the proposed Gulf States Utilities and Entergy Corporation merger. The staff concluded that the changes in the licensee's activities identified by several commenters represented enforcement or compliance issues and consequently were not relevant to the significant change licensing amendment process. The staff published its post-operating license no significant change finding in the *Federal Register* in early fiscal year 1994.

The staff received a 10 CFR 2.206 petition from the Florida Municipal Power Agency (FMPA) alleging that Florida Power and Light Company (FP&L) had refused to provide certain transmission services required by antitrust license conditions attached to the St. Lucie Unit 2 license.

As a result of FMPA's Section 2.206 petition for an enforcement action, the staff initiated a compliance proceeding to review FMPA's allegations.

In early fiscal year 1993, the Atomic Safety and Licensing Board denied the requests by licensees Ohio Edison Company, Cleveland Electric Illuminating Company, and Toledo Edison Company to suspend certain antitrust license conditions attached to the Davis-Besse and Perry licenses. The Commission declined to review the decision by the Licensing Board and the proceeding was terminated.

## INDEMNITY, FINANCIAL PROTECTION, AND PROPERTY INSURANCE

### The Price-Anderson System

Under NRC regulations implementing the Price-Anderson Act (September 2, 1957, extended August 20, 1988), a three-layered system was established to pay public liability claims in the event of a nuclear incident causing personal injury or property damage. The provisions of the system involve a sharing of liability by the individual reactor licensee, the nuclear industry, and the Federal Government. Government indemnity for large power reactors was phased out in 1982.

### 1993 Insurance Premium Refunds

The two private nuclear energy liability insurance pools—American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid policyholders a 27th annual refund of premium reserves, under their Industry Credit Rating Plan. Under the plan, a portion of the annual premiums is set aside as a reserve, either for payment of losses or for eventual refund to policyholders. The amount of the reserve available for refund is determined on the basis of the loss experience of all policyholders over the preceding 10-year period.

Refunds paid in 1993 totaled $16,968,821, which is approximately 52.4 percent of all premiums paid on the nuclear liability insurance policies issued in 1983 and covers the period 1983–1993. The refunds represent 74.4 percent of the premiums placed in reserve in 1983.

### Property Insurance

The 11th annual property insurance reports submitted by power reactor licensees indicated that, of the 75 sites insured, 70 are covered for at least the $1.06 billion required in the revised property/accident recovery insurance rule, published on April 2, 1990. The remaining five sites have sought or have been granted exemptions from the full amount of required coverage, because of their small size or their operating status. Thirty-three sites carry the maximum $2.625 billion currently available.

## ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS), established by statute in 1957, by revision of the Atomic Energy Act of 1954, provides advice to the Commission on potential hazards of proposed or existing reactor facilities and the adequacy of proposed safety standards. The Atomic Energy Act also requires that the ACRS advise the Commission with respect to the safety of operating reactors and perform such other duties as the Commission may request. Consistent with the Energy Reorganization Act of 1974, the committee will review any matter related to the safety of nuclear facilities specifically requested by the Department of Energy. Also, in accordance with Public Law 95-209, the ACRS is required to prepare an annual report to the U.S. Congress on the NRC Safety Research Program.

The ACRS reviews requests for pre-application site and standard plant approvals, each application for a construction permit or an operating license for power reactors, 10 CFR Part 52 license applications, and applications for licenses to construct or operate certain test reactors.

With respect to reactors that are already licensed to operate, the committee is also involved in the review and evaluation of any substantive licensing changes and corrective action resulting from operating events and incidents and the resolution of generic safety issues associated with the operation of these plants.

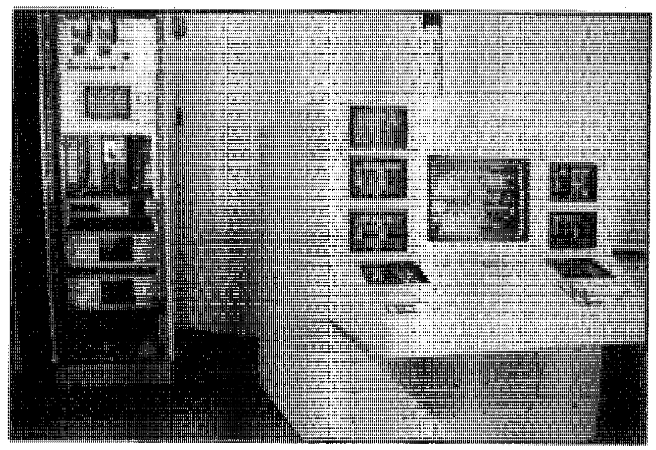
Consistent with the statutory charter of the committee, all ACRS reports, with the exception of classified reports, are made part of the public record. Activities of the committee are conducted in accordance with the Federal Advisory Committee Act, which provides for public attendance at and participation in committee meetings. The ACRS membership necessary to conduct a balanced review is drawn from scientific and engineering disciplines and includes individuals experienced in conducting safety-related reviews of nuclear plant design, construction, and operation.

During fiscal year 1993, the ACRS completed its annual report to Congress on the overall NRC Safety Research Program and other closely related matters. It also reported to the Commission on the following project-related matters:

- General Electric Nuclear Energy Advanced Boiling Water Reactor Design.
- Licensing issues related to the PRISM, MHTGR and PIUS designs.
- Use of digital instrumentation and control systems in evolutionary plant designs.
- Policy, technical, and licensing issues for evolutionary and advanced light water reactor designs.

The committee also provided special topical reports to the NRC and others on a variety of issues, including:

- Implementation of the Safety Goal Policy.
- Use of risk assessment in the regulatory process.
- Consistency of present NRC regulations with the Safety Goal Policy.
- Risk-based regulation.
- Prioritization of generic safety issues.
- Regulatory analysis guidelines.
- Digital instrumentation and control systems.
- Diesel generator reliability.
- On-line testability of protection system.
- Leakage through electrical isolators.
- Incident investigation.
- Human performance.
- Organizational factors.
- Systematic Assessment of Licensee Performance Program.
- Technical Specifications improvements program.
- Valve performance.



**One of the advanced reactor designs under review by the NRC for "design certification" is the System 80+, submitted by Combustion Engineering, a manufacturer of pressurized water reactors. Members of the ACRS and staff are shown above, at left, inspecting the advanced digital control panel in the representative System 80+ control room. The full control panel is on the right.**

- Interfacing systems LOCA.
- Effects of fire-protection system actuation on safety-related components.
- Availability of chilled water systems and room cooling.

The committee also provided advice to the NRC on proposed rules, criteria, and regulatory guides related to:

- 10 CFR Part 52 licensing reviews.
- Implementation of the License Renewal Rule.
- Advanced light-water reactor severe accident performance.
- Implementation of the Maintenance Rule.
- Operator licensing and requalification.
- Implementation of 10 CFR Part 20.
- Implementation of the reactor pressure vessel pressurized thermal shock rule.
- Reactor pressure vessel annealing.
- Proposed changes to the Backfit Rule.
- Protection against electrical transients.

In performing the reviews and preparing the reports cited above, the ACRS holds monthly full committee meetings and subcommittee meetings as required during the year.

# Operational Information/Investigations And Enforcement Actions

Chapter 3

This chapter deals with the activities of three NRC offices concerned with (1) gaining the fullest possible understanding of every aspect of operations at facilities licensed by the NRC, in particular of unplanned and unforeseen occurrences from which safety lessons may be drawn; (2) investigating alleged wrongdoing by licensees, applicants for licenses or vendors to licensees, or their contractors; and (3) taking appropriate enforcement action against licensees for violations of NRC regulations, through the issuance of notices of violation, assessment of civil penalties, and orders for the modification, suspension or revocation of licenses. The three offices dedicated to these tasks are the Office for Analysis and Evaluation of Operational Data, the Office of Investigations, and the Office of Enforcement, respectively.

## Analysis And Evaluation Of Operational Data

The NRC Office for Analysis and Evaluation of Operational Data (AEOD), created in 1979, provides the NRC with an independent capability for the analysis of operational data. The office serves as the NRC's center for the independent assessment of operational events, and it manages the review, analysis and evaluation of both reactor and non-reactor safety performances. It is also responsible for the NRC's Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and the Incident Investigation Program. The AEOD office provides support for the work of the Committee to Review Generic Requirements (see below).

AEOD undertakes the review and evaluation of operating experience in order to identify (1) significant events and associated safety concerns and root causes; (2) the trends and patterns displayed by these events; (3) the adequacy of corrective action taken to address the concerns; and (4) generic implications of these events and concerns. Specific AEOD functions include:

- Analysis of operational safety data associated with all NRC-licensed activities and identification of safety issues calling for NRC staff actions.
- Development and implementation of the agency program on reactor performance indicators, for use by senior managers.
- Development of the NRC program for diagnostic evaluations of licensee performance and direction of "diagnostic evaluation" teams.
- Development of policy, program requirements, and procedures for the NRC's investigations of significant operational events.
- Identification of needed operational data to support safety analyses, and development of agency-wide operational data reporting and retrieval methods and systems.
- Analysis of selected operating events using the Accident Sequence Precursor (ASP) program to gain insight into events and improve understanding of them from risk perspective.
- Conduct studies of the impact of human performance during selected power reactor events.
- Development of a coordinated system for the feedback of operational safety information to NRC offices, licensees, and other organizations, as appropriate.
- Preparation of the Abnormal Occurrence Reports to Congress.
- Development in consultation with other NRC offices, of NRC policy for responding to incidents and emergencies, as well as assessing the NRC response capabilities and performance.
- Tracking the recommendations and staff actions contained in the AEOD studies and Incident Investigation Team reports until they are resolved.
- Development of an agency-wide technical qualifications programs covering a broad range of technical positions within the NRC staff, and provision for technical training needed by NRC personnel, through operations of the NRC's Technical Training Center at Chattanooga, Tenn.
- Continuous staffing of the NRC Operations Center, to screen reactor and nonreactor events, and any

other information reported to the Center, in order to assure appropriate NRC reaction to reported events.

- Serving as the point of coordination for generic operational safety information and data systems with industry, foreign governments, and other agencies involved with the collection, analysis and feedback of operational data.

## Committee to Review Generic Requirements

All generic requirements proposed by the NRC staff related to one or more classes of reactors must be reviewed by the Committee to Review Generic Requirements (CRGR). The Committee is made up of senior NRC managers who review proposed new requirements for the purpose of advising the Executive Director and Operations (EDO) as to whether or not the requirements should be imposed.

The members of the CRGR, as of the end of fiscal year 1993, are:

Edward L. Jordan (Chairman), Director, Office for Analysis and Evaluation of Operational Data.

Guy A. Arlotto, Deputy Director, Office of Nuclear Material Safety and Safeguards.

William F. Kane, Deputy Administrator, Region I.

Frank J. Miraglia, Jr., Deputy Director, Office of Nuclear Reactor Regulation.

Janice E. Moore, Deputy Assistant General Counsel for Advanced Reactors and Special Proceedings, Office of the General Counsel.

Brian W. Sheron, Director, Division of Systems Research, Office of Nuclear Regulatory Research.

In making its evaluations of proposed requirements, the CRGR seeks assurance that a proposed requirement (1) is necessary for the public health and safety, (2) is needed for compliance with existing requirements or written licensee commitments, or (3) is likely to have an impact on the public, industry, and government which is consistent with and justified by the safety improvement to be realized.

From its inception in November 1981 through September 1993, the CRGR has held 249 meetings and taken up a total of 425 separate issues. In fiscal year 1993, the CRGR held 20 meetings and considered 34 issues, including nine generic backfits (incorporated into five rules), two generic letters and two bulletins. A listing of the 34 issues considered by CRGR at its regular meetings follows.

- Proposed rule amendment on reactor operator requalification examinations.
- Generic letter on Thermo-Lag fire barriers.
- Proposed rule amendment to require submittal of nuclear transaction data in computer readable form.
- Proposed regulatory guide endorsing use of industry (NUMARC) guidance for implementing the maintenance rule.
- Proposed revision of NRC regulatory analysis guidelines.
- Generic letter on allowed modifications to administrative controls in existing technical specifications related to emergency and security plans.
- Safety analysis of topical report on allowed relaxation of existing technical specification limits on steam relief valve setpoints.
- Proposed rule amendment to incorporate by reference subsections IWE and IWL of Section XI, ASME Code.
- Proposed rule amendment to allow reduced random test rate in licensees' fitness-for-duty program.
- Generic letter regarding allowed modification of existing technical specifications to reduce surveillance testing during power operations.
- Final rule amendment to approve the VSC–24 dry concrete storage cask for spent reactor fuel at nuclear power plant sites.
- Proposed resolution of unresolved safety issue regarding emergency diesel generator reliability by acceptance of industry initiatives.
- Proposed rule amendment on exemption from criticality monitoring for unirradiated reactor fuel under specific conditions.
- Regulatory guide on control of access to high and very high radiation areas.
- Proposed regulatory guide on bioassay programs.
- Generic letter on relocation of technical specification tables on instrument response time limits.
- Generic letter supplement regarding inaccuracy of motor-operator valve diagnostic equipment.
- Safety evaluations of owner's group responses to generic letter on safety implications of control systems, including steam generator overfill.
- Final regulatory guide (and industry guidelines) on implementation of the maintenance rule.
- Generic letter regarding solutions for instability in boiling water reactors.

- Administrative letter on a new form of generic communications.
- Urgent bulletin on debris plugging of emergency core cooling system strainers.
- Urgent bulletin on boiling water reactor level instrumentation.
- Supplement to generic letter on fire endurance testing for fire barriers.
- Supplement to generic letter on motor-operated valves.
- Urgent generic letter on rod control system failure.
- Regulatory guide on evaluation of reactor vessels with Charpy upper shelf energy less than 50 foot-pounds.
- Regulatory guide on calculational and dosimetry methods for determining reactor vessel fluence.
- Proposed rule amendments on streamlining the regulatory process.
- Generic letter on removal of accelerated testing and special reporting requirements from technical specifications.
- Proposed rule amendments on reactor vessel toughness and annealing of reactor vessels.
- Proposed rule amendment on protection against malevolent use of vehicles at nuclear power plants.
- Supplement to generic letter on guidance for inservice testing of pumps and valves.
- Generic letter on modifications to technical specifications to reflect revised rules on standards for protection against radiation.

## Analyses of Operational Data

**Domestic.** AEOD analyzes and evaluates the operational experience of nuclear power plants as reflected in the reports submitted by plants to the NRC in compliance with the "Immediate Notification Requirements for Operating Nuclear Power Reactors" (10 CFR 50.72) and the "License Event Report System" (10 CFR 50.73), and also in the voluntary reports on component failure submitted to the Nuclear Plant Reliability Data System (NPRDS), which is managed by the industry's Institute of Nuclear Power Operations (INPO). AEOD also examines plant operating profiles and shutdown data found in the licensees' Monthly Operating Reports, in order to generate a context for event analysis and also to establish data from which to gauge normalization of events (e.g., to keep track of reactor trips-per-1,000 critical hours).

One of the primary sources of operational event data is the Licensee Event Report (LER) system. In the early 1980's, a major effort was undertaken to prepare a rule (10 CFR 50.73) governing the content and the submission of LERs. The rule clarified reporting requirements and established a more uniform threshold for event reporting. The threshold included consideration of infrequent events of significance to plant and public safety, as well as of the more frequent events of lesser significance that are more amenable to statistical analysis and trend detection. Since the implementation of the rule in 1984, events that met the threshold have provided a basis for assessing the performance trends of the industry as a whole and those of individual licensees.

In fiscal year 1993, AEOD completed a minor rulemaking exempting certain types of events from reporting requirements (pursuant to 10 CFR 50.72 and 10 CFR 50.73)—primarily those involving invalid actuations of certain narrowly defined engineered safety features (ESFs). Such events would include the invalid actuation, isolation or realignment of the following ESFs: the reactor water clean-up system; the control room emergency ventilation system; and the reactor building, fuel building, or auxiliary building ventilation systems, or their equivalents. Also excluded from reporting are invalid ESF actuations that occurred after the safety function had already been completed, and invalid ESF actuations that occurred when the system was properly removed from service.

At the first public meeting on draft NUREG-1022, Revision 1, Event Reporting Guidelines, which took place in May 1992, about half of the open issues were resolved. In April 1993, AEOD issued a *Federal Register* notice detailing the staff's positions on the remaining issues and the reasons for those positions. A second public meeting was held in May 1993, in which industry representatives still expressed substantial disagreement with the staff's proposals regarding voluntary reporting of certain ESF actuations and plant conditions outside the design basis. The NRC staff has redrafted the document and, in response to industry requests, will reissue it for public comment. Following the public comment period, CRGR review and ACRS comments will be sought and the final document will be issued.

AEOD uses the Sequence Coding and Search System (SCSS) for storage and retrieval of LER data. The system was developed in the early 1980's and is maintained under contract at the Oak Ridge National Laboratory (ORNL), at Oak Ridge, Tenn. It contains on average 150 pieces of information on each LER submitted since 1980. The primary purpose of the SCSS is to facilitate the storage and retrieval of information relevant to each event (e.g., causal and time aspects of occurrences within the event sequence).

**Foreign.** AEOD also employs foreign event data in its comparative studies of reactor operational experience. Reports of operational events received from the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development, from the International Atomic Energy Agency, and from bilateral exchange programs with over 20 countries supplement these domestic data. The NRC continues to assess foreign operational experience for its applicability to nuclear power plants in the United States.

During 1993, the AEOD staff and contractors reviewed about 60 reports on foreign events submitted to the NEA/Incident Reporting System (NEA/IRS). The NRC continued to participate in the NEA/IRS to share U.S. reactor operational experience with the world nuclear community. In fiscal year 1993, about 55 reports were submitted to the NEA/IRS (see "International Programs" in Chapter 7).

## OPERATING EXPERIENCE FEEDBACK

### Operating Experience

AEOD collects, analyzes, and disseminates a wide range of operational data. A subset of this information is used in the NRC Performance Indicator (PI) Program.

Selected industry trends are developed by analysis of industry average operational experience data. (These industry average values are subject to change as the result of revised licensee event reports submitted by licensees and continuing quality checks on the data. The values shown in the report for the current year are projections based upon nine months of data and are subject to change when the final data are available.) The industry averages over the last five years—for seven specific types of events that AEOD monitors as indicators of plant performance—indicate that the incidence of four of these indicators may have stabilized.Those indicators are: safety system actuations, significant events, safety system failures, and forced outage rate. Based upon projections from nine months of data, the annual industry average number of automatic reactor scrams while critical and equipment-forced-outages-per-1,000 hours have again begun to decline after several years of little or no improvement in these indicators. In calculating the industry-wide averages, data for certain periods were excluded for plants that either (1) have ceased commercial operation, or (2) were in extended shutdowns requiring Commission approval before startup or operation above low power.

**Reactor Scrams.** As an essential element of basic reactor safety systems, a reactor shutdown or "scram" can result from initiating events that range from relatively minor incidents to events that are precursors of accidents. The 1992 industry average data shown in last year's NRC Annual Report was a projection based upon nine months of data. The actual year-end value for 1992 was slightly lower. The 1993 data show an decrease in the incidence of scrams. The result is that this indicator has slowly continued to improve.

In 1993, equipment failure remained the leading cause of scrams, causing over twice as many scrams as the next leading cause (personnel error). For scrams occurring at operating plants during 1993, the systems initiating the most scrams, in descending order, were the feedwater and the reactor protection systems (tied), the turbine, and the electrical systems.

**Safety System Actuations.** AEOD monitors a subset of engineered safety feature (ESF) actuations of two types, designated safety system actuations (SSAs); they are (1) actuations of certain emergency core cooling systems and (2) actuations of the emergency a.c. electrical system caused by loss of power to an emergency bus. In general, plant systems designated as ESFs vary considerably among nuclear plants, as do the conditions initiating actuations. The SSAs focus on two key ESFs found at all plants, in order to provide a fairly standardized measure of challenges to ESF systems.

The revised annual industry average data for 1990 through 1992, and the projected data for 1993, indicate a leveling off of this indicator.

**Significant Events.** Significant events are events that the NRC staff identifies through the application of certain criteria. The identification process includes a daily review and discussion of selected operating reactor events. Significant events are normally identified according to one or more of the following criteria: (1) the degradation of important safety equipment; (2) an unexpected plant response to a transient, or a major transient itself; (3) a degradation of fuel integrity, the primary coolant pressure boundary, or important associated structures; (4) a reactor trip with complications; (5) an unplanned release of radioactivity exceeding plant Technical Specifications (TS) or regulations; (6) operation outside the limits of TS; and (7) other events or aspects of an event considered significant.

The annual industry average number of significant events-per-plant decreased from 1989-to-1991. Since then, significant event indicator has been constant.

**Safety System Failures.** AEOD monitors safety system failures (SSFs), which include any event or condition that could prevent the fulfillment of the safety function of structures or systems; the oversight encompasses 26 safety systems, subsystems, and component groups. Unsatisfactory conditions in these areas are generally found during testing, special inspections, and engineering design reviews, rather than following commands to operate. For a system that consists of multiple redundant subsystems or trains, inoperability of all trains constitutes an SSF. SSFs can have implications for a plant's readiness to respond to anticipated events and postulated accidents.

From 1989 through 1993, the trend in the average number of SSFs-per-plant was essentially constant with some fluctuations. The projected data for 1993 show a decrease in this average number from 1992. Whether or not this represents an improving trend or is within the statistical variation for this indicator cannot be determined until the final data are available.

**Forced Outage Rate.** The forced outage rate indicator is the number of forced outage hours in a period divided by the sum of the unit service hours (i.e., generator on-line hours) plus the forced outage hours. For performance monitoring purposes, forced outages are defined as those outages required to be initiated by the end of the weekend following the discovery of an off-normal condition. The trend in forced outage rate can provide a perspective on overall plant operating performance. The annual industry average forced outage rate has remained between 7.2 percent and 9.9 percent for the past five years.

**Equipment-Forced-Outages-per-Thousand Commercial Critical Hours.** The equipment-forced outage (EFO) indicator is a measure of the number of forced outages caused by equipment failures-per-1,000 hours of commercial operation, while the reactor is critical. The EFO rate is the inverse of the mean time between forced outages caused by equipment failures. AEOD monitors the EFO rate as an indicator that can provide perspective on the effects of equipment problems on overall plant performance.

The industry average EFO rate has again declined in 1992 and 1993 after several years of no improvement.

## Performance Indicator Enhancements

The AEOD staff has taken steps to improve the Performance Indicator (PI) Program through (1) the use of peer groups for comparing individual plant performance to that of the average performance of a group of similar plants, (2) the development of a methodology to account for the cyclic nature of some of the indicators during the operating cycle, (3) sponsorship of the development of a risk-based indicator of safety system unavailability by the NRC's Office of Research (RES), and (4) participation in the International Atomic Energy Agency (IAEA) program for development of safety indicators.

During fiscal year 1993, the AEOD staff completed development of the enhanced Performance Indicator Report, which incorporates both peer groups and the effects of operating cycle characteristics. In a Commission paper entitled "Performance Indicator Program—Peer Group and Operating Cycle Phase Enhancements" (SECY-92-425), the staff proposed that the enhanced version of the PI report be approved to replace the existing version. The Commission approved the new format and the first official enhanced PI report was produced for the first quarter of 1993 (published in June 1993). At the same time, quarterly production of the report was changed to semiannual production. The enhanced PI report provides information on both shutdown and operating performance, compares plant performance to that of a peer group of similar plants, and displays the statistical significance of the observed trends and deviations.

The risk-based indicator of safety system unavailability developed by the Office of Nuclear Regulatory Research (RES) was transferred to AEOD for evaluation. This indicator is the product of the fractions of time during plant operation that each train of selected safety systems was unable to perform its safety function. AEOD has begun the development of a trial program to produce data from this indicator, and will continue to work with RES to assess the usefulness of this candidate indicator in the coming year.

Since 1986, AEOD has provided the International Atomic Energy Agency (IAEA) with consultants to contribute to the development of performance indicators. In November of 1992, the IAEA convened a Technical Committee Meeting (TCM) on "Development of Predictive Indicators to Monitor Operational Safety" to discuss draft documents on both plant-specific and risk-based safety indicators. The TCM was followed by a Specialist Meeting on "Safety Indicators for Use by Regulatory Organizations" to review the current status of, identify future developments in, and exchange experience in the use of PIs by regulatory bodies around the world. AEOD represented the United States at these meetings.

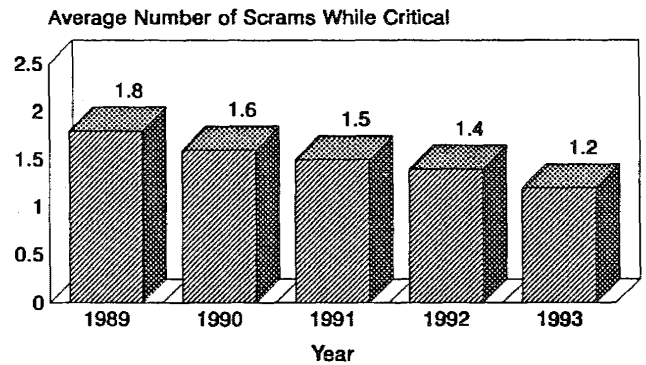
## Collective Radiation Exposure

Data on the industry's collective occupational radiation exposure for 1993 were not available at the close of the report period. The industry's collective radiation exposure declined from 1988 through 1992. Although the NRC receives radiation exposure data on an annual basis, INPO routinely receives radiation exposure data from the plants on a quarterly basis. AEOD uses the INPO data to disseminate information, without duplicating their effort.

## Radiation Exposures From Reactor and Non-reactors

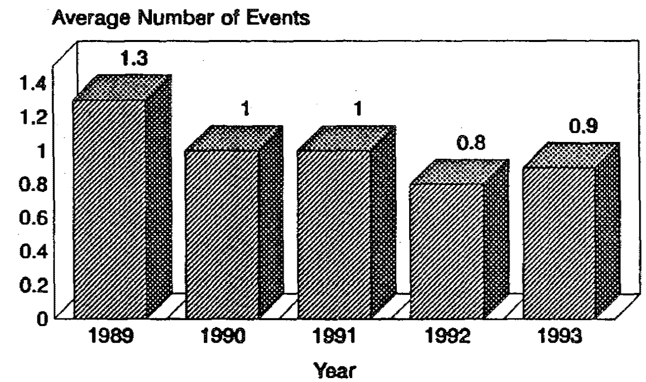
People are exposed to naturally occurring radiation and to radiation from man-made applications of radioactive materials including medical diagnosis and therapy, industrial and commercial activities, nuclear production of

## Average Number of Reactor Scrams While Critical



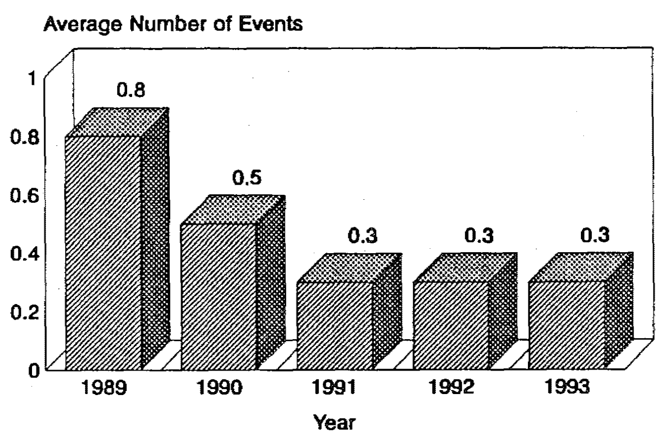

(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Number of Safety System Actuations



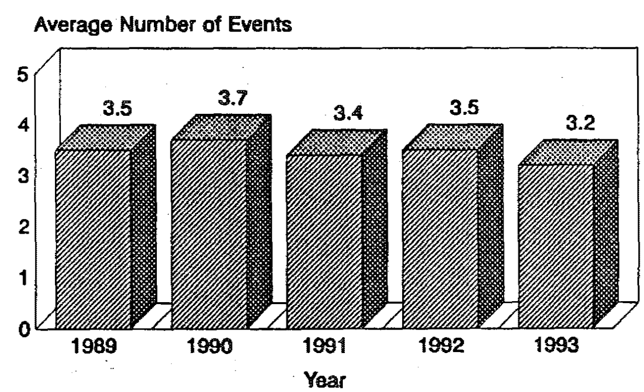

(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Number of Significant Events



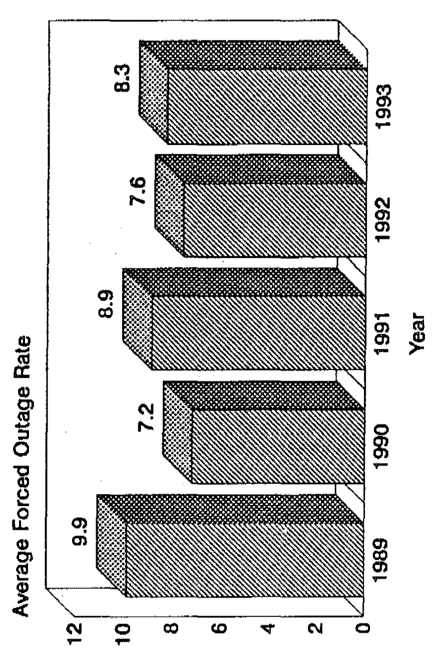

(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Number of Safety System Failures



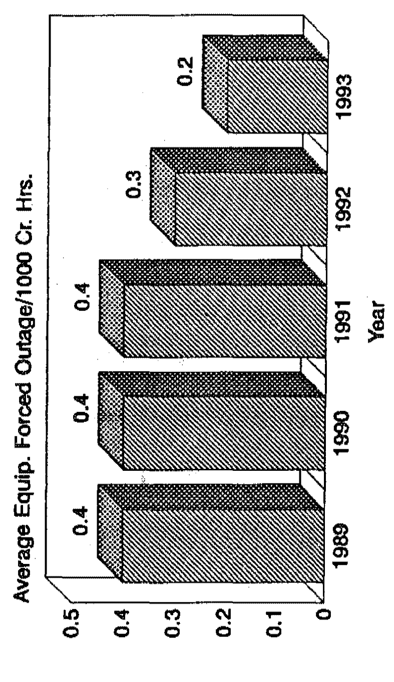

(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Forced Outage Rate



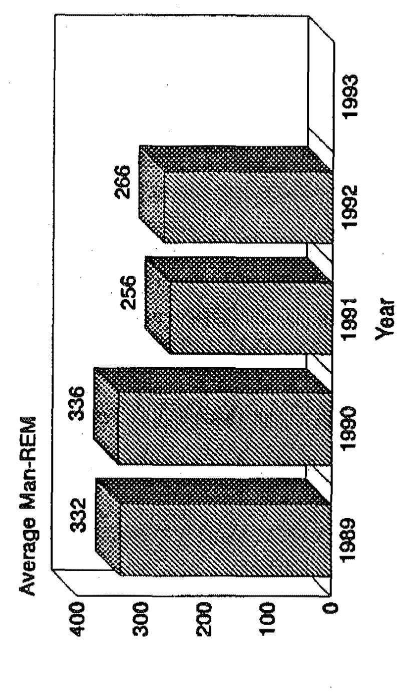

(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Equipment Forced Outage Rate Per 1000 Critical Hours




(1993 PROJECTED FROM 9 MONTHS OF DATA)

## Average Collective Radiation Exposure (Man-REM)




(DATA FOR 1993 NOT YET AVAILABLE)

**The graphs on these two pages display five-year trends (1989 through 1993) for Performance Indicators (PIs) 1-to-7 (the eighth indicator, "Cause Codes," is not subject to industry-wide calculation), beginning top-left and top-right for PI-1 and PI-2 and proceeding to PI-7. The averages shown do not include data for a period when a plant (1) was in an extended shutdown that required Commission approval before either a startup or operation above low power, or (2) was no longer in commercial operation.**

electricity, environmental radiation other than naturally occurring sources, and consumer products. According to the National Council on Radiation Protection and Measurements, the total average effective dose equivalent to a person in the United States is approximately 3.6 millisieverts (mSv) (360 millirem (mrem))-per-year. The average person in the United States receives an effective dose equivalent of about 0.5 mSv (50 mrem)-per-year from medical applications. The whole fuel cycle, including operation of reactors, contributes less than 0.01 mSv (one mrem)-per-year. All the other human-controlled sources of radiation combined add up to an effective dose equivalent of approximately 0.06 mSv (six mrem)-per-year.

Almost all of the radiation dose from nuclear power plants is occupational dose, that is, the dose to the nuclear power plant employees and their contractors who work at the plant. Because the economics of operating a plant creates a strong impetus to lower exposures and achieve ALARA (As Low As Reasonably Achievable) objectives, utility violations of NRC limits on personnel exposure are rare, and the vast majority of nuclear power plant personnel have annual exposures far below NRC regulatory limits specified in 10 CFR Part 20. The actual mean value has been reduced from 19 millisieverts (mSv, equal to 1.9 rems) in 1973 to 10 mSv (one rem) in 1985, and to 1.5 mSv (.15 rem) in 1991. The reduction is believed to be primarily the result of the licensees' extensive dose-reduction efforts. Some measures that reduce collective exposures are the licensees' efforts to have an effective maintenance program, experienced and well-trained personnel, a good water chemistry control program, effective decontamination and cleanup practices, good fuel cladding integrity, effective radiation exposure control programs, good housekeeping, and an alert health physics staff.

The NRC regulates both reactors and non-reactor applications of nuclear materials. All NRC licensees are required to provide appropriate personnel monitoring equipment to each individual who has the potential of receiving a dose in any calendar quarter in excess of 25 percent of the allowable limits specified in Part 20 of the Title 10 of *Code of Federal Regulations* (10 CFR Part 20), "Standards for Protection Against Radiation." Certain licensees, namely reactor operators and those involved with industrial radiography, manufacturing and distribution of radioactive materials, low-level radioactive waste disposal, and independent spent fuel storage installation and processing, are required to provide annual summaries of exposure data for individuals for whom personnel monitoring has been required.

Exposure data for licensee categories for 1992 show that—of the six categories of licensees that are required to report collective exposures for monitored individuals—the 114 reactor licensees that reported (111 operating), by virtue of the large number of employees, had the highest collective exposure (293 Sieverts (Sv), or 29,313 rems, to 194,693 people), followed by radiographers (16 Sv, or 1,600 rems, to 4,974 people), manufacturers and distributors (4.63 Sv, or 463 rems, to 3,815 people), and fuel fabrication licensees (5.29 Sv, or 529 rems, to 8,264 people). Low-level waste disposal (0.37 Sv, or 37 rems, to 467 people) and independent spent-fuel storage (0.11 Sv, or 11 rems, to 279 people) licensees had relatively low collective doses. Of the categories that report collective radiation exposures for monitored individuals, industrial radiography has the highest average measurable dose-per-worker. For each category of licensee, including industrial radiography, the average measurable dose-per-worker is far below the allowable limits established in 10 CFR Part 20.

Although worker occupational exposures have been maintained at a low level, a few over-exposures continue to occur. Between 1988 and 1992, licensees reported 13 events at nuclear power plants involving 14 individuals who received exposures that exceeded the quarterly limits specified in 10 CFR Part 20. Usually more people receive occupational overexposures from materials applications than from being at reactor sites.

## Results of AEOD Studies

In 1993, the AEOD staff continued to analyze and evaluate operating experience, publishing a major study of human performance in operating events, and several technical reports describing equipment problems. Emergency diesel generator performance continued to be studied and several reports were issued on this topic. Considerable effort was expended on the quantitative analysis of risk associated with operational events and conditions, and on better understanding human performance.

In the evaluation of operational experience, the AEOD staff reviews a broad variety of operating data. These data include reports submitted by licensees to the NRC in compliance with 10 CFR 50.72 ("Immediate Notification Requirements for Operating Nuclear Power Reactors"), 10 CFR 50.73 ("Licensee Event Report [LER] System"), and the data base of component failures in the Nuclear Plant Reliability Data System (NPRDS), a system managed by the Institute of Nuclear Power Operations (INPO). Other operational experience reviewed includes 10 CFR Part 21 reports ("Reporting of Defects and Noncompliance"), NRC regional inspection reports, preliminary notifications (PNs) of events or unusual occurrences issued by the NRC, and data on foreign reactor events.

Based on review and analysis of these data, several reports were written and broadly distributed both within the NRC and to the regulated industry. These reports are publicly available. Table 1 provides a list of 1993 reports.

**Operating Experience Feedback Report—Human Performance in Operating Events** (Case Study AEOD/C92-01—Published as NUREG-1275, Volume 8). AEOD began a program in 1990 to conduct on site, in-depth studies of human performance that affected reactor safety during selected power reactor events. The purpose of the program is to identify the factors that have contributed to good operator performance during events, as well as the factors that have hindered performance, and to feed this information back to industry.

Each study was conducted by a multi-disciplinary team, led by an AEOD staff member, with additional NRC headquarters, regional, or Idaho National Engineering Laboratory (INEL) personnel. The studies focused on those factors that helped or hindered operator performance. The team usually spent 1-to-3 days on site interviewing plant personnel and gathering records. Individual reports of each site study were prepared and distributed within the NRC, the site involved in the study, industry groups, and the public. This case study describes generic observations and conclusions drawn from 16 such studies.

Six studies were performed in 1990, seven in 1991, and three in 1992. Of these 16 events, nine occurred at pressurized-water reactors and seven occurred at boiling-water reactors. Ten events occurred while the reactor was at power and six occurred while the plant was in a standby or shutdown mode. Fifteen separate sites were visited. Five studies were performed as part of augmented inspection team (AIT) inspections, while 11 were performed solely under AEOD auspices.

The events represented a wide variety of event or accident scenarios, including stuckopen safety-relief valve, reactor trip with safety injection, reactor scram resulting from positive reactivity insertion, reactor scram resulting from control rod withdrawal, pressurizer spray valve failure, partial loss of instrument air in the containment, turbine building pipe rupture, loss of shutdown cooling, excess steam demand, main steam isolation, defeat of reactor water cleanup isolation, relief valve lifting, loss of annunciators and plant computer, and loss of electro-hydraulic fluid.

The study summarized each event and the findings drawn, observations discerned from multiple events, and conclusions concerning overall human performance. These fell into four groups: control room organization, procedures, human-machine interface, and industry initiatives. Finally, the categorization of events by latent factors compared the similarities among the events. The primary observations and conclusions of the special study included the following.

**Control Room Organization.** Control room staffing level, division of responsibility, and degree of teamwork significantly affected crew response to events. Control room management was overburdened during emergencies when task, supervision, and technical oversight were not appropriately allocated.

The use of the "dual-role" shift technical advisor impaired crew performance by overloading other senior reactor operators when one senior reactor operator assumed the shift technical advisor role. The dual-role shift technical advisors sometimes lacked independent "fresh eyes" because of involvement in shift activities. Other tasks, *such as notifications, also detracted from the shift technical advisor's safety function.

Teamwork during events improved human performance in complex, high-stress situations. Training and teamwork were shown to be useful in increasing the effectiveness of knowledge-based performance.

In this study, the staff concluded that an examination of control room staffing and structure versus emergency functions would result in better utilization of shift resources and allocation of tasks so that no individuals were overburdened. This would be especially worthwhile with regard to the dual-role shift technical advisor function.

**Procedures.** Some operators acted during events without using a procedure. Procedure content, ease of use, and management policy and practices influenced procedure use. Procedure problems were key contributors in the less successful events, but not during the more successful events when the procedures were accurate and complete and management required their use.

Operators experienced difficulty in applying their knowledge to unusual plant conditions during events, which resulted in delays in recognizing and responding to events. Some knowledge-based performance is necessary in every event, so that personnel recognize the significance of the situation, initiate use of the appropriate abnormal operating procedures or emergency operating procedures, and follow those procedures to respond to events.

Preconditioning from past experience, training, or management direction strongly affected how operators recognized and responded to events and in some cases led operators to disbelieve valid indications or take inappropriate actions. Preconditioning to bypass engineered safety features actuation has a high potential for harm.

In two events, operators inappropriately defeated the automatic operations of engineered safety features during valid system demands. Some licensees have not provided criteria and sufficient guidance that limits the defeating of engineered safety features. Improper defeating of engineered safety features in two events during a 4-month period showed that the NRC and industry efforts to appropriately control engineered safety features have not been completely effective and that further action would have a high safety return in the reduction of risk of operator error.

**Human-Machine Interface.** A lack of appropriately arranged, direct-reading control room instrumentation to monitor reactor pressure, temperature, and coolant-level has caused operators to have difficulty in recognizing and responding to shutdown events, when operator actions

# Table 1. AEOD Reports Issued During FY 1993

*CASE AND SPECIAL STUDIES*

| *Designation* | *Subject* | *Issued* |
|---|---|---|
| C92-01 | Human Performance in Operating Events | 12/92 |
| S92-07 | Pressure Locking and Thermal Binding of Gate Valves | 12/92 |
| S93-01 | Review of Auxiliary Feedwater System Reliability | 3/93 |
| S93-02 | Operating Experience Feedback—Reliability of Safety-Related Steam-Driven Pumps | 3/93 |
| S93-03 | Operating Experience Feedback—Service Water System Failures and Degradations | 4/93 |
| S93-05 | Operational Data Analysis of Shutdown and Low Power Licensee Event Reports | 4/93 |

*ENGINEERING EVALUATIONS*

| *Designation* | *Subject* | *Issued* |
|---|---|---|
| E92-02 Supplement 1 | Insights From Common-Mode Failure Events | 2/93 |
| E93-01 | Human Factors Aspects of Boiling Water Reactor Reactivity Management Events During Power Operations | 3/93 |
| E93-02 | Loss of Off-site Power Due to Plant-Centered Events | 3/93 |

*TECHNICAL REVIEWS*

| *Designation* | *Subject* | *Issued* |
|---|---|---|
| T92-08 | Emergency Diesel Generator Start Frequency | 10/92 |
| T92-09 | Review of Manual Valve Failures | 11/92 |
| T92-10 | Prospective Trend of Low Reliability Emergency Diesel Generators | 12/92 |
| T93-01 | Primary System Integrity, Pressurized Water Reactor Coolant System Leaks | 6/93 |
| T93-02 | Tardy Licensee Actions | 8/93 |

were required to accomplish the safety functions of disabled automatic safety systems.

Annunciator and computer alarms are important operator aids in recognizing and responding to events. Operators failed to recognize conditions that were off-normal, but that were not alarmed during events.

**Industry Initiatives.** The effectiveness of individual licensee's studies of human performance during operating events varies widely. While some licensees have initiated worthwhile plant specific corrective actions because of their follow-up on these events, other have missed such opportunities.

Industry groups are engaged in many efforts to improve human performance and human reliability. These efforts have resulted in improvements to plant performance, procedures and programs. With the perceived reduction in the number of events caused by equipment failures, INPO and other industry groups and human performance experts agree that a key to continued improvement in plant performance and safety is improved human performance.

**Pressure Locking and Thermal Binding of Gate Valves** (Special Report AEOD/S92-07). In July 1991, at FitzPatrick, the inboard injection valve for the low-pressure coolant injection system (LPCI) became pressure locked following a hydrostatic test of the piping between the inboard and outboard injection valves. The test pressure had pressurized the injection valve bonnet cavity and the cavity did not completely depressurize during the subsequent 10 hours before to the valve operability test. That resulted in failure of the motor operated valve (MOV) to operate. The motor operator burned out because the force required to open the MOV exceeded motor capability. Although the valve pressure locking was revealed during a hydrostatic test, the licensee determined that the cause was a design problem that was shared by the LPCI inboard injection valves in both loops and both inboard injection valves of the core spray system. The problem could prevent the operation of all four low-pressure emergency core cooling systems.

The nuclear industry has been aware for many years of the potential for gate-valve inoperability caused by pressure locking or thermal binding. The staff found that valve binding caused by pressure locking or thermal binding is a common-mode failure mechanism. This not only can prevent a gate valve from opening on demand, but may damage the motor winding and could render redundant trains of safety systems, or multiple safety systems, inoperable. In spite of numerous generic communications issued in the past by both the NRC and industry, pressure locking and thermal binding continue to occur in gate valves installed in safety-related systems of both boiling-water reactors and pressurizedwater reactors. The previous generic communications have not led to effective industry action to fully identify, evaluate, and correct the problem.

The AEOD staff identified (1) conditions when the failure mechanisms have occurred, (2) the spectrum of safety systems that have been subjected to the failure mechanisms, and (3) conditions that may introduce the failure mechanisms under both normal and accident conditions. The valve binding is a result of inadequate design considerations under specific system conditions. Most valve binding events occurred during plant evolutions, system transients, or unusual system alignments. Hence, the inadequacy in design or installation will not necessarily be found during plant startup testing or regular surveillance testing. The staff concluded that comprehensive system evaluations and analyses, including consideration of plant system conditions and ambient conditions during all modes of plant operations, are needed to identify the valves susceptible to binding and determine the effect on safety system function.

As a result of the study, NRC plans to conduct a workshop for the industry and to either issue a supplement to Generic Letter 89-10 or a new Generic Letter. NRC inspectors plan to verify the adequacy of licensees' evaluations and corrective actions on the potential binding mechanisms of safety-related gate valves.

**Special Study—Review of Auxiliary Feedwater System Reliability** (Special Report AEOD/S93-01). The Trends and Patterns Branch in the Division of Safety Programs of AEOD revised its Trends and Patterns Analysis Program at the beginning of fiscal year 1993. The central activity of the program is the performance of trending analyses. In the past, risk insights were not routinely incorporated into these analyses, nor were they used to identify which components or systems should be analyzed. In the revised program, these trending analyses consist of a disciplined, systematic process for analyzing operating experience data for trends and patterns. In order to make the most effective use of staff and resources, risk insights from past probabilistic risk assessments (PRAs), NUREG-1150 studies, the Individual Plant Examinations, and other relevant resources are employed to determine which systems and components items should be trended. From systems identified as being risk-important, reliability analyses of system performance are being performed to obtain a baseline for future trending. The report documents the results of the first of the system reliability studies, which consisted of a pilot study of industry-wide pressurized-water reactor (PWR) auxiliary feedwater (AFW) system performance at the train level using licensee event report (LER) data.

The AFW system unavailability was calculated using a system failure model which required failure of the two redundant motor-driven trains and one turbine-driven train. The study produced the following results:

- There were no total loss of auxiliary feedwater system events reported during the five-year period analyzed.
- For the loss of one train of AFW (26 failures in the five-year period), the most significant was the failure of the turbine-driven train (19 failures).
- No common-cause failures were observed among the failures of the motor-driven train or the turbine-driven train.
- The motor-driven train was approximately ten times more reliable than the turbine-driven train.
- The trend in train unavailability for a motor-driven train appears to be declining (i.e., reliability is increasing), while there is no discernible trend for the turbine-driven train.
- The trend in total AFW system unavailability appears to be decreasing. That is, the reliability of the AFW system seems to be improving. However, plant-specific system improvements or degradation of turbine drivers were not within the scope of the review and, therefore, were not analyzed.

**Operating Experience Feedback—Reliability of Safety-Related Steam-Driven Pumps** (Special Report AEOD/S93-02). This report, issued for peer review in March 1993, documents a detailed analysis of failure initiators, causes and design features for steam turbine assemblies (turbines with their related components, such as governors and valves) which are used as drivers for standby pumps in the auxiliary feedwater systems of U.S. commercial PWR plants, and in the high pressure coolant injection and reactor core isolation cooling systems of U.S. commercial boiling-water reactor (BWR) plants. These standby pumps provide a redundant source of water to remove reactor core heat as specified in individual plant safety analysis reports.

The period of review for this report was from January 1974 through December 1990 for LERs and January 1985 through December 1990 for Nuclear Plant Reliability Data System (NPRDS) failure data.

The study confirmed the continuing validity of conclusions of earlier studies by the NRC and by the U.S. nuclear industry that the most significant factors in failures of turbine-driven standby pumps have been the failures of the turbine-drivers and their controls. Inadequate maintenance and the use of inappropriate vendor technical information were identified as significant factors which caused recurring failures.

**Operating Experience Feedback—Service Water System Failures and Degradations** (Special Report AEOD/S93-03). This report, issued in April 1993, documents a study performed by AEOD on service water system (SWS) failures and degradations. The study is an update of the 1988 AEOD study published in November 1988 as NUREG-1275, Vol. 3, "Operating Experience Feedback Report—Service Water System Failures and Degradations." Besides using LERs for 1986 through 1991, this updated study also performed a corollary review, using component failure histories for major components within the SWS and for interfacing system heat exchangers for 1987 through 1991.

In performing the updated study, the nine cause categories identified in the 1988 study were consolidated into six cause categories and then used as a basis for review and analysis of operational data from LERs. The revised cause categories are (1) silting/sedimentation, (2) biofouling, (3) corrosion/erosion, (4) foreign material/debris, (5) personnel/procedural errors, and (6) design/seismic deficiencies. A review of Nuclear Plant Reliability Data System (NPRDS) was also performed for SWS equipment failures using the four mechanistic cause categories.

For the updated study, 361 LERs were reviewed, of which 64 considered safety significant were also evaluated and sorted into the following system failure/degradation categories according to relative safety significance: (1) Total Failure of SWS—Actual, (2) Total Failure of SWS—Conditional, (3) Potential Total Failure/Degradation of SWS—Inoperable, (4) Partial Failure/Degradation of SWS—Actual, (5) Partial Failure/Degradation of SWS—Conditional, and (6) SWS Caused Failure/Degradation of Another System. No operating events were identified for Total Failure of SWS—Actual (i.e., Category 1) in this study. Although a limited number of events were identified for the other failure/degradation categories (i.e., Categories 2 through 6), none of the events reviewed significantly affected SWS ability to perform its safety function.

The AEOD staff concluded that analyzing and developing trends for SWS events, including events related to interfacing system heat exchangers and SWS component failures and interfacing system heat exchangers, would provide a useful method for monitoring whether industry is resolving failure and degradation problems associated with the SWS. Use of the first four (mechanistic) cause categories, appears to provide data that are more directly applicable to trending this aspect of SWS performance.

The updated study also revealed that a clear majority of events and component failures occurred geographically at plant sites located in Regions I and II. And the study indicated that baseline operational event data for 1986 through 1991 have not provided conclusive evidence that the issuance of GL 89-13 has resolved SWS degradation problems.

**Operating Data Analysis of Shutdown and Low Power Licensee Event Reports** (Special Report AEOD/S93-05). This report, issued in April 1993, documents the methodology and findings of a special study performed by AEOD on shutdown and low power (SD/LP) events from the beginning of 1987 through the first quarter of 1992. The study represents the AEOD contribution to the staff's evaluation of SD/LP operations at commercial nuclear power plants in the United States, and supports the basic findings as documented in the staff's February 1992 draft report, NUREG-1449, "Shutdown and Low Power Operation at Commercial Nuclear Power Plants in the United States."

The study included a technical review of four PRAs, an earlier AEOD report, an ASP study, and a review of all applicable Nuclear Regulatory Commission (NRC) generic communications. Safety significance criteria were developed and used to assess 1,800 LERs. A data base was then developed to retain, organize, and help analyze the SD/LP operational data.

Several weaknesses in SD/LP operating performance were identified. The important findings include:

- There are few safety significant low power events and no notable trends.
- Based on a qualitative assessment, PWRs experienced 21 percent and 7 percent more high and medium safety significant events, respectively, than BWRs. BWRs have a higher rate of low and total safety significant events than PWRs.
- Pre-Three Mile Island General Electric-designed plants, all Babcock & Wilcoxdesigned plants, and new Westinghouse-designed plants, as defined by the AEOD Performance Indicator Program, have the highest rate of safety significant events.
- Human factor concerns caused 82 percent of the safety significant SD/LP events analyzed.
- Loss-of-shutdown-cooling events, loss-of-coolant-related events, and loss of off-site power (LOOP) are the dominating groups of initiating events related to safety significant SD/LP operational performance. Loss of power supplies other than those initiated by LOOP is also important.

AEOD plans to continue monitoring industry performance using the same techniques developed for this study. Future efforts will consider the effectiveness of the initiatives implemented by the staff and industry to reduce risk during shutdown and low power operations.

**Insights From Common-Mode Failure Events** (Engineering Evaluation AEOD/E92-02 Supplement 1). Common-mode failures were studied because they are a major uncertainty in the bottom-line value in probabilistic risk assessment. Sixty-two licensee event reports (LERs) that described actual or potential common cause failures were analyzed. Most of these events occurred in 1990 and were limited to those situations judged not recoverable (in the event of a coincident accident) or not self-revealing during normal plant operation. Sixteen of the events reviewed were identified as precursors under the Accident Sequence Precursor program and thus exemplify the importance of this issue.

The intent of this work was to identify dominant corrective actions that would preclude or reduce the likelihood of common-cause failures at operating nuclear power plants. Each of the events was reviewed against a set of eight corrective actions. The events were divided into two groups--maintenance and design/installation related failures. The results are dominated by errors that occur at the design, fabrication, and installation stage which go undetected for extended periods of time. Examination of the events indicated that using equipment with larger design margins might have prevented 56 percent of the events; performing comprehensive systems tests, insuring adequate train separation, and the use of diverse equipment each had a potential impact on about 27 percent of the cases; and the staggered surveillance testing could impact about 20 percent of the events.

The effectiveness or practicality of applying any of these corrective actions to all important safety components or systems at operating plants was not explored.

**Human Factors Aspects of Boiling Water Reactor Reactivity Management Events During Power Operations** (Engineering Evaluation AEOD/E93-01). The study identified the human factors aspects of 17 boiling water reactor reactivity management events that resulted in a reactor scram during power operations. Over half of the reactivity events occurred during startup, and about 30 percent occurred during plant shutdowns. Only 18 percent occurred during steady state operation at high power. Every event had human factors aspects, while less than half of the events involved equipment failure. Human performance weaknesses were found in procedures; operator knowledge; command, control, and communications; human-machine interface; and impromptu operator actions. The physical causes of reactivity insertion events were pressure transients, cold feedwater injection, plant cooldown, recirculation flow increase, and control rod withdrawal. The equipment failures involved feedwater or condensate system valves, pressure regulators, turbine stop valve and bypass valve position indication, and a safety relief valve. The findings of the study, as well as, licensee corrective actions were studied because of their potential usefulness in improving licensee performance in this area.

**Loss of Off-site Power Due to Plant-Centered Events** (Engineering Evaluation AEOD/E93-02). The study investigated plant-centered events (events caused by failure or malfunction of equipment or systems inside the plant) involving loss of off-site power (LOOP) at the medium voltage (between two and 15 kilovolts) Class 1E buses. There were 86 identified events between 1985 and 1989 with approximately 30 percent involving total LOOP with the plants at either power operation or shutdown conditions. Analysis of the events indicated 48 percent were caused by personnel errors, 28 percent by equipment malfunctions or failures, 14 percent by design deficiencies, and 10 percent by inadequate maintenance practices.

The study found that 26 plant-centered total LOOP events within the study timeframe exhibited a median LOOP duration of two-hours-and-eight-minutes, which is significantly different than the 18-minute median duration reported in NUREG-1032, "Evaluation of Station Blackout Accident at Nuclear Power Plants." Many conclusions in NUREG-1109, "Regulatory Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," are based on the 18 minute median duration stated in NUREG-1032. Twelve plant-centered LOOP events had been evaluated with conditional core probabilities in the E-5 to E-4 range and thus, are considered risk significant and precursors to accidents.

**Emergency Diesel Generator Start Frequency** (Technical Review AEOD/T92-08). The AEOD staff reviewed the frequency of emergency diesel generator (EDG) starts from 1988 through 1991. Although the number of starts varied significantly among the licensees, there was no significant difference in EDG reliability between groups of EDGs with the most frequent starts and those that were started least. The results of the technical review, and also of AEOD/T92-10, were used in resolution of Generic Issue B-56, "Diesel Reliability."

**Review of Manual Valve Failures** (Technical Review AEOD/T92-09). The AEOD staff conducted a review of manual valve failures in light water reactors. It obtained data on these failures using the NRC Nuclear Document System, Sequence Coding and Search System, and Nuclear Plant Reliability Data System. Results indicated there were 20 plant systems with 20 or more reported failures for the same kind of manual valve.

No events were discovered that severely compromised plant safety. The review of manual valve failures also indicated that, in general, large valves (18-inch) fail by normal wear, six-inch valves fail as a result of random mechanical failures, and two-inch valves fail because of packing problems.

On the basis of the evaluation, the staff concluded that current industry practices with respect to maintenance and surveillance of manual valves were adequate.

**Prospective Trend of Low Reliability Emergency Diesel Generators** (Technical Review AEOD/T92-10). The AEOD staff reviewed EDG start data to identify EDGs with less than 95 percent reliability to start and load-run; this accounted for approximately 7 percent of the EDGs in any given year. The analyses showed that in almost all cases, start and load reliability of these problem EDGs improved in succeeding years, to match the industry average. However, yearly EDG rotation occurs; after one year, a new set of low reliability EDGs was identified. In each succeeding year, the group of problem EDGs was replaced by a new group, with few repeat offenders.

The staff did not identify any EDGs with consistently on-going problems, and no station appears to have an inordinate number of low reliability EDGs.

**Primary System Integrity, Pressurized Water Reactor Coolant System Leaks** (Technical Review AEOD/T93-01). This study was initiated and prepared by a foreign assignee from the Czechoslovak Atomic Energy Commission (CSKAE) during an assignment at NRC that resulted from bilateral agreements between NRC and CSKAE. The purpose was to identify issues, problems, or lessons learned from U.S. operating experience that might be applicable to Czech and Slovak nuclear power plants. The study included development of a data file containing information about reactor coolant system (RCS) leaks; categorization of the leaks; and evaluation of RCS leaks; their causes, and corrective actions taken. The study's results offer insights into the number and extent of RCS leaks, systems and components most likely to leak, and the causes of the leaks.

**Tardy Licensee Actions** (Technical Review Report AEOD/T93-02). A study of delayed licensee actions was initiated following a delayed replacement of batteries in uninterruptible power supplies at the Nine Mile Point Unit 2 (N.Y.) nuclear power plant. The study examined 25 situations identified as violations in inspection reports in 1991 and 1992. Nine of these violations resulted in civil penalties. Delays occurred because known plant deficiencies were not brought to the attention of persons having responsibility and authority for initiating action, because schedules were not adhered to for completing maintenance work requests, and because schedules were not adhered to for implementing commitments to the NRC.

Most of the violations were based on 10 CFR 50, Appendix B, Section XVI, which addresses timely corrective actions as part of the quality assurance program. These types of deficiencies generally do not get reported in LERs, so that the NRC must rely on the resident inspectors to monitor the licensees in this area. The plants generally have numerous tracking systems and, occasionally, management meetings that can be monitored periodically to understand how swiftly these items are being completed.

## Analyses of Human Performance in Operating Events

AEOD continued a program to expand the staff's understanding of human performance during reactor events (previously described under Case Study AEOD/C92-01). During fiscal 1993, four additional studies were completed:

(1) Palo Verde Unit 3 (Ariz.)—Loss of Main Feedwater Pump with Reactor Trip and Safety Injection (2/4/93).

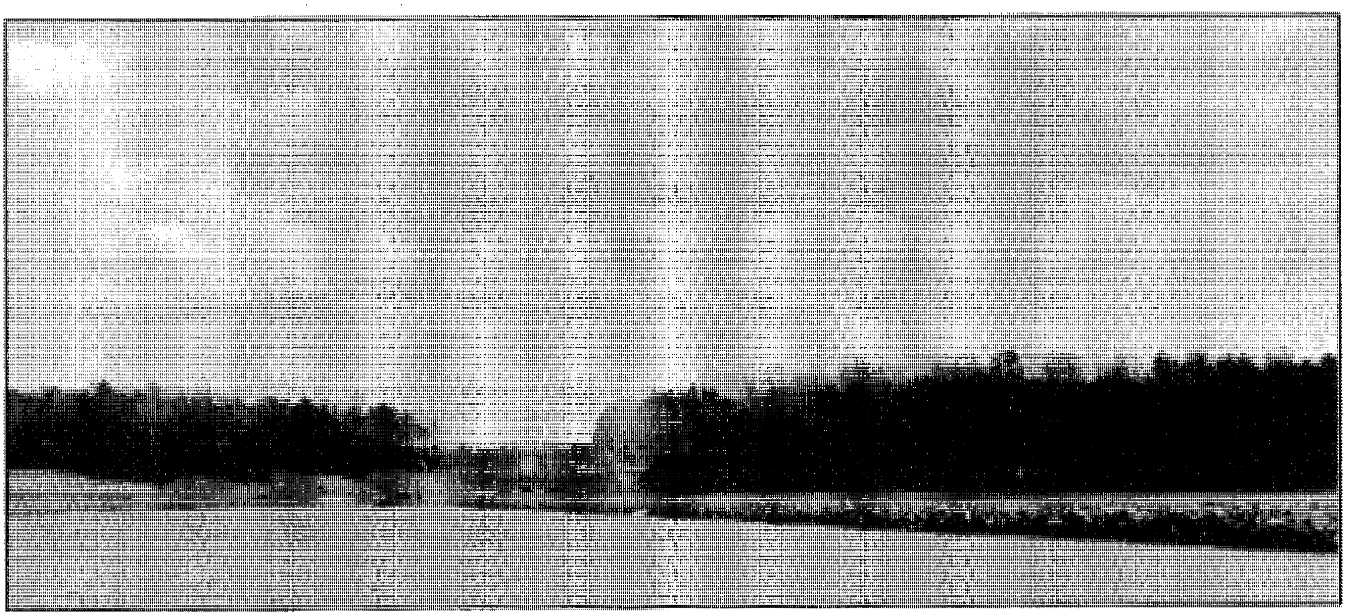
(2) North Anna Unit 2 (Va.)—Disabling Auxiliary Feedwater System During Reactor Trip Recovery (4/16/93).

(3) Browns Ferry Unit 2 (Ala.)—ARI/RPT Actuation Due to Shutdown Overpressurization (5/11/93).

(4) Fermi Unit 2 (Mich.)—Spurious Reactor Scram with Loss of Condenser Vacuum (8/13/93).

The first AEOD human performance study during the period was of a February 4, 1993 event at Palo Verde Unit 3 involving a loss of a main feedwater pump, reactor trip, and safety injection. While the plant was at 100 percent power, a main feedwater pump's speed decreased for unknown reasons. Before the reactor operator could get permission to manually trip the pump to initiate an automatic reactor power cutback, the reactor automatically tripped on low steam generator level. The reactor coolant system temperature and pressure decreased causing safety injection and containment isolation actuations. Operators manually stabilized steam generator levels and terminated safety injection. The AEOD study found that the auxiliary feedwater and safety injection system flows were not recorded so that the amount of water injected could not be readily identified. Emergency procedures did not include using feedwater to feed the steam generators and prolonged the time from automatic initiation to shutdown of the emergency diesel generators. During a similar event two weeks later, the operators successfully tripped the feedwater pump to initiate an automatic reactor power cutback and avoid a reactor trip.

The second human performance study was an April 16, 1993 event at North Anna Unit 2 that involved main generator, turbine, and reactor trips. This study was performed as part of an NRC Region II special inspection. Following the reactor trip, the crew implemented their emergency operating procedures for plant recovery. During this recovery, the motor driven auxiliary feedwater (AFW) pumps were placed in "pull-to-lock" and the steam supply valves to the turbine driven AFW pump were closed for about 18 minutes. The system misalignment and inoperability were discovered by the procedure reader when he reached a later step that returns AFW to a standby configuration. Shift supervision immediately returned AFW to a standby, operable configuration. Required heat sink conditions for the reactor were maintained while AFW was inoperable. The AEOD team found that: AFW was disabled, unknown to shift supervision, while a valid AFW automatic initiation signal was present and prior to meeting the criteria specified in the applicable emergency operating procedure (EOP); despite a prior NRC Information Notice (IN) to the industry regarding bypassing of plant protective features, operators at North Anna did not have a clear understanding of the relevant corporate policy and that, although North



**Among the four AEOD human performance studies undertaken during the report period was one associated with an event at the North Anna Unit 2 (Va.) nuclear power plant. The April 1993 event involved the disabling of the auxiliary feedwater system during recovery from a reactor "trip," or inadvertent shutdown. Required heat sink conditions were maintained while the feedwater system was inoperable for a period of about 18 minutes. The North Anna plant is on North Anna River, in north central Virginia, between Richmond and Fredericksburg.**

Anna was vulnerable to similar events, actions were not identified or taken in response to the IN to further preclude such events; command, control, and communications during the event were weak, and weaknesses in procedures and training contributed to the event.

The third study was of a May 5, 1993 event at Browns Ferry Unit 2 involving an alternate rod insertion (ARI) and recirculation pump trip (RPT) resulting from shutdown over-pressurization, which occurred during hydrostatic testing and testing of instrument line excess flow check valves (EFCV). During the EFCV testing, instrument and control (I&C) technicians isolated the pressure channel that was being monitored in the control room for control of hydrostatic pressure. The control room operator responded to an indicated pressure decrease on this isolated (the isolation was unknown to the operator) instrument and attempted to raise pressure. Actual reactor coolant system pressure increased to where an ARI/RPT occurred from operable pressure channels. The AEOD study found that: there was a lack of effective verbal communication between the unit operators and the I&C technician within the control room; only one of the three unit operators on the crew had participated in these tests previously; although operators periodically cross-checked other pressure indicators, at the time of the event, the operator did not cross-check pressure instruments; procedural inconsistencies contributed to the event; and the licensee had no practice of tagging out of service instrumentation in the control room during testing.

The fourth study was of an August 13, 1993 event at Fermi Unit 2 involving a spurious reactor scram from 93 percent power. When an operator unseated a valve on a common reactor pressure vessel instrument reference leg while removing tape residue, a spurious high water level signal was generated, which scrammed the reactor. The reactor pressure vessel water level decreased, which initiated high pressure coolant injection and reactor core isolation cooling, tripped the recirculation pumps and isolated the reactor water clean up system. The operators were distracted by several equipment failures, the smell of burning insulation in the control room, and about 100 control board lamps burnt out either prior to or during the event. The licensee had not provided training or guidance on how extra reactor operators should assist during an event; the third reactor operator took over many of the actions of another operator, which resulted in short-term, ineffective operator performance. The operators did not take the manual actions required to operate the gland seal steam system, which resulted in loss of condenser vacuum and subsequent main steam isolation valve closure, which made the event more complicated than necessary. For about 1 hours, the operators had to maintain reactor pressure control by cycling nine safety-relief valves. The AEOD study found that, although the gland seal steam system had been operated in manual for the past four years because of a design problem, it was not addressed in reactor or turbine trip procedures. It was also found that procedures and simulator training did not address the timeliness of restoring the reactor water clean up and throttling control rod drive system after a loss of forced circulation.

## Analysis of Non-reactor Operational Experience

One of the activities of the Office for Analysis and Evaluation of Operational Data (AEOD) is the review and evaluation of operating experience of programs involving the use of materials licensed by the NRC and the Agreement States, such as reactor-produced isotopes, natural and enriched uranium, and other special nuclear materials.

As part of operational experience feedback, the AEOD staff prepared two video tapes: "Good Practices in Preparing and Administering Radiopharmaceuticals," and "Good Practices in Co-60 Teletherapy"; these were distributed in February 1992 and April 1993, respectively. Copies of these videos were sent to each NRC medical licensee and to the NRC Office of State Programs which provided copies to the Agreement State licensees. A third video tape is being produced on good work practices for radiographers. The video tape will demonstrate "lessons learned" through re-enactment of radiography over-exposure incidents reported to the NRC. The video, "Taking Control: Safety Procedures for Industrial Radiography," is scheduled to be completed by the end of 1993. The staff has been preparing the videotapes with support from Argonne National Laboratories.

The AEOD Annual Report (NUREG–1272, Vol. 7, No. 2) includes a review of 1992 non-reactor events and medical misadministrations reported by NRC licensees and Agreement States.

**1992 Non-reactor Events.** The primary concern is the excess exposure to the whole body and/or critical organs that has the potential for causing cancer, or in cases of severe over-exposures, even death. The potential for radiation-induced long term genetic mutations is also an important consideration. Extremity or localized skin exposures (from hot particles) are a lesser health concern, but are still important to NRC in assessing how adequately byproduct materials are controlled.

NRC nuclear material licensees, excluding medical misadministrations, reported 623 events for 1992. In eight events, 56 individuals received exposures that were greater than those permitted by NRC regulations. One event, a therapeutic misadministration at Indiana Regional Cancer Center (IRCC), Indiana, Pa., resulted in 94 individuals receiving radiation exposures, of whom 49 individuals received over-exposures, and the patient under treatment died. The event was thoroughly investigated by an NRC

Incident Investigation Team (IIT) and the findings of the investigation were documented in NUREG-1480, "Loss of an Iridium-192 Source and Therapy Misadministration at Indiana Regional Cancer Center (IRCC), Indiana, Pennsylvania, on November 16, 1992." The 49 over-exposures from the IRCC event, and an over-exposure from another event, happened to individuals not employed as radiation workers.

The 29 Agreement States reported 641 events for 1992, excluding medical misadministrations. Of these events, 31 resulted in radiation exposures to 32 individuals in excess of regulatory limits. More than 60 percent of these over-exposure events were the result of industrial radiography operations. Most of the remaining over-exposures involved medical or academic activities. One individual, not employed as a radiation worker, received an over-exposure in a radiography event.

As noted earlier, in 1992, as well as in the preceding four years, the industrial radiography licensees had the highest individual and collective average exposures. The radiological problems of industrial radiography have been recognized for many years. Fuel fabrication and processing licensees were one of the group of licensees that showed an appreciable increase in the average dose to a worker in 1992, even though it is generally lower than the other monitored categories. Compared to 1991, the number of individuals monitored by fuel fabrication and processing licensees decreased by about 70 percent in 1992. The sudden decrease reflects the closing of several fuel facilities. The average measurable dose-per-person for the low-level waste disposal licensees has been steadily increasing over the last years, with a sudden surge in 1992, while maintaining a relatively constant total collective dose. The rise in the average individual dose may have been caused by the decrease of approximately 50 percent of individuals monitored in this group.

**Medical Misadministration.** The 1991 amendment to Part 35 of the Title 10 of the *Code of Federal Regulations* (10 CFR 35), which became effective on January 27, 1992, included the *Quality Management Program* and a revised definition of *Misadministration*. The amendment included a new classification of misadministration, which was defined to include two types of sodium iodide misadministrations, therapeutic and diagnostic. The rule also raised the threshold for diagnostic misadministrations from merely administering a radiopharmaceutical to the wrong patient to administering a radiopharmaceutical that would result in an unscheduled exposure of 500 mSv (50 rems) or more to an organ. This change effectively eliminated the reporting of diagnostic misadministrations that had no safety significance, and therefore, resulted in significantly fewer misadministration events reported for 1992.

For 1992, NRC received 36 medical misadministration reports (excluding diagnostic misadministrations) from its licensees which involved 52 patients. Of those reported events, seven involved sodium iodide misadministrations to seven patients, and 29 involved therapeutic misadministrations to 45 patients. The number of misadministrations reported by the NRC licensees during 1992 was less than 1/12th of the misadministrations reported in 1991, primarily, because of the change in reporting requirements.

For 1992, all 29 Agreement States submitted annual summary reports. Of these, 17 Agreement States reported misadministrations, excluding diagnostic misadministrations. Of these events, seven involved sodium iodide misadministrations to seven patients, and 10 involved therapeutic misadministrations to 10 patients. The 29 Agreement States reported significantly fewer misadministrations for 1992 than were reported by 19 Agreement States in 1991 (three of the 19 Agreement States submitted data after the 1991 AEOD Annual Report was published). Although the population of the Agreement States is almost twice that of the NRC regulated area, only one the 16 medical Abnormal Occurrences were from Agreement State licensees.

## ABNORMAL OCCURRENCES

The NRC prepares a quarterly Report to Congress on Abnormal Occurrences, (NUREG-0090 series), which also serves to promulgate significant event information to licensees, other government agencies, and the public. (These reports may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, D.C. 20013-7082, or the National Technical Information Service, 5285 Port Royal Road, Springfield, Va., 22161. Copies are also available for public inspection or copying for a fee at the NRC Public Document Room, 2120 L Street (Lower Level), N.W., Washington, D.C. 20555, or at Local Public Document Rooms (LPDRs) throughout the country (see Appendix 3 for list of LPDRs)).

Four abnormal occurrence (AO) reports were issued during fiscal year 1993: NUREG-0090, Vol. 15, No. 3 (July-September 1992); Vol. 15, No. 4 (October-December 1992); Vol. 16, No. 1 (January-March 1993); and Vol. 16, No. 2 (April-June 1993). The four reports describe three AOs at a nuclear power plant, 16 AOs at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and five AOs reported by the Agreement States. There were no AOs reported at fuel cycle facilities. The four reports also provide updated coverage of certain AOs previously reported.

The AOs covered in the reports listed above are listed in Table 2, and each is described below. Nine of the events (AOs 92-10, 92-14, 92-15, 92-19, 93-2, 93-3, 93-4, 93-6, and 93-7) resulted in civil penalties proposed by the NRC.

(See Appendix 6 for a list of all civil penalties proposed by the Office of Enforcement, with capsule descriptions of the reasons therefor).

## Abnormal Occurrences at Nuclear Power Plants

**Operation With Degraded Steam Generator Tubes at Arkansas Unit 2 and McGuire Units 1 and 2.** A major degradation of fuel integrity, primary coolant pressure boundary, or primary containment boundary can be considered an abnormal occurrence.

The NRC received notifications from two nuclear power plant licensees indicating that some of their steam generator (SG) tubes had degraded to a point where they no longer had adequate margin for structural integrity. SG tubing constitutes more than half of the reactor coolant pressure boundary. Thus, its integrity is important in minimizing the loss of primary coolant and release of radioactive fission products. Requirements for periodic inservice inspection of the SG tubes have been established in the plant Technical Specifications to ensure that the SG tubes retain adequate structural margin as defined in NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The structural margins that a SG tube must possess at the end of an operating cycle, as described in the NRC Regulatory Guide, take into account inspection measurement uncertainty and flaw growth between inspections.

A partial inservice inspection of the SG tubing is typically performed every refueling outage with eddy current probes. During the past year, two utilities identified SG tubes that failed to satisfy the structural margin criteria of NRC Regulatory Guide 1.121. These SG tubes did not retain adequate structural margins to ensure their integrity for the full range of normal operating, transient, and postulated accident conditions.

On March 9, 1992, Arkansas Unit 2 was shut down upon detection of a primary-tosecondary coolant leak of 360 gallons-per-day, which is approximately half of its Technical Specifications limit. Entergy Operations, Inc., the licensee for the facility, conducted an eddy current inspection of the SG tubing and identified the location of the leak. The licensee reviewed the eddy current test data from the previous inspection (April 1991) and found that the tube had exhibited a flaw-like indication at that time. The licensee also identified six additional tubes that had been incorrectly analyzed in 1991. To further characterize the degradation mechanism affecting this particular location on the SG tubes, the licensee removed several tube sections from the SG for analysis.

On August 7, 1992, the licensee reported to the NRC that for three of the tubes, the examination revealed circumferentially oriented intergranular stress corrosion cracking (IGSCC) beginning on the outer surface of the tubes. The cracks extended 360 degrees around each of the tubes and had average depths ranging between 88 and 94 percent of the tube wall thickness. Because of the extent of the degradation, these SG tubes did not retain adequate structural margins, consistent with NRC Regulatory Guide 1.121, at the time of the March 1992 shutdown of the plant.

On January 16, 1992, McGuire Unit 1 (N.C.) was shut down after the primary-tosecondary coolant leak rate approached 250 gallons-per-day. The licensee subsequently identified the source and location of the leak using eddy current techniques. The licensee reviewed the eddy current test data from the previous inspection (October 1991) for the leaking tube and found that the tube had exhibited a flaw indication at that time. The tube, however, was not removed from service, as required by the plant's Technical Specifications, because of improper classification and administrative handling of the indication. The licensee identified another indication on the tube that had not been identified in the previous inspection of October 1991. The licensee, therefore, reviewed all of the eddy current data from the 1991 inspection. Since McGuire Unit 2 was in a scheduled refueling outage at this time, the licensee decided to further characterize the degradation mechanism affecting this particular location of the SG tubing by removing several tube sections for analysis. The eddy current indications in these sections exhibited similar characteristics to those in the Unit 1 SGs.

In April 1992, Duke Power reported the preliminary results from the examination of the SG tube segments from McGuire Unit 2 to the NRC. One tube was found to contain axially oriented IGSCC beginning on the outer surface of the tube. The crack was approximately 1.4 inches long with an average through-wall depth of 72 percent of the wall thickness. Furthermore, the burst pressure for this tube section was lower than specified in NRC Regulatory Guide 1.121. Both McGuire Units 1 and 2 were shut down in May 1992 for further examination of the SG tubing. At this time, several tube sections were removed from the Unit 1 SGs, including the tube that had leaked and was subsequently plugged in January 1992. The destructive examination of a segment of the tube that leaked in January 1992 revealed an axially oriented stress corrosion crack initiating from the outer surface of the tube. The tube was found not to retain structural integrity consistent with the NRC Regulatory Guide.

The licensee for ANO-2 attributed the missed eddy current flaw indications from the 1991 inspection to (1) lack of training for the eddy current data analysts in guidelines for SG tube damage mechanisms specific to ANO-2 and similar sites; (2) lack of a performance demonstration test by the data analysts using actual site data; and (3) inherent difficulties in analyzing signals at the location where the defect was found to be caused by interference from surrounding structures, the geometry of the tube at this location, and deposits on the SG tubes.

Although the problems cited above may have been contributing factors, the NRC staff believes inappropriate test probes were used for tube locations susceptible to circumferential cracking. Circumferential cracks can only be detected satisfactorily with specialized probes, as discussed in NRC Information Notice 90-49, "Stress Corrosion Cracking in PWR Steam Generator Tubes," and in NRC Information Notice 92-80, "Operation with Steam Generator Tubes Seriously Degraded." During the ANO-2 SG tube inspections in 1991, the licensee did not use the type of probe discussed in Information Notice 90-49. However, during subsequent plant outages in 1992, the licensee did use an appropriate probe for the detection of circumferential cracking.

The primary-to-secondary leakage observed at McGuire Unit 1 in January 1992 was attributed primarily to a SG tube that had not been removed from service during the previous refueling outage (October 1991) as required by the plant's Technical Specifications. The failure to plug the tube to remove it from service was due to misclassification of the tube's flaw indication during the data analysis process. Although the tube was reinspected during the process with a more sensitive probe, the process still failed to assure that the tube was plugged.

The licensee for ANO-2 has taken several corrective steps to address the causes of the missed eddy current flaw indications including: (1) enhancement of the eddy current guidelines; (2) implementation of an analyst training and performance demonstration program; (3) broadening of the SG tube inspection scope; and (4) the use of a probe appropriate for the detection of circumferential cracking. The licensee plans to perform a mid-cycle inspection of SG tubes beginning at the end of April 1993.

As a result of the missed eddy current indications during the 1991 outage at McGuire 1, Duke Power Company took several corrective steps to address the causes of the missed indications including: (1) re-evaluation of eddy current data from the previous outage using enhanced guidelines; (2) implementation of administrative controls to address the manner in which information on SG tubes is conveyed for tube disposition; and (3) reducing the work schedule of the eddy current data analysts. As a result of the metallographic examination of the tube sections in April 1992, the licensee: (1) further enhanced their eddy current guidelines and reinspected the SGs; (2) performed additional training; and (3) administratively lowered the allowable amount of primary-to-secondary leakage.

NRC Information Notice 92-80, "Operation with Steam Generator Tubes Seriously Degraded," was issued on December 7, 1992, to inform licensees of findings from SG tube inspections and investigations at ANO-2. As a result, recipients can consider, as appropriate, actions to avoid similar problems at their facilities. The generic implications of undetected degradation of SG tubes will continue to be actively studied by the NRC.

**Engineered Safety Features Actuation System Design Deficiency—Single Failure Vulnerability at Millstone Unit 2.** A major degradation of essential safety-related equipment can be considered an abnormal occurrence.

On July 6, 1992, during a planned outage at the Millstone Unit 2 (Conn.), with the fuel removed from the reactor pressure vessel and stored in the spent fuel pool, the licensee, Northeast Nuclear Energy Company, was preparing to replace two vital inverters. Millstone Unit 2 uses four inverters, two on each vital d.c. bus, to power two trains of engineered safety features actuation system (ESFAS), comprised of four sensor cabinets and two actuation cabinets. Operators removed power from one actuation train, which caused a false loss of normal power signal and a false start signal for the emergency core cooling system. The effect of this action was similar in consequence to the complete loss of one of the two vital d.c. buses.

One emergency diesel generator (EDG) started and tied onto the bus. The second EDG did not start because it was out of service for maintenance. After the one EDG started, the safety loads failed to sequence onto the bus because of a continuous false load shed signal. Operators recovered from the event by stopping the EDG and restoring power to one of the sensor cabinets. This action removed the false loss of power signal and thus the load shed signal.

The licensee reviewed the event and concluded that an unblocking feature of the automatic test insertion (ATI) system had caused the continuous load shedding signal. The ATI system, a continuous, on-line, logic tester that is common for both trains, was still energized and permitted the spurious loss of power signal to continue to shed the loads.

In reviewing the event, the licensee determined that the ESFAS could also cause other unintended actions under certain power supply failure conditions. These automatic actions are not related to the ATI modification.

(1) If power is lost to either one of the two d.c. vital buses, both the safety injection actuation signal and sump recirculation actuation signal are simultaneously initiated. The recirculation actuation signal results in tripping all low pressure injection pumps. Also, the spurious sump recirculation actuation signal would cause one of the containment sump outlet valves to open.

(2) If power is lost only to the sensor cabinets in one actuation train, both containment sump outlet valves open. If this occurred during a loss-of-coolant accident, high pressure in containment could shut both

refueling water storage tank check valves, inhibiting flow to all emergency coolant injection pumps.

(3) The loss of all d.c. power to one actuation train would cause a power operated relief valve in the other train to open. Also, when control power alone is lost to only the sensor cabinets in a single actuation train, spurious high pressurizer pressure signals would cause the relief valves in both trains to open. Both cases would result in a loss of primary coolant.

The design deficiency in the on-line testing feature could have prevented both emergency diesels from accepting emergency loads under certain single failure conditions. The licensee investigated the event and found several design vulnerabilities related to loss of a vital d.c. bus which may apply to ESFAS at other plants. Although the described event resulted from an ATI modification, the other vulnerabilities are inherent in the ESFAS design and its power supplies.

The licensee is planning modifications to correct these problems and is reviewing the design of Unit 2 for other similar problems. The NRC reviewed the licensee's corrective actions prior to plant restart.

The event was caused by a failure to correctly transfer design package requirements into the plant modification. The implementation plan identified the proper sequence that the inverters would have to be replaced and turned on, but when the work order was prepared the plant sequence was not followed.

Some plant design vulnerabilities were known to the licensee prior to the event. In 1990, the licensee discovered a long standing Technical Specifications interpretation that had permitted indefinite operation of an emergency electrical bus on the non-safety-related backup supply.

The NRC reviewed the design change, the outage plan, and the operation implementation procedures. The NRC also interviewed the engineer who was responsible for the design change, the personnel who prepared the work order, and the operations personnel who released the work order. The NRC concluded that there was ample opportunity for the licensee personnel to identify the error in the work order that caused the partial loss of normal power. The staff considers the subsequently discovered design vulnerabilities in the ESFAS to be significant because, had the plant been operating at the time of the event, the licensee would have been required to take immediate remedial action.

Implementation of the pertinent aspects of the Performance Enhancement Program will enhance pre-performance review of design change records and help resolve plant requirements to support multiple work activities in outage planning.

An ESFAS analysis has thus far identified three areas in which design changes were needed. These include:

(1) Eliminate the interaction of the Automatic Test Insertion Unit with the Load Shed Actuation Module.

(2) Modify the action of the simultaneous Safety Injection Actuation Signal and the Sump Recirculation Actuation Signal to eliminate the LPSI pump trip and to prevent premature pump suction shift to the containment sump.

(3) Modify the PORV control relay logic to prevent inadvertent opening on loss of control power.

These changes are complete. Satisfactory operation of the ESFAS was demonstrated during recent testing performed as a part of startup preparations following the steam generator replacement project. An Independent Review Committee was formed to investigate the event. The recommendations from the committee have been evaluated and, where appropriate, implemented.

The NRC conducted an inspection to investigate the circumstances of the July 6, 1992 event. On February 4, 1993, the NRC issued Information Notice No. 93–11, "Single Failure Vulnerability of Engineered Safety Features Actuation Systems," that described the Millstone Unit 2 event.

**Steam Generator Tube Rupture at Palo Verde Unit 2.** A major degradation of the primary coolant pressure boundary can be considered an abnormal occurrence.

At 4:34 a.m., while at 98.8 percent power, Unit 2 experienced a tube rupture in steam generator (SG) No. 2. A substantial and continuous decrease in pressurizer level and pressure were observed by the operators in the control room. In response, the operators started the third charging pump and isolated letdown at 4:37 a.m. Pressurizer level and pressure continued to decrease and the operators manually tripped the reactor at 4:47 a.m., at which time pressurizer level was about 25 percent. At 4:48 a.m., the safety injection and containment isolation signals actuated when the pressure reached 12.7 megapascals (MPa) absolute (1,837 pounds-per-square inch absolute (psia)). All engineered safety features equipment actuated as required.

The licensee declared an Unusual Event at 4:58 a.m., because of the safety injection system actuation. The licensee upgraded the emergency classification to an Alert at 5:02 a.m., based upon the operators' observation that reactor coolant system leakage was in excess of 167 liters-per-minute (L/min) or 44 gallons-per-minute (gpm). At a later date, the maximum tube leak rate was estimated to be about 909 L/min (240 gpm), which exceeded the makeup capacity (485 L/min (128 gpm)) of the three positive displacement charging pumps that were operating.

Immediately after the reactor trip, pressurizer level decreased to less than the 0 percent reference mark and remained less than 0 percent for about 1.5 minutes until high pressure safety injection restored the level to about 8 percent. The pressurizer did not empty and the minimum pressure reached was 11.6 MPa absolute (1,687 psia). The operators commenced cooldown and depressurization at 6:03 a.m. and isolated the faulted SG at 7:28 a.m. The unit entered Mode 4 (hot shutdown) at 10:29 a.m. and the operators placed the unit on shutdown cooling at 10:35 p.m. on March 14, 1993. The Alert was terminated at 1:15 a.m. on March 15, 1993, and the unit entered Mode 5 (cold shutdown) at 5:56 a.m.

An Augmented Inspection Team (AIT) was sent by the NRC to investigate the Palo Verde Unit 2 event. The AIT concluded that the operating crew performed competently. However, weaknesses were identified in the licensee's implementation of emergency plan actions, including event classification, activation of the emergency response facilities, and promptly determining accountability for on-site personnel. Weaknesses were also found in the procedures, equipment, and training associated with responding to a SG tube rupture. The tube failure did not result in a radiological release to the environment that exceeded regulatory limits. The event did not result in exceeding a Technical Specifications (TS) safety limit. All notifications to the NRC and off-site agencies were made in a timely manner. The AIT report, was issued on April 16, 1993.

On July 22, 1993, the NRC issued Information Notice 93-56: "Weakness in Emergency Operating Procedures Found as Result of Steam Generator Tube Rupture" to all PWR licensees. Enforcement action resulting from the AIT in the area of emergency preparedness was issued as Severity Level IV (Severity Levels I and V range from the most significant to the least significant, respectively) violations by the NRC on July 1, 1993. The licensee responded on July 30, 1993, with an admission of the violations and a corrective action plan. Two Severity Level IV violations were issued related to chemistry and radiation monitoring concerns, and two Severity Level IV violations were identified related to the review of crack growth rates and Emergency Operating Procedure inadequacies.

The licensee issued its response to the NRC Confirmatory Action Letter, its Unit 2 Steam Generator Tube Rupture Analysis Report, and the basis for restart of the facility on July 18, 1993. The report concluded that the damage mechanism for the steam generator tubes was inter-granular attack and inter-granular stress corrosion cracking, exacerbated by a caustic-sulfate environment, crevice formation, and residual and applied stresses. The NRC issued the Safety Evaluation Report on August 19, 1993, concluding that Unit 2 could safely resume operation for six months before the next steam generator tube inspection. The licensee restarted the facility on August 27, 1993, and achieved 100 percent power on September 6, 1993.

It should be noted that several SG tube rupture events occurred at other nuclear power plants, and were previously reported as abnormal occurrences. North Anna Unit 1 had a SG tube plug failure February 25, 1989 (AO 89-1 in NUREG-0090, Vol. 12, No. 1), and another on July 15, 1987 (AO 87-15 in NUREG-0090, Vol. 10, No. 3). McGuire Unit 1 had a SG tube rupture on March 7, 1989 (AO 89-2 in NUREG-0090, Vol. 12, No. 1), while Ginna had one on January 25, 1982 (AO 82-4 in NUREG-0090, Vol. 5, No. 1).

## Abnormal Occurrences Involving Other NRC Licensees

**Medical Therapy Misadministration at Cooper Hospital/University Medical Center in Camden, N.J.** A therapeutic misadministration affecting two or more patients at the same facility can be considered an abnormal occurrence.

On January 27, 1992, the NRC Region I Office was notified by telephone that five therapeutic misadministrations involving iridium-192 (Ir-192) wire occurred at Cooper Hospital/University Medical Center at Camden, N.J., from November 11, 1991, to January 7, 1992. The licensee had discovered the error on January 24, 1992, after the review of patient charts in preparation for the Quality Management Program submittal. The error caused a 12.2 percent underdosing of the patients.

Four patients received external beam therapy (Linear Accelerator), in addition to the radiation received from the Ir-192 implants. Patient A was to receive 1043 rads from an Ir-192 intra-cavitary bronchial implant for the treatment of lung cancer and received 916 rads on November 11, 1991. Patient A later received 5,576 rads from external beam therapy. Patient B was to receive 1,266 rads to the head and neck from an Ir-192 interstitial implant for the treatment of cancer and received 1,112 rads on November 12, 1992. Patient B later received 4,600 rads from external beam therapy. Patient C was to receive 2,150 rads from an Ir-192 interstitial implant for the treatment of breast cancer and received 1,888 rads on December 2, 1991. Patient C later received 5,940 rads from external beam therapy. Patient D was to receive 2,000 rads to the tongue for the treatment of cancer from an Ir-192 interstitial implant and received 1,756 rads on January 7, 1992. Patient D later received 5,940 rads from external beam therapy. The licensee has determined that the patient's treatments were not compromised by the small decrease in the total dose received when the external beam therapy treatment is factored into the assessment.

One patient did not receive external beam therapy. On November 21, 1991, Patient E was prescribed to receive

4,628 rads to the pelvis for the treatment of cancer from an Ir–192 interstitial implant and received 4,063 rads. Patient E's attending physician had originally calculated a desired dose between 4,000 and 4,500 rads and wanted to include hyperthermia treatment. Hyperthermia treatment required insertion of interstitial microwave antennae so that heat treatment was terminated within one hour before the implants were inserted and was initiated within one hour after the implants were removed. The attending physician was informed by the licensee's staff that the implants would have to be removed at unreasonable times in order to fall within the attending physician's desired dose range. The attending physician then agreed to give 4,628 rads so that the second hyperthermia treatment could be given at a more reasonable time. Since the actual delivered dose fell within the attending physician's initial range, the licensee does not foresee any adverse effects for Patient E.

It was determined that the cause of the misadministration was an input error into the treatment planning computer. Specifically, the source calibration factor was in non-System International (SI) units (non-metric), however, the computer was set to receive the data in SI units and the setting was not changed. The licensee's corrective action was to include the calibration factor that is used during treatments in their records for Implant Source Inventory--Source Type Characteristics so that the licensee can verify that the proper factors are used.

An NRC Region I inspector conducted an inspection of the incident on August 5, 1992, to determine the circumstances associated with the misadministration. The inspector's findings were in agreement with the licensee concerning the cause of the misadministration. The inspector determined that the licensee's corrective actions were adequate to prevent recurrence. Inspection findings regarding the misadministration are documented in Inspection Field Notes approved September 9, 1992.

**Extremity Over-exposure of a Radiographer at MQS Inspection, Inc., Field Site in Trenton, Mich.** An exposure of the feet, ankles, hands, or forearms of any individual to 375 rems or more of radiation can be considered an abnormal occurrence.

On July 6, 1992, a licensee radiographer was assigned to radiograph various pipes at a construction site. Radiography is a nondestructive testing technique which uses a sealed radiation source to make x-ray-like images of heavy metal objects. The configuration of the job required that the radiography exposure device (camera) be suspended 20 feet above the floor. The radiation source is exposed using a remote cable to make the film image and then is retracted into the shielded camera. After an exposure, the radiographer used an aerial platform to reach the camera. He performed a radiation survey as he approached to assure that the source was in the shield. The radiographer was wearing his audible alarm radiation measuring device, but it was turned off.

The radiographer then moved the camera to reach the camera port to lock the radiation source inside. When he removed the tube which guides the source, he discovered the radiation source was exposed about four inches outside the camera. The source had apparently shifted into the unshielded position when the radiographer moved the camera to lock it. The source was locked into place in its exposed condition. The radiographer immediately returned to ground level, but later returned to the camera to unlock it so that the radiation source would be retracted into its shield.

The incident was subsequently re-enacted by the licensee's Radiation Safety Officer and NRC inspectors to evaluate the radiation exposure received by the radiographer. The calculation by the Radiation Safety Officer, based on a series of re-enactments, indicated a minimum 440 rems exposure to the individual's hand. NRC inspectors estimated that the dose was about 880 rems. The radiation measuring device worn by the worker indicated a whole body radiation exposure of about 250 millirems.

The worker's hand was evaluated and monitored by medical radiation specialists at an area medical center. No short-term physical changes to the skin of the hand were observed. The NRC limit for extremity exposures is 18.75 rems in a calendar quarter. Therefore, the re-enactment showed that the exposure received was substantially over the limit. The whole body radiation exposure was within the NRC limit of three rems in a calendar quarter.

The over-exposure occurred as a result of the failure of the radiographer to use an audible alarm exposure measuring device as required by NRC regulations. The locking mechanism allowed the source to be locked in place while it was still exposed. The radiographer was wearing an audible alarm device required by the NRC for radiography work, but the device was turned off. The device had been turned off to conserve battery power while the radiographer was doing paperwork, but had not been turned back on for the remainder of the day. Use of an operable alarm device could have avoided or minimized the over-exposure.

The licensee alerted its staff to the potential problem with the locking mechanism of this type of radiography camera. It also provided augmented training on the use of the required audible alarm radiation devices and included verifying that the devices are turned on during routine internal audits of radiography activities. The radiographer was restricted indefinitely from further work with radioactive materials.

NRC Region III conducted a special inspection of the licensee's activities on July 8–10, 1992. The inspection identified three violations of NRC requirements associated with the over-exposure incident: (1) the extremity exposure in excess of the 18.75 rems limit for a calendar quarter; (2) failure of the radiographer to wear an operable audible radiation monitoring device; and (3) failure to perform an adequate radiation survey of the radiography camera in that the radiographer did not survey the full circumference of the camera. The first two violations were classified as a Severity Level I problem, and the third as a Level IV violation. On October 9, 1992, a $5,000 fine was proposed for the first two violations. No fine was proposed for the third violation. On November 2, 1992, the licensee paid the civil penalty.

**Medical Therapy Misadministration at the Medical Center of Delaware, Incorporated, in Wilmington, Del.** A therapeutic exposure to a part of the body not scheduled to receive radiation can be considered an abnormal occurrence.

On August 12, 1992, the NRC Region I Office was notified by telephone by the Medical Center of Delaware, Incorporated's radiation safety office that a therapeutic misadministration involving a cobalt-60 teletherapy unit occurred on August 11, 1992.

The physician's written directive called for 3015 rads in 15 fractions to be delivered to the central area of the pelvic region with the teletherapy machine set up in a fixed modality. During the 14th fraction, the radiation therapy technologist (RTT) did not ensure that the teletherapy machine was set in the fixed modality and started the treatment. The previous patient had received treatment in the rotational modality and the setting of the machine was not changed. The patient received a total of 160 rads to the pelvic treatment area instead of the prescribed 200 rads. The licensee also estimated that the patient received a dose of 80 to 110 rads to the left side of the pelvis outside of the treatment area and from 60-to-70 rads to the right side of the pelvis outside of the treatment area. Both the patient and the physician were immediately notified of the misadministration and informed as to the physiological consequences that could follow from this event. The licensee has determined that the patient will not suffer any adverse effects in the areas that received an unintended radiation dose. The licensee will increase the prescribed dose for the 15th fraction to make up for the underdosing during the 14th fraction.

It was determined that the cause of the misadministration was the failure of the licensee to follow the department's Quality Management (QM) Program. The licensee's QM Program calls for two RTTs to be present when a patient is being set up to ensure that the setup is done properly. The first RTT did not ensure that the setup was done correctly and the second RTT was out of the department getting another patient.

The licensee's corrective and preventive actions included: (1) training for all Radiation Therapy staff on August 12, 1992, to review standard operating procedures for external beam therapy; (2) increased supervisory review and evaluation of existing procedures to ensure comprehension and implementation; and (3) strengthening of other existing procedures to ensure that periodic reviews of the Radiation Therapy Technologist's activities are conducted.

The NRC Region I staff conducted an inspection on November 19, 1992, and held an enforcement conference with the licensee on December 17, 1992, to discuss the inspection findings. As a result, three violations of NRC requirements that related to the licensee's implementation of its Quality Management Program and that contributed to the misadministration were identified. The violations were classified in the aggregate as a Severity Level III problem. No fine was proposed for the violations. The licensee's corrective and preventive actions will be reviewed during the next inspection of the licensee's program.

**Medical Therapy Misadministration and Unplanned Exposure at St. Clares Riverside Medical Center in Denville, N.J.** A therapeutic exposure to a part of the body not scheduled to receive radiation can be considered an abnormal occurrence.

On October 2, 1992, St. Clares Riverside Medical Center notified the NRC by telephone that a therapeutic misadministration involving the implant of two iridium-192 ribbons had occurred that day at its facility. At 2:30 p.m. on October 1, 1992, a patient was implanted with 48.25 millicuries of iridium-192 contained in two nylon ribbons. The ribbons were inserted into catheters that extended from the patient's abdomen into the common bile duct. The procedure was scheduled to last 20 to 23 hours during which a dose of 1,500 to 2,000 rads would be delivered to a colon tumor obstructing the common bile duct. After implanting the iridium-192 ribbons into the two catheters, the implant site was dressed and instructions were given to nursing personnel not to change the dressing. These instructions were not detailed on the patient's chart. Because of excessive drainage of bile at the implant site during the evening and early morning hours, the patient's dressings were changed several times and then reinforced with additional absorbent. At 4:15 a.m. on the morning of October 2, 1992, the nurse on duty noted that the dressing was completely displaced and acted to replace the dressing. The nurse noticed that the two ribbons were displaced but, not knowing what they were, coiled the ribbons in her hand and taped the ribbons to the patient's abdomen. A routine x-ray identified that the seeds were no longer implanted, and the coiled ribbons were removed from the surface of the patient's abdomen by a physician at approximately 12:00 p.m. on October 2, 1992.

The licensee estimated that the patient received 1,145 rads to the targeted tumor site, between 172 and 1,032 rads to the skin of the abdomen, 19.9 rads to the liver and small bowel, 12.7 rads to the kidneys, 50.9 rads to the colon, and 6.7 rads to the testes. The licensee estimated that the nurse who coiled the ribbons and taped them to the patient's abdomen received approximately 7.6 rads to her hands. The licensee expects no adverse effects as a result of the misadministration since this brachytherapy treatment was a booster to the external beam dose that was yet to be administered.

Both the patient and the nurse were notified of the misadministration. The misadministration was caused by (1) lack of oversight of the procedure by the licensee's Radiation Safety Officer (RSO); and (2) inadequate training of the nursing staff in that they were unable to identify the brachytherapy source ribbon.

The licensee initiated an expanded training program that includes familiarization of personnel with the size and appearance of the radioactive sources used in brachytherapy treatments at the licensee's facility. The licensee stated that a manager will be responsible for ensuring that personnel on all shifts involved in the care and treatment of radiation therapy patients receive this training. The licensee decided to name a new RSO because the current RSO was unable to devote sufficient time to the radiation safety program, considering his other responsibilities. The licensee's actions also included: 1) committing that a new RSO would be in place before another brachytherapy procedure is performed; 2) developing a nurses' procedure manual; 3) conducting formal inservice training in radiation safety with all nursing unit workers; and 4) requiring a written directive be initiated before ordering radioactive material.

NRC Region I conducted an inspection on October 5, 6, 7, and 9, 1992, and held an Enforcement Conference on November 5, 1992, to discuss the inspection findings. The licensee's corrective and preventive actions will be reviewed during the next inspection of the licensed program. Several violations of NRC requirements were identified including: (1) failure to adequately train nursing personnel to recognize brachytherapy procedures, (2) failure to train personnel on potential radiological emergencies for brachytherapy procedures, and (3) failure to implement radiation safety and quality management programs to ensure adequate safety. A civil penalty of $10,000 was proposed in a letter dated January 11, 1993. The licensee paid the civil penalty on February 5, 1993.

**Medical Therapy Misadministration at the Lahey Clinic Medical Center in Burlington, Mass.** A therapeutic exposure to a part of the body not scheduled to receive radiation can be considered an abnormal occurrence.

On October 19, 1992, the Lahey Clinic Medical Center notified the NRC Operations Center of a therapeutic misadministration involving a high dose rate remote afterloader (HDR) that occurred at the facility on October 14, 1992. A patient was scheduled to receive brachytherapy treatment to the right main stem bronchus in three fractions using a Gamma Med HDR. Each fraction was to deliver 700 rads to the targeted tumor site. On October 7, 1992, the patient was administered the first treatment as prescribed. On October 14, 1992, the therapist made an error during input of the offset distance into the treatment computer, entering an offset distance of seven millimeters, rather than seven centimeters as required. The error resulted in the second fraction delivering 90 percent of the prescribed fractionated radiation dose to unintended tissues away from the tumor site and underdosing the tumor site. The underdose was made up during the administration of the third fraction on October 22, 1992. The physician stated that he expected no adverse clinical effect on the patient to result from the misadministration as the dose was made up in the third and final fraction.

The referring physician and patient were both notified of the misadministration. The licensee followed established procedures; however, the procedure did not include a mechanism to verify data entries on the HDR console at the time of treatment. The licensee instituted a new procedure that requires that a second individual verify the data input on the HDR console prior to administration of the therapy.

NRC Region I conducted a routine inspection at the facility on December 3, 1992. The inspection resulted in the identification of six apparent violations: (1) failure to have a quality management program to meet the regulatory requirements, (2) failure to make timely notification to the NRC, (3) failure to provide radiation safety training to workers, (4) failure to perform the required tests of the dose calibrator, (5) failure to perform radiation surveys, and (6) failure to maintain the prior exposure record of a new employee. The licensee stated that there were no adverse effects to the patient as a result of the misadministration.

The NRC enforcement action consisted of issuance of a Notice of Violation with one Severity Level III violation, three Severity Level IV violations and two Severity Level V violations. On August 4, 1993, the NRC decided to mitigate the civil penalty in its entirety.

**Medical Therapy Misadministration at Indiana University Medical Center in Indianapolis, Ind.** A therapeutic dose that is greater than 1.5 times the prescribed dose can be considered an abnormal occurrence.

A 31-month old patient, being treated for a brain tumor, was to receive two cobalt-60 teletherapy treatments of 150 rads each for a total dose of 300 rads to reduce swelling behind the patient's eye. The dosimetrist mistakenly prepared the dose calculations for 300 rads-per-treatment. The patient was treated on November 13 and 14, 1992, at the Indiana University Medical Center with 300 rads-per-treatment for a total dose of 600 rads.

Prior to the treatment, the treatment plan was reviewed by the treating physician. Following the treatments, the dose calculations were reviewed by a medical physicist and approved. The error was discovered by a student technologist during a monthly chart review on December 2, 1992.

Both the patient's referring physician and guardian were informed of the misadministration. The treatment accomplished its intended purpose and the swelling was reduced. The licensee reported that no adverse medical effects were expected because of the additional radiation exposure.

The error was caused by the mistaken calculations by the dosimetrist and by the apparent inadequate review by the physician before the treatment began. The doses normally used for this type of treatment are 300 rads-per-treatment, and this further contributed to the failure to identify the error before the treatments occurred. There was also a problem with the legibility and format of the treatment plan.

The licensee has provided further training to treatment personnel to eliminate the types of problems that contributed to the misadministration. The licensee also intends to revise the treatment form to make it more understandable.

The NRC retained a medical consultant to review the case and to provide clinical assessment of the misadministration. NRC Region III conducted a special inspection on December 14–15, 1992, to review the circumstances surrounding the misadministration. Enforcement action on the inspection findings is pending.

**Loss of Iridium-192 Source and Medical Therapy Misadministration at Indiana Regional Cancer Center in Indiana, Pa.** A therapeutic dose that is greater than 1.5 times the prescribed dose can be considered an abnormal occurrence. An exposure to an individual in an unrestricted area occasioning a whole body dose in excess of 0.5 rem in one calendar year may be considered an abnormal occurrence.

On December 1, 1992, the licensee, Oncology Services Corporation (OSC), notified NRC Region I of the loss of an approximately 4.3-curie sealed iridium-192 source from a high dose rate (HDR) remote afterloader unit at the Indiana Regional Cancer Center (IRCC), Indiana, Pa. The licensee stated that they were notified by a local nursing home that a manager from Browning-Ferris Industries (BFI), a biological and hazardous waste handler, found radioactive material in the biowaste that was picked up from the nursing home. The licensee performed radiological surveys of the HDR and noted that the iridium192 source was missing.

On December 1, 1992, Region I dispatched a section chief and inspector to the IRCC to ascertain the facts surrounding the loss of the iridium-192 source and how it was transferred to BFI facilities. On December 3, 1992, the NRC upgraded its response to an Incident Investigation Team (IIT). On February 8, 1993, the IIT presented its findings (NUREG–1480) to the Nuclear Regulatory Commission. The following are synopses of the Region I inspection and IIT findings.

On November 16, 1992, an elderly patient was treated for anal carcinoma at the IRCC in Indiana, Pa., using HDR brachytherapy. The patient died on the evening of November 21, 1992, five days after the treatment. Before the treatment, five catheters were placed in the tumor. During the treatment, an approximate 4.3-curie iridium-192 source was placed at various positions in each catheter to irradiate the tumor by use of a remotely controlled Omnitron 2000 afterloader. The treatment was the first of a series of three 600-rad treatments planned by the physician, and the five catheters were to remain in the patient for subsequent treatments.

On November 16, 1992, after a trial run through the five catheters with a dummy wire, the iridium source wire was placed in four catheters without difficulty. After several unsuccessful attempts to insert the source wire and the dummy wire into a fifth catheter, the treatment was terminated. An area radiation monitor in the treatment area was observed in an alarm condition at various times when the source should have been retracted during the unsuccessful attempts to insert the source wire through the catheter. Although three technologists and the physician attending the patient were aware of the alarm condition, no one conducted a survey for radiation levels with the available portable radiation survey instrument. The only action taken was to check the control console of the HDR remote afterloader. Because the console indicator showed "safe," they believed the source to be fully retracted into the lead shield and assumed the area radiation monitor was malfunctioning. They were unaware the source wire had broken, leaving the source in one of the catheters in the patient. The patient was transported by ambulance, with the source, to a local nursing home.

The source remained in the patient's body for almost four days. The catheter with the source came loose on the fourth day and, eventually, the catheter fell out early on the morning of November 20, 1992. It was placed in a medical biohazards bag (red-bag) in a storage room by nursing home personnel who did not know it contained the radioactive source. Later, on the same day, the catheter containing the source was moved to another storage location at the nursing home and placed in a box with other red bags. From November 16 through November 25, 1992, numerous residents, employees, and visitors to the nursing home were unknowingly irradiated. The ambulance staff who returned the patient to the nursing home were irradiated along with employees and patients at the IRCC.

On November 25, 1992, a driver from BFI picked up the red-bag biowaste and transported it to a BFI facility in Carnegie, Pa., and from there, it was transported to a BFI medical waste incinerator in Warren, Ohio. At the Warren facility, fixed radiation monitors identified radiation emanating from the trailer, and, on facility personnel direction, the trailer was returned to Carnegie the same day. It was left over the weekend and on Monday, November 30, 1992, the BFI staff searched the truck for the radiation source. They identified the box with the radiation source and looked at individual red bags to identify the origin of the waste. On December 1, 1992, BFI successfully identified a name found with the red-bag waste in the box, and traced it to the nursing home.

After being notified by BFI, the nursing home called the IRCC on December 1, 1992. The cancer center had not used the HDR afterloader after the single treatment on November 16, 1992. Upon being informed of the source discovery, the medical physicist determined that no source was present in the HDR afterloader and informed NRC Region I of this fact. The physician and the medical physicist drove to Carnegie and retrieved the source.

A second Omnitron 2000 source wire broke at the Greater Pittsburgh Cancer Center (GPCC) of OSC on December 7, 1992. This wire broke in the same approximate location as the first wire. The GPCC medical physicist who was conducting the treatment was aware of the first incident and immediately recognized the problem and promptly and appropriately intervened, thereby preventing significant dose consequences to the patient or the cancer center staff.

An NRC medical consultant concluded that an analysis of the medical records and physical dosimetry would indicate that the massive radiation dose was a probable contributing cause of death in this patient. The licensee reported the prescribed dose at one centimeter was 1,800 rads, to be delivered in three treatments, and that the delivered dose was 1,600,000 rads to the same point, an overdose of about three orders of magnitude. The licensee stated the effect on the patient would be significant local tissue damage and possible significant tissue damage to organs outside the treatment area, depending upon the progression of radiation damage over time, before the patient expired. The licensee stated the dose was of sufficient magnitude that it believed it was highly probable that the radiation exposure was at least a contributing factor to the patient's subsequent death. In a press release dated January 26, 1993, the Indiana County Coroner stated that the cause of death listed in the official autopsy report was "Acute Radiation Exposure and Consequences Thereof."

Besides scrutinizing this case, the team evaluated the radiation doses to 94 persons associated with the IRCC event. Radiation doses received by these individuals ranged between 40 millirems and 22 rems. Cytogenetic studies were also performed on a number of these exposed individuals and the results were found to be consistent with calculated doses, within the limits of accuracy of both techniques. The highest extremity dose was calculated to be between 73-to-160 rems to the hands of one of the Certified Nursing Assistants.

No personnel or property contamination occurred and no occupational worker received a whole body radiation dose above the NRC occupational limit of 1.25 rems. While members of the public received radiation doses above applicable limits, no one received a dose at which acute radiation injury or clinical signs are expected to occur.

The event was caused by the following:

(1) OSC had weaknesses in its radiation safety program that were a major contributing cause of the seriousness of the event and radiation exposure consequences. Some of these were a result of a rapid expansion in the HDR brachytherapy program from one facility to ten facilities in less than a year. The Radiation Safety Officer (RSO) failed to ensure that the staff at all facilities received adequate radiation safety training and that all management instructions related to HDR were being followed.

    Informal and unwritten procedures that may have been adequate when the licensee possessed one HDR unit under the direct control of the RSO were ineffective for the expanded program.

(2) A number of weaknesses were found in the design and testing of the Omnitron 2000. Weaknesses were identified in the testing and validation of source-wire design, and in the design of certain safety features of the HDR afterloader. These could allow the undetected retraction and further use of a broken wire with no warning to the user. Although not contributing to the event, weaknesses were found in Omnitron's quality assurance/quality control program. The cause of the wire failure is not known with certainty at this time. The vender believes it has evidence to show that storage of the source wire in teflon, if moisture is present, causes degradation of the teflon with release of fluorine or hydrogen fluoride that causes degradation of the Nitinol (nickel-titanium alloy) wire. The NRC and its consultant are still evaluating this hypothesis and conducting further studies.

(3) The safety culture at IRCC contributed significantly to the event. Technologists routinely ignored the PrimAlert-10 alarm. Its problems were worked around and not fixed. Technologists did not survey patients, the afterloader, or the treatment room following HDR treatments. No one was sure who was responsible for radiation safety training or the radiation safety program. The authorized user failed to wear a film badge on both occasions when the source was encountered.

(4) Overall regulatory oversight was weak. NRC regulations do not directly address HDR brachytherapy to the extent that teletherapy and low dose rate brachytherapy are addressed. Licensing guidance for HDR has been unchanged since 1986 in spite of significant changes in medical regulations and other medical

licensing guidance. Inspection guidance for medical licensees does not specifically address HDR brachytherapy. Although inspected by Region I within a year of initial licensing, the inspection program does not require early reinspection in cases where licensees significantly expand the scope of their program through license amendments. The regulatory interaction between the NRC, the Food and Drug Administration (FDA), and the involved Agreement States in the regulation and authorization of the Omnitron 2000 HDR afterloader is poorly defined.

Licensee actions to prevent recurrence are still undergoing NRC staff review.

The NRC issued Bulletin 92-03 to users of Omnitron 2000 HDR afterloaders, Information Notice 92-84 to all NRC licensees, and Confirmatory Action Letters curtailing the use of Omnitron 2000 HDR and providing safety precautions. On January 20, 1993, the NRC issued an Order Suspending License (Effective Immediately) to preclude the licensee from performing licensed activities at any of its facilities pending further order. Issuance of this order does not preclude further enforcement action.

The manufacturer's (Omnitron) actions to prevent recurrence are still undergoing FDA review.

The licensee hired a consultant to assess its radiation safety program immediately after the event occurred. The consultant provided the licensee with audit findings and suggested program upgrades. The licensee addressed the audit findings, created new operating and emergency procedures, trained personnel on procedures and the NRC requirements, and requested a Management Meeting with NRC to discuss the implemented program upgrades. The licensee met with the NRC on January 27, 1993. The NRC issued a meeting report on February 19, 1993, discussing all commitments the licensee made during the meeting. The licensee requested on February 9, 1993, permanent relaxation of the order suspending license to treat patients at the Greater Harrisburg Cancer Center and the GPPC. The licensee submitted its action plan to NRC in a letter dated February 15, 1993. The NRC reviewed the action plan and issued a deficiency letter to the licensee on March 5, 1993. The licensee again requested a management meeting to discuss the issues described in the NRC's March 5, 1993, deficiency letter. The licensee met with NRC on March 23, 1993. On March 29, 1993, the NRC issued a report discussing all commitments the licensee made during the meeting. On April 8, 1993, the licensee submitted its upgraded action plan and invited NRC to inspect its Harrisburg and Pittsburgh facilities to verify that all licensee procedures and NRC requirements were being followed as required. On April 22, 1993, NRC acknowledged receipt of the licensee's April 8, 1993 letter and agreed to inspect the Harrisburg and the Pittsburgh facilities.

The licensee submitted its program upgrades—in letters dated February 15, 1993, March 26, 1993, and April 8, 1993—implemented to address all items outlined in the Order suspending license. NRC Region I conducted an inspection at the licensee's facility in Harrisburg, Pa., on April 26, 27, and on May 3, 1993, and an inspection at the licensee's facility in Pittsburgh, Pa., on April 28 and 29, 1993. The inspectors concluded that the licensee had addressed all procedure requirements, order suspending license issues, Bulletin requirements, and other regulatory requirements. The inspectors also concluded that all personnel were trained on all licensee and NRC requirements as they pertain to their job.

The licensee requested that its NRC license be amended to include its current program as described in its February 15, 1993, March 26, 1993, April 8, 1993, May 7, 1993, and May 11, 1993, letters. The NRC issued a license amendment on May 26, 1993, and an inspection report on May 28, 1993. On June 3, 1993, the NRC Region I Regional Administrator approved a full "Order relaxation" request to treat patients, limited to OSC's facilities in Harrisburg and Pittsburgh.

On April 20, 1993, the NRC issued Bulletin 93-01: "Release of Patients After Brachytherapy Treatment With Remote Afterloading Devices" to licensees authorized to use afterloading brachytherapy units.

**Medical Therapy Misadministration and Temporary Loss of Brachytherapy Source at Yale-New Haven Hospital in New Haven, Conn.** A therapeutic exposure to a part of the body not scheduled to receive radiation can be considered an abnormal occurrence.

On December 2, 1992, the NRC was notified by the Yale-New Haven Hospital that it had recovered a 35 millicurie brachytherapy source that was discovered to be missing earlier that day. On December 3, 1992, NRC Region I was notified that the source had probably been lost before or during a brachytherapy treatment, resulting in a therapeutic misadministration. A female patient, approximately 39 years old, was to receive 1,848 rads to the cervix for cancer treatment. One of the sources that was prescribed was either never inserted or was removed from the applicator during treatment. Assuming maximum deviation from the planned treatment, the actual dose to the patient was only 1,235 rads. The licensee stated that a source was also misplaced and was in contact with one of the patient's legs for a period of time, resulting in an estimated dose to the leg of 260 rads. The physicians responsible for the treatment, after reviewing the dose estimates, decided no further treatments were necessary.

The misplaced source was inadvertently put with hospital linen. The linen with the brachytherapy source was taken to an off-site laundry facility, from which it was subsequently recovered. The referring physician and patient were notified of the misadministration.

The licensee failed to recognize the significance to radiation safety of a procedural change that eliminated the use of disposable pads in favor of reusable linen pads. Previously, the licensee disposed pads by putting them in infectious waste, which stayed in the room until after the final radiation survey was performed, after removal of the radiation sources. The reusable pads, when changed, were placed in laundry bags in the hallway, which were taken to the laundry facility daily. The nursing staff failed to follow the procedure that prohibited removing anything from the patient's room that had not been checked for the presence of a brachytherapy source.

The licensee has taken the following steps:

(1) Physicians have been instructed to visually confirm that sources are properly loaded into applicators.

(2) Dosimetrists have been instructed to observe the loading process and confirm that applicators are correctly loaded.

(3) A linen hamper will be placed in each brachytherapy patient's room so that linen will not, generally, be removed until after the final room survey to confirm that no sources have been lost.

(4) Soiled linen that cannot be left in the room until the end of treatment will be surveyed to ensure that no sources are in the linen prior to its removal from the patient's room.

(5) Physicians have been instructed to visually check for the presence of sources at the time they are removed from the patients.

The NRC retained a medical consultant to review the case to provide clinical assessment of this misadministration. NRC Region I conducted a special inspection on December 3–4, 1992, and three violations of NRC requirements were identified: (1) failure to survey soiled linen pads prior to removing them from a patient's room; (2) loss of control of the radioactive source; and (3) existence of radiation levels above the regulatory limit in unrestricted areas.

An Enforcement Conference was held on January 6, 1993. On January 13, 1993, NRC Region I recommended to the NRC Office of Enforcement that a Severity Level III violation with a Civil Penalty be issued with the Notice of Violation. A Notice of Violation and Proposed Imposition of Civil Penalties, and Confirmatory Order Modifying License were issued on April 26, 1993. The enforcement action was based on this event and another AO, discussed below. The cumulative amount of $10,000 for the violations was based on the combined events.

**Medical "Sodium Iodide" Misadministration at Ingham Medical Center in Lansing, Mich.** A diagnostic dose of a radiopharmaceutical that is greater than five times the prescribed dose can be considered an abnormal occurrence.

The referring physician's staff telephoned the Ingham Medical Center's nuclear medicine department on May 5, 1992, to schedule a thyroid scan to detect or rule out thyroid cancer. There was a miscommunication between members of the support staff. The technologist who received the call understood that the referring physician wanted a whole body scan to rule out thyroid metastasis and to look at a thyroid nodule. The medical technologist entered a whole body scan into the scheduling record.

On May 11, 1992, a 47-year old patient received 366.3 megabecquerels (MBq) (9.9 millicuries (mCi)) of iodine-131 (I–131) in capsule form, as preparation for a whole body scan. This procedure is normally used after a patient with thyroid cancer has had the thyroid removed or ablated to determine if the cancer originating in the thyroid has spread elsewhere in the patient's body. The patient still had an active thyroid and the patient's physician intended that the patient receive a thyroid scan to help determine if a thyroid nodule was cancerous. The thyroid scan is a different procedure from a whole body scan and as performed at the licensee's facility uses technetium-99m, a different radiopharmaceutical than I–131.

On May 12, 1992, the patient returned to the licensee's nuclear medicine department for the scan. The image of the initial scan showed that the patient's thyroid was intact and that an error had been made. The technologist performing the scan immediately reported the situation to the supervising physicians. The licensee's procedures for an I–131 whole body scan specified that the diagnostic procedure be used only on individuals whose thyroid had been removed.

The referring physician and the patient were notified of the misadministration. The licensee has been monitoring the patient and has observed decreased thyroid function.

Initially, the licensee decided that the incident was not a misadministration and did not report it to the NRC. That conclusion was reached because the correct dosage and procedure were used for a study, as understood by the technologist to have been what was requested. The licensee contacted NRC Region III about the incident after reading about a similar case in an NRC Office of Nuclear Material Safety and Safeguards newsletter. A licensee consultant reviewed the case with NRC Region III on February 19, 1993. Following that discussion, the consultant reported the incident as a misadministration because the procedure requested by the patient's physician, a thyroid scan, would normally employ a different radiopharmaceutical, namely, technetium-99m.

The basic causes of the misadministration were a miscommunication between the referring physician's office and the licensee, and a failure of the licensee to follow its Quality Management (QM) Program for procedures using radioactive pharmaceuticals. The licensee's QM

Program, which was implemented in January 1992, requires that a written directive be prepared for procedures using more than 1.11 MBq (30 microcurie (Ci) of I-131. However, no written directive was prepared for this procedure.

The licensee's procedure for a whole body I-131 scan required that the patient's thyroid had been removed previously. The licensee's procedures were not effective in determining if the patient had an intact thyroid. The nuclear medicine department staff had not received training on the requirements of the licensee's QM Program which included the provision that a written directive had to be issued for a whole body scan (using more than 1.11 MBq (30 Ci) of I-131).

The licensee has revised the procedures for thyroid cancer studies and provided training for nuclear medicine personnel in the QM Program requirements.

A special NRC inspection was conducted from February 25-to-26, 1993, to review the circumstances surrounding the I-131 misadministration. The NRC also retained a medical consultant to review the case. The NRC consultant concluded that the most probable effect of the misadministration would be permanent hypothyroidism, and he noted that evidence suggested that this condition had already occurred. No other health effects would be expected as a result of the misadministration. Several violations of NRC requirements were identified in the inspection.

On March 2, 1993, NRC Region III issued a Confirmatory Action Letter to the licensee documenting its agreement to provide training to the nuclear medicine staff on the requirements of the QM Program, NRC regulations, and NRC licensee requirements. No procedures using more than 1.11 MBq (30 Ci) of I-131 were to be performed before the training was completed. The licensee also agreed to make certain that its procedures for I-131 studies are consistent with the QM Program.

On September 9, 1993, the NRC issued a notice of violation and proposed imposition of a fine for $11,250 to the licensee. The licensee was cited for failing to have the physician authorized to use radioactive materials prepare a written directive as required for the dosage of I-131 involved in a whole body scan and for failing to follow the hospital's written instruction that I-131 whole body scans be used only for patients who had had their thyroids removed. Since the patient in this case had an intact thyroid, the whole body I-131 scan should not have been performed.

**Medical Therapy Misadministration Involving the Use of a High Dose-Rate Remote Afterloader Brachytherapy Device at Yale-New Haven Hospital in New Haven, Conn.** A therapeutic exposure, if parts of the body receiving radiation improperly would have normally received radiation anyway, had the proper administration been used, and the actual dose is greater than 1.5 times that intended to the above described body part, the event can be considered an abnormal occurrence.

A patient was prescribed to receive three treatments of 700 centigrays (cGy.) (700 rads)-per-treatment to the vagina using a Gamma Med high dose-rate remote afterloader brachytherapy device (HDR) at Yale-New Haven Hospital. During the first treatment on January 21, 1993, the physician mistakenly inserted the HDR applicator into the patient's rectum instead of the vagina, as prescribed. The licensee discovered the error immediately after the treatment was completed and the patient was immediately notified. The licensee estimated that the patient received approximately 350 cGy. (350 rads) to the vagina and 700 cGy. (700 rads) to the rectum. At the time of the NRC inspection on January 22, 1993, the licensee had planned to make up the dose to the vagina during the remaining two treatments and to add shielding to the applicator to prevent significant additional dose to the rectum.

The patient's physician, the physician who delivered the therapy, and an NRC Medical Consultant are presently evaluating the probable consequences of the misadministration.

The event occurred because the licensee did not confirm the treatment site before the treatment was given as required by its Quality Management (QM) Program.

The licensee added a procedure requiring physicians to visually insert applicators. And the licensee committed to a complete program assessment by an outside expert. This commitment was formalized by the NRC in a Confirmatory Order Modifying License issued on April 26, 1993.

NRC Region I conducted a special inspection at the facility on January 22, 1993. An NRC medical consultant was contacted to provide a clinical assessment of the effects of the misadministration. The licensee was offered the opportunity to participate in an Enforcement Conference but declined, believing that it would not be able to provide the NRC with any further information. The NRC recommended an enforcement action. A Notice of Violation and Proposed Imposition of Civil Penalties, and Confirmatory Order Modifying License were issued on April 26, 1993. The enforcement action was based on this event and another AO, discussed above. The cumulative amount of $10,000 for the violations was based on the combined events.

**Medical Therapy Misadministration at Papastavros' Associates Medical Imaging in Wilmington, Del.** A therapeutic dose that is less than 0.5 times the prescribed dose can be considered an abnormal occurrence.

On February 1, 1993, NRC Region I was notified by telephone that a therapeutic misadministration of I-131 occurred at the Papastavros' Associates Medical Imaging facility. In early January, the nuclear medicine technologist received a telephone call from the referring physician requesting that a patient be scheduled for a third treatment for hyperthyroidism and that 1.11 gigabecquerels (GBq) (30 millicuries (mCi])) of I-131 be administered. On January 13, 1993, the technologist ordered a 1.11 GBq (30 mCi) dose from the radiopharmacy. The dose was received on January 14, 1993. The technologist noted that the label on the lead container indicated 1.07 GBq (29 mCi) of I-131, but did not note that the label indicated that two capsules were present in the vial. A second technologist who removed the vial from the lead container and placed it in the dose calibrator for assay also failed to note that labels on both the lead container and the vial indicated the presence of two capsules. The assayed dose was consistent with the activity noted on the label. The technologist transferred the dose from the supplier's vial to a glass vial for administration to the patient. Only one of the capsules came out of the vial. The presumed empty lead container that still contained the plastic vial and remaining capsule was placed in the nuclear medicine hot laboratory for storage. The licensee discovered the remaining capsule on February 1, 1993, when the technologist was preparing lead containers for disposal. The patient was administered 0.56 GBq (15.1 mCi) of I-131, instead of the intended dose of 1.11 GBq (30 mCi). The misadministration was reported as required on February 1, 1993. The patient and the patient's physician were notified of the error and the patient was scheduled for follow-up therapy on February 10, 1993. The licensee's physician expected no adverse effects as a result of the misadministration. While the therapeutic dose administered was actually about 0.5 times the prescribed dose, the staff believes that this misadministration should still be considered an abnormal occurrence.

The misadministration was caused by failure of the licensee to establish and implement a Quality Management (QM) Program as required by 10 CFR 35.32(a). In particular, failure of the licensee to establish procedures to ensure that each therapy administration is in accordance with the written directive contributed to the misadministration.

The licensee's plan for preventing recurrence of the misadministration includes three steps: (1) to prepare and implement a written QM Program and provide training; (2) to have the radiopharmaceutical supplier indicate the number of capsules in each vial on the packing slip provided with I-131 therapy doses; and (3) to require the nuclear medicine technologists to read the label on the vials and lead containers to determine the number of capsules present in the vial, and then verify that the required number of capsules are administered to the patient. Also, the vial into which the capsules are transferred after initial assay will be reassayed to ensure that all capsules are transferred. The written QM Program was received by the NRC on February 11, 1993.

NRC Region I conducted an inspection on February 3, 1993. Because the misadministration resulted in an underdose to the patient and the therapy could be completed, the NRC did not contact a medical consultant to review the misadministration. A Confirmatory Action Letter (CAL) was issued to the licensee on February 5, 1993, which described the commitments made by the licensee to establish and implement a QM Program. An Enforcement Conference was held on March 1, 1993, to discuss the inspection findings and actions taken by the licensee in response to the CAL. On March 18, 1993, NRC Region I issued a Notice of Violation with a Severity Level III violation and $250 Civil Penalty. The licensee paid the Civil Penalty. The licensee's corrective and preventive actions will be reviewed during the next NRC inspection of the licensed program.

**Medical Brachytherapy Misadministration at Parkview Memorial Hospital in Fort Wayne, Ind.** A therapeutic exposure to a part of the body not scheduled to receive radiation should be considered an abnormal occurrence.

On December 9, 1992, a 62-year-old patient was scheduled to receive a 500 centigrays (cGy) (500 rad) radiation dose for vaginal cancer using a high-dose-rate brachytherapy treatment device at Parkview Memorial Hospital. The device uses a 296,000 megabequerel (MBq) (8 curie (Ci)) iridium-192 (Ir-192) source. The brachytherapy treatment was the final part of a curative radiation treatment series.

The location of the treatment area was unusual for vaginal treatments and required a different starting position for the Ir-192 source than is normally used for such treatments. Both the dosimetrist and the medical physicist performed the treatment calculations working together (the second series of calculations was not an independent check) and both used the incorrect starting location for the source position. The error was not detected, and the treatment was performed as scheduled. As a result, the intended 500 cGy. (500 rads) radiation dose was delivered to an area 5.25 centimeters (2.07 inches) away from the intended treatment area. A small portion of the intended treatment area received a radiation dose ranging from 50-to-300 cGy. (50-to-300 rads), according to the licensee.

On January 6, 1993, the error was discovered during a record review by a dosimetrist. The referring physician and the patient were informed of the error. The licensee reported the misadministration to NRC on January 7, 1993. The incident constitutes a misadministration because the radiation dose was administered to the wrong treatment site. On January 18, 1993, the patient received an additional treatment using the high dose rate brachytherapy treatment device. The treatment plan was revised to meet the intended objectives of the earlier treatment, taking into account the lower dose already received by a portion of the treatment area.

The licensee reported that no physical effect was observed as a result of the misadministration. The NRC retained a medical consultant to evaluate the circumstances of the misadministration. The consultant concluded that no noticeable biological effect is expected as a result of the misadministration.

Because of the unusual configuration of the treatment area, the standard treatment parameters used for vaginal brachytherapy treatment were not applicable. A medical physicist and a dosimetrist prepared the dose calculations working together and made the same error in assuming the initial position of the treatment source. The licensee's Quality Management Program requires that an independent check of the dose calculations be performed by a qualified individual before the treatment is initiated. Such an independent check was not performed.

The licensee has revised its procedures for preparing the treatment plans for the high-dose-rate brachytherapy procedures. It has made improvements in the calculation notebook and other related data used in preparing the treatment plans and the dose calculations.

NRC Region III conducted a special inspection on January 28 and 29, 1993, to review the circumstances surrounding the misadministration. An NRC medical consultant was also retained to review the case. The NRC inspection determined that the licensee failed to follow its Quality Management Program requirement for an independent check of brachytherapy dose calculations. Other violations were identified which did not directly relate to the misadministration. A notice of violation was issued to the licensee.

**Inoperable Research Reactor Scrams at University of Virginia in Charlottesville, Va.** A major deficiency in operating, management, or procedural controls that impact safety should be considered an abnormal occurrence.

Since November of 1992, the University of Virginia's research reactor had been experiencing a series of spurious scrams. The scrams were occurring without any annunciator indication. Because of the design of the scram annunciator system, the licensee staff did not believe that the unannunciated scrams were being caused by electrical supply line noise. A member of the licensee's staff who was in charge of the electronic maintenance at the facility concluded that the most likely source of the problem was in the scram logic system. Therefore, when he experienced unannunciated scrams on April 28, 1993, while performing the duties of the Senior Reactor Operator (SRO), he independently began trouble-shooting the problem to try to isolate the source of the scrams. There was no specific procedure in place to provide guidance for the trouble-shooting activities.

With the reactor shutdown, the SRO first interchanged some of the electronic equipment in the reactor control console. That action did not remedy the situation so he interchanged some other equipment, i.e., two mixer/driver (MD) modules. The MD modules appeared to be identical in their external appearance and both had the same identification number. After approximately 30 minutes, no further scrams were received so the SRO briefly conferred with the Reactor Administrator about the situation, and the reactor was restarted. Neither the SRO nor the Reactor Administrator realized that the trouble-shooting actions (exchanging the MD modules) were maintenance activities. Therefore, no postmaintenance testing was performed to ensure that the safety systems were operating as required.

The reactor was operated at full power for the next 5.5 hours with a change in SROs every two hours. No scram signal was received during that period. During a normal shutdown of the reactor at the end of the day on April 28, another SRO, who was then in charge of reactor operations, decided to complete the shutdown by introducing an electronic period scram. The scram logic, however, failed to produce the expected period scram and the SRO manually scrammed the reactor, which resulted in safe shutdown of the reactor.

The principal cause of the incident was the SRO exchanging the MD modules in the reactor control console. This inadvertently defeated five of the scrams required for reactor operation. Other contributing causes were not recognizing the exchanging of the modules as a maintenance activity, lack of adequate procedures defining maintenance and troubleshooting activities, and failure to perform post-maintenance testing of the safety system prior to restarting reactor operations.

The Reactor Director was notified of the problem when no scram was received the evening of April 28 and an investigation was begun into the cause of the problem. As a result of the investigation, the licensee initiated various corrective actions including: (1) maintaining the reactor in safe shutdown until the problem was investigated, understood, and reviewed with the Reactor Safety Committee and with the NRC; (2) notifying the University, the community, and the NRC of the problem; (3) requesting a peer review from the National Organization of Test, Research, and Training Reactors; (4) determining the root cause(s) of the event and taking corrective actions; (5) determining if there were any problems with the hardware, schematics, and Standard Operating Procedures which may have contributed to the event and taking actions to correct the problems noted; and (6) determining if any administrative corrective actions were needed.

A reactive inspection was conducted on May 3, 1993. Staff members from NRC Region II and headquarters participated in the inspection. A follow-up inspection was conducted on June 3 and 4, 1993, again with participation from NRC Region II and Headquarters. Apparent violations of regulatory requirements were identified and discussed with licensee management and the SRO involved in the incident during a June 29, 1993 enforcement conference held in the NRC Region II Office. The licensee presented its perspective on the significance of the event, its causes, and the licensee's corrective actions. A notice of violation and proposed imposition of civil penalty was issued by the NRC on July 28, 1993. Violations were proposed for operating the reactor without five safety system channels required by the Technical Specifications and for failing to verify that the safety system channels were operable following maintenance, as required by the Technical Specifications. These were categorized in the aggregate as a Severity Level II problem and a civil penalty of $2,000 was proposed. The licensee paid the civil penalty on August 26, 1993.

**Medical Brachytherapy Misadministration at Mercy Memorial Medical Center in St. Joseph, Mich.** A therapeutic exposure to a part of the body not scheduled to receive radiation should be considered an abnormal occurrence.

On February 16, 1993, at 5:00 p.m., a patient was undergoing a brachytherapy procedure using cesium-137 (Cs–137) sources at Mercy Memorial Medical Center. The radiation oncologist involved in this procedure failed to properly rotate the insert of the brachytherapy device containing the sources, and one source containing 862.1 megabecquerels (MBq) (23.3 millicuries (mCi)) of Cs–137 fell out of the insert onto the patient's bed. The source landed on an absorbent pad that was placed between the patient and the surface of the bed. The loss of the source was not observed by the oncologist or the medical physicist who was assisting him.

On February 17, at about 8:20 a.m., a nurse observed a small piece of metal between the patient and the absorbent pad. The nurse thought it was a small screw and retrieved it, placing it in a paper cup on the bedside table. The radiation oncologist and the medical physicist were notified, and they identified the object as a Cs–137 source. Using tongs, they placed it in a shielded storage container. The dislodged source was subsequently placed in the treatment device, and the treatment plan was revised to reflect that the source was implanted for a reduced period of time. The revised treatment plan indicated that this implant time reduction for the one source would result in an underdose of about 6 percent to the intended treatment site.

The licensee calculated that the dislodged source resulted in a radiation dose of about 45.8 centigrays (cGy) (45.8 rad) to the perineum, an area different from the intended treatment site. The licensee also stated that there is no evidence of clinical effects on the patient as a result of the radiation exposure from the dislodged source. The incident is considered a misadministration because a part of the patient's body received unscheduled radiation exposure. The licensee reported that both the patient and the referring physician had been notified of the incident.

The NRC staff calculated the dose to the nurse who discovered and handled the dislodged source. Based on information supplied by the nurse on her handling of the source, NRC calculated that she received a 4.25 cGy. (4.25 rads) radiation exposure to the surface of the hand in contact with the source.

The cause of the misadministration was the radiation oncologist's failure to properly rotate the Cs–137 source insert while loading the source into the treatment device. Also, the nurse who discovered the dislodged source had not received any training on the size and appearance of the brachytherapy sources.

To prevent recurrence, the licensee conducted refresher training for its nurses to explain brachytherapy procedures and provided them with instructions.

NRC Region III conducted a special inspection from March 26 through April 7, 1993, to review the circumstances surrounding the misadministration. An NRC medical consultant was also retained to evaluate the circumstances of the event.

The inspection identified several apparent violations of NRC requirements including: (1) substantial failure to implement a Quality Management Program for brachytherapy procedures; (2) failure of the RSO to adequately investigate the accident to identify a misadministration, and to assess over-exposure to the nurse's hands; (3) failure to adequately instruct nurses caring for brachytherapy patients; and (4) failure to make evaluations to assure compliance with NRC exposure limits for occupational workers. On August 2, 1993, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $6,250. The licensee paid the civil penalties on August 12, 1993.

**Medical Brachytherapy Misadministration at Keesler Medical Center, Keesler Air Force Base in Biloxi, Miss.** A therapeutic exposure to a part of the body not scheduled to receive radiation should be considered an abnormal occurrence.

On June 14, 1993, the United States Air Force Radioisotope Committee Secretariat (RIC) notified NRC Region IV of an incident involving a brachytherapy treatment which occurred at Keesler Medical Center on June 10, 1993. The permittee's Radiation Safety Officer (RSO) was not present at the facility on June 10. The permittee staff involved with the treatment did not initially recognize the incident as a misadministration. The incident and related facts were reported to the RSO when he returned to the medical center on June 14. The RSO subsequently notified RIC of the incident, who in turn formally notified NRC. RIC's initial report noted that a patient who was to receive a total dose of 1,400 centigray (cGy) (1,400 rads) to the right lower lung had also received an unintended dose of approximately 2.09 cGy. (2.09 rads) to the facial area.

The incident involved a brachytherapy treatment using an iridium-192 high-dose-rate remote afterloading device. The written directive prepared by the authorized user prescribed two treatment doses of 700 cGy. (700 rads) each to be delivered to the lower lobe of the patient's right lung. The first treatment dose was administered on June 2, 1993, using a single endobronchial catheter, as prescribed in the written directive. The second treatment dose was to be administered on June 10, 1993, using two endobronchial catheters, one positioned in the lower lobe of the right lung and the second positioned in the middle lobe of the lung. The fractional dose prescribed for the lower lobe was delivered as intended. The fractional dose for the middle lobe was not delivered as prescribed in the written directive because of the incorrect positioning of the source.

The mispositioning of the source resulted from an error in entering the length of the catheter into the treatment planning computer. The treatment plan established by the authorized user called for use of two catheters, each with a length of 150 centimeters (cm) (59.1 inches (in.)). The length of the first catheter and source position was properly entered at the treatment planning computer console. The permittee's dosimetrist believed that the length and source position for the second catheter were properly entered. However, it was later discovered that, because of an erroneous keystroke, a default value of 100 cm. (39.4 in.) was entered as the length of the second catheter. This resulted in an error in the source position since the actual distance of source travel is determined by subtracting an "offset" value from the length of the catheter. The error in the source position was recognized by the authorized user as the treatment was underway and the treatment was promptly stopped.

Following consultation with the device manufacturer and review of the treatment planning computer data and the data available from the remote afterloading device control console, permittee representatives determined that the source had been positioned approximately 10 cm. (3.9 in.) in front of the patient's face for a period of approximately 46 seconds. The estimated dose to the patient's face was determined to be approximately 2.09 cGy. (2.09 rads). In the absence of the licensee's RSO, and based on advice provided by the manufacturer's representative, the permittee's staff determined that the incident did not constitute a therapeutic misadministration. The remainder of the prescribed treatment dose was delivered to the middle lobe of the patient's right lung later that day. Through discussions with the RSO, the RIC, and the NRC staff, the permittee subsequently determined on June 14 that a misadministration had occurred and reported the incident to NRC and the patient as required.

NRC inspectors were at the medical center on June 23 and 24, 1993, to review the circumstances associated with the misadministration and its probable cause(s).

Based on interviews with permittee representatives and re-enactment of the treatment planning and setup, the apparent root cause of the misadministration was determined to be an erroneous keystroke at the treatment planning computer console. The permittee's dosimetrist demonstrated for the NRC inspectors the sequence of steps taken during treatment planning, noting that the correct value of 150 cm. (59.1 in.) had been entered for both catheters on June 10. However, the dosimetrist believed that after the length of the second catheter was entered, she depressed the "F2" function key to enter another treatment parameter and accidentally touched the "F1" function key with her hand at the same time. This caused the catheter length value to change to the default value of 100 cm. (39.4 in.) with only the sound of a "beep" to warn the operator. Through repetitive testing of different keystroke sequences, the dosimetrist determined that this was the only sequence that would reproduce a reset of the catheter length to the default value once the length was manually entered at the treatment planning console. This sequence of steps was repeated for the inspectors several times during the inspection and in each instance, the catheter length defaulted to 100 cm. (39.4 in.).

A contributing factor to the misadministration was the failure of permittee staff to verify the dwell positions for each catheter prior to performing the treatment as required by an "Operating Instruction" established by the permittee. Although this operating instruction was not incorporated in the permittee's Quality Management Program, it did require that individuals administering patient treatments using the high-dose-rate remote afterloading device verify both the source dwell time and source dwell positions prior to administering a treatment. This requirement was established to ensure that treatment parameters entered in the device control unit matched the parameters entered in the treatment planning computer. Both the dosimetrist and medical physicist who administered the treatment on June 10 acknowledged that they had only verified the source dwell times noted on the treatment planning and device control computer printouts. Although the dwell position value on both records was incorrect (because the error was propagated in both computer systems), the dosimetrist and physicist stated that they probably would have identified the error if they had verified the dwell positions prior to treatment.

Following the misadministration, the permittee modified a checklist that had been used by the staff to verify that certain actions were completed prior to treatment. The modifications included requirements to (1) physically measure each catheter prior to use for patient treatments and document the measured length of the catheter on the checklist form, (2) document the planned distance from the end of the catheter to the first dwell position on the checklist form, (3) have the authorized user and medical physicist verify the documented catheter length and dwell positions and sign the checklist for approval, and (4) include a review of the checklist in the permittee's Quality Management Program.

An inspection was conducted on June 23 and 24, 1993, to review the misadministration and its probable cause(s). Based on the results of the inspection, two apparent violations were identified with respect to the permittee's Quality Management Program. These included (1) a failure to implement and maintain a Quality Management Program that met the objective of ensuring that radiation from byproduct material was administered in accordance with a written directive, and (2) failure to indicate the radioisotope to be used for brachytherapy treatments in 22 written directives. Several weaknesses were also identified in the permittee's written Quality Management Program. The inspection findings indicated that the failure to verify the source dwell positions prior to performing a patient treatment was an isolated event and that the permittee staff had complied with the applicable operating instruction during previous patient treatments. A Notice of Violation was issued on July 20, 1993. A Civil Penalty was not proposed.

## Abnormal Occurrences Involving Agreement State Licensees

**Medical Diagnostic Misadministration at Southwest Texas Methodist Hospital in San Antonio, Tex.** Administering a diagnostic dose of a radiopharmaceutical that is greater than five times the prescribed dose can be considered an abnormal occurrence. This account is based on information provided to the NRC in October 1992 by the Agreement State of Texas.

On January 30, 1992, an iodine-131 (I-131) thyroid scan was requested for a patient to further evaluate a suspected right paratracheal mass to determine if the mass was a substernal goiter. The technologist confused the thyroid scan requested with a whole body scan because the mass to be imaged was in the chest. As a result, the patient was administered five millicuries of I-131 for a whole body scan instead of 100 microcuries of I-131 for the prescribed procedure for a thyroid scan with substernal mass.

Because of the high activity in the thyroid at the time of the imaging on January 31, 1992, a doctor was asked to review the examination. He discovered the dose error. The doctor reported that based on a normal thyroid uptake of 15 percent for I-131, a dose of five millicuries would deliver exposures of 4,000 rads to the thyroid and 2.35 rads to the whole body.

The misadministration was reported to the patient's referring physician, and he was advised that a radiation dose of this magnitude to the thyroid could result in development of hypothyroidism. The referring physician plans to follow the patient accordingly.

The misadministration occurred because a nuclear medicine technologist confused the requested partial body thyroid scan procedure with a whole body scan because of the location of the mass to be imaged.

The licensee established a policy that the administration of any dosage of I-131 greater than 100 microcuries must be reviewed by a staff radiologist licensed to administer radioactive materials with full knowledge of the clinical problem. The significance of the error was discussed with the technologist.

The licensee was cited by the Texas Bureau of Radiation Control for the misadministration in violation of license procedures.

**Contamination of Pool Irradiator Facility Owned by Radiation Sterilizers, Inc., in Decatur, Ga.** Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas should be considered an abnormal occurrence.

This event occurred in June 1988 in Georgia, an Agreement State by a joint Georgia and NRC incident Evaluation Task Force. The investigated incident was documented in NUREG-1392, "Leakage of an Irradiator Source—the June 1988 Georgia RSI Incident," published in February 1990. At that time, neither the State nor NRC identified the event as an AO. The state re-evaluated the incident against the reporting criteria in 1993 and concluded that the event should be classified as an abnormal occurrence.

On June 6, 1988, the Radiation Sterilizers, Inc. (RSI) facility in Decatur, Georgia, ceased sterilizer operations utilizing its pool irradiator because of the detection of dissolved radioactive cesium-137 (Cs-137) in a 25,000 gallon pool of water in which 252 stainless steel encapsulated radioactive sources were stored. The Cs-137 sources, which were leased from the Department of Energy (DOE), had a total activity of approximately 444,000 terabecquerels (TBq, or 12 megacuries). The sources were Waste Encapsulation Storage Facility (WESF) capsules manufactured by DOE, under the By-product Utilization Program (BUP). This DOE program was initiated with a mission to develop the means for application of radioactive fission products for the benefit of society. Under BUP, the sources were designed for waste storage, not as gamma radiation sources. In 1985, however, NRC agreed that use of WESF sources could be authorized in a limited number of commercial demonstration facilities, including irradiators such as RSI's that operate in the "wet load, wet storage, dry irradiation" mode.

Operators of the RSI pool irradiator notified the State of Georgia Radiation Control Program that the safety system had prevented them from raising sources from the storage pool. Subsequently, radiation levels of 600 microsieverts (Sv.) (60 millirem)-per-hour at the surface of the pool water were found, which indicated that one or more of the 252 Cs-137 source capsules used in the irradiator were leaking. Discrete samples of pool water were collected and analyzed and the analytical results showed elevated levels of Cs-137 dissolved in the pool water, confirming the presence of one or more leaking sources. This was the first recorded instance of a leaking WESF capsule. A joint Federal/State task force, consisting of the Georgia Department of Natural Resources (DNR), the Georgia Department of Human Resources, and NRC, was established to assist with the RSI incident.

After review and recommendation by the joint task force and upon discussion with RSI, on June 11, 1988, the State of Georgia formally requested that DOE manage the effort to identify the leaking capsule, develop a plan for the safe removal of the leaking capsule, manage the removal of the damaged capsule, and oversee the cleanup and recovery activities at RSI. DOE responded immediately to the State's request and dispatched resources from the Westinghouse Hanford Corporation. The joint Federal/State task force was also expanded to include representatives from the Food and Drug Administration and the U.S. Environmental Protection Agency.

The incident generally was confined to the RSI facility, with no evidence of major discharges to the environment. There was no evidence of over-exposure, although areas of minor contamination were found in a warehouse, the office carpet, one automobile seat, one individual's pants and spots on a carpet at a residence, all of which were decontaminated. Some medical products sterilized at the RSI facility before the incident was discovered were contaminated; however, the only contaminated products released from the RSI facility were in a shipment that was recalled before it reached its destination.

Five suspected leaking (damaged) sources were initially removed from the storage pool. One was confirmed to be leaking. Subsequently, the remaining 247 source capsules were examined for leakage. None was found to be leaking. On November 19, 1990, the last of the capsules was shipped off site and final decontamination of the facility began. On September 11, 1992, the DOE contractor completed decontamination of the facility and began the survey. DOE estimated cost of the cleanup to be 45 million dollars. On November 11, 1992, the DOE contractor completed the free release survey and the DNR contractor completed the confirmatory survey. By December 16, 1992, DNR received the free release survey. On January 5, 1993, after review and evaluation of the reports, DNR returned control of the facility to the owner, Sterigenics International (formerly RSI), and terminated its radioactive materials license.

The facility contamination resulted from one stainless steel Cs-137 source capsule, out of a total of 252 capsules, leaking in the source storage pool. DOE has not identified the exact cause of failure of the Cs-137 source capsule.

The licensee requested that DOE (the source manufacturer and the source lessor) manage the effort to identify the leaking capsule, develop a plan for its safe removal, manage its removal, and oversee the cleanup and recovery activities at RSI.

Following the incident, NRC re-evaluated the WESF sources and determined in early 1991 that WESF sources were not appropriate for long term use in commercial irradiator facilities and ensured that the remaining commercial users were so notified and advised to cooperate with DOE in scheduling removal of WESF sources from the facilities. At the close of the report period, WESF capsules remained in place in two licensed irradiators, one in Virginia and one in Colorado (licensed by the State of Colorado). According to DOE staff, if certain technical matters are resolved, DOE plans to begin removing the remaining WESF sources from these facilities by the end of 1993.

The State of Georgia secured the services of an independent consultant to verify the results of decontamination efforts by the DOE contractor. Once it was verified that the facility met Federal and State regulatory standards for decontamination, the State terminated RSI's material license and returned control of the facility to its owner. Georgia will no longer license highly soluble cesium for this application.

**Medical "Sodium Iodide" Misadministration at Grenada Lake Medical Center in Grenada, Miss.** Administering a diagnostic dose of a radiopharmaceutical that is greater than five times the prescribed dose should be considered an abnormal occurrence. This account is based on information provided to the State of Mississippi on April 3, 1992.

On April 1, 1992, a patient scheduled to receive 3.7 megabecquerels (MBq) (100 microcuries ($\mu$Ci)) of iodine-131 (I-131) for a thyroid uptake study was administered 218.3 MBq (5.9 millicuries (mCi)) of I-131. The 218.3 MBq (5.9 mCi) dosage of I-131 was to be administered to another patient. The technologist immediately discovered the error and notified the physician (authorized user). Vomiting was induced within five minutes of administering the I-131 capsule. The patient was also administered a thyroid blocking agent, 1.2 milliliters (ml) (0.04 fluid ounces (fl. oz.)) of potassium iodide. The patient was also instructed to take additional thyroid blocking agent, 0.3 ml (0.01 fl. oz.) of potassium iodide, once a day for 14 days. A thyroid uptake and scan were performed 24 hours after the incident. The thyroid uptake was 0.3 percent. The referring physician and the patient were informed of the misadministration.

The misadministration occurred because the nuclear medicine technologist failed to identify the patient prior to the administration of the radiopharmaceutical.

The licensee's Radiation Safety Officer has implemented new procedures for verification of patient identification and has committed to improve the supervision of personnel. The licensee also stated that patients who are prescribed radiation therapeutic procedures will no longer be included in the same schedule with patients who are prescribed diagnostic procedures.

The State agency staff has reviewed the circumstances of the misadministration and will evaluate the licensee's corrective actions during the next inspection to be conducted in the near future.

**Medical Brachytherapy Misadministration at Maine Medical Center in Portland, Me.** A therapeutic exposure to a part of the body not scheduled to receive radiation should be considered an abnormal occurrence.

A patient was prescribed a brachytherapy treatment using 13 seeds of iridium-192 in a nylon ribbon. The catheter used for the treatment developed a kink and stopped 26 centimeters (cm) (10.24 inches (in.)) from the prescribed treatment area. This resulted in a dose to the patient's hypopharynx area of 3,500 centigrays (cGy) (3,500 rads), which was the prescribed dose to the lung. The intended treatment area of the lung was estimated to have received less than 10 cGy. (10 rads).

Prior to implantation of the radioactive seeds, non-radioactive sources were implanted for visualization and dosimetry/treatment planning. The licensee performed x-rays which showed that the dummy seeds had reached their desired location. The active seeds were to be inserted immediately after withdrawal of the dummy seeds. However, because of scheduling difficulties with the patient's room, and not wanting a patient with radioactive seeds to remain in the therapy department for an undetermined period of time, the dummy seeds were withdrawn but the catheter remained in the patient. The radioactive seeds were implanted sometime later. In retrospect the licensee estimated that the kink in the catheter developed during the interval of removing the dummy seeds and inserting the active seeds.

After the treatment was completed and while removing the catheter and the nylon ribbon together, the doctor and the Radiation Safety Officer both noticed the kink. The licensee stated that no long term effects are expected. The patient was notified of the misadministration.

The licensee implemented the following actions: (1) measuring the non-radioactive seed strand when properly inserted (verified by x-ray) and marking the distance on the active strand; (2) the dummy strand or similar wire will be left in the catheter until immediately prior to insertion of the radioactive strand; and (3) a film will be taken of the area to be treated after the active seeds are inserted to ensure that they are in the correct location.

The State agency has reviewed and approved the corrective actions taken by the licensee as a result of the misadministration and therefore considers this case closed.

**Industrial Radiographer Over-exposure Event at Murphy Oil Refinery in Meraux, La.** Any loss of licensed material in such quantities and under such circumstances that substantial hazard may result to persons in unrestricted areas should be considered an abnormal occurrence.

While working at a temporary job site at the Murphy Oil refinery on May 7, 1993, a 21 year-old industrial radiographer employed by Inspection Specialists, Inc., using 3,700 gigabecquerels (GBq) (100 curies) of iridium-192 in a SPEC 2–T exposure device, received a 276.6 millisieverts (mSv.) (27.66 rems) whole-body exposure as indicated by a thermoluminescent dosimeter badge. Re-enactment of the events appears to indicate that the radiographer received the whole body dose, and that there was probably no extremity dose that would cause acute (short-term) injury. A preliminary physical examination with blood tests indicated no indication of excessive exposure. The radiographer's assistant is estimated to have received a dose of 9.6 mSv. (0.96 rems).

Radiography operations were being conducted on a large, open-top steel tank. The radiographers and camera had to be moved from place to place along the side of the tank in personnel baskets. The radiographer failed to lock the exposure device, so that when the radiographer's assistant moved toward the device with the control handle, the source moved slightly out of the shielded position. The radiographer apparently failed to read the survey meter while the source was exposed. The radiographer and assistant realized after several more exposures that their dosimeters were off-scale.

The licensee provided retraining to the entire staff with special counseling for the Operations Manager, who apparently did not follow written operating procedures.

The Louisiana Radiation Protection Division recommended to the licensee that routine physical examinations and blood work be performed. Enforcement actions included citations for violations associated with whole body and extremity over-exposures and a lack of adequate training. A civil penalty is being considered for lack of management control.

## DIAGNOSTIC EVALUATION PROGRAM

The Diagnostic Evaluation Program (DEP) provides an independent assessment of licensee performance at selected reactor facilities. The program evaluates the involvement of licensee management and staff in ensuring safe plant operations, the effectiveness of their actions, and the root causes of safety-related performance problems. The DEP supplements the licensee assessment information provided by the NRC's Systematic Assessment of Licensee Performance (SALP) Program, Performance Indicator (PI) Program, and the routine and special inspections performed by NRC Headquarters and the

# Table 2. Abnormal Occurrences Reported During FY 1993

*OCCURRENCES AT NUCLEAR POWER PLANTS*

| *AO Number* | *Subject* | *NUREG–0090 Issue* |
|---|---|---|
| 92–12 | Operation With Degraded Steam Generator Tubes at Arkansas Nuclear One Unit 2 and McGuire Nuclear Station Units 1 and 2 | Vol. 15, No. 4 March 1993 |
| 92–13 | Engineered Safety Features Actuation System Design Deficiency—Single Failure Vulnerability at Millstone Power Station Unit 2 | Vol. 15, No. 4 March 1993 |
| 93–1 | Steam Generator Tube Rupture at Palo Verde Unit 2 | Vol. 16, No. 1 June 1993 |

*OCCURRENCES AT FUEL CYCLE FACILITIES*

None reported in FY 1993

*OCCURRENCES AT OTHER NRC LICENSEES*
*(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)*

| *AO Number* | *Subject* | *NUREG–0090 Issue* |
|---|---|---|
| 92–9 | Medical Therapy Misadministration at Cooper Hospital/ University Medical Center in Camden, New Jersey | Vol. 15, No. 3 December 1992 |
| 92–10 | Extremity Over-exposure of a Radiographer at MQS Inspection, Incorporated, Field Site in Trenton, Michigan | Vol. 15, No. 3 December 1992 |
| 92–11 | Medical Therapy Misadministration at the Medical Center of Delaware, Incorporated, in Wilmington, Delaware | Vol. 15, No. 3 December 1992 |
| 92–14 | Medical Therapy Misadministration at Memorial Hospital of Laramie County in Cheyenne, Wyoming | Vol. 15, No. 4 March 1993 |
| 92–15 | Medical Therapy Misadministration and Unplanned Exposure at St. Clares Riverside Medical Center in Denville, New Jersey | Vol. 15, No. 4 March 1993 |
| 92–16 | Medical Therapy Misadministration at the Lahey Clinic Medical Center in Burlington, Massachusetts | Vol. 15, No. 4 March 1993 |
| 92–17 | Medical Therapy Misadministration at Indiana University Medical Center in Indianapolis, Indiana | Vol. 15, No. 4 March 1993 |
| 92–18 | Loss of Iridium-192 Source and Medical Therapy Misadministration at Indiana Regional Cancer Center in Indiana, Pennsylvania | Vol. 15, No. 4 March 1993 |
| 92–19 | Medical Therapy Misadministration and Temporary Loss of Brachytherapy Source at Yale-New Haven Hospital in New Haven, Connecticut | Vol. 15, No. 4 March 1993 |

# Table 2. Abnormal Occurrences Reported During FY 1993
## (continued)

| *AO Number* | *Subject* | *NUREG-0090 Issue* |
|---|---|---|
| 93-2 | Medical "Sodium Iodide" Misadministration at Ingham Medical Center in Lansing, Michigan | Vol. 16, No. 1 June 1993 |
| 93-3 | Medical Therapy Misadministration Involving the Use of a High Dose-Rate Remote Afterloader Brachytherapy Device at Yale-New Haven Hospital in New Haven, Connecticut | Vol. 16, No. 1 June 1993 |
| 93-4 | Medical Therapy Misadministration at Papastavros' Associates Medical Imaging in Wilmington, Delaware | Vol. 16, No. 1 June 1993 |
| 93-5 | Medical Brachytherapy Misadministration at Parkview Memorial Hospital in Fort Wayne, Indiana | Vol. 16, No. 2 September 1993 |
| 93-6 | Inoperable Research Reactor Scrams at University of Virginia in Charlottesville, Virginia | Vol. 16, No. 2 September 1993 |
| 93-7 | Medical Brachytherapy Misadministration at Mercy Memorial Medical Center in St. Joseph, Michigan | Vol. 16, No. 2 September 1993 |
| 93-8 | Medical Brachytherapy Misadministration at Keesler Medical Center, Keesler Air Force Base in Biloxi, Mississippi | Vol. 16, No. 2 September 1993 |

*OCCURRENCES AT AGREEMENT STATE LICENSEES*

| *AO Number* | *Subject* | *NUREG-0090 Issue* |
|---|---|---|
| AS92-1 | Medical Diagnostic Misadministration at Southwest Texas Methodist Hospital in San Antonio, Texas | Vol. 15, No. 3 December 1992 |
| AS93-1 | Contamination of Pool Irradiator Facility Owned by Radiation Sterilizers, Incorporated, in Decatur, Georgia | Vol. 16, No. 2 September 1993 |
| AS93-2 | Medical "Sodium Iodide" Misadministration at Grenada Lake Medical Center in Grenada, Mississippi | Vol. 16, No. 2 September 1993 |
| AS93-3 | Medical Brachytherapy Misadministration at Maine Medical Center in Portland, Maine | Vol. 16, No. 2 September 1993 |
| AS93-4 | Industrial Radiographer Over-exposure Event at Murphy Oil Refinery in Meraux, Louisiana | Vol. 16, No. 2 September 1993 |

Regional Offices. The DEP provides in-depth and detailed information for the decision-making of senior NRC management in their oversight of nuclear plant safety.

When a diagnostic evaluation is approved for a specific facility, a Diagnostic Evaluation Team (DET) is established by the NRC's Executive Director for Operations (EDO). The DET is composed of technical staff members from NRC's Headquarters and Regional Offices, resident inspectors and contractors, as appropriate. Team members who are selected for a DET have not had previous significant involvement in recent inspections or reviews of the facility so as to provide an unbiased and independent assessment of plant performance. A DET provides the broad-based assessment of licensee safety performance at the plant selected for the evaluation. Within the overall broad scope of the review, emphasis and focus of the DET is dependent on areas of special interest to NRC management. The evaluation process involves observations of plant activities, in-depth technical reviews, licensee employee interviews, equipment "walkdowns," and programmatic reviews in a number of functional areas important to safety, such as maintenance, surveillance and testing, management involvement, engineering and technical support, conduct of operations, safeguards and security, plant modifications and design changes, radiation protection, quality assurance, and corrective action.

**Diagnostic Evaluation of the South Texas Project.** In January 1993, the EDO directed that a diagnostic evaluation of the South Texas Project nuclear power plant be conducted. The decision to conduct the evaluation was based on an apparent decline in the performance of maintenance and surveillance, engineering and technical support, security, and safety assessment/quality verification. A 16-member team spent approximately three weeks evaluating activities at the South Texas Project site. The evaluation was performed in March and April of 1993. Some members visited the licensee's headquarters in Houston. The areas evaluated included operations and training, maintenance and testing, engineering support, and management and organization. The team's evaluation report was issued in June 1993. The findings and conclusions of the DET were discussed with he licensee at a public meeting on June 3, 1993.

The DET identified several performance deficiencies in the areas of operations and training, maintenance and testing, and engineering support, and found that weaknesses in management had contributed to these deficiencies. The team found that although management was aware of many of these problems for some time, they had not been effective in resolving underlying root causes and in improving performance. Specifically, the team found that: staffing levels in operations were strained; planning, scheduling and work control processes were ineffective and inefficient; material condition of some equipment was poor; system engineering was ineffective; problem identification, root cause determination and corrective actions for some equipment failures were inadequate; accident mitigation capability of the safety-related essential chillers under low heat load conditions had not been adequately analyzed or demonstrated; and management had not provided effective direction or support.

The DET concluded that the underlying root causes for declining performance were: failure of management to provide adequate support, ineffective management direction and oversight, failure to effectively utilize self-assessment and quality oversight functions, and ineffective root cause and corrective action processes.

**Diagnostic Evaluation of the Quad Cities (Ill.) Nuclear Power Plant.** In June 1993, the EDO directed that a diagnostic evaluation be conducted for the Quad Cities nuclear power plant. The decision to conduct the evaluation was based on an apparent decline in performance indicated by increased personnel errors and procedural problems, lack of a questioning attitude by plant personnel, and weaknesses in management oversight and control. A 16-member team spent approximately three weeks evaluating activities at the Quad Cities site and visited the licensee's corporate offices in both Downers Grove and Chicago, Illinois. The areas evaluated included operations and training, maintenance and testing, engineering support, and management and organization. The evaluation was performed in August and September of 1993. The team's evaluation report was issued in November 1993. The findings and conclusions of the DET were discussed with he licensee at a public meeting on November 8, 1993.

The team identified performance deficiencies in the areas of operations and training, maintenance and testing, and engineering and technical support, and found that weaknesses in management had contributed to these deficiencies. Although senior site managers had been aware, for some time, of many of the problems described in the DET report, they had not been effective in resolving underlying root causes and improving performance. Specifically, the team found that: management was willing to accept equipment problems without aggressively pursuing corrective actions; operations management rarely formally evaluated operability of degraded equipment; engineering assessments of degraded plant hardware were not rigorous; the work control process was ineffective and inefficient; the effects of vibration on several plant systems had not been evaluated; the large number of uncorrected component problems resulted in the degradation of safety systems; there were significant leadership weaknesses in site and corporate management; and a number of previous initiatives and self-assessments to improve performance had not been successful.

The team found the root causes of Quad Cities' performance to be: (1) ineffective corporate leadership, oversight, involvement, and follow through; (2) site management's failure to resolve identified safety problems; (3)

low standards of performance; and (4) site management's failure to exercise effective leadership.

## INCIDENT INVESTIGATION PROGRAM

The Incident Investigation Program (IIP) exists to ensure that the NRC's investigation of significant events is timely, thorough, well coordinated, and formally administered. The scope of the program covers the investigation of significant operational events involving both reactor and materials licensees. The IIP's primary objectives are, in general, to ensure that: operational events are investigated in a systematic and technically sound manner; all available information pertaining to the causes of the events is collected, including events involving the NRC's own activity; and to provide appropriate feedback regarding what has been learned from the events, to the NRC, the industry and the public.

By focusing on the causes of operating events and the identification of associated corrective action, the IIP process provides for a more complete technical and regulatory understanding of significant events. The IIP involves two types of investigatory response, based on the safety significance of the operational events. The objectives of both responses are to identify the event circumstances and to ascertain the causes. For an event of potentially major significance, an Incident Investigation Team (IIT) is established by the Executive Director for Operations (EDO), comprised of a Headquarters-directed team complemented by regional staff, and may include both industry representatives and contractors as appropriate. The investigation of less significant operational events is conducted by an Augmented Inspection Team (AIT), consisting of a Region-directed team complemented by headquarters personnel and, in some cases, by personnel from other Regions.

To maintain a high state of readiness to dispatch a highly qualified IIT at any time, an IIT training program was developed to provide prospective IIT members with detailed knowledge of the guidelines and methods used for conducting a systematic and technically sound investigation. The training program was developed by the Office for Analysis and Evaluation of Operational Data (AEOD) following discussion with representatives of the National Transportation Safety Board, Federal Aviation Administration, and National Aeronautics and Space Administration, and has been continually refined over the years.

The fifth in a series of two-week IIT training courses was conducted from January 25 through February 5, 1993. Two more one-week IIT refresher training courses have also been conducted since the inception of the IIP. The 1993 IIT training course emphasized training on the incident investigation process and techniques, and included simulated investigations of reactor and non-reactor incidents for the course participants. Training was also provided on the recent changes to the IIP.

Of the reportable events which occurred during fiscal year 1993, two were judged to have a sufficiently high level of safety significance to warrant an IIT response. The first IIT response was for the Loss of an iridium-192 Source and Therapy Misadministration at the Indiana Regional Cancer Center, Indiana, Pa., on November 16, 1992. The second IIT response was for the Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 (Pa.) on February 7, 1993. Significant findings and conclusions from the associated IIT reports are provided below. In addition, a total of 14 reactor and one materials licensee events resulted in AITs being established.

IIT investigated the loss of an iridium-192 source and therapy misadministration at the Indiana Regional Cancer Center, Indiana, Pa. On December 1, 1992, the Indiana Regional Cancer Center reported to the NRC's Region I Office that they believed a 1.37 E+11 becquerel (3.7 curie) iridium-192 source from the center's Omnitron 2000 high dose rate (HDR) brachytherapy afterloader had been found at a biohazard waste transfer station, operated by Browning-Ferris Industries, in Carnegie, Pa. The Indiana Regional Cancer Center is one of several operated by the licensee, Oncology Services Corporation. The source was first detected when it triggered radiation alarms at a waste incinerator facility in Warren, Ohio. After notifying the NRC, cancer center personnel retrieved the source, and Region I dispatched an inspector and a supervisor to investigate the event. The licensee informed the NRC that the source wire had apparently broken during treatment of a patient on November 16, 1992, leaving the source in the patient. The patient subsequently died on November 21, 1992. On the basis of the seriousness of the incident, the Executive Director for Operations elevated the NRC's response by establishing an IIT. The IIT initiated its investigation on December 3, 1992. During the investigation, on December 7, 1992, a very similar source wire failure occurred with an HDR afterloader at another Oncology Services Corporation facility located in Pittsburgh, Pa. Although the latter incident had minimal radiological consequences, it was included in the scope of the IIT's investigation.

The IIT's investigation included visits to: the Browning-Ferris Industries facilities in Carnegie, Pa., and Warren, Ohio; Oncology Services Corporation facilities in Pittsburgh and Indiana, Pa.; the Omnitron facilities in Texas and Louisiana; and NRC Regional and Headquarters Offices. The team also utilized outside medical, technical and laboratory contractor services for the investigation. The IIT coordinated and cooperated with local, state, and federal agencies, including the Pennsylvania State Police and the U.S. Food and Drug Administration.

The team found that the patient who died had received a serious misadministration and that over 90 other individuals had been exposed to elevated levels of radiation from November 16 to December 1, 1992. In a press release dated January 26, 1993, the Indiana County Coroner stated that the cause of death listed in the official autopsy report was "Acute Radiation Exposure and Consequences Thereof." The IIT also found:

- Weaknesses in Oncology Services Corporation's Radiation Protection Program were a contributing cause of the seriousness of the event and the radiation exposure consequences.
- Weaknesses existed in the design and testing of the Omnitron 2000 remote afterloader system and its source wire.
- Oncology Services Corporation and Indiana Regional Cancer Center lacked critical safety awareness with respect to high dose rate brachytherapy.
- Regulatory oversight weaknesses existed in areas such as HDR afterloader use, licensing and inspection of the licensee's rapidly expanding treatment programs and the regulation of vendors of devices that use licensed nuclear material. The IIT concluded, however, that none of the weaknesses directly caused the incident or increased the significance of the consequences.
- NRC regulatory guidance did not exist for non-radioactive waste collectors; and, Browning-Ferris Industries personnel failed to follow their existing radiation control policies.

**Unauthorized Forced Entry into the Protected Area at Three Mile Island Unit 1 (Pa.).** On February 7, 1993, with Unit 1 operating at full power, an intruder drove into the site owner-controlled area, through a gate into the protected area of the Three Mile Island Unit 1 (Pa.) nuclear power plant (TMI–1) and crashed through a roll-up door for the turbine building. The plant operators in the control room were notified of the event by a call from the off-going operations shift foreman and site protection officers. The control room personnel responded by implementing emergency response procedures and classified the event as a Site Area Emergency within twelve minutes of the intrusion. The licensee notified the Commonwealth of Pennsylvania and the NRC who responded by activating their respective response organizations. The shift supervisor declared a Site Area Emergency and notified the NRC headquarters operations officer of this action 18 minutes later. Upon considering the possible significance to physical security and the regulatory questions that could result from the event, the Executive Director for Operations established an IIT to determine what had happened and make appropriate findings and conclusions. The team included an industry consultant, two observers from the Commonwealth of Pennsylvania, and one observer from the NRC Office of the Inspector General.

The IIT ultimately concluded that no actual adverse reactor safety consequences had occurred and the event was of minimal safety significance. The team also concluded that the decision to maintain power operations was appropriate for this event and the licensee's security force responded appropriately to the specific challenge presented by the intruder. However, a number of salient findings were reported by the IIT. Among them were:

- There were conflicts between operations, emergency response, and security actions that resulted from limited key card access, the locking of the control room fire doors and personal safety concerns.
- The licensee focused on re-establishing the security of the facility and eliminating the intruder threat. TMI management departed from the E-Plan and procedures to address the immediately known conditions and did not fully consider the possibility of radiological sabotage which could warrant full scope emergency response capabilities.
- The need to deviate from the security and emergency plan implementing documents may have been appropriate during this event, however, compensatory alternatives were not considered and the use of 10 CFR 50.54(x) and (y) was not properly implemented.
- The decision to maintain stable, steady-state reactor power operations was in accordance with an established emergency procedure and was appropriate for this event. However, the procedure did not contain qualifying guidance and may not have been appropriate in all security events.

The following regulatory-related concerns were also identified:

- The NRC focused its response on security concerns and did not fully staff response facilities in preparation to address the broader implications of any radiological sabotage.
- Previous TMI events, drills, and other reports identified weaknesses that were also evident during this event.
- The NRC requirements for establishing and maintaining a physical protection system and as used during the security program licensing process do not consider use of a vehicle to breach the protected area barrier. In this event, the use of a vehicle reduced the amount of time the security force had to assess and respond to the threat.
- The NRC's security inspection program was not effective in revealing and evaluating the types of challenges demonstrated by this event.

The IIT did not interview the alleged intruder, and thus was unable to establish a motive for his actions. However, an interview was subsequently conducted by the Office of Investigations (OI). Based on the OI interview, it has been concluded that the TMI plant was not specifically targeted for the intrusion, and there was no basis to revise any of the conclusions documented in the IIT's report.

## TECHNICAL TRAINING PROGRAM

The NRC Technical Training Center (TTC) coordinates with the NRC Headquarters and Regional Offices in the development and implementation of NRC staff technical qualification programs. Technical training is provided for NRC personnel, selected NRC contractors, and other government organizations, as appropriate. Initial training is provided to NRC inspectors, operator licensing examiners, reviewers, project managers, operations officers, technical managers, and other NRC personnel with the level of knowledge of reactor technology and other specialized technical training necessary to perform assigned agency functions. Refresher training is provided for NRC inspectors, examiners, and operations officers. Principles of the systems approach to training are routinely used throughout the life cycle of courses managed by the TTC. Although located in Chattanooga, Tenn., the TTC is part of the NRC headquarters organization within the Office for Analysis and Evaluation of Operational Data (AEOD).

The reactor technology curriculum continued to consist of a spectrum of courses involving both classroom and simulator training covering the General Electric, Westinghouse, Combustion Engineering, and Babcock & Wilcox reactor vendor designs. Reactor technology courses are typically presented by TTC staff members. The TTC continued management of the operation, maintenance, and upgrade of full scope reactor training simulators for each vendor design and associated computer equipment in support of established training needs.

The core of the reactor technology training provided in support of initial qualification programs for NRC staff continued to be an integrated series of reactor technology courses, consisting of a three-week technology course, a two-week advanced technology course, a one-week reactor simulator course, and a one-week emergency operating procedure (EOP) simulator course. This integrated reactor technology training for each reactor vendor design was available numerous times throughout the year. A variety of other stand-alone reactor technology courses were made available to support other parts of NRC staff qualification programs. Simulator refresher training was provided in all reactor technology areas on numerous occasions to maintain staff qualification.

The specialized technical training curriculum continued to consist of a number of courses in engineering support, health physics, safeguards, and inspection or examination techniques. Specialized technical training was provided by means of customized courses developed by TTC staff or contractors, by coordination of slots (training opportunities) in courses that were presented by other government agencies, and by identification and promotion of appropriate commercially available courses that NRC personnel attended as individual training opportunities. For many of the contracted courses, NRC perspectives were provided by specifically designated individuals within the NRC staff.

During fiscal year 1993, the TTC conducted or coordinated a total of 94 courses in the reactor technology areas and 88 more in the specialized technical training areas. These courses represented a total of 221 course-weeks, 116 of which were associated with reactor technology training and 105 of which were associated with specialized technical training. All courses falling under the TTC program element and listed in the TTC Syllabus of Courses are included in these totals. This level of training represented 69,479 instructional hours, of which 32,352 were associated with reactor technology training and 37,127 with specialized technical training. (An instructional-hour is a one hour period of training devoted to any of the following activities: lectures, seminars, discussions, problem solving sessions, quizzes, examinations, on-the-job training, laboratory exercises, programmed learning, and simulation exercises. For example, a course of 16 hours for 10 students would constitute a total of 160 instructional hours.)

Besides the technical training in support of qualification programs for NRC technical staff, the TTC provided reactor technology training in association with the Orientation and PRA Technology Transfer Programs managed by the Office of Personnel.

Engineering support training in many forms was provided at various locations and times throughout the year to meet agency needs. The Power Plant Engineering Course continued to be needed for initial training of inexperienced technical personnel such as interns. Specialized seminars on motor operated valve diagnostic methodologies were held to support the NRC inspection program. Examples of other courses presented in this area included Emergency Diesel Generator, Motorized Valve Actuators, Fire Protection for Power Plants, Welding Technology and Codes Nondestructive Examination Technology and Codes, Eddy Current Testing and Inservice Inspection.

With the presentation of the Health Physics Topical Review Course, all technical courses required by NRC Inspection Manual Chapter (IMC) 1245, Inspector Qualifications, for health physics inspectors have been implemented. The course, intended as a refresher course for health physics personnel, included sessions on survey instrument calibration and counting statistics. Future course presentations will be rotated through the Regional Offices and will have periodic changes in the course content to reflect important and emerging issues.

Skin Dosimetry Workshops were conducted in each Region. The workshops provided instruction on radiobiology and dose assessment methodologies and included a review of agency guidance from information notices as well as summaries of guidance from national and international standards.

A multi-year contract for the Transportation of Radioactive Materials Course was established to replace the previous interagency agreement, and the first course was presented. The course will remain the same with hands-on exercises and field trips to the Barnwell Waste Disposal Site. Course attendance will continue to be shared by NRC and Agreement State personnel.

Development and initial presentation of a Nuclear Criticality Safety Course was accomplished during the year. The course was developed by the Oak Ridge Institute for Science and Education with significant cooperation between NMSS and the TTC. An abbreviated version of the Fuel Cycle Technology Course was presented for incident response center personnel. These courses are the first in an effort to develop training courses necessary to support NRC personnel having inspection responsibility for various stages of the fuel cycle.

Contracted presentations of Site Access Training (SAT), Site Access Refresher Training (SART) and NMSS Radiation were presented a number of times during the year. The training manuals associated with these courses have been updated to incorporate the revised 10 CFR Part 20. Actions are underway to update the course lesson plans and to update the computer-based SART.

The TTC staff, supported by personnel from NRR, NMSS and SP, started a second round of training on the new 10 CFR Part 20. Two-day sessions attended by both NRC and Agreement State personnel were conducted in three Regional and Headquarters Offices. Significant interest has been shown in the 10 CFR Part 20 training manual which was updated to integrate the approved question and answer sets. Production of a training video on the revised Part 20 was also completed. Copies of the video have been distributed to various NRC Headquarters Offices and each State.

The TTC continued to work closely with the DOE Central Training Academy (CTA) to provide training for security and safeguards personnel. Basic and Advanced Weapons Familiarization courses were conducted for NRC personnel at the CTA facility.

The Fundamentals of Inspection Course (FOIC) Work Group, co-chaired by personnel from the TTC and NRR, completed revision of the FOIC student manual and lesson plans. Course materials were extensively revised to incorporate regulatory impact issues and current policies and practices. Significantly more information on nuclear materials was provided and increased student involvement was attained through the use of case studies. The pilot course, using the updated material, was presented in Region I in June 1993. Instructor Guides for presentation of the Fundamentals of Inspection Refresher Course were distributed to the Regions and Headquarters in May 1993. The first course presentations are planned for Region II in November 1993.

Techniques courses for operator licensing examiners were provided three times to meet NRR needs. The course focused on techniques to be applied during the performance of operating and written examinations for licensed candidates.

Inspection techniques training was provided several times. Courses included the Incident Investigation Team (IIT) Training Course, Root Cause Workshops, Root Cause/Incident Investigation Workshops, and the Inspecting for Performance Course.

Replacement, multi-year contracts were put in place for both the Inspecting for Performance Course and the IIT Training Course. The IIT training contract also includes presentation of the IIT Refresher Training Course, Root Cause/Incident Investigation Workshops and the Human Performance Investigation Process (HPIP) Course.

The first presentation of the Reactor Safety Course (R-800) was conducted in February 1993. The course provided a broad perspective of important reactor safety concepts with emphasis on topics important to reactor risk. Five major areas were covered. The Historical Overview module included design for safety, defense in depth strategy, ECCS rulemaking, and severe accident and safety goal policy. The Accident Sequence module included safety risk concepts and terminology, accident sequence development, important accident sequences, and IPE and IPEEE programs. The Accident Progression in the Reactor Vessel module included fission product inventory and decay heat and core melt progression. The Accident Progression in the Containment module included containment phenomena, reactor cavity and vessel breach phenomena, and hydrogen and combustion events. The Radiological Releases and Consequences module included radionuclide groups, environmental transport, EPA protective action guidelines, and emergency response. There were also discussions on the Three Mile Island and Chernobyl events, accident management principles, and other historical perspectives.

Revision of the content and structure of Technical Managers Courses for all reactor technology areas was completed during the year, in response to changes suggested by NRC senior management. Other topics (including electrical distribution, emergency operating procedures, and shutdown risks) were incorporated, and the course length was extended from three to five days.

Technical assistance was provided by the TTC Staff in a number of diverse areas throughout the year. Technical assistance was provided to the Operator Licensing Branch of NRR for the Examination Techniques Courses. Both GE and Westinghouse Technology instructors played the role of license candidates, provided assistance in setting up the exam scenarios and validated the simulator scenarios of the examiners-in-training.

Staff of the TTC conducted an accident scenario simulation on the Westinghouse simulator at the request of NRR. The scenario consisted of a main steamline break concurrent with a large steam generator tube leak in the faulted steam generator. The purpose of the simulation was to provide information to aid NRR assessment of the consequences of a bounding case for the subject scenario using a best estimate simulation with expected operator actions. During the simulation, plant parameters were obtained using a computerized on-line Data Acquisition System developed by the TTC simulation staff.

Technical assistance was provided to NRR in evaluating the effectiveness and acceptability of the operator actions requested by the new emergency procedure guidelines (EPGs) proposed by the BWR Owners Group (BWROG). This effort included running a set of test scenarios proposed by ORNL. These scenarios were run on the BWR/6 simulator and included scenarios using both the existing EPGs/EOPs and those proposed by the BWROG. Analysis of the results was documented in a joint report co-authored by NRR, ORNL, and TTC. The recent major upgrade of this simulator's software which had added an advanced thermal-hydraulic model, a three dimensional core physics model, and high fidelity multi-node containment model made this evaluation possible.

The TTC began publishing a special training document referred to as a "Technical Issue Training Bulletin" (TITB). The concept of the TITB was identified as a method for expedited development and distribution of training on emerging technical issues. The TITBs are developed by a project team at the TTC using the most current information available to the NRC. TITB subjects are recommended to the Director of AEOD from those issues that would require prompt licensee action through generic letters or bulletins or by the Program Office Directors. The first TITB, "BWR Level Instrumentation Noncondensible Gas Release," was issued on June 25, 1993. The TITB was developed to convey the critical safety issues and the relevant engineering considerations and phenomena. The targeted NRC technical staff for this TITB included BWR resident inspectors, engineering support inspectors, and BWR operator licensing examiners in the Regions, headquarters operations officers and BWR instructors in AEOD, BWR operator licensing examiners, project managers, engineering support inspectors, and technical reviewers in NRR and the supervisors of these staff members.

One meeting of the Training Advisory Group (TAG) was conducted during the year. The TAG is a group of agency managers who provide field and program office feedback and advice on a variety of issues important to agency technical training programs. The meeting included discussions of strategic planning issues associated with future technical training needs, emerging technical issues, refresher training, simulation program updates, training contract updates, and health physics curriculum updates. It resulted in development and implementation of a comprehensive, agency-wide, technical training needs survey to take into account the staffing, hiring, attrition, and program changes anticipated within NRC for at least the next two years. Significant changes to the NRC technical training program are anticipated to address the changing needs within the agency.

## INCIDENT RESPONSE

**Events Analysis.** The NRC maintains a 24 hour-a-day, 365 day-a-year Operations Center in Bethesda, Md. The Operations Center provides a focal point for NRC communications with licensees, State agencies, and other Federal agencies about significant events. The center receives notifications each year from licensees primarily nuclear power plant operators pursuant to the immediate notification requirements contained in the Code of Federal Regulations. Typically, only a small subset of these notifications are considered by the licensee to meet the criteria of an emergency classification.

An "Unusual Event," the lowest emergency level, involves an off-normal condition that is of no immediate threat to public health and safety, but requires licensees to notify appropriate state and local agencies. In fiscal year 1993, there were 97 Unusual Events declared at commercial power reactors and four Unusual Events declared at fuel facilities. The next higher level of emergency, an "Alert," is declared in the event of an actual or potential degradation of plant safety. There were six Alerts declared in fiscal year 1993.

A "Site Area Emergency," the second highest emergency level, involves the actual or likely failures of plant functions needed for protection of the public. There were two Site Area Emergencies declared in fiscal year 1993, one at the Three Mile Island Unit 1 nuclear plant and one at the Sequoyah Fuels facility. The Three Mile Island Site Area Emergency was declared when an intruder drove his station wagon through a protected area fence and crashed into a turbine building roll-up door. The intruder was later found unarmed in a nonvital area of the plant. The operators of the Sequoyah Fuels uranium hexafluoride conversion facility declared a Site Area Emergency when a chemical reaction caused by a valve failure resulted in the (non-radiological) release of hazardous nitric oxide gases.

A "General Emergency," the highest level of emergency, involves actual or imminent core degradation with potential for loss of containment integrity. There have been no General Emergencies declared since the NRC developed its emergency classification scheme after the 1979 Three Mile Island accident.

The staff at the Operations Center evaluates telephone notifications as they are received and, depending on the safety significance of the event, notifies appropriate NRC personnel and other Federal agencies. In all cases, the NRC Regional Office in the area from which the facility is reporting the event is notified. Response to an event may vary from the simple documentation of the circumstances of the event for later evaluation to an immediate activation of response organizations within Headquarters and in the affected NRC Region.

Upon activation, these response organizations evaluate and monitor the event, to ensure that appropriate actions are being taken to protect the health and safety of the public. The NRC recognizes that, at this stage, the agency's role is secondary to that of the licensee and off-site authorities, whose immediate responses are defined in their emergency plans. Each event reported to the Operations Center by licensees is analyzed to determine whether it has generic implications for other nuclear facilities. Event reports are screened for this purpose on the first working day after receipt. Follow-up of plant-specific events is carried out by the appropriate Region.

When an event exhibits significant systems interaction, or otherwise raises questions as to plant safety, an Augmented Inspection Team (AIT) or an Incident Investigation Team (IIT) may be formed. (See discussion under "Incident Investigation Program," earlier in this chapter.) Events that may be significant from a generic standpoint receive an in-depth evaluation and, when appropriate, the NRC issues a generic communication, such as an Information Notice or Bulletin, to all potentially affected licensees or construction permit holders.

**International Nuclear Event Scale.** The International Nuclear Event Scale (INES) is a ranking system that is used to promptly and consistently communicate to the public the safety significance of reported events at nuclear installations worldwide. It was designed by an international group of experts convened jointly by the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA). The international scale is currently in use throughout the world.

In January 1993, NRC became a limited participant in the INES program. Only events classified at the Alert level or higher, according to the U.S. emergency classification system, are reported within the INES. And only events at commercial nuclear power facilities are considered for INES reporting. Reporting under the INES is only made after careful consideration of the facts and circumstances surrounding the event. This helps to avoid confusion with the existing four-level emergency response scale used in the United States. The NRC issued a Generic Letter to nuclear power reactor licensees on December 31, 1992, explaining its policy on limited participation in the INES program.

In fiscal year 1993, the NRC submitted INES reports for seven power reactor events. A summary of these events along with their INES rating is provided in the table.

The NRC also reviewed INES reports received from the other participating countries. Of particular note in fiscal 1993 were station blackout (loss of all on-site and off-site power) events at a pressurized heavy water reactor in India and at a pressurized water reactor in Russia.

**Operations Center.** A prompt incident response capability entails continuous staffing by well trained specialists with the appropriate resources to receive, assess, and communicate information swiftly and reliably with other involved parties.

The NRC entered the "Standby" response mode once during the year, when Palo Verde Unit 2 (Ariz.) declared an Alert, because of a steam generator tube leak. The Operations Center facilities were also employed to monitor several other events, including an internal plant flood at the Perry (Ohio) plant caused by a plant service water pipe break, an external flood at Cooper (Neb.), a loss of off-site power at LaSalle (Ill.), and a number of other concerns resulting from extreme weather conditions (e.g., Hurricane Emily, the Atlantic Coast snowstorm in March).

During fiscal year 1993, the Operations Center participated in five full participation exercises. Exercises dealt with various accident scenarios to confirm and maintain the capabilities of the agency's response personnel and were concerned with incidents at fuel cycle facilities as well as power reactors. The facilities for which exercises were conducted were Maine Yankee, Susquehanna (Pa.), Fort Calhoun (Neb. and Iowa), Babcock & Wilcox Lynchburg Fuel Facility (Va.), and Comanche Peak (Tex.). Computer generated "Nuclear Plant Analyzer" accident simulations were also conducted in several Regional Offices.

Telecommunication capabilities of the Operations Center were regularly used by NRC management for teleconference discussions concerning incidents of widespread technical and media interest, and events of potential significance, which, as they transpired, did not prove sufficiently serious to warrant staffing of the Operations Center.

## Table 3. FY 1993 INES Reports

| *Plant* | *Event Description* | *Date* | *Rating** |
|---|---|---|---|
| TMI 1 | Unauthorized Intruder into Plant Protected Area | | Out of Scale |
| PVNGS 2 | Steam Generator Tube Rupture | 3/14/93 | 1 |
| Zion 2 | Loss of NSSS Control Room Annunciators | 3/15/93 | 0 |
| Perry 1 | Plant Service Water Pipe Break | 3/26/93 | Out of Scale |
| North Anna 2 | Feedwater System Water Hammer | /24/93 | 0 |
| Robinson 2 | Minor Fire on EDG Lagging | 8/16/93 | Out of Scale |
| LaSalle 1 | Loss of Off-Site Power | 9/14/93 | 0 |

* Events are classified on the scale at seven levels. The lower levels (1-3) are termed incidents, and the upper levels (4-7) accidents. Events which have no safety significance are classified as below scale/level 0 and are termed deviations. Events which have no safety relevance are termed "out of scale".

Throughout the year, representatives of other Federal agencies, industry, State and local governments, and foreign countries were given tours of the Operations Center and detailed descriptions of the NRC response role and of typical activity within the Operations Center.

**New Operations Center.** The new NRC Operations Center, under construction at TWFN, is expected to be operational by July 1994. The new Center will have a totally integrated Information Management System for the collection, processing, dissemination, storage and display of information needed during both normal and emergency response operations. State-of-the-art equipment will be utilized for most of the Center's supporting systems.

**Regional Response Capability.** Each Regional Office maintains its own incident response capabilities and its own Incident Response Center to support agency response when the NRC enters "Standby" response mode. A Regional Base Team and a Regional Site Team are assembled for significant events. Both Headquarters and the Region monitor licensee action until a decision is made whether to dispatch a team to the site. An Initial Site Team of 14–23 specialists, can usually be at the site within eight hours from dispatch. After the Site Team has been fully briefed by licensee management and by their Headquarters counterparts, and is prepared to carry out their assignments, the NRC Chairman (or designee) can transfer the requisite responsibilities and authorities to the Regional Administrator, who would then be designated the NRC's Director of Site Operations.

In the event that extended NRC response is indicated, the initial Site Team will be augmented by personnel from Headquarters and/or other Regions. Procedures in this area allow coordination at the major response facilities identified in the Federal Radiological Emergency Response Plan (FRERP) and the Federal Response Plan (FRP).

Each Region has prepared its own supplement to the NRC Incident Response Plan, with specific implementation details. Regional response capabilities are assessed, and the Regions participate in several exercises each year, at least one of which includes participation by headquarters personnel. The Regions have also made major contributions to the State Outreach program (see below).

**The NRC Operations Center took part in five full-participation exercises during the report period, dealing with various accident scenarios designed to confirm and maintain the capabilities of the agency's response personnel. One of the facilities for which the exercises were conducted was the Maine Yankee nuclear power plant, located in Wiscasset, about 18 miles south of the State capital of Augusta.**

**Emergency Response Training.** During fiscal year 1993, staff response training was conducted for NRC Headquarters, each Regional Office, and other Federal and State response organizations. The training included:

- NRC Headquarters and Regional Office training on computer codes used for consequence projection.
- One course on the operation of the Federal Radiological Monitoring and Assessment Center (FRMAC) for NRC, and other Federal, State and utility response personnel.
- Seven courses on emergency response involving Headquarters, Regional Offices, EPA, DOE, FEMA, USDA and HHS. Topics included NRC response procedures and interfaces with other Federal response organizations.

**Emergency Response Technical Tool Development.** A program is ongoing to augment the assessment capabilities of the Reactor Safety team (RST) during its response to nuclear power plant emergencies. The program involves the development of an expert system, known as the Reactor Safety Assessment System (RSAS). During an event at a reactor site, RSAS will be used as an independent tool to monitor and display the status of the plant's Critical Safety Functions (those plant conditions without which core damage becomes a possibility). Assessment information derived from RSAS will be limited to use by the RST to confirm their assessment and/or identify potential inconsistencies.

Technical tool development for the protective measures response function centered around the development of a revision to the Response Technical Manual (RTM-92, NUREG/BR-0150). The change addressed accident response in the areas of:

- Projection of reactor accident consequences
- $UF_6$ Accident Assessments
- Determination of protective actions for the public
- Application of EPA/FDA guidance on re-entry and ingestion issues
- Conduct of airborne monitoring.

Work also continued on the RASCAL model, a computer code used to project radiological consequences during accidents. The present version of the code (RASCAL 2.0) has been distributed to the Regional Offices and is available to the public. RASCAL 2.1, which is currently under development, will include provisions to assess the consequences of fuel pool accidents and to project doses based on containment radiation conditions, as measured by monitors and air samples.

A Geographic Information System is under development with DOE. This system will provide demographic and other map-based response information for NRC Reactor sites and the surrounding 50-mile area.

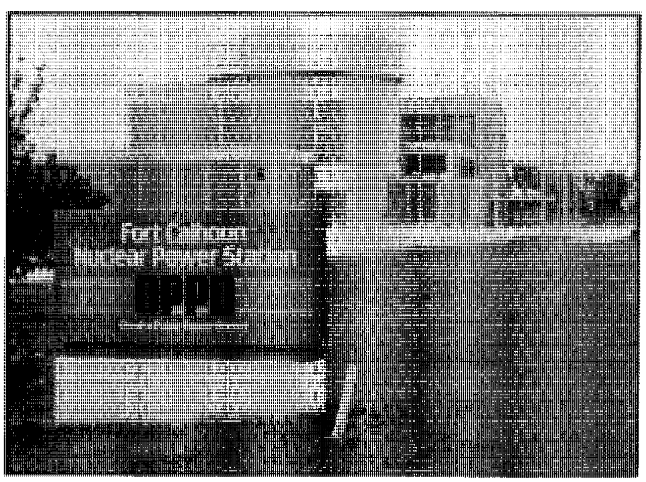
**Emergency Response Data System.** The Emergency Response Data System (ERDS) provides for licensee activated transmission to the NRC of pre-selected plant data from onsite computers during emergencies at commercial nuclear power plants. NRC's ERDS computer receives, sorts, and stores the licensee data and provides output displays to users in the Operations Center, as well as to remote users at NRC Regional Offices, the Technical Training Center, and various State emergency response facilities.

Implementation of ERDS began in 1988 under a voluntary program. As of August 13, 1991, implementation by all licensees was required by regulation. All licensees completed ERDS implementation before the February 13, 1993 deadline, with the exception of two units which were granted schedular exemptions because of planned computer system upgrades. State governments interested in receiving ERDS data during plant emergencies are required to establish Memoranda of Understanding (MOU) with the NRC. To date, MOUs have been established with the States of Alabama, Arizona, Georgia, Maryland, Massachusetts, Michigan, New Jersey, New York, North Carolina, Ohio, Pennsylvania, Tennessee, and Washington. MOUs are currently being developed with Arkansas, Connecticut, Kansas, and South Carolina.

**Coordination with Other Federal Agencies.** The NRC participated actively in the development of the Federal Response Plan (FRP). The FRP was developed by the Federal Emergency Management Agency (FEMA) as an umbrella plan for coordinating the Federal response to major emergencies and disasters. To support this, the NRC participated in the Agency Planning Leaders and Catastrophic Disaster Response Group meetings for developing implementation procedures for the FRP. The NRC participated on the FEMA- chaired Federal Radiological Preparedness Coordinating Committee (FRPCC) and six subcommittees. At the request of FEMA, the NRC also participated in the Federal Response Planning Task Force, including chairing the Operations Work Group. This task force expedited the development of detailed implementing procedures for the FRP.

The NRC co-sponsored, with DOE, a major exercise in conjunction with the regularly scheduled emergency planning exercise at Fort Calhoun. During this exercise the licensee, the States of Nebraska and Iowa, DOE, EPA, HHS, USDA, FEMA, and NRC tested their procedures for conducting and coordinating radiological monitoring and assessment in the event of a major accident. A Federal Radiological Monitoring and Assessment Center (FRMAC) was established at a National Guard Armory in Omaha. A wide range of radiological monitoring capabilities including over 20 field teams, mobile laboratories, and aircraft were deployed during this exercise. The lessons learned from this exercise were incorporated in the DOE FRMAC program and tested in a DOE exercise conducted several months later. NRC personnel also participated in this DOE exercise as FRMAC technical staff. Subsequently, NRC personnel have been permanently incorporated into the FRMAC response staff and may be called on in the event of an accident to support the FRMAC.

In April 1993, NRC conducted the U.S. portion of the first International Offsite Emergency Exercise (INEX1) sponsored by the Nuclear Energy Agency (NEA) of the Organization Economic Cooperation and Development (OECD). INEX1 was a tabletop exercise to address cross boundary issues such as notification, protective actions, field monitoring, customs, cleanup criteria, radiological waste, international trade, public exposure and international assistance. The NRC arranged for the participation




**The NRC and DOE co-sponsored a major exercise in conjunction with the regularly scheduled emergency planning exercise at the Fort Calhoun (Neb.) nuclear power plant. The facility, shown above, is a pressurized water reactor plant, located about 20 miles north of Omaha. A Federal Radiological Monitoring and Assessment Center (FRMAC) was set up at a National Guard Armory in Omaha for the exercise. Officials from Nebraska and Iowa, as well as a number of Federal agencies, took part. A wide range of radiological monitoring capabilities, including over 20 field teams, mobile laboratories, and aircraft were deployed during the FRMAC exercise. At right is a FRMAC monitoring team taking field measurements to determine the type and amount of radioactive material deposited on the ground. The helicopter in the background is used to conduct aerial radiation monitoring of the area, in the hypothetical scenario.**

of other Federal agencies (DOE, EPA, HHS, USDA and FEMA), a nuclear power plant operator (Detroit Edison Company), State organizations (Michigan State Police, Health Department and Agriculture Department), a neighboring country (Canada Atomic Energy Control Board and Atomic Energy of Canada Limited) and a neighboring province (Ontario Solicitor General) as well as NRC Headquarters and Region III personnel. The NRC presented the United States lessons learned to the international community at NEA in Paris. NRC staff was requested to summarize the monitoring lessons learned from the all of the INEX1 participants.

During 1993, improvements continued to be made among Federal agencies concerning the coordination necessary during a reactor accident. These were mainly the result of:

- Sponsoring of a FRMAC courses (See Emergency Response Training Section).
- On-scene participation in exercises with regional Federal emergency responders to demonstrate the NRC's role as the lead Federal agency in a radiological emergency and to specify its expectations from supporting Federal agencies.
- Participation on FEMA's Federal Response Planning Task Force.

Finally, an NRC liaison staffed the FEMA emergency response center during the 1993 Midwest flooding. The liaison provided status on NRC licensed facilities in the flooded areas and responded to questions from other Federal representatives.

**State Outreach.** During the year, the NRC continued to implement its State Outreach program which is designed to increase and improve the NRC's interaction with States. The program emphasizes increased exercise participation frequency, attempting to exercise with each State on a three-year cycle. The NRC is also working to expand its participation in meetings, workshops, and other vehicles that help describe the available NRC assessment tools, response capabilities, and accident assessment training courses. During 1993, at the headquarters and regional levels, the NRC coordinated and conducted 12 exercises with States to demonstrate NRC interfaces and capabilities. The NRC also worked with 10 other States to explain the NRC Headquarters interfaces and capabilities during an accident, and conducted State Outreach briefings for Region I and with the States of California, Iowa, Nebraska, New Jersey, New York, Pennsylvania, and Tennessee. The NRC spoke at a meeting sponsored by the State of California, the National REP Conference, the National Emergency Management Agency Conference and FEMA State Conferences in New Jersey and Missouri.

# Office Of Investigations

The Office of Investigations (OI) conducts investigations of alleged wrongdoing by individuals or organizations other than employees of the Nuclear Regulatory Commission (NRC) or NRC contractors. (Allegations involving NRC employees or NRC contractors come under the purview of the NRC Office of the Inspector General). Thus, OI is concerned with the activities of NRC licensees, applicants for licenses, licensee contractors and vendors.

In fiscal year 1992, the process was revised by which suspected wrongdoing matters are referred to OI. This new procedure resulted in OI's becoming involved in potential wrongdoing matters at an earlier stage and increased the number of investigations opened significantly. OI opened 271 investigations and closed 216 investigations in fiscal year 1993. Of these, 26 cases were referred to the Department of Justice (DOJ) for prosecutorial review. During fiscal year 1993, OI supported two Federal grand juries, two trials in Federal courts, and one major Federal task force investigating counterfeit parts. In fiscal year 1993, OI investigations resulted in seven indictments, two convictions, and eight guilty pleas in Federal courts.

During fiscal year 1993, OI continued to focus attention on the sale of counterfeit and substandard parts, such as circuit breakers, valves, and fasteners, to utilities operating nuclear power plants. OI remained a principal participant in the interagency working group on problem parts and suppliers, as well as a subgroup composed of Federal investigative personnel. Major convictions in this area resulted from investigations conducted jointly with other Federal agencies.

## Department of Justice/Office of Enforcement Actions

Following are examples of significant OI investigations on which subsequent action was also taken by the U.S. Department of Justice or the NRC Office of Enforcement, during fiscal year 1993.

*CASES INVOLVING THE JUSTICE DEPARTMENT*

On June 21, 1993, Gordon Finlay, President of Finlay Testing Laboratories, was convicted on nine counts and his company on 19 counts of conspiracy to defraud the United States, making false statements, and falsifying documents regarding the transport of radioactive materials. The conviction was a result of an extensive OI investigation culminating in a four-week criminal trial in U.S. District Court in Hawaii. Sentencing is scheduled for February 7, 1994.

On August 30, 1993, Timothy Carroll, former Manager and Radiation Safety Officer for Finlay Testing Laboratories, was sentenced to two terms of 5-years probation, to run concurrently, and a $5,000 fine. Carroll cooperated with OI and testified against Finlay during the June 1993 trial.

As a result of a joint investigation by OI and the Defense Criminal Investigative Service, on September 10, 1993, Ricardo Contreras, owner of Molded Case Circuit Breakers, Inc., pled guilty in U.S. District Court, Boise, Idaho, to one count of fraud involving the sale of counterfeit circuit breakers. On the same day Contreras was sentenced to 60-days confinement, 3-years probation, and a $3,000 fine.

On November 23, 1992, Travis Miller, former president of Stanford Mining Company (SMC), was sentenced in Federal court to 3-years probation, and fined $3,600. This sentencing was the result of an OI investigation into the providing of false information to the NRC concerning the improper transferring and disposing of three nuclear weigh scales containing cesium-137, and the making of false statements to the NRC concerning the whereabouts of the scales. OI:Region I was able to recover two of the three missing scales. The corporation (SMC) had been previously fined $30,000 on similar charges.

An extensive investigation of the American Inspection Company, Inc. (AMSPEC), a radiography company charged with falsifying NRC-required training records and with violations of other regulatory requirements, has resulted in the indictment of five officers of the company. Richard Odegard, a vice president, was indicted, pled guilty, and was sentenced to 120 days confinement, a $3,000 fine, and 1-year supervised probation. Paul Bowman, a former vice president, was indicted, pled guilty, and was sentenced to 30 days confinement, a $3,000 fine, 125 hours of community service, and 1-year supervised probation. Steven Oliver, the project manager, St. Croix, was indicted, pled guilty, and was sentenced to pay a fine of $2,500 and 2-years probation. The probation was suspended. Larry Ladner was indicted and pled guilty. He will be sentenced at a later date. Daniel McCool, the president of AMSPEC, has been indicted. Another company official is also pending indictment. Further action against these two individuals is pending.

On March 5, 1993, Joseph Satin, CEO of Satin American Corporation, and the company's vice president, Daniel Casotti, each pled guilty in U.S. District Court in Hartford, Conn., to a criminal conspiracy charge for their roles in providing re-manufactured circuit breakers and other electrical components bearing counterfeit General Electric, Westinghouse, and other manufacturers' nameplates to various nuclear power plants throughout the United States. On May 5, 1993, Satin was sentenced to 3-years probation, fined $250,000, and prohibited from engaging in safety-related business with the nuclear industry for a period of five years. Casotti was also sentenced to three-years probation, fined $5,000, and prohibited from engaging in safety-related business with the nuclear industry for a period of three years. Full restitution to the affected companies was also required.

As a result of an OI investigation, on March 16, 1993, Jack D. Smith, a partner of Coffeyville Valve, Inc., pled guilty to one count of conspiracy to traffic in counterfeit goods. The investigation determined that Smith had ordered counterfeit nameplates which were affixed to two refurbished used valves, that were represented as new, and subsequently installed at Indian Point nuclear power plant in Buchanan, N.Y. On May 18, 1993, Smith was sentenced to 3-years probation and fined $15,000.

OI has completed an investigation at Houston Lighting & Power Company, South Texas Project, to determine whether a former contract employee at the South Texas Project was fired as a result of his having reported safety concerns to the NRC. A Federal grand jury has been convened in Houston, Tex., to investigate possible criminal wrongdoing in this matter. Testimony was heard by the grand jury in August 1993, and additional testimony is scheduled before the grand jury in December 1993.

*CASES INVOLVING OFFICE OF ENFORCEMENT*

An OI investigation involving Northeast Nuclear Energy Company and its Millstone Unit 3 (Conn.) nuclear power plant disclosed that utility officials, up to and including the former senior vice president-nuclear, had engaged in the harassment and intimidation of a supervisory engineer over the raising of safety concerns involving Rosemount transmitters. Based on the OI investigation, on May 4, 1993, the NRC issued a $100,000 civil penalty.

An OI investigation involving a medical doctor who was licensed by the NRC established that the doctor willfully administered excessive radiopharmaceutical doses to patients. The NRC issued a $3,800 civil penalty to the doctor.

An OI investigation determined that South Dakota Department of Transportation (DOT) employees deliberately provided false information to the South Dakota DOT Radiation Safety Officer regarding the alleged theft of a moisture density gauge. These employees knew that this false and misleading information was subsequently submitted to the NRC, but took no action to correct known inaccuracies in the submitted report. Based on the OI investigation, on December 22, 1992, the NRC issued a Notice of Violation and imposed a civil penalty of $3,400 for violations of NRC requirements.

An OI investigation disclosed that Gray Wireline Services deliberately conducted radiography in NRC jurisdiction under reciprocity without paying the required fee. The investigation also disclosed that the President of Gray Wireline intentionally lied to an NRC inspector who questioned him about Gray Wireline's activities within

NRC jurisdiction. Based on the OI investigation, on June 9, 1993, the NRC issued a Notice of Violation and imposed a civil penalty of $1,500 for violations of NRC requirements.

An OI investigation concluded that a Tulsa Gamma Ray, Inc., radiographer lost control of a radiography camera when it fell off their truck. The camera was subsequently recovered by a member of the general public. Tulsa Gamma Ray failed to report the camera's loss to the NRC. As a result of the OI investigation, on July 28, 1993, the NRC issued a Notice of Violation and imposed a civil penalty of $5,000 for violations of NRC requirements.

An OI investigation determined that Southwest X-Ray Corporation personnel failed to use ratemeters while performing radiography within NRC jurisdiction. On April 9, 1993, as a result of the OI investigation, the NRC issued a Notice of Violation and imposed a civil penalty of $2,500 for violations of NRC requirements.

An OI investigation found that N. V. Enterprises, Inc., personnel deliberately failed to use ratemeters while performing radiography within NRC jurisdiction. Based on the OI investigation, on May 7, 1993, the NRC issued a Notice of Violation and imposed a civil penalty of $4,000. The licensee, in lieu of paying the civil penalty, has requested that the license be terminated.

An OI investigation determined that Edwards Pipeline, Inc., deliberately failed to conduct required quarterly inspections of its radiographers. This was a recurring violation. During the conduct of the OI investigation, the licensee admitted that he was aware of the requirement, but because of "logistics, time, and money" was unable to comply with it. Based on the OI investigation, on September 1, 1993, the NRC issued a Notice of Violation and imposed a civil penalty of $12,000. The licensee has requested mitigation of the civil penalty. That request is currently under review by the NRC Office of Enforcement.

As a result of an OI investigation which concluded that licensee officials had provided inaccurate information regarding failure rates for emergency lighting at Palo Verde (Ariz.) nuclear power plant, a Notice of Violation was issued, on April 5, 1993, to the Arizona Public Service Company.

On April 16, 1993, a Notice of Violation was issued to Portland General Electric as a result of an OI investigation which determined that security guards were not reporting guards who were sleeping on duty because of intimidation by security managers.

On August 6, 1993, a Notice of Violation was issued to Arizona Public Service Company (APS) as a result of an OI investigation which found that the APS security training program was inadequate and that required training records had been falsified.

# Office Of Enforcement

The NRC Office of Enforcement is responsible for managing the Commission's enforcement program. The office is subject to oversight by the Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations, and Research for enforcement actions related to reactor licensees, and by the Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support for enforcement actions involving any other licensees.

Appendix 6 provides a listing and brief summary of the civil penalties proposed, imposed, and/or paid during fiscal year 1993; and a listing and brief summary of the nine orders issued during fiscal year 1993. Recognizing that enforcement actions can sometimes span several fiscal years, there were a total of 134 civil penalties acted upon in fiscal year 1993. Of these, 120 cases were proposed for a total of $4,115,900; 20 were imposed for a total of $741,925; and 111 were paid (including the total amount for those civil penalties being paid over time) for a total of $4,187,150. Fifty-four cases were issued as escalated enforcement actions without a civil penalty for reasons unique to each case.

## NRC ENFORCEMENT PROGRAM

The NRC's Enforcement Program seeks to protect the public health and safety by ensuring compliance with the Atomic Energy Act, the Energy Reorganization Act, NRC regulations, and license conditions; obtaining prompt correction of violations and conditions adverse to quality; deterring future violations; and encouraging improvement of licensee performance. Violations are identified through inspections and investigations. All violations are subject to civil enforcement action and may also be subject to criminal prosecution. After an apparent violation is identified, it is assessed in accordance with the NRC Enforcement Policy. This policy has been approved by the Commission and is published as Appendix C to 10 CFR Part 2.

There are three primary enforcement sanctions available: Notices of Violations, civil penalties, and orders. A Notice of Violation (NOV) summarizes the results of an inspection and formalizes a violation. A civil penalty is a monetary fine issued under authority of Section 234 of the Atomic Energy Act. That section provides for penalties of up to $100,000-per-violation per-day. NOVs and civil penalties are issued based on violations. Orders may be issued for violations, or in the absence of a violation, because of a public health or safety issue.

The Commission's order issuing authority is broad and extends to any area of licensed activity that affects the public health and safety. Orders may modify, suspend, or revoke licenses. Orders may also be issued to individuals who are not themselves licensed if they violate the regulations concerning deliberate misconduct.

The first step in the enforcement process is assessing the severity level of the violation. Severity levels range from Severity Level I for the most significant violations to Severity Level V for those of minor concern. Severity levels may be increased for cases involving a group of violations with the same root cause, repetitive violations, or willful violations.

Enforcement conferences are normally held for violations assessed at Severity Levels I, II, or III, and may be held for violations assessed at Severity Level IV if increased management attention is warranted (e.g., repetitive violations). An enforcement conference is a meeting between the NRC and the licensee to: (1) discuss the apparent violations, their significance, the reason for their occurrence, including the apparent root cause, and the licensee's corrective actions; (2) determine whether there were any aggravating or mitigating circumstances; and (3) obtain other information that will help the NRC determine the appropriate enforcement action. The decision to hold an enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. In fiscal year 1993, the NRC conducted 206 enforcement conferences.

On June 23, 1992, the Commission approved implementation of a two-year trial program to allow certain enforcement conferences to be open for public observation. This is being done so that members of the public can have the opportunity to gain a full understanding of the agency's regulatory process. In fiscal year 1993, 39 conferences were open to the public.

Civil penalties are normally issued for Severity Level III or higher violations, absent any mitigating factors, and may be issued for violations at Severity Level IV if the violations are repetitive or similar to previous Severity Level IV violations. Civil penalties are normally issued for any willful violation.

The NRC imposes different levels of civil penalties based on a combination of the type of licensed activity, the type of licensee, the severity level of the violation, and certain escalation and mitigation factors. These factors are: (1) who identified the violation, (2) was the corrective action prompt and extensive or untimely and only marginally acceptable, (3) was the violation a reflection of prior licensee performance, (4) did the licensee have prior opportunity to identify the violation, (5) were there multiple occurrences of the violation, and (6) how long did the violation or its impact endure.

If a civil penalty is to be proposed, a written Notice of Violation and Proposed Imposition of Civil Penalty is issued and the licensee has 30 days to respond in writing, by either paying the penalty or contesting it. The NRC considers the response and, if the penalty is contested, may either mitigate the penalty or impose it by order. If the civil penalty is to be imposed by order, the order is published in the Federal Register. Thereafter, the licensee may pay the civil penalty or request a hearing.

Besides their use in imposing civil penalties, orders may be used to modify, suspend, or revoke licenses. Orders that modify a license may require additional corrective actions, such as removing specified individuals from licensed activities or requiring additional controls or outside audits. The NRC issues a press release with a proposed civil penalty or order.

# Nuclear Materials Regulation

Chapter

4

The Nuclear Regulatory Commission (NRC) Office of Nuclear Material Safety and Safeguards (NMSS) and the NRC's five Regional Offices administer the regulation of nuclear materials, as distinct from nuclear reactor facilities (covered in Chapters 2 and 3). The NRC conducts materials regulation under several broad programs: material safety (including the storage and transport of nuclear fuel), discussed in this chapter; fuel facility safety and safeguards, discussed in Chapter 5; and waste management activities, discussed in Chapter 6.

Activities covered in this chapter include licensing, certification, inspection, and other regulatory actions concerned with: (1) storage of spent reactor fuel; (2) transportation issues associated with the fuel, and (3) production and use of reactorproduced radioisotopes (byproduct material).

Nuclear materials regulation during fiscal year 1993 comprised:

- Over 5,000 licensing actions on applications for new byproduct materials licenses, amendments, and renewals of existing licenses and reviews of sealed sources and devices.
- Approximately 2,400 materials licensee inspections.
- Over 100 fuel storage and transportation package reviews and 15 route approvals for transporting special nuclear material and spent fuel.
- 11 inspections of supplier quality assurance (QA) programs.

## FUEL STORAGE

### Interim Spent Fuel Storage

Under the Nuclear Waste Policy Act of 1982, utilities are responsible for interim storage of their spent fuel until a Federal repository or monitored retrievable storage (MRS) is available. Utilities are continuing to develop plans for increasing storage capacity as they approach the limits of their storage pools. Where possible, utilities rerack spent fuel pools, a measure that has extended storage capacity for most reactors. On-site dry storage of spent fuel, in casks or concrete vaults, is also employed by an increasing number of utilities to meet storage needs.

In 1993, the NRC completed a rulemaking to amend 10 CFR Part 72 in the regulations, adding two storage cask models to the list of approved casks—the TN-24, designed by Transnuclear, Inc., and the VSC-24, designed by Sierra Nuclear Corporation. These additional models make a total of six cask models that any utility can use at its reactor site without a specific license for that site. The reactor licensee must ensure that there are no unreviewed safety questions, nor changes needed, before using the casks. The licensee also must comply with the conditions of a Certificate of Compliance related to a given cask and must develop operating procedures for the use of the cask. Consumers Power Company became the first utility to store spent fuel under the general license when, following approval of the VSC-24 cask, it began using it for storage at its Palisades (Mich.) nuclear plant.

In November 1992, NRC staff completed its technical review and issued a storage license to the Baltimore Gas and Electric Company for an independent spent fuel storage installation (ISFSI) at the Calvert Cliffs facility (Md.). As of the close of fiscal year 1993, the licensee expected to begin loading spent fuel in the ISFSI by the end of calendar year 1993.

During fiscal year 1993, NRC staff reviewed an application from Northern States Power (NSP), which led to issuance of Materials License SNM-2506 in October 1993. The license authorized receipt and storage of spent fuel in an ISFSI located at its Prairie Island nuclear power plant site in Goodhue County, Minn. The ISFSI is to provide interim storage for up to 1,920 fuel assemblies in 48 TN-40 casks built by Transnuclear, Inc. However, NSP stopped construction of the casks after the Minnesota Court of Appeals ruled that the Minnesota legislature must vote on the matter before use of an ISFSI. As the year ended, NSP planned to load its first cask in the middle of 1994, provided that the State legislature voted to approve it.

The NRC inspected the quality assurance records of Pacific Nuclear Fuel Services (PNFS) with respect to its Nutech Horizontal Modular System (NUHOMS) spent fuel storage design. To inspect the records, inspectors visited three separate sites—(1) the place where the casks were manufactured; (2) PNFS headquarters, and (3) the

Oconee (S.C.) nuclear power plant, where the casks are to be used.

## Monitored Retrievable Storage

The NRC has provided comments to the Department of Energy (DOE) on two revisions to an annotated outline for the Safety Analysis Report for DOE's Monitored Retrievable Storage (MRS) facility; the outline will be the basis for the MRS license application. DOE subsequently informed the NRC that it was suspending work on the MRS annotated outline until a suitable site for the facility was proposed. Meanwhile, nine Indian tribes have written to the Nuclear Waste Negotiator expressing their readiness to enter formal negotiations leading to an agreement for siting an MRS.

The NRC met with the DOE to discuss plans and schedules for DOE's development of a multi-purpose canister for the storage, transportation, and disposal of the nation's nuclear reactor fuel.

# TRANSPORTATION OF RADIOACTIVE MATERIALS

The Federal Government regulates safety in the transportation of radioactive materials primarily through the NRC and the Department of Transportation (DOT). The regulatory responsibilities of the two agencies in this area are delineated in a Memorandum of Understanding (MOU). For international shipments, the DOT is the designated United States Competent Authority and is responsible for implementing International Atomic Energy Agency (IAEA) standards. The NRC advises the DOT on technical matters.

## Status of Transportation Activities in 1993

**Plutonium Air Shipment Criteria Development.** Section 5062 of Public Law 100-203 imposes requirements regarding the air transport packages used to ship plutonium from one foreign country to another through U.S. air space. The law requires that the NRC certify the safety of plutonium air transport package designs to the Congress. During fiscal year 1993, the NRC completed feasibility studies related to the testing of such packages; the studies and testing had been requested and funded by the Power Reactor and Nuclear Fuel Development Corporation (PNC), on behalf of the Japanese Government. A final technical report was provided to PNC in December 1992, and a final cost accounting report was provided in September 1993. These actions complete all activities under the Agreement with PNC. Contract support for this effort was provided by Lawrence Livermore National Laboratory.

**Quality Assurance Inspection Activities.** The NMSS continued its inspection activities in fiscal year 1993 to ensure that transportation packaging and dry spent fuel storage systems certified by the NRC are designed, fabricated and tested in accordance with an NRC-approved QA program. This inspection program was initiated in 1989. Inspections were conducted at eight transportation packaging and three dry spent fuel storage system suppliers. These inspections were selected to represent a broad spectrum of the industry. The supplier inspection program encompasses designers, fabricators, and distributors taking part in NRC-approved QA programs, Certificates of Compliance for transportation packaging, and Special Nuclear Material licenses for dry spent fuel storage systems. The inspection program is structured to provide information as to whether transportation packaging and dry spent fuel storage systems are fabricated, procured, and maintained in compliance with 10 CFR Part 71 and Part 72 requirements, respectively.

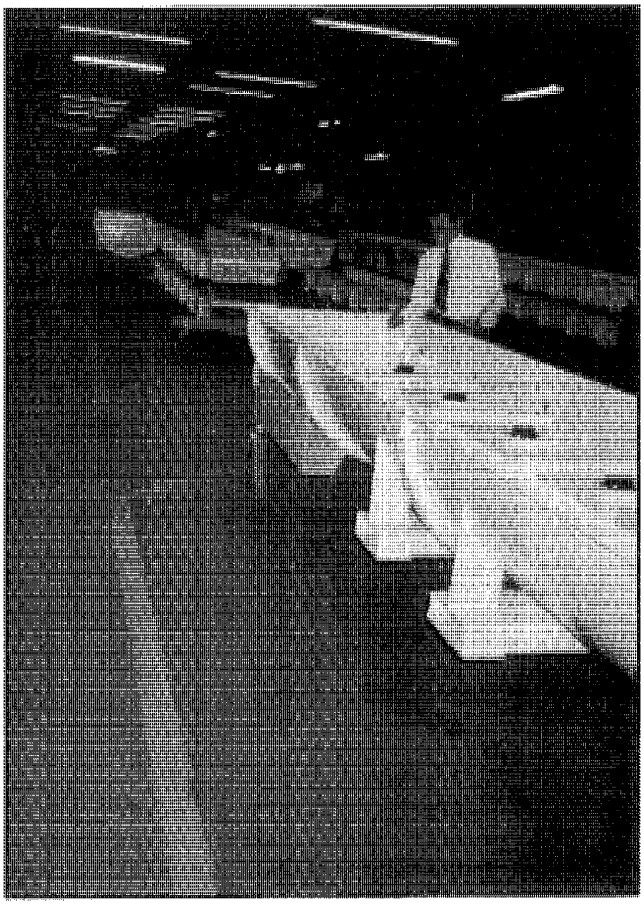
# MATERIALS LICENSING AND INSPECTION

The NRC currently administers approximately 6,850 licenses for the possession and use of nuclear materials in medical and industrial applications. This total represents a reduction of about 350 licenses in the past year. Table 1 shows the distribution of licenses by Region. The 29 Agreement States administer about 15,000 licenses.

The program is designed to ensure that activities involving these uses of radionuclides do not endanger the public health and safety. NRC regional staff completed approximately 2,400 inspections of materials facilities in fiscal year 1993. The NRC Regional Offices administer almost all materials licensees, with the exception of "exempt distribution" licenses, sealed source and device design reviews, and licenses for companies that extract other metals from ores and slags containing uranium and thorium. The excepted licenses are handled by NRC Headquarters.

The NRC completed 5,043 licensing actions during the fiscal year. Of this total, 366 were new licenses, 3,217 were amendments, 1,088 were license renewals, and 372 were sealed source and device reviews.

**Human Factors.** Human error associated with the production and non-reactor use of byproduct material (e.g., medical and industrial use) is a significant contributor to incidents that result in unnecessary or excessive public and occupational exposures. Successful reduction of human error begins with an in-depth knowledge of its causes. Human factors evaluations designed to acquire



The NRC conducts quality assurance inspections of the fabrication of transportation packaging and dry spent fuel storage systems. In the photo above, NRC inspector Thomas Matula examines fresh fuel shipping packaging, newly fabricated for the Babcock & Wilcox Company by the Erie Engineered Products in Tonawanda, N.Y.

such knowledge with respect to applications in teletherapy and brachytherapy using remote afterloaders continued during 1993, and contractors for the projects have completed data collection and analysis. Human factors problems—tasks with a high potential for human error that can adversely affect system performance, along with the factors that can contribute to those errors—have been identified and assigned priorities. Alternative means for resolving those problems have also been identified and evaluated. Final reports summarizing critical tasks and identifying and evaluating alternatives for resolving problems associated with those tasks are expected to be ready for publication in early 1994.

Human error in the use of medical devices, including devices using nuclear byproduct material, can be reduced by means of improved human factors engineering guidance provided to designers. As a member of the Human Engineering Committee of the Association for the Advancement of Medical Instrumentation (AAMI), an NRC human factors analyst continues to participate in revision of the document, "Human Factors Engineering Guidelines and Preferred Practices for the Design of Medical Devices." The document received approval by AAMI's Standards Committee in late summer of 1993, and approval as an American National Standard by the American National Standards Institute in the fall of 1993. The revised guidelines will be published in early 1994.

An NRC project to evaluate information in reports of nuclear medicine misadministrations continued during 1993. A key element of the project is a computerized data base. The data base now contains information on misadministrations that occurred in 1989 and 1990. A chapter entitled *Radiopharmaceutical Misadministrations: What's Wrong?*—summarizing some of the information in the data base—was prepared for inclusion in a book entitled *Human Error in Medicine*. The book is being prepared for publication in 1994 by Lawrence Erlbaum Associates, Inc.

The NRC took part in an Intra-Governmental Workshop on Human Error, sponsored by the U.S. Food and Drug Administration, which focused on ways in which human error in medicine might be addressed. Emphasis was given to identifying the root causes of human error in medicine and proposing means for reducing such error.

Human error in fuel cycle facilities can lead to inadvertent criticalities, personnel contamination, and off-site radiation releases. A draft human factors section to be included in the Standard Review Plan for Fuel Cycle Facilities was prepared during the report period. The draft section describes a program for evaluating fuel cycle facilities for human factors problems and for addressing those problems.

**Regulatory Impact Survey.** In May 1992, the staff submitted a plan to the Commission to conduct a regulatory impact survey of fuel facility and materials licensees (SECY-92-166). The plan proposed a three-phase approach to determine the impact of the NRC's regulatory program on these licensees. The survey would seek to determine if there is an appropriate balance between the burden imposed on licensees by NRC requirements and the level of safety actually achieved. Phase I included a pilot series of nine on-site interviews at selected fuel cycle and major materials facilities, and these were completed between August and October 1992.

The staff submitted a report to the Commission on May 13, 1993 (SECY-93-130), recommending a number of changes in staff practices and a plan for obtaining a broader range of licensee views. The Commission instructed the staff to present a plan for obtaining additional information from licensees and for evaluating and incorporating it into the regulatory program. The Commission

## Table 1. Distribution of NRC Nuclear Materials Licenses

(as of October 1, 1993)

| | |
|---|---|
| Region I | 2,388 |
| Region II | 893 |
| Region III | 2,410 |
| Region IV | 706 |
| Region V | 240 |
| Headquarters | 216 |
| Total: | 6,853 |

approved the plan recommended by the staff in SECY-93-268. As fiscal year 1993 ended, the staff was making plans to survey several hundred licensees through mail questionnaires. The staff expected to report its final conclusions and its recommendations to the Commission by September 1994.

## Industrial Uses

**Industrial Radiography.** As described in previous NRC Annual Reports (see *1989 Annual Report*, pp. 74–5; *1990 Annual Report*, p. 81; *1991 Annual Report*, p. 95, and *1992 Annual Report*, pp. 102–3), NRC staff has been involved for some time with an initiative to develop a certification program for industrial radiographers. During the report period, the NRC staff continued its support of the American Society for Nondestructive Testing (ASNT) in implementing ASNT's "Industrial Radiography Radiation Safety Personnel" (IRRSP) certification program. The staff also worked toward developing a proposed rule that would mandate radiographer certification. The proposed rule has now been combined with another 10 CFR Part 34-related rulemaking effort, seeking an overall revision of 10 CFR Part 34 and proposing several new requirements to be added to the regulation. The NRC staff anticipates publishing a combined proposed rule in early 1994.

**Irradiator Rule.** On February 9, 1993, the NRC staff published, in final form, a new part to NRC regulations, designated 10 CFR Part 36, which became effective on July 1, 1993. The new part specifies radiation safety and licensing requirements for the use of high-activity sources in irradiators. Irradiators use gamma radiation (usually from cobalt-60) to irradiate various products, changing their condition or characteristics in some way. More than 90 percent of irradiator capacity in the United States is used for the sterilization of disposable medical supplies (e.g., disposable syringes and gloves) and packaging materials. The staff is preparing a licensing guide to assist those preparing applications for irradiator licenses and expected to publish the draft guide for comment by late 1993. For more information on irradiators, see the *1990 NRC Annual Report*, pp. 82-83, the *1991 NRC Annual Report*, p. 95, and the *1992 NRC Annual Report*, p. 103.

**Petition by Indian Orchard Citizens Council.** On May 7, 1993, the Director of NMSS issued a decision on a petition, dated June 28, 1992, which had been submitted by Indian Orchard Citizens Council, regarding Interstate Nuclear Services, Inc.'s (INS) Indian Orchard, Mass., facility. INS is authorized by the NRC to conduct commercial nuclear laundry operations at this facility. The petition requested NRC response or action on 10 matters or requests and made four demands with respect to INS's activities.

In the decision, petitioners' requests and demands were granted in part and denied in part. The petition was granted with respect to eight of the 10 matters or requests, and the petition was denied with respect to the remaining two matters or requests. The petition was denied with respect to three of the demands, and the fourth demand was rendered moot by the voluntary action of the licensee. The reasons underlying the decision are set forth in a document entitled, "Director's Decision under 10 CFR § 2.206" (DD-93-09), which is available for public inspection in the Commission's Public Document Room located at 2120 L Street, N.W., Washington, D.C. 20555.

**Source/Device Registration.** Manufacturers and distributors of radiation sources and devices containing radiation sources are required to submit safety information about their products and information about their QA programs to the NRC or an Agreement State. The NRC or the Agreement State evaluates the information to ensure that each product is adequately designed to protect the public health and safety and meets all applicable radiation safety requirements, and that the company's QA program is adequate to ensure that the product meets the design specifications. The regulatory agency then issues a Certificate of Registration to the vendor. The certificate is used by the NRC or the Agreement State in its issuance of specific licenses to users of the products.

The NRC maintains a nation-wide registry of sealed source and device designs, including those registered by the NRC and the Agreement States, and also sources and devices listed in the radioactive materials reference manual of the U.S. Food and Drug Administration's Center for Devices and Radiological Health. These sources and devices contain naturally-occurring or accelerator-produced radioactive material. The NRC maintains copies of the registrations and a computerized registry that includes summary information about the sources or devices.

**Sealed Sources Exceeding 10 Part 61, Class C.** Licensees with certain sealed sources are experiencing problems disposing of the sources when they are no longer needed. Certain well-logging sources, gauges, irradiators, and teletherapy sources are not accepted for disposal at commercial burial sites because, when packaged for disposal, concentrations of radioactivity exceed the limits for Class C low-level waste, as set forth in 10 CFR Part 61.

Under Federal law, ultimate disposal of these wastes is the responsibility of the Department of Energy (DOE), and licensees must pay the full cost for disposal. The DOE is in the process of establishing a disposal facility for these kinds of wastes, but the facility may not be available for many years. Several thousand NRC and Agreement State licensees possess sealed sources that will have to be stored until a disposal facility is available. The NRC staff has generated recommended options and procedures for licensees who possess these wastes and are unable to dispose of them. The NRC and the DOE have also set out procedures for the acceptance and storage or disposal of abandoned radioactive material by the DOE. The NRC staff recommended that each Agreement State establish procedures for requesting assistance from the DOE for the retrieval of abandoned radioactive material located in its State, and has agreed to assist the States with such requests until their procedures are in place.

The DOE has retrieved and stored or disposed of several sources that were abandoned in the public domain and were found to meet the eligibility criteria for emergency acceptance established by the NRC and the DOE. The majority of these sources exceeded the limits for Class C low-level waste, and DOE is storing these sources at its laboratories. Several sources were under the limits for Class C low-level waste and were disposed of in a commercial near-surface disposal site or transferred to another licensee. The NRC and DOE staffs are continuing discussions to establish additional eligibility criteria for accepting sources for interim storage or disposal by the DOE; NRC staff continues to monitor DOE's progress in identifying an interim storage facility.

**General License Program.** Under 10 CFR Part 31, a general license may be issued for possession and use of certain measuring and gauging devices containing byproduct materials. The generally licensed device usually consists of radioactive material, contained in a sealed source, within a shielded device. The device is designed with inherent radiation safety features, so that it can be used by persons with no radiation training or experience.

During the year, the staff evaluated comments received in response to a proposed rule concerning general licensees and prepared a final rulemaking package. The purpose of the rule is to make general licensees more aware of the NRC requirements and to ensure that they are accountable for their generally licensed devices. The rule would require general licensees to respond to requests by the NRC for information pertaining to their possession of generally licensed devices. The rule would also require the distributors of generally licensed devices to provide the NRC and Agreement States with additional information about general licensees receiving the devices, and to provide the general licensees with additional information on regulatory requirements for the possession, use, transfer and disposal of the device.

During fiscal year 1993, the NRC published a proposed rule concerning the maximum air gap between a source housing and its detector unit, for generally licensed devices. The rule is intended to reduce the number of exposures caused by general licensees' inadvertently inserting body parts into radiation beams and subjecting them to higher radiation levels. The staff has evaluated the comments received in response to the proposed rule and has worked on development of the final rulemaking package. The staff expects the final rulemaking package to be forwarded to the Commission in fiscal year 1994.

**Quality Assurance and Control Manual for Source/Device Vendors.** The staff continued to update its draft Quality Assurance and Control Manual for manufacturers and vendors of sources and devices containing byproduct material. Revisions to the draft were based on information obtained during the pilot evaluation program and comments from NRC Regions, Agreement State programs, and NRC and Agreement State licensees. In fiscal year 1993, the staff continued its pilot evaluation program by visiting vendors and manufacturers of sealed sources and devices, and continued developing the draft manual into a regulatory guide. The staff expects a draft regulatory guide to be published for public comment in fiscal year 1994.

## Medical Uses

**Management Plan for the Medical-Use Regulatory Program.** During an NRC Senior Management Conference held on August 3 and 4, 1992, senior management decided to prepare a staff management plan to guide the conduct of the medical-use regulatory program. In September 1992, the staff identified a number of actions to address the more pressing problems associated with the regulatory program, and developed a medical issues paper that identified certain program areas to be reviewed to determine whether and what modifications were necessary for improvement. The issues identified in the paper were discussed during meetings with representatives from the Agreement States, the NRC's Advisory Committee on the Medical Uses of Isotopes (ACMUI; see below) and NRC regional management. At that time, the staff anticipated submitting the management plan to the Commission in January 1993 for approval. However, these efforts were delayed as a result of a need for staff responses to three events: a November 1992 radiation therapy misadministration and associated patient fatality; the findings of the resultant Incident Investigation Team (IIT); and a series of articles published in the *Cleveland Plain Dealer*, December 13–17, 1992.

In a Staff Requirements Memorandum dated March 31, 1993, the Commission directed the staff to continue with development of the medical management plan, the internal management review of current practices for implementing the medical-use regulatory program, and the external review of the program. Subsequently, the staff developed a master agenda for improvement of the regulatory program, identifying 71 action items dealing with the issues raised by the IIT report on the patient fatality, and with other matters previously identified in the medical issues paper. The staff identified 19 other action items deriving from the senior manager review of the medical-use program. These were combined with the 71 master agenda items and taken up in the development of the staff's proposed management plan for the medical-use regulatory program. Each of these 90 action items was analyzed according to category and the scope of the work involved, its priority, the projected time-frame for completion, and inter-connections with other items. In some cases, action items were consolidated to coordinate work efforts and increase efficiency. On a number of items, action has been completed or will be completed shortly. The plan ultimately identified nine major program areas and incorporated current management direction for the regulatory program over the next five years. The development of the management plan included contributions from the ACMUI, representatives from Agreement States, professional organizations, other regulatory agencies, the medical community, and NRC senior management.

The staff anticipates that the medical management plan will be a dynamic, requiring periodic review and modifications which take account of new initiatives deriving from periodic reassessments, unforeseen events, and changes to projected completion dates, particularly when the outcome or completion date of one task affects another.

**Medical Visiting Fellows.** In 1990, the NRC selected a nuclear medicine physician and a radiopharmacist to participate as NRC Medical Visiting Fellows on a full-time basis. The radiopharmacist joined the NRC in December 1991 and completed his fellowship in June 1993. The nuclear medicine physician joined NRC in October 1991 and is scheduled to complete his fellowship in December 1995. The Fellows have been involved with a variety of medical issues, such as development of a proposed rule on the practice of radiopharmacy, in response to a petition; implementation of the 1992 Quality Management Program (QMP) and Misadministration rule, which affects most medical licensees; enforcement issues and misadministration cases; and exchanges of information with the regulated industry on medical issues of mutual interest, by participating in various professional meetings, including meetings of the ACMUI.

**The Advisory Committee on Medical Uses of Isotopes.** The Advisory Committee on Medical Uses of Isotopes (ACMUI) met in October 1992 and in February, May, and July 1993. Topics discussed at these meetings included an analysis of a medical issues paper (preliminary to staff preparation of a management plan for medical-use regulation), the radiopharmacy petition for rulemaking, the administration of byproduct material to pregnant or breast-feeding women, petitions regarding patient release criteria, patient notification and follow-up in cases of medical incident, brachytherapy regulation, training and experience requirements, and the ACMUI's views, presented to an NRC task force, regarding issues related to radiation safety in the uses of ionizing radiation.

ACMUI members serve two-year terms and are limited to three terms. In July 1993, four members, Gerald Pohost, M.D., Edward Webster, Ph.D., Capt. William Briner, and Steven Collins completed their terms. Five new members have been appointed: Daniel Berman, M.D., Wil B. Nelp, M.D., Robert Quillin, Judith A. Stitt, M.D., and Dennis P. Swanson, MS. Current membership of the committee is shown in Appendix 2.

**Quality Management Rule Implementation.** On January 27, 1992, regulations became effective that require medical licensees to establish and implement a Quality Management Plan (QMP), in compliance with 10 CFR 35.32. The rule is performancebased and focuses on the therapeutic uses of byproduct materials. Since that time, the NRC has contracted with Lawrence Livermore National Laboratory (LLNL) to review the QMPs submitted by the applicable licensees. The NRC expects the review of all QMPs to be completed in 1994.

The NRC, along with the LLNL, performed a pilot study that compared the licensee-submitted QMPs with the implemented program. Ten licensees, with multiple modality programs, were inspected. When the reviewed QMPs were compared to the implemented program, the implemented programs proved better than the submitted programs, in all but one case. NRC inspectors are also performing limited inspections of the implemented programs during routine inspections. A full inspection of the licensee's QMP is performed as part of the investigation of medical incidents and/or misadministrations. Enforcement action has been taken by the NRC for failure of the licensee to establish and/or maintain a QMP.

The enforcement policy regarding the QM rule was modified on April 2, 1993, to focus on programmatic weaknesses, rather than on isolated mistakes of limited consequence. From November 1992 to October 1993, the NRC convened a committee to review all QM enforcement cases each week. The QM Review Committee consisted of representatives from NMSS, the Office of Enforcement, the Office of the General Counsel, and the involved Regional Offices. The NRC staff was scheduled to brief the Commission on its findings in January 1994.

**Petition for Rulemaking: Traditional Nuclear Medicine and Pharmacy Practice.** On June 8, 1989, the NRC received a Petition for Rulemaking from the American College of Nuclear Physicians and the Society of Nuclear Medicine. The petition proposed changes to certain sections of the NRC regulations in 10 CFR Parts 30, 32, 33, and 35, affecting NRC medical-use licensees' receipt and use of byproduct radioactive drugs.

The NRC published a final rule for public comment on June 17, 1993 (58 FR 33396). The proposed rule is intended to provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians greater discretion to prepare radioactive drugs containing byproduct material for medical purposes. The proposed rule would allow research involving human subjects and using byproduct material, and also the medical use of radio-labeled biologics. The proposed rule also contains miscellaneous amendments necessary to clarify or update the current regulations.

In addition, the NRC published, on July 22, 1993 (58 FR 39130), an extension of the expiration date for the interim final rule on the subject, from August 23, 1993 to December 31, 1994. The extension is necessary to maintain the relief provided by the interim final rule. The action allows licensees to continue to use byproduct material under the provisions of the interim final rule, until the NRC completes a related rulemaking to address broader issues for the medical uses of byproduct material.

## EVENT EVALUATION AND RESPONSE

The NRC continues to review and analyze operational safety data from nuclear fuel facilities and materials licensees, and to maintain its ability to respond to events at these facilities.

**Transportation Incidents.** NMSS continues to monitor transportation incidents. Fifteen transportation incidents were documented during calendar year 1992, of which four were accidents, one was a handling event, three were thefts or loss of packages, and seven were classified as "other" events. Of the four accidents that occurred, one involved a type B package and three involved type A packages. There was no release of contents in the accident involving the type B package. Only one of the accidents involving type A packages resulted in a release of radioactivity. (Type A packages are not designed to withstand accidents, because of the limited amount of radioactive materials they contain.)

**Brachytherapy Accident in Indiana, Pa.** On December 1, 1992, the Indiana Regional Cancer Center reported to the NRC that it believed an iridium-192 source from its Omnitron 2000 high dose rate remote brachytherapy afterloader had been found at a biohazard waste transfer station in Carnegie, Pa. After notifying the NRC, the center, one of several operated by the licensee, retrieved the source, and Region I dispatched an inspector and a supervisor to investigate the event. The source was first detected when it triggered radiation alarms at a waste incinerator facility in Warren, Ohio. The licensee informed the NRC that the source wire had apparently broken during treatment of a patient on November 16, 1992, leaving the source in the patient. Considering the seriousness of the incident, the NRC elevated its response to an Incident Investigation. The Incident Investigation Team initiated its investigation on December 3, 1992. The team concluded that the patient, who died on November 21, 1992, had been subjected to a serious misadministration, and that over 90 other individuals had been exposed to radiation from November 16 to December 1, 1992. In a press release dated January 26, 1993, the Indiana County Coroner stated that the cause of death, as listed in the official autopsy report, was "Acute Radiational Exposure and Consequences Thereof."

An almost identical source wire failure occurred with an afterloader in Pittsburgh, Pa., on December 7, 1992, but with minimal radiological consequences. Again, a source became separated from the drive cable on a Omnitron Model 2000 Unit during brachytherapy treatment of a patient. In this incident, the source separation was detected; the catheter was cut and the patient was immediately removed from the treatment room. The remaining portion of the catheter was then removed from the patient, and both the catheter and patient were scanned with a survey instrument to confirm that no part of the source remained within either the catheter or the patient.

# Fuel Cycle Safety and Safeguards

Chapter 5

The Nuclear Regulatory Commission's Office of Nuclear Material Safety and Safeguards (NMSS) and the NRC's five Regional Offices administer the regulation of fuel cycle safety and safeguards.

The NMSS Division of Fuel Cycle Safety and Safeguards (FCSS) is responsible for the development, implementation and evaluation of overall agency safety and safeguards policy for fuel cycle facilities licensed under the Atomic Energy Act of 1954, as amended, or certified in accordance with the Energy Policy Act of 1992 (the 1992 Act). FCSS activities include the principal licensing, certification, inspection and regulatory activities associated with these facilities to ensure adequate safety and safeguards. The FCSS develops the NRC's design basis threats and assesses threats to the domestic environment affecting all of the NRC-licensed activities. The FCSS directs the NRC contingency planning and emergency response operations dealing with accidents, events, incidents, threats, thefts or radiological sabotage related to licensed activities under its responsibility. Technical support is provided to the International Atomic Energy Agency with respect to export/import requests and also in the review of safeguards issues related to the transportation of nuclear material. The FCSS coordinates with the NRC's Office of Nuclear Reactor Regulation, to ensure consistency in the implementation of the NRC's safeguards program for reactors.

In February 1993, certain functions and organizational elements in NMSS were reorganized, in order to more efficiently and effectively conduct the NMSS mission, through consolidation of fuel cycle facilities activities (both safety and safeguards) in FCSS. The changes allow for more focused management attention within NMSS to time-sensitive matters and high-visibility tasks, such as fuel facility safety issues and new enrichment activities.

## FUEL CYCLE LICENSING AND INSPECTION

### Fuel Cycle Action Plan

**Action Plan for Regulating Fuel Cycle Facilities.** In a May 1993 briefing to the Commission, the staff described an action plan designed to enhance the rigor of the regulatory base for the fuel cycle facility safety program, to improve the timeliness of the licensing renewal program, and to make numerous improvements in the program, as documented in the "Proposed Method for Regulating Major Materials Licensees" and the "Regulatory Impact Survey for Fuel Cycle and Materials Licensees." To accomplish these objectives, the action plan focused on improvements in the areas of regulatory development, licensing, inspection, training and licensee self-assessment. Among the efforts to clarify and upgrade the regulatory base is a major revision to 10 CFR Part 70. A draft rule is projected to be published by the end of fiscal year 1994, and a final rule within fiscal year 1995. In support of the rulemaking initiative, the Commission was informed of the staff's high priority efforts to develop a Standard Review Plan (SRP) and detailed guidance to assist licensees in performing Integrated Safety Analyses. The SRP will be useful not only to the NRC staff, in reviewing license renewal applications and amendments, but also to licensees, in formulating these documents. Public meetings with the fuel cycle facility licensees are planned to obtain input toward the development of the SRP and the standard format and content guidance document. With respect to the review of pending license renewal applications, the Commission was advised that licensing staff intends to continue ongoing reviews while contributing to the development of the SRP. In the interim, until the revision to 10 CFR Part 70 becomes effective, the developing SRP will be used in the review of license renewals and amendment applications, wherever relevant.

Upgrading of the inspection program is achieved primarily through formation of a new Headquarters Inspection Section, providing for an increased focus on inspection activities and more efficient use of limited technical expertise for performing nuclear criticality and chemical safety inspections, along with ongoing headquarters material control and accounting (MC&A) inspections. Headquarters will provide senior technical expertise to address difficult design, integration and adequacy concerns.

The Commission was advised that improvements in training and licensee self-assessment programs are under development. A standard training program is being developed for the licensing and inspection staffs. Regarding licensee self-assessment, the staff proposed letting industry take the lead in proposing a program to the NRC for consideration. The staff will coordinate closely with industry representatives to monitor progress on this initiative.

## Fuel Cycle Licensing Activities

By the end of fiscal year 1993, the NRC had completed 120 fuel cycle licensing actions. Table 1 shows licensing actions by category.

# FUEL CYCLE SAFETY

## Fuel Cycle Safety Licensing

**Combustion Engineering License Amendment.** On May 12, 1993, a license amendment was issued to Combustion Engineering, Inc., (CE) authorizing consolidation of the low-enriched uranium (LEU) conversion and fuel fabrication operations at the Hematite, Mo., facility. Previously, CE's Hematite facility was licensed to convert LEU hexafluoride to uranium oxide, which was subsequently pressed into pellets. These pellets were then shipped to CE's Windsor, Conn., facility to be sealed into fuel rods and encapsulated into fuel assemblies for use at commercial power reactors. In connection with this action, CE terminated its Windsor fuel manufacturing operations in September 1993, and plans to decontaminate and decommission the buildings used for these operations. CE does plan, however, to continue laboratory research and development activity at the site. The NRC is reviewing an amendment request to reduce the possession limits for special nuclear material (SNM) and to remove the authorization for conducting uranium-bearing nuclear fuel manufacturing operations for the Windsor facility.

**Nuclear Fuel Services (NFS).** Several license amendments were requested by NFS and ultimately approved by the NRC. In May 1993, the NRC approved a license amendment application from NFS to authorize the dilution of high-enriched uranium (HEU), in liquid form, to enrichments suitable for use in the commercial nuclear power industry. This approval would allow NFS to process HEU that could come from dismantled Russian nuclear weapons. However, at the end of the report period, arrangements with the Republics of the Commonwealth of Independent States had not been consummated.

A request to allow restart of the NFS LEU recovery facility was reviewed and approved, providing the capability for processing materials of various physical and chemical compositions to recover LEU.

Another NFS initiative for which a license amendment was reviewed and approved was its request for authorization to recover HEU from unirradiated fuel elements. This would permit NFS to extract uranium from unirradiated fuel from the Fort St. Vrain (Colo.) power reactor and to convert the fuel to an oxide form for reuse.

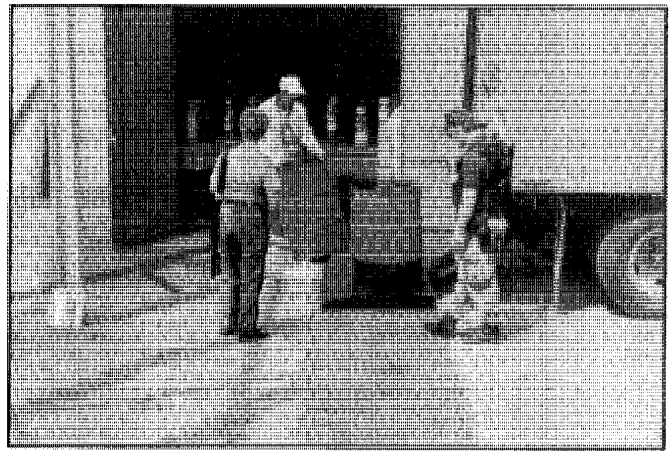
A final significant amendment request by NFS was for a change in possession limits to include uranium-233 and plutonium for use in research and development projects. These projects would include engineering studies and process evaluations for the remediation of contaminated sites and laboratory analyses of environmental samples.

**West Valley Demonstration Project Oversight.** Throughout fiscal year 1993, the NRC staff continued safety oversight at the Department of Energy's (DOE) West Valley Demonstration Project (WVDP), near Buffalo, N.Y. The purpose of the WVDP is to demonstrate the solidification and preparation of high-level waste from spent nuclear fuel reprocessing, for disposal in a Federal repository. Removal of dissolved cesium from the supernatant (liquid) portion of the waste, which began in early 1988, was declared complete in November 1990. The cesium will be combined with the solid portion of the high-level waste, which contains most of the other radionuclides. Before combination, the solid portion of the high-level waste will be processed to remove salts, in a process called "sludge washing." Beginning in 1996, the combined wastes will be solidified (vitrified) in borosilicate glass.

The NRC staff monitors public health and safety aspects of the WVDP through inspections at the West Valley site and review of Safety Analysis Reports (SARs) submitted by the DOE. The DOE normally submits a separate SAR for each segment of the waste process, including solidification in glass. The staff reviews each submittal and issues a corresponding Safety Evaluation Report (SER), giving its conclusions regarding the public health and safety implications of that process segment.

In fiscal year 1993, the staff monitored continued construction and installation of equipment for the vitrification process building. The staff also continued to assess data from cement produced through "sludge washing." As an agency cooperating in the preparation of an Environmental Impact Statement (EIS) for site decommissioning, the NRC continued discussions with the DOE to develop decommissioning criteria to be addressed by the DOE for various aspects of the WVDP under NRC oversight. A draft EIS is expected to be published by the DOE and the State of New York in 1994.

**Gaseous Diffusion Uranium Enrichment.** The Atomic Energy Act of 1954 (AEA) was amended under the 1992 Act to establish a wholly owned government corporation to enrich uranium, conduct related programs, and participate in uranium fuel cycle activities as a commercial enterprise. Specifically, the 1992 Act established the United



**The NRC, through the Office of Nuclear Material Safety and Safeguards, regulates safeguards for licensed nuclear materials, facilities and activities. In the regulatory context, "safeguards" denotes measures taken to deter, prevent or respond to unauthorized possession or use of certain nuclear materials through theft or diversion, and to protect against sabotage involving nuclear facilities or materials. Safeguards activity typically entails the guarding of entrances to prevent unauthorized entry and inspection of vehicles transporting nuclear materials, as above.**

States Enrichment Corporation (USEC) and authorized the USEC to lease on July 1, 1993, the two DOE gaseous diffusion plants (which enrich uranium for use in reactor fuel) located in Portsmouth, Ohio, and Paducah, Ky. Further, the Act placed the plants under the NRC's regulatory purview. In February 1993, the NRC reorganized to accommodate its new regulatory responsibilities for the these plants, creating the Enrichment Branch (FCEB) in the NMSS and a gaseous diffusion plant inspection organization in Region III (Chicago).

The Act requires the NRC to promulgate safety and safeguards standards governing the enrichment plants within two years of the date the legislation was signed into law. Annually, the NRC is required to make findings on compliance, certify compliance with the standards (or approve a compliance plan), and report the findings to Congress. The DOE will retain regulatory purview for the plants until the NRC's initial certification.

A proposed new rule (10 CFR Part 76) has been developed to establish technical, legal, and administrative requirements for the NRC's regulation of the enrichment plants. The rule establishes standards for adequate protection of public health and safety and the environment, as well as for safeguarding nuclear materials in the interest of national security. The proposed rule, published for public comment in 1993, is to be issued as an effective regulation by October 1994. An application for the NRC certification from USEC is anticipated by April 1995. The NRC is coordinating with the DOE, the Environmental Protection Agency, the Occupational Safety and Health Administration, the Federal Emergency Management Agency, and State and local governments to identify and resolve technical, legal and administrative issues concerning the NRC's regulatory oversight of the plants.

Together the two enrichment plants have annual sales totaling $1.3 billion. They are the world's largest single supplier of LEU hexafluoride ($UF_6$), slightly less than 50 percent of the world market share. The plants have operated continuously since the early 1950's. Each plant site encompasses nearly 3,000 acres, with nearly 100 acres under roof. The Portsmouth design electrical power utilization is 2,260 megawatts (maximum). The plant employs approximately 2,740 people. The Paducah design electrical power utilization is 3,040 megawatts (maximum), and that plant employs approximately 1,825 people.

**Gas Centrifuge Uranium Enrichment.** In November 1990, the President signed into law the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990 (Public Law 101-575). This law amended the Atomic Energy Act to establish new requirements for regulation of commercial uranium enrichment facilities. The NRC published rule changes implementing the amendment in the *Federal Register* on September 16, 1991.

In January 1991, the Louisiana Energy Services, L.P., submitted an application for a license to construct (at a projected cost of over $800 million) and operate a gas centrifuge uranium enrichment plant, to be known as the Claiborne Enrichment Center. It will be located in Claiborne Parish near Homer, La., and would have a capacity of 1.5 million kilograms of "separative work units-per-year," about 15 percent of the annual requirement of United States nuclear utilities for enrichment services.

In July 1991, a preliminary meeting took place in Homer, La., as part of the process leading to preparation of the required environmental studies. During 1993, staff continued review of the license application and preparation

## Table 1. Fuel Cycle Licensing Actions Completed in FY 1993

| *Category* | *No. of Actions* |
|---|---|
| Uranium Fuel Fabrication | 43 |
| Uranium Hexafluoride Production | 9 |
| Critical Mass Materials | 7 |
| Fuel Research & Development & Pilot Plant | 4 |
| Other Source Materials (Metals Extraction) | 4 |
| Fuel Facility Decommissioning | 3 |
| Fresh Fuel Storage | 2 |
| Material Control and Accounting | 35 |
| Physical Security | 13 |

of the draft EIS and SER. The draft and final EIS documents will be published in late 1993 and 1994, respectively. The SER will be published in early 1994. The required hearings on technical and environmental issues will begin following publication of the final SER and EIS documents, respectively.

### Fuel Cycle Safety Inspection

**Headquarters-Based Inspection Activities.** As part of the February 7, 1993 reorganization of fuel cycle activities within NMSS, several fuel cycle facility inspection activities have been consolidated in Headquarters, in a phased approach. During fiscal year 1993, headquarters staff provided technical expertise to address difficult design, integration and adequacy concerns in the areas of criticality and chemical safety.

**Region-Based Inspection Activities.** The five Regional Offices conducted more than 100 safety inspections at 15 operating and decommissioning fuel cycle facilities during fiscal year 1993. The inspections included resident inspector activities at two of these fuel cycle facilities. The inspections covered the areas of criticality safety, radiation protection, emergency preparedness, environmental safety, and transportation.

## FACILITIES AND TRANSPORTATION SAFEGUARDS

### Fuel Cycle Safeguards Licensing

There were 13 active, licensed nuclear fuel cycle facilities subject to NRC comprehensive safeguards requirements during fiscal year 1993. Of these, eight were major fuel fabrication facilities. Two of the 13 facilities contain significant quantities of HEU, requiring extensive physical security and MC&A measures. One of these two facilities—NFS, of Erwin, Tenn.—essentially phased out its naval reactors program work during calendar year 1993. An agreement with the Russian Federation, involving the conversion of HEU from the former Russian nuclear weapons program into light water reactor fuel, did not lead to any subsequent activity during 1993. If the NFS Erwin facility eventually becomes involved in this conversion work, or other sources of work are procured, the facility will continue to be operated under NRC license.

The fully implemented physical protection requirements established in 1991 provide for performance testing through the use of mandated tactical drills and exercises. Besides the additional assurance that physical protection at these sites is effective, both of the two Category I facilities cited above have increased performance and provided more effective implementation of physical protection measures, as a result of lessons learned during performance testing.

The NRC continues to support the DOE on the storage of spent reactor fuel at the monitored retrievable storage and the monitored geologic depository storage facilities, in anticipation of the eventual submittal of a license application for these sites. Development of regulations, guidance, and certification modules for these sites was under way in 1993. In addition to the facilities noted above, several independent spent fuel storage installations that are not located on the site of a licensed power reactor were also subject to safeguards requirements.

### Fuel Cycle Safeguards Inspection

Headquarters staff conducted 15 comprehensive MC&A inspections, while the regional and resident inspectors continued to perform inspections for physical security at major fuel fabrication facilities. Approximately 17 physical security inspections were performed by region-based inspectors. Newly implemented physical security improvements were thoroughly inspected at the two facilities cited above as possessing significant quantities of HEU. Performance-based inspection procedures were followed by both MC&A and physical security inspectors.

## Reactor Safeguards

**Reactor Safeguards Inspection and Licensing.** Within the five NRC Regional Offices, a total of 185 safeguards inspections were conducted at licensed nuclear power reactors under NRC safeguards requirements. Approximately 227 revisions to licensee security, contingency, and guard training plans were reviewed and found acceptable by both regional and headquarters staff.

**Operational Safeguards Response Evaluations at Power Reactors.** After completion of the Regulatory Effectiveness Review Program in May 1991, the NRC staff initiated an Operational Safeguards Response Evaluation (OSRE) program at power reactors. An OSRE is an effectiveness review conducted by an interdisciplinary team consisting of a nuclear engineer and physical security specialists, assisted by U.S. Army Special Forces personnel. The team evaluates a licensee's contingency response capabilities by focusing on the interactions between operations and security personnel in establishing priorities for the protection of safety equipment, and by scrutinizing and testing the defensive strategies used. OSRE teams also conduct safety/safeguards interface reviews, to ensure that safeguards measures do not adversely affect the safe operation of the plant. Seventeen OSREs were conducted during fiscal year 1993. This has resulted in a combined total of 15 significant improvements at seven power reactor sites.

**Fitness for Duty and Access Authorization at Power Reactors.** Power reactor licensees are required to implement fitness-for-duty programs under 10 CFR Part 26. Although the existing rule appears to be achieving the desired effects, the Commission is considering changes that would reflect lessons learned during the first three years of the program.

Program performance data provided by licensees have been summarized in "Fitness for Duty in the Nuclear Power Industry: Annual Summary of Program Performance Reports, CY 1992" (NUREG/CR-5758, Volume 3). The report indicates that over 266,000 tests for the presence of illegal drugs and alcohol were conducted during calendar year 1992, of which 1,818 were positive. The majority of the positive test results (1,110) were obtained through pre-access testing (a 1.06 percent positive rate). There were 461 positive tests from random testing (0.29 percent positive rate). The positive rate also varied by worker category. For example, 0.20 percent of random tests of licensee employees were positive; for long term contractors, the rate was 0.37 percent; and for short-term contractors, the rate was 0.46 percent. The general trend of the positive rates, with some minor exceptions, is downwards. The Commission has proposed modifications to the fitness-for-duty program that would permit licensees to lower the random testing rate to 50 percent, from the current 100 percent rate.

**Access Authorization Programs at Power Reactors.** Power reactor licensees are required to implement access authorization programs, under 10 CFR 73.56. The programs—by means of background investigations, psychological assessments, and behavioral observations—are intended to ensure that individuals granted unescorted access to protected and vital areas at nuclear power plants are trustworthy and reliable, and do not constitute an unreasonable risk to the health and safety of the public, including a potential to commit radiological sabotage.

Sixteen inspections of licensee access authorization programs have been conducted under a temporary inspection program (TI 2515/116), to assess initial implementation of selected programs to determine whether they meet regulatory requirements and to identify program strengths and weaknesses. The results of these inspections are being evaluated to determine if changes to the program requirements are needed and if modifications need to be made to the scope and depth of the inspection program.

**Non-Power Reactors.** The NRC conducted 34 safeguards inspections of non-power reactors (NPRs) during fiscal year 1993. Efforts are continuing toward converting 25 NPRs from the use of HEU to LEU fuel. NRC regulations governing this project continue to be predicated on (1) the availability of DOE funding, (2) the availability of a suitable replacement fuel, and (3) whether a reactor has a "unique purpose" requiring the use of HEU. The status of the conversion program at the end of the fiscal year is as follows: one license has been terminated; two licensees have been issued decommissioning orders; one licensee is planning to decommission its reactor, and seven reactors have been converted from the use of HEU to LEU fuel. Of the 14 reactors that are still operating with HEU, nine have received funding from the DOE for the purpose of evaluating the operational effects of the conversion and the writing of an SAR. Also, two "unique purpose" applications are being reviewed by the Commission. There are two commercial reactor licensees that are not scheduled to receive DOE funding.

## Transportation Safeguards

**Spent Fuel Shipments.** Safeguards requirements were applied to 29 shipments of irradiated spent reactor fuel made over approved routes during fiscal year 1993, including nine rail shipments to the spent fuel pool at the Harris

(N.C.) nuclear power plant, which is configured to store a large number of spent fuel assemblies. These shipments, planned by the Carolina Power and Light Company, will, over a five-year period, transfer approximately 1,170 fuel assemblies from other reactors to the Harris pool for storage. One of the shipments was an export.

**Strategic Special Nuclear Material Shipments.** Four domestic shipments of less than five but more than one kilogram of HEU were completed during fiscal year 1993. Two export shipments of five or more kilograms were also made during fiscal year 1993.

Tracking International Shipments of SNM. NRC regulations require licensees to notify the NRC of international shipments of SNM and natural uranium. During fiscal year 1993, the NRC received about 200 such notifications. When appropriate, these were forwarded to the Department of Transportation, for notification of international authorities.

## INTERNATIONAL ACTIVITIES

(See Chapter 8 for detailed coverage of NRC "International Cooperation.")

### International Safeguards

The NRC is responsible for implementation of International Atomic Energy Agency (IAEA) safeguards at licensed nuclear facilities in the United States. Although there are no U.S. nuclear facilities under IAEA inspection at this time, the IAEA has discussed selecting U.S. facilities for inspection. In this connection, the NRC assures that U.S.-licensed facilities maintain their MC&A systems and conduct their reporting responsibilities to meet the terms of the U.S./IAEA Agreement for the Application of IAEA Safeguards in the United States, as specified in 10 CFR 75. The United States continues to report to the IAEA all accounting information required by the Protocol to the U.S./IAEA Safeguards Agreement.

In response to concerns with Iraqi nuclear activity, the IAEA is looking to broaden its safeguards activities, to include measures to detect undeclared nuclear facilities. The NRC is supporting this effort and is contributing to the evaluation and implementation of new measures. In this regard, the IAEA Board of Governors decided, during 1992 and 1993, with the support of the United States, to request that Member States report certain additional information. The information is to include (1) early provision of design information on new facilities; (2) early provision of information on major modifications and additions to existing facilities; (3) expanded reporting of exports, imports, and production of nuclear materials; and (4) reporting of the import and export of certain nonnuclear materials and equipment. During fiscal year 1993, the NRC took measures to satisfy these requests.

The NRC is also assisting the IAEA in conducting two Short Notice Random Inspection (SNRI) tests at NRC-licensed facilities. One SNRI application is for the shipments and receipts at an LEU fabrication plant, and the other application is related to the randomization of interim inspections and surveillance on spent fuel at five reactors.

The NRC is responsible for the licensing of exports and imports of nuclear facilities, equipment, material, and related substances, as set forth in the Atomic Energy Act (AEA), as amended. Further, under amendments to the AEA adopted in the Nuclear Non-Proliferation Act of 1978, the NRC must be consulted by the Department of State regarding new agreements for nuclear cooperation. Also, the NRC must be consulted by the Department of Energy (DOE) before the authorization of subsequent arrangements for the retransfer of U.S.-obligated nuclear materials from one country to another, and before the provision of technological assistance to foreign nuclear energy activities. During 1993, 113 technical international safeguards reviews were performed regarding export applications, agreements for nuclear cooperation, subsequent arrangements, and technology transfers.

In keeping with the NRC responsibility to ensure application of IAEA safeguards to U.S. nuclear material exported, the NRC supports the improvement of effective international safeguards. The NRC continues to contribute to U.S. Government efforts to strengthen IAEA safeguards and to maintain effectiveness of implemented safeguards. During 1993, special studies continued for the development of a technique that contributes to the safeguarding of the nuclear material in the "head end" of reprocessing plants, for an analysis of the safeguards approach for a centrifuge enrichment plant, and for the establishment of international criteria for the termination of IAEA safeguards on nuclear materials contained in highlevel waste.

The NRC continues to contribute to the total U.S. support of IAEA safeguards through interagency efforts involving the DOE, the Arms Control and Disarmament Agency, the Department of State, and the NRC. These interagency activities serve to coordinate U.S. Government technical safeguards support to the IAEA.

The NRC continues to provide support to the interagency Comprehensive Threat Reduction Program. This initiative, originally called the Safe and Secure Dismantlement program, is to coordinate support to the republics of the former Soviet Union in dismantling their nuclear arsenals and stemming proliferation of weapons of mass destruction. The NRC's role is to supply assistance to these republics in setting up national regulatory systems for material control and accounting (MC&A) and physical protection, as well as to assist individual facilities in developing and evaluating site-specific MC&A and physical protection plans.

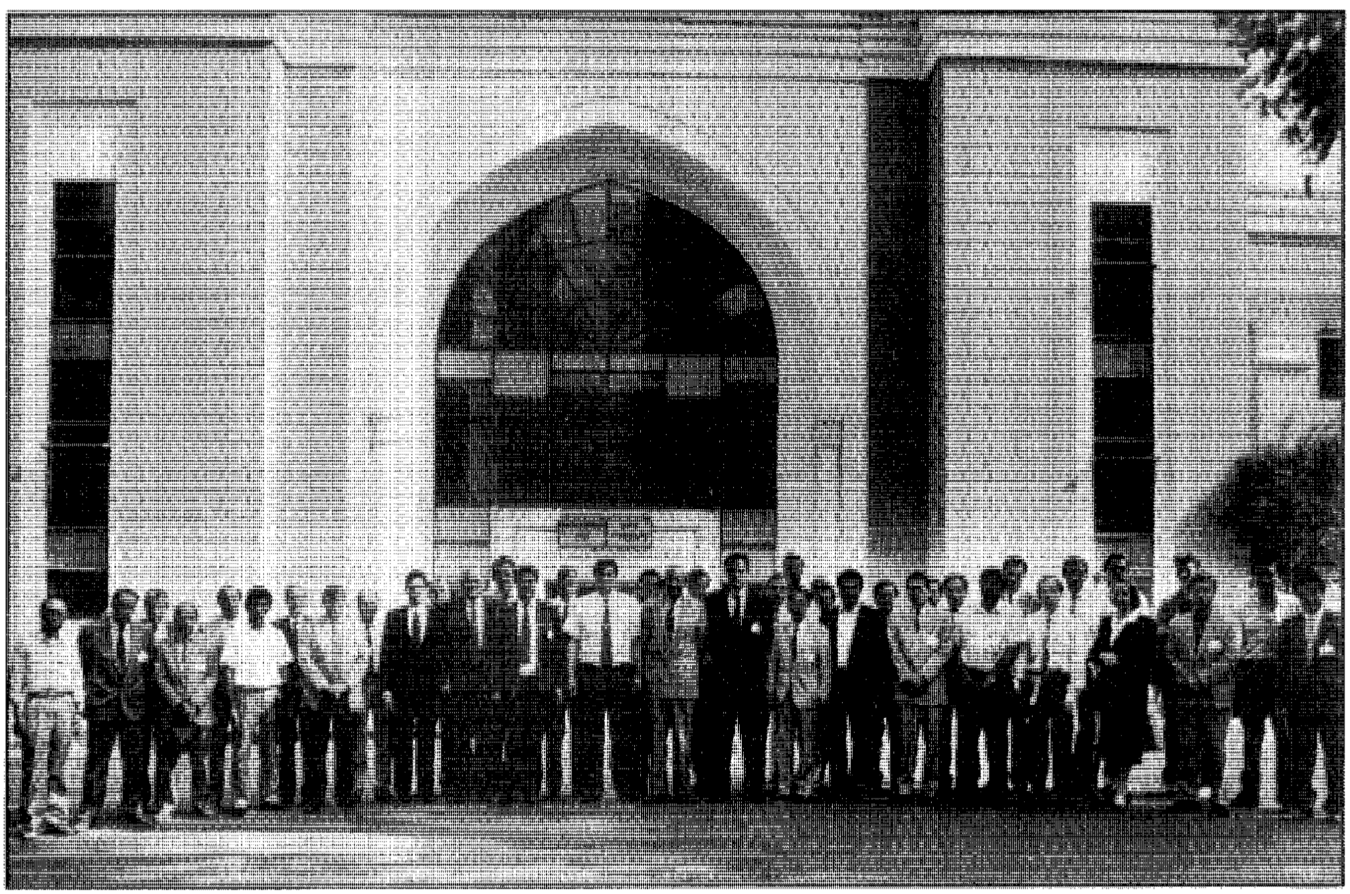
Russia signed the MC&A implementing agreement in September 1993. Kazakhstan has agreed to the text of the MC&A implementing agreement and is expected to sign in January 1994. Ukraine is still waiting for parliamentary approval of several of its agreements, including MC&A. Belarus, which has no reactors or fuel facilities, has also requested U.S. assistance in setting up a national regulatory program, but discussions in this area have been limited to date.

In February 1993, the United States and the Russian Federation reached agreement on the disposition of HEU recovered from decommissioned Russian nuclear warheads. The bilateral agreement allows the United States to purchase approximately 500 metric tons of HEU extracted from dismantled nuclear weapons and blended, in Russia, down to low-enriched form. The material will be fabricated in the United States, by NRC licensees, for use as light water reactor fuel. The NRC's role is to ensure that "transparency" measures in U.S. facilities are practical, and, in this context, the NRC solicited comments from fuel fabricators and ensured that their concerns were considered in the agreements. The United States and Russia are negotiating final details related to transparency, with the intent of starting the blending process in 1994.

In August 1993, in response to an invitation by the government of the People's Republic of China (PRC), a senior safeguards specialist presented seminars on the NRC's safeguards programs related to the commercial nuclear industry in the United States. The visit resulted in a better understanding, by the Chinese, of the regulations for the U.S. safeguards system. The PRC is trying to improve its state system for regulating its nuclear industry for peaceful purposes.

## International Physical Protection

In connection with its export licensing program, the NRC participates in an interagency program to visit and to



**The NRC continued during the report period to help coordinate support to Russia and other nations formerly in the Soviet sphere in dismantling their nuclear arsenals and in setting up national regulatory systems. Above are delegates to a meeting, including NRC personnel, held in June 1993 in Almaty, Kazakhstan, to help frame nuclear safety programs for former republics of the Soviet Union and the Baltic States.**

exchange information on physical protection of nuclear materials and facilities with all countries that have imported a significant amount of nuclear material from the United States, or have received significant retransfers of U.S.-obligated material. During fiscal year 1993, a visit for this purpose was made to Switzerland. Similarly, teams from Japan, France and the United Kingdom visited the NRC and NRC-licensed facilities.

## NUCLEAR MATERIALS MANAGEMENT AND SAFEGUARDS SYSTEMS

Jointly funded by the DOE and the NRC, the Nuclear Materials Management and Safeguards System (NMMSS) is an accounting system encompassing all licensed SNM and foreign source material in the United States, including materials that originated both in the United States and elsewhere. Material is tracked from facility to facility, on a continuous basis, from original refinement to eventual disposal. Export/import transactions are also tracked. Selected data, based on NMMSS output, are then furnished to the IAEA, in fulfillment of U.S. international obligations and bilateral agreements. On January 26, 1993, the NRC published a notice making licensee submittal of information in computer-readable form mandatory. The 90-day comment period ended April 26, 1993. The general response was approval and readiness to take advantage of today's technology. After final testing of the concept, the NRC plans to publish the final rule during fiscal year 1994.

## SAFETY AND SAFEGUARDS EVENT EVALUATION AND RESPONSE

### Reporting of Nuclear Criticality Safety Events

On October 18, 1991, the NRC Bulletin 91-01, "Reporting Loss of Criticality Safety Controls," was issued to all NRC-licensed facilities whose activities include hot cell operations, enriched uranium operations, uranium fuel research and development, and critical mass operations. The bulletin requested that licensees inform the NRC of their criteria and procedures to ensure prompt evaluation and reporting of conditions and events involving nuclear criticality safety. Following the review of licensees' responses and comments, the NRC issued a supplement, dated July 27, 1993, to clarify the immediate and 24-hour reporting criteria for degraded nuclear criticality safety controls. Safety-related events are reported to the NRC Operations Center.

Following the creation of the Fuel Cycle Safety and Safeguards (FCSS) Division of NMSS, in February 1993, the staff established a team, consisting of technical experts and management, to assess the safety significance of licensee reported events. The staff also initiated an effort to compile event reports for tracking and trend analysis. During the period from February through October 1993, 13 events were reported that met the Bulletin 91-01 criteria. A small number of these resulted in enforcement actions by the NRC Regional Offices. The majority of the events were reported within 24 hours and involved less significant degradations of criticality safety controls.

### Threat Assessments and Incident Response

**Threat Assessment and Liaison.** The NRC staff assesses threats to NRC-licensed facilities, materials and activities, and prepares the NRC's safeguards incident response plans for responding to actual thefts of nuclear material or radiological sabotage of nuclear facilities or activities. The safeguards staff maintains close and continuing contact with the intelligence community, including participation in regular interagency meetings of Federal agencies that are concerned with, and prepared to deal with, terrorism. Other liaison activity includes briefings and consultations with the representatives of other governments regarding the NRC threat assessment and incident response activities. In March 1993, the NRC staff provided training at a Federal Bureau of Investigation (FBI) In-Service training program at Quantico, Va. During fiscal year 1993, the NRC continued its participation in a variety of other training sessions for intelligence community threat analysts and others to increase their understanding of nuclear-related matters. Finally, the NRC worked closely with the DOE and other interested agencies on reported attempts to sell alleged illicit nuclear materials. The joint NRC/DOE Communicated Threat Credibility Assessment Team, a multi-discipline assessment team, was activated twice during fiscal year 1993 to assess written threats against NRC licensees. Both threats were assessed as being not credible.

**Design Basis Threat.** Two events in February 1993 caused the NRC to reconsider the vehicle intrusion and bomb threat issues associated with its design basis threat for radiological sabotage (10 CFR 73.1(a)1)), which previously had not specifically addressed the malevolent use of vehicles. The first event was a forced vehicle entry into the security protected area at Three Mile Island Unit 1 (Pa.). Although the entry did not threaten public health and safety, it demonstrated that a vehicle could be used to gain quick access to vital areas of the plant. The second event was the vehicle bombing at the World Trade Center in New York City.

The NRC staff and interested parties briefed the Commission in two public meetings, and the staff held another public meeting to seek the involvement of those expected to be burdened with additional requirements. Continual staff evaluation concluded that there is no indication of an actual vehicle threat against the domestic commercial nuclear industry. Nonetheless, in light of the two incidents, the staff concluded that a vehicle intrusion or bomb threat to a nuclear power plant could develop without warning in the future. Accordingly, in order to maintain a prudent margin between what is the current threat estimate (low) and the design basis threat (higher), the staff proposed a modification of the design basis threat for radiological sabotage to include protection against malevolent use of vehicles at nuclear power plants. A design basis threat was developed based on staff analysis of over 500 past vehicle bomb attacks worldwide. The NRC is proceeding with expedited rulemaking to provide protection against such threats. The design basis threat for theft of nuclear materials was not modified as a result of the above activities.

As part of these efforts regarding the vehicle threat, the Commission was briefed by the Central Intelligence Agency and the FBI regarding current threat considerations that could assist the Commission in its deliberations. At the May 10, 1993 public meeting noted above, formal presentations were made by the NRC and representatives of the Committee to Bridge the Gap, the Nuclear Control Institute, the Nuclear Management and Resources Council, and a private citizen from the Harrisburg, Pa., area. All attending were provided with an opportunity to express their views during a panel discussion. A range of opinions were put forth regarding the need for vehicle bomb protection at nuclear power plants, and equally diverse views were expressed regarding other aspects of the design basis threat for radiological sabotage. In response to a Commission request, the staff also solicited estimates regarding the cost of the various vehicle protection options.

In addition to its routine continuing evaluation of events worldwide, the staff was directed by the Commission to focus on both design basis threat statements, to ensure that no trend in adversary attributes (e.g., group size or weaponry) has developed undetected during recent years that would require a change to the threat statements. As part of this special reexamination, the staff is also reviewing changes in the commercial nuclear power industry that have occurred since the design basis threat statements were promulgated in the late 1970's and analyzing those changes in terms of the existing design basis threats. The re-examination of adversary characteristics and industry changes was scheduled for completion in January 1994, with the results to be reported to the Commission.

**Incident Response.** During the report period, the fuel cycle safeguards incident response plan was reviewed and updated. In February 1993, the reactor Safeguards Incident Response Team was activated and placed in standby mode during the intrusion event at the Three Mile Island (Pa) nuclear power plant. In May 1993, specialized training was held for the NRC Headquarters Duty Officers regarding NRC threat assessment procedures, and an exercise involving power reactor safeguards was conducted in June.

### Safeguards Summary Event List

The staff continued to analyze safeguards events related to threats and incidents, to identify trends, patterns and anomalies. During fiscal year 1993, the staff published the "Safeguards Summary Event List" (NUREG–0525, Volume 2, Revision 1), a compilation of brief summaries of several hundred safeguards-related events involving nuclear materials or facilities regulated by the NRC, which occurred and were reported from January 1, 1990 through December 31, 1992. Volume 1, which summarized events that occurred and were reported from pre-NRC through December 31, 1989, was published in July 1992. The list is intended to provide a broad perspective on the nature of safeguards incidents in the licensed nuclear industry, both unusual and routine, and is distributed to the licensed nuclear community, foreign governments, the Congress, and other Federal agencies.

### Safeguards Event Log Analysis

The NRC data base, which contains all safeguards events reported to the NRC by power reactor and Category I fuel cycle licensees, in accordance with 10 CFR 73.71, "Reporting of Safeguards Events," is used by the agency to perform analyses. From those data, a number of reporting trends and indicators of precursors to equipment failure and human error have emerged.

The event logs and data analysis indicated a number of repeatedly occurring events, which licensees have reduced through equipment or procedural changes. During the report period, two reports were issued which included a summary of solutions to common problems developed by some licensees, in order to help other licensees prevent similar or identical events.

Information gleaned from the logs and provided by the NRC inspectors and licensees indicate that cost-effective and sound, long term solutions to equipment failure and human error are proving successful in providing effective security. (NRC analysis of the log data has been discontinued because of budgetary constraints.)

## SAFETY AND SAFEGUARDS REGULATORY ACTIVITIES

### Guidance Documents

**Integrated Safety Analysis of Fuel Cycle Facilities.** After a potential criticality incident in May 1991, the Director, NMSS, appointed a Materials Regulatory Review Task Force to conduct a broad-based review of the Commission's current licensing and oversight programs for fuel cycle facilities. The resulting Task Force report (NUREG-1324) recommended, among other items, revision of the regulations in 10 CFR Part 70 to require a hazard analysis for each such facility—an analysis that should be detailed enough that each system or component that contains radioactive material or that serves as a barrier to the release of radioactive material is examined for potential failure leading to an accident.

The staff is considering amending 10 CFR Part 70 to require that the licensee or license applicant perform an Integrated Safety Analysis (ISA) of its plant and submit the results as part of the license application. In anticipation, the NRC is developing a document to provide guidance to the industry on acceptable ways of performing an ISA. As part of this effort, the NRC organized a publicly attended workshop on August 27, 1993. The objective was to obtain information from the operators of the facilities regarding current techniques of safety analysis followed by the industry and its current capabilities and resources that can be brought to bear on the performance of an ISA. An outline of the proposed guidance document was discussed at this workshop. Another such workshop is planned at the completion of the document. A draft product will be published with the proposed modifications to 10 CFR Part 70.

**Physical Security Guidance Documents.** As part of a continuing initiative to keep regulatory guidance documents current and up-to-date, work continued during fiscal year 1993 to update a number of physical security guidance documents. These documents are used by the NRC staff, licensees and members of the public and have been particularly useful in providing guidance to foreign countries on how the NRC implements many of its programs. Documents issued in fiscal year 1993 include guidance on locking systems for physical protection and control and entry/exit control components for physical protection systems.

**Fire Protection of Fuel Cycle Facilities.** Draft Regulatory Guide DG-3006, "Standard Format and Content for Fire Protection Sections of License Applications for Fuel Cycle Facilities," was published in April 1993. The document provides guidance to license applicants and licensees on the information that should be presented in the fire protection sections of applications for new licenses, renewals of existing licenses, or amendments to existing licenses that have impact on the fire safety of the facility. The Guide presented a standard format for submitting the information, to facilitate timely and uniform review of these applications. This Guide was based on the Branch Technical Position on Fire Protection for Fuel Cycle Facilities published on August 10, 1992 (57 FR 35607-13). A 90-day period was allowed for the public to comment on the Guide. The staff is presently considering the comments received in the context of 10 CFR Part 70 regulatory review criteria under development.

## Proposed Rules

The following rulemaking actions were initiated during fiscal year 1993:

- **Design Basis Threat.** Work was initiated on a proposed power reactor rulemaking to modify the design basis threat for radiological sabotage (10 CFR 73.1) to include a land vehicle for the transport of personnel, hand-carried equipment, and/or explosives. The rule would also modify 10 CFR 73.55 to reflect the change to the design basis threat and would allow for consideration of reasonable alternative security measures, when establishing "stand-off" distance. The proposed rule is expected to be published for comment early in fiscal year 1994.

- **Independent Spent Fuel Storage Installations (ISFSIs).** Work was initiated on a proposed rulemaking to amend 10 CFR Parts 72 and 73, to provide physical protection requirements for ISFSIs with a possession-only license. The rule would affect only those sites where spent fuel is stored away from an operating power reactor. Pending finalization of policy, the rule would also apply to the DOE monitored retrievable storage sites and, possibly, the permanent storage geological repository.

- **Fuel Cycle Plant Safety.** As a result of the repeated occurrence, over a period of several years, of fuel cycle plant safety-related incidents, ranging in level of safety significance from minor to major events, a review group was formed to analyze safety practices at these facilities and to recommend changes to the regulatory base needed to correct deficiencies. Its report, NUREG-1324, resulted in an action plan, approved by the Commission, which includes a substantial revision of the regulations. Two principal changes are the use of an Integrated Safety Analysis of plant processes and changes to plant processes, and the application of upgraded management control systems. A rule incorporating the group's recommendations has been drafted and is expected to be published for public comment during 1994.

- **Certification of Gaseous Diffusion Plants.** In accordance with the provisions of the Act, work was initiated on a proposed new 10 CFR Part 76 that would include the requirements for certification and operation of the DOE plants that enrich uranium. This new part will include procedural requirements, generally applicable NRC health and safety standards, technical safety requirements, and safeguards requirements specific to the gaseous diffusion plants.

Besides the proposed new 10 CFR Part 76, there are a number of minor conforming changes also being proposed to implement the new rule.

The following rulemaking actions continued during fiscal year 1993:

- **Strategic Special Nuclear Material (SSNM) in Transit.** Work is continuing on a rulemaking to upgrade the requirements for physical protection of SSNM in transit. Presently, the DOE is making commercial shipments of SSNM. The proposed rule would upgrade the NRC regulations, to make the NRC criteria for commercial transport comparable to that provided by the DOE.

- **Computer-Readable Reports on SNM.** Work is continuing on a proposed rule to amend 10 CFR Parts 40, 72, 74, 75 and 150. These amendments propose that licensees now satisfying reporting requirements on SNM transactions using paper forms make such reports in computerreadable form. The proposed amendments are intended only to take advantage of current computer technology, which enables the data collection process to be both more efficient and less costly. The 90-day comment period ended April 26, 1993. The general response was approval and encouragement to take advantage of today's technology. After field testing the concept, the NRC will publish the final rule during fiscal year 1994.

- **Annual Physical Fitness Performance Testing.** On October 6, 1993, a proposed rule was republished (originally part of the proposed rule published on December 13, 1991) to amend 10 CFR Part 73, "Physical Protection of Plants and Materials." This proposed rule would require Tactical Response Team (TRT) members, guards and other armed response personnel at Category I licensees to participate in a continuing physical fitness program and pass an annual performance test. As an alternative, licensees would be permitted to develop a content-based sitespecific test, to be administered quarterly, and to verify that this test duplicates the response duties that are expected of TRT members, guards and other armed response personnel in the event of a strenuous tactical engagement.

## Final Rules

The following rulemakings were completed and published in fiscal year 1993:

- **Clarification of Physical Protection Requirements at Fixed Sites.** On March 15, 1993, a final rule to amend 10 CFR Part 73 to clarify physical protection requirements was published. The amendment clearly states that each licensee shall provide physical protection at a fixed site, or at contiguous sites, where licensed activities are conducted, against radiological sabotage or theft of SNM, or against both, in accordance with applicable sections of 10 CFR Part 73, for each specific class of facility or material license. A new 10 CFR 73.60(f) has been added, stating that the Commission may require, depending on the individual facility and site conditions, any alternate or additional measures deemed necessary to protect against radiological sabotage at non-power reactors licensed to operate at or above a power level of two megawatts-thermal.

- **Unannounced Safeguards Inspections.** A final rule was published, in June 1993, to ensure that the presence of NRC safeguards inspectors at Category I fuel cycle facilities is not announced or otherwise communicated to licensees and contractor personnel, without the inspector's express request that this be done. The intent is to make inspections at these sites more effective.

- **Fitness-for-Duty Requirements.** On June 3, 1993, a final rule to amend 10 CFR Parts 26, 70 and 73 was published requiring licensees who are authorized to possess, use or transport formula quantities of SSNM to institute fitness-for-duty programs. The amended regulation is limited to licensees who are authorized to possess, use or transport unirradiated Category I material. This action was necessary to provide greater assurance that individuals who have a drug or alcohol problem do not have access to or control over SSNM.

- **Daylight Firing Qualification.** On August 31, 1993, a final rule to amend 10 CFR Part 73 was published that requires that TRT members, armed response personnel, and guards at Category I licensees be qualified for daylight firing of their assigned weapons, using the updated qualification courses specified in Appendix H of 10 CFR Part 73. This modification provides greater assurance that these security force members will possess sufficient marksmanship and weapons manipulation skills to protect Category I facilities against adversaries possessing capabilities that can be ascribed to the design basis threat specified in 10 CFR 73.1.

# Waste Management

Chapter 6

The Office of Nuclear Material Safety and Safeguards (NMSS) of the Nuclear Regulatory Commission (NRC) manages and coordinates regulation of all commercial high-level and low-level radioactive waste and of uranium recovery facilities. This chapter deals with the NRC's high-level and low-level nuclear waste programs, uranium recovery and mill tailings management, and the decommissioning of nuclear facilities, including reactors transferred to NMSS from the Office of Nuclear Reactor Regulation.

## HIGH-LEVEL WASTE PROGRAM

### Regulatory Development Activities

During the fiscal year, the NRC continued to take steps to ensure that the regulations (10 CFR Part 60) governing the safe disposal of high-level waste (HLW) are clear and complete. As part of this activity, the NRC staff undertook a rulemaking on the siting and performance requirements for the U.S. Department of Energy's (DOE's) proposed geologic repository. The proposed rule, entitled "Clarification of Assessment Requirements for the Siting Criteria and Performance Objectives," would revise 10 CFR Part 60, for the purpose of clarifying that the adequacy of DOE's investigations and evaluations will be judged in terms of their significance to compliance with the postclosure performance objectives. In addition, provisions of the rule concerning the presentation of information in DOE's license application would be completely separated from the technical criteria of 10 CFR 60.122 and moved to 10 CFR 60.21, the section that defines the required contents of the license application. The proposed rule was published in the *Federal Register* on July 9, 1993; action on a final rule is expected during fiscal year 1995.

The NRC staff also analyzed 10 CFR Part 60 to determine whether all of the repository functions dealing with the issue of pre-closure radiological health and safety were covered in sufficient depth. As a result of this analysis, the NRC staff is developing a draft proposed rulemaking, "Design Basis Events for the Geologic Repository Operations Area." The draft proposed rulemaking would clarify the relationship of 10 CFR Part 60 requirements to potential accident conditions and provide consistency among NRC regulations governing similar activities by including a "controlled-use area," and by revising the definition of "important to safety." The draft proposed rulemaking would also address an April 19, 1990 petition for rulemaking (PRM-60-3) by the DOE, requesting that 10 CFR Part 60 be amended to include quantitative dose criteria for a design basis accident. The NRC expects to publish a proposed rulemaking in the *Federal Register* for public comment in 1994.

In the Energy Policy Act of 1992, the Environmental Protection Agency (EPA) is directed to promulgate health-based standards for protection of the public from releases of radioactive materials from a repository at Yucca Mountain, Nev. As directed by the Act, the EPA has contracted with the National Academy of Sciences (NAS) to conduct a study and provide recommendations to the EPA on the appropriate technical bases for such standards. Although the NAS may consider a range of issues, its recommendations must address: (1) whether a standard based on doses to individuals is reasonable; (2) whether post-closure oversight and active institutional controls can effectively ensure that exposures of individuals will be maintained within acceptable limits; and (3) whether scientifically-supportable probability estimates of human intrusion into a repository over 10,000 years can be made.

After the EPA promulgates its standard under the Act, which must be consistent with the findings and recommendations of the NAS, the NRC will have to modify its technical criteria in 10 CFR Part 60 to conform to the new standard. At the request of the committee conducting the NAS review, the NRC presented its views on the issues before the committee and provided documentation of the history and bases of NRC regulations, as well as current NRC staff and contractor work in repository performance assessment. Two NAS Committee meetings were held in fiscal year 1993, and at least four more are anticipated in fiscal year 1994, before publication of the Committee's findings and recommendations in December 1994.

### Regulatory Guidance Activities

NRC regulatory guidance issued during this fiscal year included one final Staff Technical Position (STP) and one draft STP. STPs provide guidance to the DOE on selected topics in the form of criteria for methods acceptable to the NRC staff for demonstrating compliance with the requirements of 10 CFR Part 60. The final STP on "Geologic Repository Operations Area Underground Facility Design—Thermal Loads" was published as NUREG-1466 in December 1992. The STP provides the DOE with a methodology acceptable to the staff for demonstrating compliance with the requirement for thermal loads design criteria specified in 10 CFR 60.133(i). The staff's position is that the DOE methodology for modeling thermal loads for the repository should include evaluation and development of appropriately coupled models to account for the thermal, mechanical, hydrological and chemical processes that are induced by repository-generated thermal loads.

The availability of the draft STP, "Consideration of Fault Displacement Hazards in Geologic Repository Design," for public comment, was announced in the *Federal Register* in March 1993. The STP addressed those situations in which geologic faults of regulatory concern exist, or are assumed to exist, at the location of structures, systems, or components important to safety or important to waste isolation. Specifically, the draft STP would recognize the acceptability of designing the geologic repository to take into account the attendant effects (e.g., fault displacement) of faults of regulatory concern and would identify the information the DOE must provide if the DOE chooses to locate structures, systems or components important to safety or important to waste isolation in areas that contain faults with Quaternary-age displacement. The STP also stated that the DOE should seek early resolution, at the staff level, of fault-related design and performance issues, before submitting a license application to construct and operate a geologic repository. Action on a final STP is expected during fiscal year 1994.

## NRC/DOE Revised Procedural Agreements

On June 3, 1993, the NRC and the DOE signed the revised "Procedural Agreement between the Nuclear Regulatory Commission and the U.S. Department of Energy Identifying Guiding Principles for Interface during Geologic Site Investigation and Site Characterization" (Procedural Agreement) and the "Agreement Between the U.S. Department of Energy Office of Civilian Radioactive Waste Management and the Nuclear Regulatory Commission Division of High-Level Waste Management During Site Characterization Programs Prior to the Submittal of an Application for Authorization to Construct a Repository" (Project-Specific Agreement). The Procedural Agreement was originally signed in 1983, and the Project-Specific Agreement was originally signed in 1984. Together, these agreements provide the bases for conducting most NRC/DOE interactions. The substantive changes to the Procedural Agreement incorporated new or revised guidelines for conducting technical exchanges, site-visits, licensing and management meetings, and quality assurance audits and surveillances. Similarly, major changes in the Project-Specific Agreement included revised guidelines for preparing interaction reports, maintaining and distributing site characterization data, communications between points of contact from NRC and DOE project offices, acquisition of samples by NRC contractors from the DOE during site characterization activities, and specific NRC On-Site Licensing Representative responsibilities and authority.

## Technical Assessment Capability For Repository Licensing Reviews

The NRC staff continued work on the draft License Application Review Plan (LARP, designated NUREG-1323), the comprehensive guidance document for the NRC staff's review of a potential DOE license application to construct and operate an HLW repository. The 97 individual review plans that constitute the LARP cover the NRC requirements, in 10 CFR Part 60, for which the DOE must show compliance in its license application. The review plan topics are generally consistent with the draft Format and Content Regulatory Guide for the License Application (Regulatory Guide DG-3003). Each review plan will have a standard structure with separate sections that identify the applicable 10 CFR Part 60 requirements, and include the staff's review strategy, review procedures and acceptance criteria, implementation (interfaces and responsibilities), and examples of staff evaluation findings.

In fiscal year 1993, the NRC staff completed the first two sections of most of the 97 individual review plans, i.e., the identification of applicable 10 CFR Part 60 requirements and the description of the staff's review strategies. In addition, the more detailed review procedures and acceptance criteria, implementation, and exemplary findings were completed for two individual review plans. One important contribution of the review strategies is the staff's identification of the key technical uncertainties that it considers most important to demonstrating safe repository performance (approximately 60 key technical uncertainties). This work will also help to focus the staff's future technical prelicensing activities. Consistent with the staff's commitment to keep the public, the interested parties, and the DOE informed of the staff's activities, the first draft of the LARP is being published in fiscal year 1994. Subsequently, revisions to the draft LARP will be published periodically to incorporate new review plan sections and changes to previously published sections.

Supporting preparation of the LARP was the NRC staff's continued progress in developing an independent performance assessment (IPA) capability to review performance assessments by the DOE for an HLW repository. The DOE intends to use performance assessments in its license application, to show compliance with 10 CFR

Part 60, including, by reference, the generally applicable environmental standard to be issued by EPA. For its part, the staff's independent IPA capability will strengthen its ability to review DOE's performance assessments and other aspects of the DOE HLW program. In particular, the IPA program will aid in the development of requirements and guidance regarding output and methodologies related to the DOE performance assessment analysis, besides refining the NRC's review strategy.

Specifically, in fiscal year 1993, the staff completed the second iteration (designated Phase 2) of the demonstration of its IPA capability, using more refined predictive models and treating a more comprehensive set of phenomena and scenarios. Although any sitespecific calculations are limited by preliminary models, simplifying assumptions, and a shortage of data, accomplishments of IPA Phase 2 included: addition of a dose assessment methodology; treatment of additional scenarios; evaluation of carbon-14 releases; more refined treatment of waste dissolution, near-field transport, and waste package failure; and more extensive treatment of radionuclide transport. Documentation of the results of the IPA Phase 2 analysis will be published as NUREG–1464, in fiscal year 1994. Also in fiscal year 1993, as part of the demonstration of its independent performance assessment capability, the staff evaluated the applicability of expert judgments to performance assessments and to other aspects of the DOE HLW program.

Other activities continued in fiscal year 1993 that will support the LARP by developing various analysis methods. Particularly in the area of tectonics, analysis methods for testing alternative conceptual models of fault displacement were developed. In seismology, continued efforts were directed towards testing computer codes in the analysis of seismic hazards at a proposed geologic repository site. For the Engineered Barrier System (EBS), work continued in developing an EBS performance assessment computer modeling code and in reducing uncertainties related to the EBS performance requirements of 10 CFR Part 60.

## Licensing Support System Advisory Review Panel

The Licensing Support System (LSS) is an information management system set up to organize the documentary material generated by the DOE, the NRC, the State of Nevada and other parties or prospective parties to the licensing proceedings related to DOE's high-level radioactive waste repository. The system is currently under the purview of the Deputy Director of NRC's Office of Information Resources. The LSS Advisory Review Panel, was created to provide advice to potential users of the system with respect to the design, development, operation and maintenance of the system. The panel includes representatives of the State of Nevada, local government entities, the National Congress of American Indians, the nuclear industry, and the DOE. Representatives of other Federal agencies having significant experience in developing automated information management systems also serve on the panel. (See Appendix 2 for a listing of the LSS Advisory Panel members and coalition representatives.)

## Yucca Mountain Site-Characterization Reviews and Interactions

The NRC staff continued pre-licensing interactions with the DOE and provided guidance on DOE's ongoing site-characterization activities. DOE's activities at the Yucca Mountain site in Nevada continued to increase in fiscal year 1993. Of particular significance was the start of work by the DOE, in April 1993, on the excavation of the exploratory studies facility (ESF) north portal ramp, a 25-foot diameter tunnel. Using drill and blast methods, the DOE constructed a 200-foot-long tunnel that will serve as a staging area for a tunnel-boring machine, which is due to arrive at the site in spring of 1994.

Reflecting the increased activity at the site, the NRC staff engaged in numerous interactions with the DOE. Among these were seven meetings, six technical exchanges, four workshops, and one site-visit. Throughout the same period, the NRC On-Site Licensing Representatives kept a continuous presence to observe the DOE's ongoing site-characterization work and to provide an interface with the DOE, the State of Nevada, and affected units of local government.

Besides the technical exchanges and interactions, the NRC staff also continued its pre-licensing review of a variety of DOE's site-characterization activities and reports, including DOE semi-annual progress reports, topical reports, revisions to its license application annotated outline, and study plans. The NRC staff also continued to make progress in its follow-up on the DOE's resolution of staff concerns in its Site Characterization Analysis (SCA), dated August 1989.

On November 2, 1992, the NRC staff notified the DOE that the last remaining SCA objection, related to the adequacy of the DOE's ESF design control process and ESF design, was lifted. Lifting of the objection was considered to be a major milestone in the DOE program.

The staff's reviews of DOE site-characterization study plans continued during fiscal year 1993. Of the 56 new and 26 revised study plans submitted by the DOE for the NRC staff's review, the staff has, to date completed reviews of 50 plans; deferred review of 11, pending receipt of needed revisions from the DOE; and concluded that eight of the revised study plans submitted by the DOE needed no review, based on the limited scope of the revisions. The remaining 13 study plans remain under review by the staff.

The staff has identified no reasons to object to start-up of activities related to any reviewed study plan but has conveyed comments to the DOE related to nine of the study plans reviewed.

On March 3, 1993, the DOE submitted, for NRC staff's review, a topical report entitled, "Evaluation of the Potentially Adverse Condition of Extreme Erosion during the Quaternary Period at Yucca Mountain, Nevada." The NRC staff conducted an acceptance review of the report and found it to be acceptable for technical review, which will continue into fiscal year 1994.

In June and August 1993, the DOE submitted information related to a proposed topical report on "Methodology for Seismic Hazards Assessment at Yucca Mountain." The NRC staff reviewed the information and found that it is acceptable for a topical report.

## Interactions with Affected Governmental Units and Indian Tribes

The State of Nevada, representatives of affected units of local government, and other interested parties continued to participate in the technical exchanges and meetings between the NRC and the DOE. These participants also continued to receive notification of upcoming NRC/DOE HLW meetings, as well as NRC Advisory Committee on Nuclear Waste meetings. Furthermore, the NRC staff continued its active role in ensuring that these parties receive all correspondence and publicly-available NRC reports regarding the HLW program.

## Quality Assurance Activities

The NRC staff conducted a variety of activities in this area. The staff continued to review DOE and DOE contractor quality assurance (QA) plans and procedures (document reviews), to evaluate DOE's effectiveness in auditing its program to identify and correct problems in program implementation, and to evaluate DOE contractor effectiveness in implementing QA programs. NRC staff work in this area, for fiscal year 1993, included review of revisions to previously accepted QA plans. In addition, as part of its evaluation of DOE's effectiveness in auditing and of DOE contractor effectiveness in QA program implementation, the NRC staff observed DOE audits conducted at all major DOE contractor organizations participating in the site-characterization program for the Yucca Mountain Project. Formal NRC staff reports were issued for all of the audits observed, and the DOE will be required to respond to those reports that indicate that improvements are needed.

## Center for Nuclear Waste Regulatory Analyses

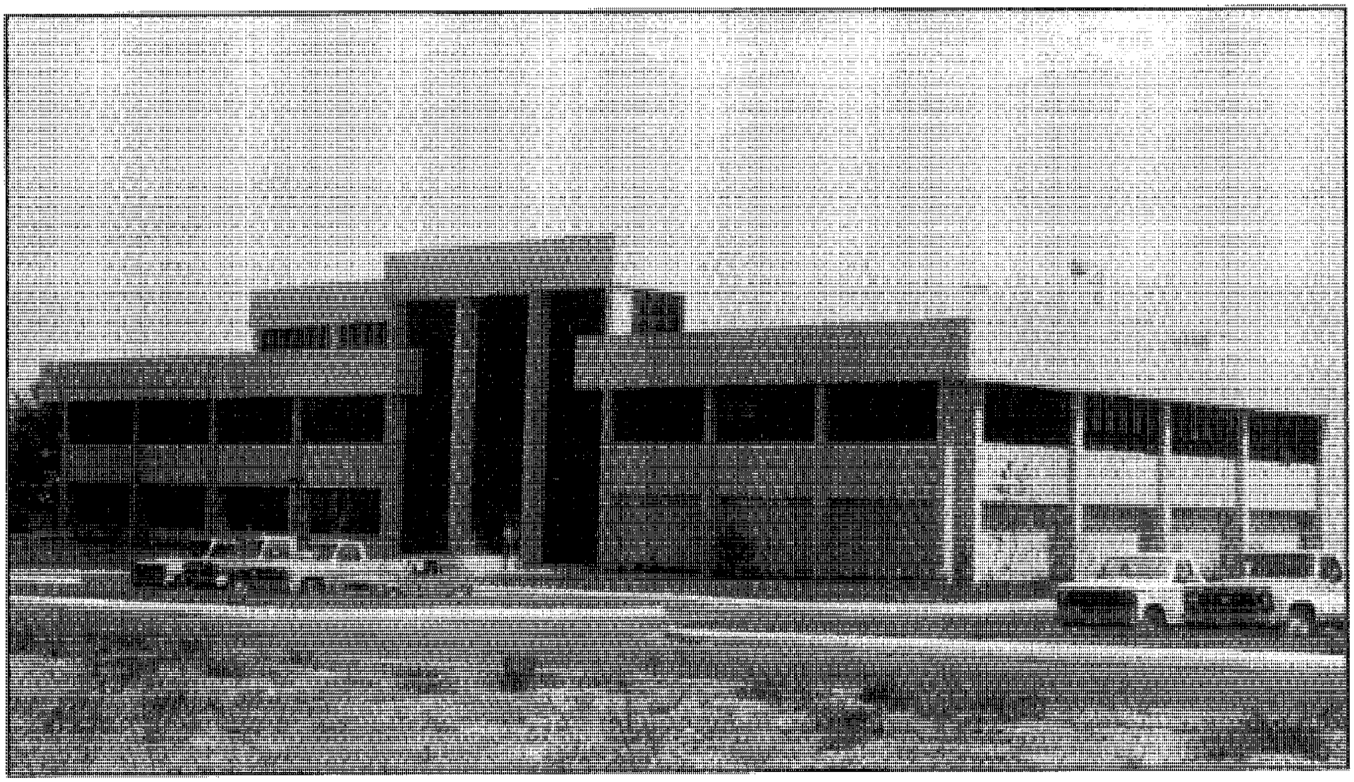
The contract with the Center for Nuclear Waste Regulatory Analyses (CNWRA), an NRC contractor, was renewed on October 15, 1992, and the CNWRA completed its sixth year of operation in October 1993. The CNWRA provides a broad range of support to NMSS and to the Office of Nuclear Regulatory Research for the HLW program. CNWRA staff are located at the Southwest Research Institute in San Antonio, Tex., and at the Washington Technical Support Office in Arlington, Va.

Together with the NRC staff, the CNWRA continued to develop and implement a computer-assisted "systems engineering approach," called the Systematic Regulatory Analysis (SRA), for the development of the staff's regulatory documents. The purpose of SRA is to identify and reduce uncertainties, to select strategies and methods for determining compliance with NRC regulatory requirements, and to define issues in licensing a HLW geologic repository. This approach is being taken by the NRC to ensure that all of its HLW activities under the Nuclear Waste Policy Act (NWPA) are planned, integrated, implemented, documented and managed as thoroughly and effectively as possible.

The CNWRA's special expertise also supports the NRC staff in such areas as review of study plans and design reports; NRC/DOE pre-licensing meetings and technical exchanges; QA observation audits; technical support to NRC rulemaking and regulatory guidance development programs; the development of analysis methods (e.g., computer codes); and research. Activities in the research program include: unsaturated mass transport (geochemistry); thermohydrology; seismic rock mechanics; integrated waste package experiments; stochastic analysis of flow and transport; geochemical analogs; modeling sorption mechanisms; regional hydrology; performance assessment issues; volcanism/seismology (review); volcanism (field); and tectonic analysis.

## Nuclear Waste Negotiator

Former Idaho Congressman Richard Stallings has been nominated by President Clinton to be the next Nuclear Waste Negotiator. The NRC staff has maintained its relationship with the Negotiator's staff and continued to support the Office of the U.S. Nuclear Waste Negotiator by responding to requests for information and meeting with interested parties to explain the NRC's regulatory responsibilities.



**The Center for Nuclear Waste Regulatory Analyses (CNWRA), an NRC contractor, completed its sixth year of operation in October 1993. The Center supports various NRC elements concerned with high-level nuclear waste handling and disposal. CNWRA staff are located at the Southwest Research Institute in San Antonio, Tex., shown above, and at the Washington Technical Support Office in Arlington, Va.**

## LOW-LEVEL WASTE MANAGEMENT

The main objective of the NRC's low-level waste (LLW) program is to provide adequate protection of public health and safety and the environment in the management of LLW, in conformance with the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA).

### Regulations and Guidance

**Regulatory Framework to Apply to On-site Storage of Low-Level Radioactive Waste (LLW) Beyond January 1, 1996.** During 1992, the Commission proposed to amend its regulations for reactor, material, fuel cycle, and independent spent fuel storage licensees to address on-site storage of LLW. A proposed rule was published in the *Federal Register* on February 2, 1993 (57 FR 6730), containing the procedures and criteria that would apply to on-site storage of LLW beyond January 1, 1996. The Commission took this action because of potential health and safety concerns associated with increased reliance on indefinite, long term, on-site storage of LLW. The proposed rule also is intended to support the goals that have been established by the LLRWPAA. Comments were requested on the proposed rule, and on strategies and options that the Commission might pursue in addition to the proposed rulemaking, that would encourage the States and Compacts to move forward with development of LLW disposal facilities. The public comment period expired on April 5, 1993, and 55 comment letters were received. Some of the public comments concerning the rule, and staff analysis of the rationale for the rule, have raised concerns that the rule may not adequately accomplish its objectives. Final action on the rulemaking is expected in 1994. The staff also summarized the comments dealing with other options the Commission might take to encourage the development of new disposal facilities and presented them in a memorandum to the Commission.

**Standard Review Plan.** During fiscal year 1993, the Low-Level Waste Management and Decommissioning (LLWM) staff prepared Revision 3 to the "Standard Review Plan (SRP) for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility" (NUREG-1200). The SRP provides guidance to regulatory personnel performing safety reviews of applications for licenses to construct and operate an LLW disposal facility. The draft revisions cover:

- The Licensing Process
- Design Considerations for Normal and Abnormal/ Accident Conditions
- Guidance on Soil Cover Systems Placed over LLW
- Receipt and Inspection of Waste
- Waste Handling and Interim Storage
- Waste Disposal Operations
- NRC Staff Recommendation for Filling Void Spaces Around Waste ContainersEmplaced in LLW Land Disposal Excavations
- Transport of Radioactive Material
- Occupational Radiation Exposures
- Radionuclide Inventories
- Radiation Protection Design Features and Operating Procedures
- Radiation Protection Program.

**Performance Assessment Guidance.** The staff is carrying out a program to develop LLW performance assessment (PA) guidance and to enhance staff expertise in the area of performance assessment. The Low-Level Waste Performance Assessment (LLWPA) program plan was developed and is being implemented through the combined efforts of staff from several organizations, who are members of a Performance Assessment Working Group (PAWG).

During fiscal year 1993, staff efforts focused on two main activities: (1) developing a draft branch technical position (BTP) on LLW performance assessment, that addresses important issues in PA modeling; and (2) gaining experience with integrated PA modeling through an NRC-developed test case model. These efforts will provide license applicants with additional guidance on acceptable approaches for evaluating the long term performance of an LLW disposal facility and will further improve NRC's ability to provide technical assistance to Agreement States on LLWPA issues.

The principal guidance objective of the BTP is to provide the applicant with an acceptable methodology for performing technical analyses required in 10 CFR 61.13 to demonstrate compliance with the 10 CFR 61.41 performance objectives. This includes giving: (1) general guidance on an acceptable PA strategy that integrates site characterization and PA modeling; and (2) specific guidance on implementing the NRC's performance assessment methodology. The objectives of each of the main sections of the BTP are as follows: (1) to define LLWPA in the context of the 10 CFR Part 61 regulatory requirements for LLW facility performance; (2) to describe an overall strategy for conducting PA modeling activities; (3) to address important technical policy issues concerning interpretation and implementation of 10 CFR Part 61 technical requirements; and (4) to provide guidance on acceptable modeling approaches for addressing technical issues regarding processes controlling LLW facility performance.

The staff has continued to cooperate in international efforts concerning LLW disposal in fiscal year 1993. The staff has participated, as a consultant to the International Atomic Energy Agency (IAEA), in the Coordinated Research Program (CRP) on the Safety Assessment of Near-Surface Radioactive Waste Disposal Facilities (NSARS). The CRP is conducting test-case programs similar to NRC's PA test case modeling. Preliminary results of the second test-case problem, which is based upon NRC's test case problem data base, were presented at a CRP meeting hosted by the NRC staff in Augusta, Ga., on October 19–23, 1992. NRC staff also participated in a meeting in Vienna, Austria, to plan continued development of the NSARS test case.

**Topical Report Reviews.** The LLWM staff continued to review technical topical reports that examine high integrity containers (HICs) and stabilization technologies for those wastes that require processing or special packaging to meet stability requirements in 10 CFR Part 61, "Licensing Requirements for Land Disposal Of Radioactive Waste." Following topical report review and approval by staff, licensees may reference the processes described in the topical report and incorporate the technology for use in their operations.

In 1993, staff continued its review of three topical reports concerning HIC technology and accepted for review topical reports on cement solidification and on a waste analysis software code. Staff approved a vinyl esther resin solidification topical report on July 12, 1993, and a bitumen solidification topical report on April 19, 1993. LLWM staff also independently reviewed the DOE's cement stabilization of LLW, produced as part of the West Valley Demonstration Project, and provided consultation to the DOE concerning this effort.

**Concentration Averaging Guidance.** 10 CFR Part 61 establishes a waste classification system based on the concentration of specific radionuclides contained in the waste. Within the regulation, it is stated that "the concentration of a radionuclide [in waste] may be averaged over the volume of the waste, or weight of the waste if the units [on the values tabulated in the concentration tables] are expressed as nanocuries-per-gram." On June 26, 1992, NRC licensees were sent copies of a proposed "Concentration Averaging and Encapsulation Technical Position, Revision in Part," on which comments were solicited. A notice of availability of the proposed Technical Position was also published in the *Federal Register* on June 30, 1992 (57 FR 29105). In response, 19 comment letters were received suggesting the need for further expansions of, and several modifications to, the Technical Position. Consideration of these comments has resulted in modifications and a further expansion of the Technical Position. The modified Technical Position was issued for comment in late fiscal year 1993.

## Technical Assistance to the States

During fiscal year 1993, the LLWM staff continued to support the NRC Office of State Programs (OSP) in providing technical assistance to the States as they implement their plans for LLW disposal facility development and licensing. Technical assistance to States included:

- Meeting with representatives of the North Carolina Division of Radiation Protection (DRP) to discuss LLW disposal facility PA and NRC staff review comments on a DRP License Application Review Management Plan.

- Support to OSP in conducting a program review of the Texas Agreement State Program.

- Meeting with representatives of the Maine Low-Level Radioactive Waste Authority, to discuss NRC staff review comments on the Authority's LLW Disposal Facility Conceptual Design Report and Quality Assurance Plan.

- Participation in meetings of the LLW Forum and the Technical Coordination Committee; these are groups of State and Compact officials which meet to discuss areas of common interest in the policy and technical areas, respectively.

- Presentation on the NRC's site suitability requirements for LLW facilities at an initial meeting of the Ohio Blue Ribbon Commission on Siting Criteria.

- Participation in meetings on LLW storage with LLW generators in Maryland, Pennsylvania, Ohio, Nebraska, and New York.

**Low-Level Waste Regulators and Uranium Recovery Workshop.** On July 28–30, 1993, NRC staff hosted the annual Agreement State Regulatory Workshop in Rockville, Md. The purpose of the meeting was to enable the States and NRC to exchange information of common interest on the licensing of LLW disposal facilities. LLW topics discussed included the status of the staff's development of PA guidance, recent changes to the SRP for licensing LLW disposal facilities, LLW storage licensing inspection experience, and EPA interface activities. Uranium Recovery topics discussed included license termination, site transfer and long term care, and closure of the uranium recovery field office. NRC Chairman Ivan Selin gave a presentation entitled "Regulation of LowLevel Waste in An Uncertain Regulatory Environment."

## Cooperation with Other Federal Agencies

During 1993, the NRC continued cooperation with other Federal agencies in resolving issues associated with LLW management, decommissioning of licensed nuclear facilities and formerly used sites, and emissions of radionuclides to the general environment. These efforts have primarily involved the Environmental Protection Agency (EPA) and the Department of Energy (DOE), but they have also included other Federal and State regulatory agencies that share interests in the regulation of radioactive materials and protection of the environment.

**Cooperation with the EPA.** Cooperation with the EPA has focused on three principal areas over the last year, including "risk harmonization," regulation of air emissions of radionuclides, and development of radiological criteria for decommissioning. The agencies also cooperated in evaluating a range of related activities involving remediation, drinking water, medical waste, emergency planning, groundwater protection, uranium mill tailings, naturally occurring and accelerator-produced radioactive materials, hazardous waste, LLW, and other issues of mutual interest. The cooperative activities are generally governed by the March 1992 General Memorandum of Understanding (MOU) between the agencies on regulation of radionuclides in the environment.

In late 1992 and early 1993, the NRC and the EPA completed a comparison of risk assessment approaches used in a variety of regulatory programs that address both radiological and non-radiological hazards. Based on this comparison, the NRC/EPA cooperative effort progressed by comparing risk goals and risk management approaches used in the same programs. The NRC staff initiated the development of a White Paper on risk harmonization, scheduled to be completed in 1994, which will provide the foundation for further cooperative efforts to reconcile risk assessment and management approaches.

Regarding emissions of radionuclides to the air, the NRC and the EPA continued to cooperate in determining whether NRC's existing regulatory program under the Atomic Energy Act protects the public with an ample margin of safety, as provided under the Clean Air Act. The agencies had previously completed (in 1992) two staff-level memoranda of understanding on potential rescission of EPA's emissions standards in Subparts T and I of 40 CFR Part 61. Cooperation in 1993 included completion of an NRC Regulatory Guide (Regulatory Guide 8.37) and Inspection Procedures on evaluating whether radiation protection programs at licensed materials facilities ensure that air emissions of radionuclides are as low as is reasonably achievable.

The EPA also proposed and promulgated amendments to its standards for uranium mill tailings disposal in 40 CFR Part 192, and the NRC proposed conforming amendments to its requirements in Appendix A of 10 CFR Part 40. Further, NRC and affected Agreement States completed license amendments approving reclamation plans and schedules for almost all of the non-operational uranium mills. These and other activities have been conducted in accordance with a settlement agreement that resolved litigation against the EPA by environmental and industry groups. Collectively, these cooperative efforts are expected to provide a sufficient basis for the EPA to complete rescission of its radionuclide air emission standards in Subpart T of 40 CFR Part 61 for uranium mill tailings disposal.

**Enhanced Participatory Rulemaking on Decommissioning Criteria.** During the report period, the NRC conducted the initial phases of the Enhanced Participatory Rulemaking on radiological criteria for decommissioning. This effort included conducting seven public workshops around the country, from January through May 1993, to discuss the issues associated with development of the criteria. The EPA and the NRC cooperated fully in planning and conducting the workshops. In July 1993, the NRC also conducted, with EPA participation, a series of eight meetings in four cities on the scope of the Generic Environmental Impact Statement that will support the Enhanced Participatory Rulemaking. Further, the agencies have been actively cooperating by exchanging information and jointly evaluating technical methods necessary to support and implement the radiological criteria. The agencies actively participated in the Interagency Steering Committee on Residual Radioactivity, which was established at the request of the Office of Management and Budget.

**Cooperation with the DOE.** Cooperative efforts between the NRC and the DOE during 1993 continued to focus primarily on resolving issues associated with the management of LLW and environmental restoration programs. Under the LLRWPAA, the DOE is responsible for disposing of the so-called "Greater-Than-Class C" wastes in an NRC-licensed disposal facility. The agencies also have cooperated in developing procedures to request DOE's assistance in picking up abandoned and other radioactive materials judged by the NRC to pose a health and safety concern if left in the long term possession of certain licensees.

In fiscal year 1993, the NRC staff was active in several national performance assessment projects, in association with the DOE, the U.S. Geologic Survey (USGS), and through national meetings. NRC staff participated in the DOE Performance Assessment Task Team (PATT) meetings. The purpose of PATT is to discuss and coordinate LLW performance assessment activities at DOE sites, to identify and resolve technical issues, to alert the DOE to policy issues, and develop revised guidance for the disposal of DOE LLW. NRC staff also participated in the DOE Peer Review Panel, which reviews, evaluates, and determines the technical acceptability of LLW performance assessments for DOE sites and provides input to DOE Headquarters. Furthermore, in May 1993, under an MOU between the NRC and the USGS, the two agencies jointly sponsored a technical workshop to exchange information and identify approaches for resolving earth science issues related to radioactive waste disposal and decommissioning.

Further, the NRC, the DOE and the EPA continued cooperation in evaluating alternative approaches for modeling environmental transport of radionuclides. The agencies jointly developed guidance for project managers on the selection and use of groundwater models in support of decommissioning or remediation projects in all three agencies, including EPA's Offices of Radiation and Indoor Air and of Solid Waste and Emergency Response.

### International Cooperation

The NRC's LLWM staff reviewed Radioactive Waste Safety Series (RADWASS) documents intended to offer guidance to International Atomic Energy Agency (IAEA) Member States for management of radioactive waste. LLWM staff reviewed these documents for consistency with U.S. LLW policy, and conveyed NRC views to the IAEA through the Office of International Programs and the DOE. LLWM staff reviewed and commented on the following RADWASS documents:

- The Principles of Radioactive Waste Management
- Establishing a National Radioactive Waste Management System
- Decommissioning of Nuclear Facilities
- Near Surface Disposal of Radioactive Waste
- Siting of Near Surface Disposal Facilities
- Classification of Radioactive Waste
- Pre-Disposal Management of Radioactive Waste.

## URANIUM RECOVERY AND MILL TAILINGS

The NRC's uranium recovery and mill tailings program licenses and regulates uranium mills, commercial *in-situ* solution mining operations, uranium extraction research and development projects, as well as disposal of uranium mill tailings and wastes. This task requires the detailed health, safety, and environmental review and inspection of facilities to provide reasonable assurance of safe operation; the development of NRC's regulatory guidance to implement EPA standards for regulating mill tailings; and the site-by-site approval of licensee plans for disposal of mill tailings. The NRC also evaluates and concurs in DOE remedial action projects for inactive uranium mill tailings sites and associated properties, as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA). Functions within the program have been implemented at both NRC Headquarters and the NRC's Uranium Recovery Field Office in Denver, Colo.

Of the 28 NRC-licensed uranium recovery facilities, 19 are uranium mills, 3 are either heap leach or other byproduct recovery operations, 5 are commercial *in-situ* solution mining facilities, and 1 is a commercial laboratory. At the close of the fiscal year, three commercial *in-situ* mining operations were in operation, and two were under construction. No conventional uranium mills were in operation, three were in standby, and the remainder were in decommissioning and reclamation. Because of the low market price of uranium, no new conventional mills are expected to be licensed in the near future, and the three standby mills are likely to resume operations only for short runs. However, *in-situ* solution mining facilities are expected to remain moderately active, with one application currently under licensing review and two more applications for licenses forecast during fiscal year 1994. Over the next few years, much of the casework confronting the Uranium Recovery Program will be in the area of remedial activity for the shutdown facilities, including decommissioning of mills, reclamation of mill sites and tailings disposal areas, remediation of groundwater contamination, and the environmental assessment of such activities.

Early in fiscal year 1993, the Commission decided to close the field office in Denver and consolidate its licensing activities with those at NRC Headquarters and its inspection activities with those of Region IV (Dallas). During fiscal year 1993, the NRC began to implement closure and transition in a manner that entailed minimal impact to ongoing inspection, licensing, and policy development programs. The NRC staff met twice during the last quarter of fiscal year 1993 with representatives of the industry and States, and plans to continue to meet approximately every other month to provide oversight, and review the status of actions associated with the field office closure/transition.

## Regulatory Development and Guidance

During the report period, the NRC continued efforts to develop regulatory guidance to implement standards dealing with groundwater contamination. The use of Alternate Concentration Limits (ACLs) for contaminants in groundwater has been an area of interest to both the licensed mills and the DOE inactive mill tailings remediation program. ACLs are one of three options (along with maximum concentration limits and background levels) for demonstrating compliance with EPA groundwater protection standards. NRC staff issued a draft technical position on ACLs for uranium mills in June 1988. Workshops were held in October 1988 and in December 1990. The NRC received comments on the draft technical position, revised the draft technical position, and in December 1992, provided a copy to the EPA. In February 1993, the EPA provided comments to the NRC, which addressed the issue of health risk criteria. Since that time, the NRC has been working with the EPA to resolve the issue.

In the area of UMTRCA Title I work, the NRC completed updates of guidance related to DOE's Uranium Mill Tailings Remedial Action (UMTRA) Project surface remedial action during the report period. These included revisions of NRC Inspection Manual Chapter 2620, "On-Site Construction Reviews of Remedial Action at Inactive Uranium Mill Tailings Sites," which were issued in February, and the SRP for review of DOE's Remedial Action Plans (RAPs), which was issued in June.

## Licensing and Inspection Activities

In the fall of 1989, the NRC received an application from Envirocare of Utah, Inc., for a license to dispose of commercial uranium and thorium mill tailings and other 11e.(2) byproduct material at its existing radioactive disposal facility in Clive, Utah. Early in 1991, a *Federal Register* notice was published announcing receipt of the application, describing the regulatory requirements to be applied in the licensing review, and giving notice of the opportunity for a public hearing. The safety portion of the licensing review concluded with the issuance of the Final Safety Evaluation Report (SER) in June 1993, and an SER supplement in September 1993. The environmental portion of the licensing review was completed with the issuance of the Final Environmental Impact Statement in September 1993. The license for Envirocare of Utah, Inc., was issued on November 19, 1993.

In fiscal year 1993, the Denver field office staff performed 36 inspections of uranium recovery facilities, issued new licenses for a commercial *in-situ* solution mining operation for commercial laboratories, three license renewals, 84 license amendments, and five mill tailings reclamation plan amendments. In addition, 121 environmental and radiological monitoring report reviews were completed and pre-licensing guidance was provided to two potential license applicants.

## Remedial Action at Inactive Sites

There were 24 abandoned uranium mill tailings sites designated, under the UMTRCA, to receive remedial action by DOE. UMTRCA requires that the NRC concur with DOE's selection and performance of remedial action, so that the action meets appropriate standards promulgated by the EPA. The DOE has established the UMTRA Project to implement the remedial actions. These sites will be held by the DOE under an NRC general license, when all remedial work is completed.

During fiscal year 1993, NRC staff completed 53 review actions pursuant to its responsibilities at sites under the UMTRCA. These included 7 RAP reviews, 4 inspection plan reviews, 11 RAP modification reviews, 9 other site specific reviews, 3 Completion/Certification Report reviews, and 4 reviews of generic items. The staff prepared two Technical Evaluation Reports, documenting its review of DOE's remedial action selection for the Gunnison (Colo.) and Maybell (Colo.) sites, and two Completion Review Reports documenting its review of DOE's remedial action completion for the Lakeview (Ore.) and Lowman (Idaho) sites.

At the beginning of the fiscal year, the staff completed review of and formally concurred in DOE's "Generic Guidance for Long-Term Surveillance Plans" (LTSP). The submittal of a site LTSP to the NRC for approval is one of the final actions by the DOE before the site comes under the NRC general license for long term care, under 10 CFR 40.27. Following concurrence in the generic document, the DOE submitted, and NRC staff completed review of, seven LTSPs. One of these reviews resulted in final acceptance of the LTSP for the Spook (Wyo.) site, which made this the first UMTRA Project site subject to the NRC general license.

In support of the UMTRA Project casework, the staff visited many of the sites. Inspections of remedial action in progress were conducted at the Gunnison (Colo.), Falls City (Tex.), Mexican Hat (Utah), Grand Junction (Colo.), Ambrosia Lake (N.M.), Rifle (Colo.), and Monument Valley (Ariz.) sites. NRC technical staff also conducted site-visits associated with RAP reviews at the Maybell (Colo.), Slick Rock (Colo.) and Naturita (Colo.) sites.

The preliminary activities for the groundwater remediation phase of the UMTRA Project continued during fiscal year 1993. As part of NRC's role as a cooperating agency in DOE's development of the Programmatic Environmental Impact Statement for this phase of the remedial program, NRC staff participated in a groundwater technical working group established to develop and review programmatic documentation. Development of an NRC SRP and a Standard Format and Content Guide for groundwater remedial action plans was initiated. These guidance documents will be used to ensure compliance with the groundwater cleanup aspects of the uranium mill tailings regulations in 40 CFR Part 192, Subparts A to C.

# DECOMMISSIONING OF NUCLEAR FACILITIES

The NRC staff continued to develop the guidance that both the NRC licensing staff and licensees will need to implement the Commission's regulations with respect to the decommissioning of nuclear facilities. The staff is also performing decommissioning reviews for both materials facilities and nuclear reactors.

## Materials Decommissioning

Several hundred NRC materials licenses are terminated each year. The majority of NRC-licensed operations result in little or no contamination of buildings or soil, and decommissioning actions leading to the termination of most licenses normally proceed in a routine fashion. Nonetheless, over the past several years, the NRC has recognized the need to strengthen its decommissioning program, particularly for non-routine cases. These non-routine cases involve sites where buildings, former waste disposal areas, large piles of tailings, groundwater, and soil are contaminated with low levels of uranium or thorium (source material) or other radionuclides. Consequently, they present varying degrees of radiological hazard, cleanup complexity and associated cost.

The NRC developed the Site Decommissioning Management Plan (SDMP) in 1990, to focus efforts on identifying non-routine decommissioning cases and ensuring that generic, as well as case-by-case, issues affecting the timely decommissioning of these contaminated sites receive the appropriate level of management attention. The SDMP is updated annually. The most recent update was issued in June 1993, and contains the following elements:

(1) Criteria for listing a contaminated site in the SDMP (there are currently 46 sites listed in the SDMP).

(2) Schedules and resources needed for NRC oversight of contaminated site cleanup.

(3) Policy issues requiring resolution for SDMP implementation and minimization of problems with future contaminated sites.

The SDMP has been effective in ensuring coordination and resolution of some policy and regulatory issues affecting site decommissioning.

Over the last year, the decommissioning of the Budd Company site was completed and the site was removed from the SDMP list. The Kerr-McGee, Cimarron plant plutonium license was terminated. The AMAX and UNC Recovery Systems sites have also completed decommissioning, and the licenses will be terminated, and the sites removed from the SDMP list, once pending administrative issues are resolved. Remediation activities at the Chevron site in Pawling, N.Y., the Texas Instruments site in Attleboro, Mass., and the Old Vic site in Cleveland, Ohio, were also completed.

Sequoyah Fuels Corporation (SFC) notified the NRC, in February 1993, of its intentions to terminate all licensed activities involving materials authorized under its license. In a July 1993 follow-up letter, SFC confirmed that all licensed operations had ceased as of July 6, 1993. SFC has entered into a Consent Order with the EPA. Because of the EPA's interest in the SFC site, the EPA and NRC staffs have crafted a draft MOU to provide a basis for formulating an efficient and effective working relationship between the two Federal agencies.

Ongoing decommissioning activities at other sites include site characterization, development of site decommissioning plans, site remediation, and termination surveys. Over the next year, the NRC staff expects to complete the review of 30 characterization reports, 12 remediation plans, nine final survey reports, and eight confirmatory reports associated with sites listed on the SDMP.

The SDMP has also focused resources on the resolution of generic issues. The NRC staff initiated an Enhanced Participatory Rulemaking to establish residual contamination criteria for decommissioning. During fiscal year 1993, this effort involved the solicitation of public input on various approaches for addressing decommissioning criteria development through a series of workshops and Environmental Impact Statement scoping meetings held across the nation. NRC staff also published, in the *Federal Register*, a proposed rule, "Timeliness in Decommissioning of Materials Facilities," that would establish timeframes within which the decommissioning of materials facilities should be completed after operations have ended. NRC staff also published, in the *Federal Register*, a final rule on decommissioning record-keeping that would ensure that needed licensee records are maintained to facilitate decommissioning activities.

In November 1992, NRC staff sponsored an SDMP workshop for SDMP licensees, States, and interested parties. The workshop provided information on the SDMP program, the decommissioning regulatory process, and some of the decommissioning experiences of individual licensees.

## Reactor Decommissioning

The LLWM staff currently has project management responsibility for eight commercial power reactors undergoing decommissioning. The Shoreham (N.Y.) nuclear power plant (N.Y.) and the Fort St. Vrain (Colo.) nuclear power plant are undergoing dismantlement (DECON decommissioning). The LaCrosse (Wis.), Peach Bottom Unit 1 (Pa.), Vallecitos (Cal.), and Humboldt Bay Power Plant Unit 3 (Cal.) facilities have boiling water reactors (BWRs) in long term storage (SAFSTOR decommissioning). These reactors have been defueled and their spent fuel is being stored on-site in the plants' spent fuel pools, with the exception of Peach Bottom and Vallecitos, whose fuel has been returned to the DOE). Fermi Unit 1 (Mich.), with a sodium-cooled reactor, is also in SAFSTOR decommissioning with its fuel having been returned to the DOE. The Pathfinder (S.D.) facility, a BWR, was partially decommissioned in 1971. All fuel was shipped off-site as a part of the partial decommissioning. Following the partial decommissioning, the plant's Part 50 license was terminated and the plant's existing Part 30 license was issued. A license amendment was issued, in 1992, authorizing the unrestricted release of Pathfinder's fuel handling and reactor buildings. Demolition of Pathfinder's reactor building was approved by the Low-Level Waste Management and Decommissioning staff in November 1992. Pathfinder completed decommissioning in 1993.

The Shoreham plant, containing a BWR, had, at the time of its final shutdown in June 1989, operated the equivalent of only two effective-full-power days. The DECON decommissioning (i.e., the immediate dismantlement and decontamination) at Shoreham has been confined primarily to the reactor, radwaste, and turbine buildings. The reactor at the plant has been segmented and removed from the reactor building and shipped off site for disposal. The slightly irradiated fuel is being stored temporarily in the on-site spent fuel pool. The Long Island Power Authority (LIPA), the Shoreham licensee, has entered into a contract with the Philadelphia Electric Company under which the latter will purchase the slightly irradiated Shoreham fuel for use in the Limerick Unit 1 (Pa.) reactor. Fuel transfers to Limerick Unit 1 began in September 1993, and LIPA anticipates that the transfers will be completed in May 1994. Shoreham dismantlement is approximately 75 percent complete. The primary activities at the plant in the near future will be fuel transfers (to Limerick Unit 1) and work related to the termination survey. Before implementing the termination survey, the Shoreham plant and its environs were evaluated to identify areas of the facility that would be covered in the termination survey. The reactor, radwaste, turbine, and control buildings, along with buildings, facilities, and grounds within the secured area fence, are included in the licensee termination survey.

Fort St. Vrain (Colo.) is a high-temperature gas-cooled reactor (HTGR) that operated from January 1974 to August 1989. The DECON decommissioning plan (immediate dismantlement and decontamination) for Fort St. Vrain was approved in November 1992. Its spent fuel has been transferred to an on-site independent spent fuel storage installation. The licensee, Public Service Company of Colorado, has begun to dismantle the plant and, to date, the 1,320-ton concrete top head has been segmented and removed. Removal of 1,770 activated graphite components is in progress. The company has shipped, for disposal or volume reduction, approximately 73,000 $ft^3$ of materials containing approximately 55,000 curies of radioactivity. The LLWM staff is preparing to initiate its review and evaluation of the Fort St. Vrain final survey plan.

The LLWM staff exercises routine project management oversight over the plants in SAFSTOR (LaCrosse, Humboldt Bay, Peach Bottom Unit 1, Fermi Unit 1, and Vallecitos), and NRC Regional Office conduct regularly scheduled inspections at these plants.

Rancho Seco (Cal.), Yankee Rowe (Mass.), San Onofre Unit 1 (Cal.), and Trojan (Ore.) have been prematurely shut down. Issuance of the Rancho Seco decommissioning Order is awaiting resolution of issues before the Atomic Safety Licensing Board. The LLWM staff is providing pre-decommissioning support to the NRC Office of Nuclear Reactor Regulation for reactors that have prematurely shut down. Yankee Rowe's decommissioning plan is expected to be submitted by the licensee in January 1994. The LLWM staff will review that plan when submitted. The Trojan and San Onofre Unit 1 decommissioning plans are due to be submitted to the NRC in late 1994.

## ADVISORY COMMITTEE ON NUCLEAR WASTE

The Advisory Committee on Nuclear Waste (ACNW) was established by the Nuclear Regulatory Commission (NRC) in 1988. The ACNW is directed to report to and advise the NRC on nuclear waste disposal facilities, as directed by the Commission. This includes 10 CFR Parts 60 and 61 and other applicable regulations and legislative mandates such as the Nuclear Waste Policy Act, the Low-Level Radioactive Waste Policy Act, and the Uranium Mill Tailings Radiation Control Act, as amended. The primary emphasis is on disposal facilities. In performing its work, the committee will examine and report on those areas of concern referred to it by the Commission or its designated representatives, and will undertake other studies and activities related to those issues as directed by the Commission.

All ACNW reports, other than those which may contain classified material, are made part of the public record. Activities of the committee are conducted in accordance with the Federal Advisory Committee Act, which provides for public attendance at and participation in committee meetings. The ACNW membership is drawn from scientific and engineering disciplines and includes individuals experienced in geosciences, radiation protection, radioactive waste treatment, environmental engineering, nuclear engineering, and chemistry. (See Appendix 2 for a listing of the members of the ACNW.)

During fiscal year 1993, the ACNW reported to the Commission on a variety of issues, including the following:

- Staff Technical Position on Consideration of Fault Displacement Hazards in
- Geologic Repository Design.
- Significant issues in the high-level waste repository program.
- Iterative Performance Assessment Phase 2.
- Impact of long-range climate change in the Southern Great Basin.
- Issues raised in the Energy Policy Act of 1992, Section 801.
- Program Plan for the Advisory Committee on Nuclear Waste.
- Possible impact of the Energy Policy Act of 1992 on NRC activities to address ongoing NRC initiatives in the high-level radioactive waste program.
- Source term and other low-level waste considerations.
- Proposed rulemaking on amendments to 10 CFR Part 60 clarifying the requirements for assessment of siting criteria.
- Review of April 21, 1993, draft NRC High-Level Radioactive Waste Research Program Plan.
- Revision 1 of the Final Standard Review Plan for the review of remedial action of inactive mill tailings sites Under Title I of the Uranium Mill Tailings Radiation Control Act.

# Communicating With The Public and The Government

Chapter

7

The Nuclear Regulatory Commission maintains regular communication with a broad range of governmental entities and with the general public. Several NRC Headquarters Offices and the Regional Offices participate in the dissemination of information about NRC activities. Commissioners and senior management frequently take part in Congressional Hearings (see table), and appropriate Congressional Committees are kept regularly and fully informed of NRC decisions and actions. Liaison with the general public, the Congress, Federal and State agencies, Indian Tribes, local community organizations, and the news media is maintained mainly through four offices of the NRC: the Office of the Secretary, the Office of Congressional Affairs, the Office of Public Affairs, and the Office of State Programs. (The NRC's international programs and exchanges are carried out through the NRC Office of International Programs, whose activities are covered in Chapter 8.)

## COMMUNICATION WITH THE PUBLIC

### Commission Meetings

The NRC Commissioners meet in public session at the NRC Headquarters building, One White Flint North, Rockville, Md., to discuss agency business. Members of the public are welcome to attend and observe Commission meetings, except on those infrequent occasions when the Commission decides that a meeting should be closed. A meeting may be closed to members of the public if it is convened to deal with one or more of certain subjects specified in the Government in the Sunshine Act, which allows the closing of meetings involving such subjects or items of information as classified documents, internal personnel matters, information that is confidential by statute, trade secrets, personal privacy, investigations, or adjudicatory matters. Members of the public are not allowed to participate in public Commission meetings unless specifically requested to do so by the Commission.

The Commission endeavors to provide meaningful public observation and understanding of open meetings. The Commission's conference room is equipped with multiple overhead speakers and a closed circuit television system to ensure that those in attendance can see and hear the proceedings. A pamphlet entitled "Guide to NRC Open Meetings" is available in the conference room and in the Public Document Room (PDR), located at 2120 L Street, N.W., Washington, D.C. The guide describes the normal seating arrangement for participants at the conference table, the general functional responsibilities of these participants, Commission procedures for voting on agenda items, general rules for public conduct at Commission meetings, and sources of additional information on the Commission and its meetings. A "Handbook of Acronyms and Initialisms" (NUREG-0544, Rev. 2) is also available in the PDR to define and explain the many technical terms discussed in Commission meetings and papers.

Copies of viewgraphs and the principal staff papers to be considered at open meetings are normally made available at the entrance to the Conference Room prior to the commencement of the meeting. Transcripts of open meetings and the papers released at the meeting are also placed in the PDR at the conclusion of the meeting for inspection and copying. And copies of all papers referenced at the meeting are normally released to the public. The public is also permitted to tape-record Commission discussions at open meetings. It is the Commission's practice to allow camera and television coverage of open meetings and briefings without prior notification.

The Commission attempts in all cases to provide at least one week's advanced notice for Commission meetings. Notice of the following four weeks of Commission meetings is published each week in the *Federal Register*. An announcement is also displayed on a TV- monitor in the lobby of NRC Headquarters and is posted in the Public Document Room. The announcement discloses the time, place and subject matter of the meeting, states whether it is an open or closed meeting, and gives the name and telephone number of an official designated to respond to requests for information about the meeting. Notice of meetings is given to the press through the wire services and by mailings to individuals who have requested copies of such notices. Announcements of Commission meetings are also regularly furnished on a recorded telephone message ((301) 504-1292), providing the schedule for upcoming Commission meetings and/or voting sessions.

### Advisory Committees

The Nuclear Regulatory Commission engages the expertise and experience of a wide segment of the public through their service on the NRC's standing advisory committees and on its *ad hoc* committees. Members of NRC committees are drawn from a broad crosssection of the scientific and technical community, as well as from State and local governmental organizations, the National Congress of American Indians, and private citizens. Committee members provide advice and recommendations to the NRC on a large range of issues affecting NRC policies and programs. Appendix 2 gives a brief statement of the purpose of each of the NRC's standing advisory committees and a listing of the names and affiliations of current members.

The NRC's advisory committees meet, in accordance with the requirements of the Federal Advisory Committee Act, in public sessions, at Headquarters locations and in venues throughout the United States. Notice of advisory committee meetings is published in the *Federal Register*, in NRC press announcements, and by the posting of meeting dates and topics in the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. Transcripts and/or minutes of meetings are also available for inspection and copying at the NRC Public Document Room. Persons interested in the activities of a particular committee or in committee meetings may call or write the NRC Advisory Committee Management Officer, Office of the Secretary, Washington, D.C. 20555; telephone (301) 504-1968.

## PUBLIC INFORMATION

In keeping with the Commission's policy of ensuring open communication with the public, the Office of Public Affairs informs the public through a variety of activities: special news briefings, notifying the news media about key Commission and NRC staff meetings and decisions, and arranging press conferences.

For a third consecutive year, the five NRC Regional Administrators held special periodic news briefings on agency-wide and regional issues. The briefings were conducted in New York City; Pittsburgh, Pa.; Huntsville, Ala.; Crystal River, Fla.; St. Louis, Mo.; Little Rock, Ark.; Denver, Colo.; Phoenix, Ariz.; and Pasco, Wash.

Topics that generated reporter interest and newspaper coverage included the performance of specific nuclear power plants, the status of cleanup and decommissioning of radioactively contaminated industrial sites, decommissioning of nuclear facilities, low-level radioactive waste activities, enforcement actions and dry cask spent fuel storage.

In addition to these periodic briefings, press conferences were arranged following Commission and/or staff meetings to answer more fully questions from reporters on important NRC rulemakings, actions or policies. The news media and general public were also informed by Public Affairs through distribution of news releases, fact sheets, pamphlets, and exchanges with NRC staff at news conferences and/or open meetings dealing with special team inspections of nuclear facilities or Systematic Assessment of Licensee Performance reports.

**Enforcement Conferences.** The Commission's trial program to permit the news media and the public to observe selected enforcement conferences continued for a second year. The program was under evaluation at the close of the report period to determine if it should be expanded or otherwise modified.

**Media Seminar Workshop.** The NRC Technical Training Center and regional Public Affairs staff held a national seminar for news reporters on October 13-14, 1992. Reporters were instructed on how nuclear power plants are designed and built, how they operate, radiation and its biological effects, and emergency planning and response. The NRC training center in Chattanooga, Tenn., has four operating control room simulators (originally built for actual nuclear power plants) connected to computers for realistic duplication of actual operations.

**NRC School Volunteers Program.** In recognition of the NRC volunteer school program, the Montgomery County Public Schools cited the agency "for outstanding service to education during the 1992-1993 school year."

The role of the NRC as a regulator of nuclear safety, with an inside look at the individual career fields of school volunteers, continued to draw the substantial interest of students and faculty of elementary and high schools, as well as colleges and universities.

During the school year, NRC volunteers traveled to schools throughout the Washington Metropolitan Area as part of the national Partnerships in Education Program initiated by the President in 1983. The program includes more than 250 headquarters employees. Coordinated by the Office of Public Affairs, NRC professionals—nuclear engineers, geologists, health physicists and chemists—visited 2-to-3 schools each week and, at times, hosted teachers and students at NRC headquarters. In all, the volunteers reached out to some 4,600 students and faculty, primarily in public schools in Montgomery County, Md., but also in the District of Columbia and in northern Virginia.

During their presentations, the volunteers described what skills are needed to qualify for the jobs they hold at the NRC, how they got to where they are and how their work supports the mission of the agency.

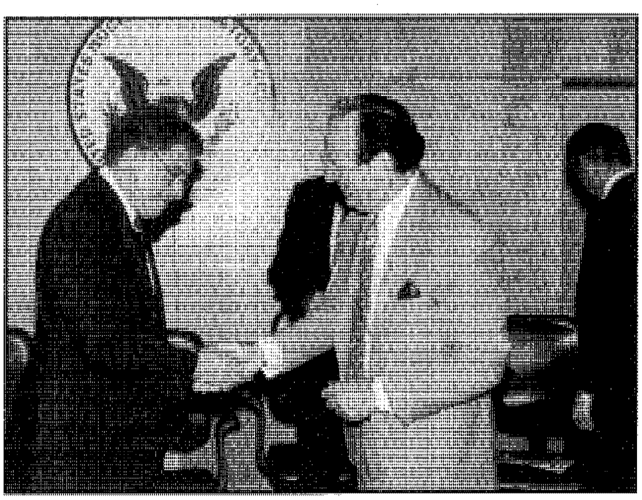
Students affected by the program ranged from the academically advanced to those at risk of dropping out, and covered all economic levels and ethnic origins, including recent immigrants. Hands-on demonstrations were provided by the volunteers, as well as academic tutoring, mentoring, assisting with science projects, and judging at science and math fairs. The volunteers also helped members of school faculties develop science and engineering activities, on-the-job math and science presentations, career awareness programs, and responses to questions that arise from student interviews of NRC volunteers.

This year, Commissioner Kenneth Rogers joined 17 NRC staff members to judge science projects of area students and presented six special awards at an annual Montgomery Area Science Fair. The winning students later presented their projects to the Commission at a meeting open to all NRC staff.

At the college and university level, NRC volunteers provided lectures and workshops for students and professionals attending Hood College and Johns Hopkins, Maryland, Georgetown, Widener and Pennsylvania State Universities. Topics covered by the volunteers included what the NRC's regulatory role is and how nuclear power reactors of the future are expected to work.

The NRC also organized and participated in a Science and Technology Program for Educators. Under that program, 15 secondary science and technology education teachers from Montgomery County schools spent a day at the NRC learning about the agency, nuclear reactors, radiation, nuclear waste and emergency preparedness. NRC Chairman Ivan Selin addressed the teachers, and NRC staff members provided hands-on demonstrations suitable for the teachers to use in their classrooms.

At a special springtime NRC ceremony, the volunteers were commended for their work in schools and their dedication to improving education.



**NRC Chairman Ivan Selin and Dr. Paul Vance, Superintendent of Montgomery County (Md.) Public Schools, renew the agreement under which NRC staff will be visiting county schools, as part of the national Partnerships in Education program that began in 1983.**

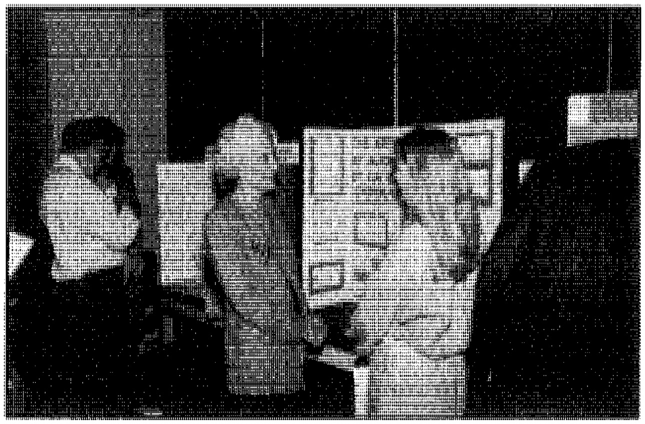
## Headquarters Public Document Room

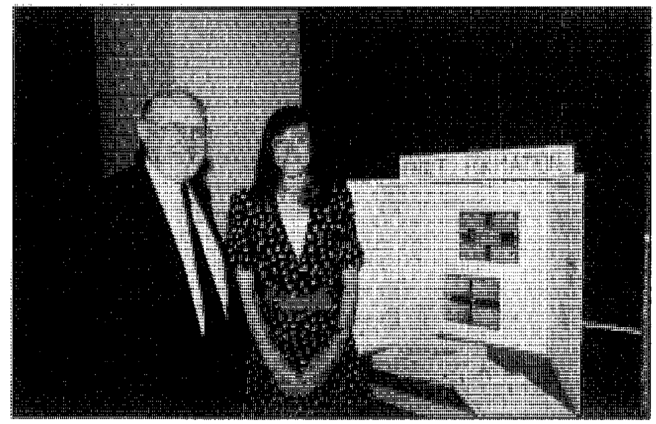
Serving as a bridge between the agency and the public, the Headquarters Public Document Room (PDR) maintains an extensive collection of documents related to NRC licensing proceedings and other significant decisions and actions, and also documents from the regulatory activities of the former Atomic Energy Commission. The computerized, online Bibliographic Retrieval System (BRS) includes extensive indices to the collection and an on-line ordering module for the placement of orders for the reproduction and delivery of specific documents. Located at 2120 L Street, N.W., Washington, D.C., the PDR is open Monday through Friday, from 7:45 a.m.-to-4:15 p.m., eastern time, except on Federal holidays. Persons interested in detailed, technical information about nuclear facilities and other licensees find this specialized research center to be a major resource. With some exceptions, documents from the collection can be reproduced in paper or microfiche form, for a nominal fee. In 1994, selected documents will also be available in electronic form.

Among the wide variety of agency documents available to the public at the PDR are NRC NUREG Reports and manuals; transcripts and summaries of Commission meetings and NRC staff and licensee meetings; existing and proposed regulations and rulemakings; licenses and amendments; and correspondence on technical, legal, and regulatory matters. Most of the documents are related to nuclear power plants—their design, construction and operation—and to nuclear materials, including the transportation and disposal of radioactive wastes. The PDR also offers a Standing Order Subscription service for selected serially published documents and reports. Certain items of immediate interest, such as Press Releases and Meetings Notices, are posted in the Reading Room at the facility. The PDR does not contain books, journals, trade publications, or documentation of industry standards.

The Headquarters PDR contains nearly two million documents. During a typical month, the PDR serves about 1,300 documented users. Reference Librarians are available to assist on-site users and those who call or write with information requests. PDR staff make the BRS data base available to users either on-site, using terminals in the Reading Room, or off-site, via modem. Off-site access (at 1200, 2400, and 9600 baud) is available for searches 24 hours a day, weekends and holidays included.

The PDR/BRS users group comprises members of Congressional staffs, media representatives, personnel from other government agencies, foreign embassies, law firms, utilities, State agencies, consulting firms, public interest groups, individual members of the public, foreign governments, and other institutions. The PDR provides the BRS, document delivery, and general reference services to the NRC's Agreement Country counterparts.





**Science projects of high-school students in the region around NRC Headquarters in North Bethesda, Md., were on display at the annual Montgomery Area Science Fair. The winners of special awards later presented their projects to the Commission at a meeting open to all NRC staff. Recipients of awards at the science fair are congratulated by Commissioner E. Gail de Planque, left, and Commissioner Kenneth Rogers, right.**

Persons wishing to visit and use the Public Document Room or obtain additional information regarding the PDR may call (202) 634–3273, Monday through Friday, between 8:30 a.m.-and-4:15 p.m. (eastern time); fax to (202) 634–3343; or write to the U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555.

## Local Public Document Rooms

At the close of fiscal year 1993, the NRC was maintaining 87 Local Public Document Rooms (LPDRs) throughout the country. These LPDRs house collections of documents related to nuclear power reactors, certain fuel cycle facilities, and low-level and high-level waste disposal facilities, both operational and prospective. Financial assistance, by means of cooperative agreements, was provided to 68 LPDRs during the report period. (See Appendix 3 for a complete listing of LPDRs, by State.)

The conversion of the 77 power reactor and two high-level waste LPDRs from paper to microfiche, for records dating from January 1, 1981, to the present has been completed. This effort reduced the shelf space required for paper records by approximately two-thirds at each library. The conversion from paper to microfiche has significantly increased the document resources available at each of these LPDRs. The collections are no longer limited to records pertaining to the local facility only, but now contain essentially all records made available to the public by the NRC since 1981. The new arrangement also reduces and stabilizes NRC's costs for support of the LPDR program. The conversion to microfiche has been favorably received by LPDR librarians and patrons.

Forty-two LPDRs currently have on-line access to NRC's computerized document management system, the NUDOCS/AD (Nuclear Documents System/Advanced Design). With this access, librarians and patrons can identify NRC publicly available records, within a data base of approximately 2,000,000 records. Microfiche of the post-1981 records are on file at the power reactor and high-level waste LPDRs.

Local librarians and their patrons may use a toll-free telephone number to request assistance and information from NRC LPDR staff on collection content, search strategies, and the use of reference tools and indices. Information on NUDOCS/AD access at LPDR libraries is also available from the LPDR staff. The telephone number is 800-638-8081.

## Commission History Program

Through the Commission History Program, the origins and evolution of NRC regulatory policies are explored and set forth in their historical context by means of research into such sources as the records maintained in archives of a number of government agencies, the personal papers of former government officials, and personal interviews with such officials. The History Office is currently conducting research for the third volume of a detailed, scholarly history of nuclear regulation. The first volume, *Controlling the Atom: the Beginnings of Nuclear Regulation, 1946–1962*, appeared in 1984. The second volume, *Containing the Atom: Nuclear Regulation in a Changing Environment, 1963–1971*, appeared in 1992. Both were published by the University of California Press. The volumes are intended to serve as historical references for the agency staff and the general public.

## COMMUNICATION WITH THE CONGRESS

The Office of Congressional Affairs is responsible for developing, managing, and coordinating relations with the Congress, and is the principal point of contact between the agency and Congress. The office coordinates the appearances and testimony of all NRC officials at hearings, monitors and tracks bills relevant to the NRC, keeps the Congress currently informed of agency activities, and keeps the NRC apprised of Congressional concerns and interests.

During fiscal year 1993, NRC witnesses testified at 10 hearings before Congressional Committees and Subcommittees, as shown in the table. Congressional Affairs staff attended and prepared summaries and reports for approximately 50 hearings and mark-ups.

## COOPERATION WITH THE STATES AND OTHER FEDERAL AGENCIES

The NRC's formal cooperation with Federal, State and local governments, interstate organizations, and Indian Tribes is administered through the Office of State Programs (OSP). The primary goal of the office is to ensure that the NRC maintains effective relations and communications with these organizations, and to promote greater awareness and mutual understanding of the policies, activities and concerns of all parties involved as they relate to radiological safety. The office's activities encompass three major areas: the Agreement State Program; State, Local, and Indian Relations; and Federal Liaison. These programs are implemented through Headquarters and the Regional Offices.

### Agreement State Program

A total of 29 States have formal agreements with the NRC by which those States have assumed regulatory responsibility over byproduct and source materials, and small quantities of special nuclear material. At the close of fiscal year 1993, approximately 15,000 radioactive materials licenses were administered by the Agreement States, representing about 69 percent of all the radioactive material licenses issued in the United States. The State of Pennsylvania is negotiating a limited agreement with the NRC which will give Pennsylvania regulatory authority over the land disposal of byproduct, source and special nuclear material only. The States of Massachusetts, Ohio and Oklahoma are negotiating full Agreement State status with NRC.

**Review of State Regulatory Programs.** The Atomic Energy Act of 1954, as amended, requires NRC to review Agreement State radiation control programs periodically; the programs are normally reviewed annually. The NRC conducts three kinds of reviews—routine reviews, review visits, and follow-up reviews. Routine reviews are complete, in-depth examinations of State regulatory programs, normally conducted every other calendar year. Review visits are usually conducted between routine reviews and serve to maintain familiarity with Agreement State radiation control programs, to provide an opportunity to discuss areas of concern on an informal basis, and confirm the satisfactory status of the State radiation control programs. Follow-up or special reviews are conducted as needed, and they tend to focus on State actions in specific areas.

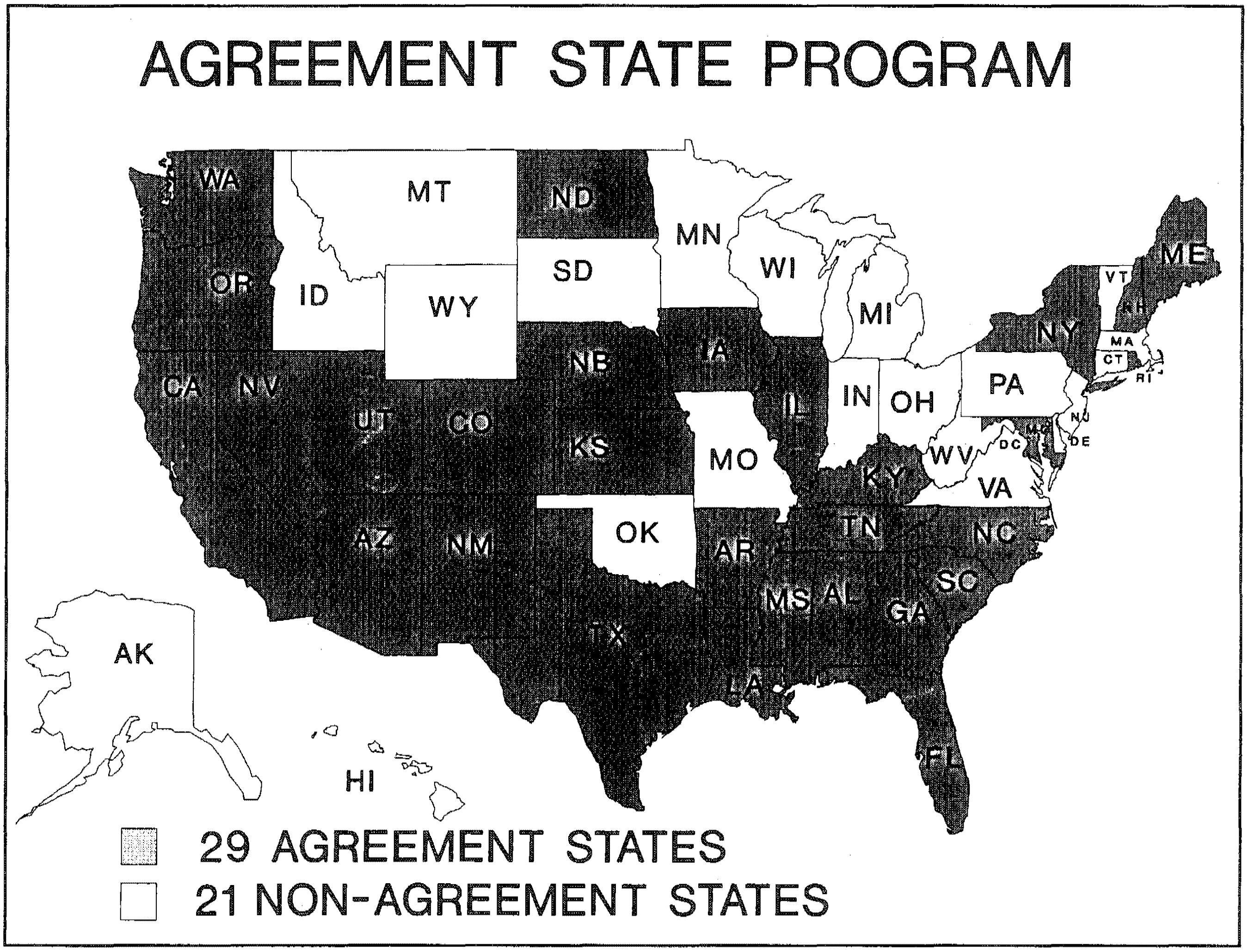
In fiscal year 1993, NRC performed 15 program reviews, 9 review visits, and 3 followup reviews. The NRC technical staff accompanied State inspectors to State-licensed facilities to evaluate inspector performance; the staff examined selected license and compliance casework in detail, in connection with these reviews. When appropriate, multi-discipline teams are sent to conduct reviews of Agreement State programs. The teams include NRC Program and Regional Office staff.

The reviews seek to identify potential problems in State programs, which are reported to high-level State management. In doing this, the NRC employs a "Category 1" designation for the more serious concerns. If no significant Category 1 comments are provided, then the program is deemed adequate to protect the public health and safety and is judged compatible with the NRC's program. If one or more significant Category 1 comments are provided, the State is notified that the program deficiencies may seriously affect the State's ability to protect the public health and safety and that the need for improvement in particular program areas is critical.

**NRC Technical Assistance to States.** The NRC continues to provide technical assistance to Agreement States in the areas of licensing, inspection and enforcement, and informs the States of proposed statutes and regulations. Technical assistance is provided by responding to requests for information, by assisting in State inspections and reviews of license applications, and by dealing with specialized or unusual radiation applications requiring specialized expertise and knowledge. The NRC provided technical assistance to the States of Washington, Utah, New York, Nebraska, North Carolina and South Carolina for the development and maintenance of low-level waste regulatory programs by States that meet the requirements of the Low-Level Radioactive Waste Policy Amendments Act of 1985. South Carolina and Washington continue to participate in the NRC review of several topical reports on

## Congressional Hearings at Which NRC Witnesses Testified – FY 1993

| *Date* | *Committee* | *Subject* |
|---|---|---|
| 10/01/92 | Committee on Interior and Insular Affairs<br>Subcommittee on Energy and the Environment<br>Testimony was supplied for the record. (House) | Decommissioning and Decontamination Standards |
| 03/03/93 | Committee on Energy and Commerce<br>Subcommittee on Oversight and Investigations<br>(House) | Fire Protection for Nuclear Power Plants |
| 03/19/93 | Committee on Environment and Public Works<br>Subcommittee on Clean Air and<br>Nuclear Regulation (Senate) | Adequacy of Nuclear Power Plant Security to Protect Against Terrorism and Sabotage |
| 04/21/93 | Committee on Appropriations<br>Subcommittee on Energy and Water Development<br>(House) | FY 1994 Appropriations |
| 04/29/93 | Committee on Science, Space, and Technology<br>Subcommittee on Energy (House) | Nuclear Energy R&D, Advanced Reactors |
| 05/06/93 | Committee on Governmental Affairs<br>(Senate) | Federal Regulation of Medical Radiation Uses |
| 05/27/93 | Committee on Natural Resources<br>Subcommittee on Energy and Mineral Resources<br>(House) | FY 1994 Authorization; Legislative Proposals |
| 06/30/93 | Committee on Environment and Public Works<br>Subcommittee on Clean Air and Nuclear Regulation<br>(Senate) | FY 1994 Authorization; Legislative Proposals |
| 07/15/93 | Committee on Environment and Public Works<br>Subcommittee on Clean Air and Nuclear Regulation<br>(Senate) | NRC's Handling of Harrassment and Intimidation Allegations |
| 08/02/93 | Committee on Government Operations<br>Subcommittee on Environment, Energy, and Natural<br>Resources (House) | Agreement States Program |




high-integrity containers, waste solidification processes, and computer codes to be used in implementing 10 CFR Part 61.

**Training Offered State Personnel by the NRC.** The NRC sponsors training courses and workshops primarily for State radiation control personnel to help them maintain high quality regulatory programs. Course subjects are diverse, covering health physics, industrial radiography safety, well-logging, radiation protection engineering, transportation of radioactive nuclear materials and low-level waste, nuclear medicine, inspection procedures, and materials licensing. Special workshops on specific areas are also held, as needed.

The NRC sponsored 20 such training courses and workshops, attended by 350 State radiation control personnel, during the fiscal year. The sessions were also attended by NRC staff and by military personnel, as well as officials from Canada and Mexico.

Representatives from the 29 Agreement States and several non-Agreement States attended the second round of five special training sessions on the revised Part 20 of Title 10 on the *Code of Federal Regulations*, which were held in August, October and November of 1993 in the NRC Regions. OSP sponsored a workshop on fees and funding of Agreement State programs in April 1992. The results of this workshop and a previous workshop on rules and regulations were published as NUREG–1479 in September 1993. The intent of the NUREG was to provide a compilation of informative methods to assist Agreement States in establishing and operating their programs more efficiently.

*Annual Low-Level Waste Regulatory Workshop.* The annual Low-Level Waste Regulatory Workshop was held in Rockville, Md., in July 1993, providing an opportunity for the NRC to discuss current regulatory issues related to low-level waste disposal with the State personnel who are expected to regulate a low-level waste facility.

*Uranium Mills Workshop.* The Agreement State Workshop on Uranium Mills was held on July 29-30, 1993, in Rockville, Md. The principal objective of the workshop was to provide a forum for the NRC and the uranium mill Agreement States to exchange information on the status of standards, implementation policy and procedures, and activities designed to result in successful termination of uranium mill licenses. Representatives from the States of Colorado, Texas, Washington, Utah, Nebraska, and New Jersey participated, along with representatives from the U.S. Department of Energy, and from NRC headquarters and regional staff.

*Regulation of Uranium Milling.* The NRC also assisted Agreement States with their programs for regulating uranium milling. Assistance was given in the areas of groundwater monitoring requirements for milling facilities, reclamation design reviews, proposed disposal units, guidance document review, and license termination determinations. This assistance was provided to the States of Colorado, Texas and Washington.

*Annual Agreement States Meeting.* The 1993 annual meeting of Agreement State radiation control program directors was held October 24-27, 1993, in Tempe, Ariz. Discussions were held on abnormal event reporting, materials regulation, operational events and radioactivity in the environment. Lengthy discussions were also held to update the attenders on the status of amended regulations and to introduce them to several new regulations. The proposed policies on compatibility and the application of performance indicators to review Agreement States was discussed in great detail.

*Operational Events in Agreement States.* Information on events in Agreement States is routinely exchanged with the NRC. Safety-significant Agreement State and NRC operational events are discussed at periodic NRC staff meetings, with an emphasis on identifying the cause of each event. During the past year, Agreement State personnel investigated events involving lost or stolen equipment, equipment failure, and incidents involving the medical use of radioactive material. When these studies lead to effective generic remedies, the information is disseminated to the appropriate regulatory agencies and users.

*Improving Cooperation With the States.* In accordance with a Commission directive to develop a process that will ensure early and substantial involvement of Agreement States in rulemaking—and in other regulatory efforts that affect facilities licensed under 10 CFR Parts 30, 40, 61, and 70, or equivalent State regulations—the NRC held two public meetings with the Agreement States, in San Francisco, Cal., in May 1993 and in Tempe, Ariz., in October 1993.

## State, Local, and Indian Relations Program

One of the goals set forth in the agency's Five Year Plan is to maintain open lines of communication and close liaison with State and local government officials and their organizational representatives, as well as with Native Americans and organizations representing American Indian Tribes. These relationships are developed in an effort to fully address concerns and to promote increased understanding of issues related to NRC regulation, inspection, and oversight activities to protect the public health and safety.

**Outreach Activities.** In keeping with the mandate of the Five Year Plan, the NRC continues cooperative activities with the States and their national organizations. Besides routine interaction with State and local government and Indian Tribe officials, NRC representatives have taken part in a number of special State-related events. NRC staff participated in discussions of spent nuclear fuel storage and disposal issues sponsored by the National Association of Regulatory Utility Commissioners (NARUC). As part of the initiative which began in 1992, a delegation from NARUC participated in a dialogue with NRC staff on October 19, 1992, in which a number of issues concerning nuclear power plant safety and economics were discussed. NRC staff met with representatives from the New England Conference of Public Utility Commissioners to discuss issues associated with the decommissioning of the Yankee-Rowe plant in Massachusetts and held other sessions with State officials on plant operations and spent fuel storage issues.

The NRC has continued to follow the activities of other State-related organizations, such as the National Governors' Association (NGA), the Western Governors' Association (WGA), and the National Conference of State Legislatures (NCSL).

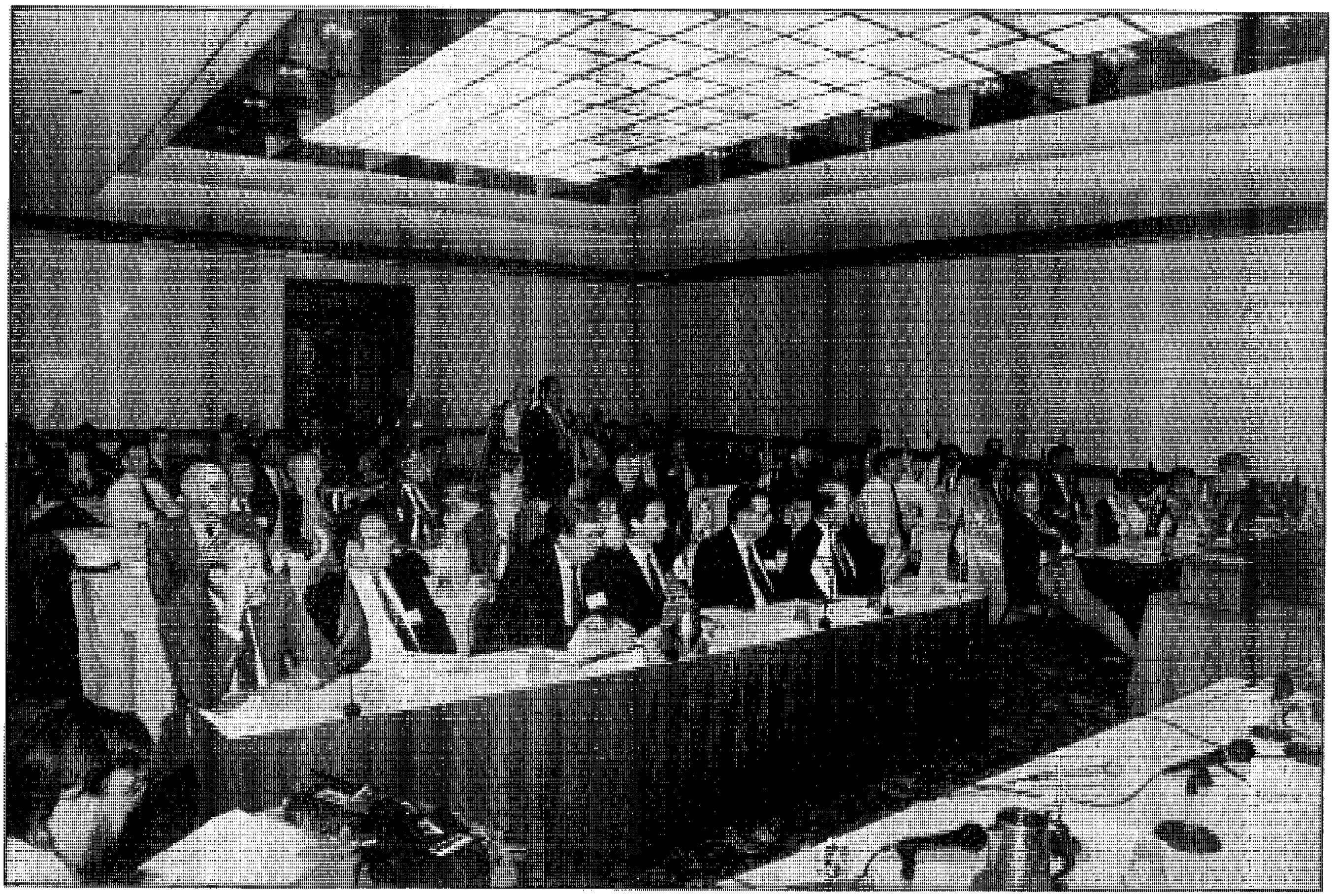
**NRC Regional State Liaison Officers.** The NRC's principal contact with SLOs and other State and local officials is through the five NRC Regional State Liaison Officers (RSLOs). The RSLOs are the coordinators for NRC activities involving State and local governments. They often attend and participate in State and local meetings when issues involving NRC are under discussion. The RSLOs work with State legislative committees and meet with State and local officials to address concerns and respond to questions. The RSLOs routinely handle requests for information from SLOs and other State officials concerning nuclear power facilities or other areas under the NRC's jurisdiction. The RSLOs attend meetings concerning regional low-level radioactive waste issues and monitor State progress in developing needed capacity for low-level waste disposal. They also participate in emergency planning exercises involving State and local governmental officials.

**Cooperation With States.** The NRC staff continues to allow States to observe NRC inspections at reactors pursuant to its policy statement on "Cooperation With States at Nuclear Power Plants and Other Nuclear Production or Utilization Facilities" (57 FR 6462, dated 2/25/92). In some cases, States may observe special inspections as well. During the year, a New Jersey official observed an Augmented Inspection Team at Salem and a Pennsylvania official observed an Incident Investigation Team at the Three Mile Island (Pa.) nuclear power plant.

**State Liaison Officer Program.** The NRC policy statement on Cooperation With States identifies the governor-appointed State Liaison Officer (SLO) as the primary State contact for all requests involving observation of NRC inspections of plants or facilities. The SLOs are also the NRC's primary point of contact with the States regarding all relevant NRC decisions and actions.

Headquarters hosted a National SLO Meeting on September 29–30, 1993, in Rockville, Md. The meeting featured State, the NRC, Congressional and interstate organization and other Federal agency representation. Topics discussed included: decommissioning of contaminated sites, medical misadministrations of nuclear materials, low-level radioactive waste management, renewal of nuclear power plant operating licenses, high-level radioactive waste management, international nuclear activities, storage of spent nuclear fuel in dry-cask storage facilities, emergency planning, communication with State legislators, and NRC/State relationships.

**The Conference of Radiation Control Program Directors.** The NRC, through the Office of State Programs, continues to be represented in the Conference of Radiation Control Program Directors (CRCPD) to help ensure that State and Commission programs for protection against the hazards of radiation are coordinated and compatible. The CRCPD was formed in 1968 to provide a forum where Federal, State and local radiation control program officials could address governmental radiation protection issues. The major work of the CRCPD is accomplished through committees and task forces. At any time there may be 50 or more groups working on specific projects. An example is the Suggested State Regulations which help promote uniformity in radiation protection



**A national meeting of State Liaison Officers was hosted by NRC Headquarters on September 29–30 of 1993, in Rockville, Md. Representatives from the States, the NRC, the Congress and other interested organizations took part in discussion of decommissioning of contaminated sites, medical misadministrations of nuclear materials, nuclear waste issues, storage of spent fuel, and many other concerns of mutual interest.**

programs throughout the United States. As many as 11 NRC resource persons are represented on approximately 24 committees and task forces which meet throughout the year. NRC contributed $110,000 in fiscal year 1993 to the CRCPD. The Conference is a valuable resource and ally when a public health and safety issue is identified. They have been highly instrumental in notifying States of immediate health concerns, and in quickly assembling a health physics network of dedicated professionals who can determine the most safe and efficient way of solving a problem.

The NRC hosts National SLO meetings every three years and hosts regional meetings as needed, in the off years.

**Low-Level Waste Compacts.** States have been slow to develop new low-level radioactive waste disposal facilities as measured by the milestones established by the LowLevel Radioactive Waste Policy Amendments Act of 1985. Nevertheless, 42 States have formed nine compacts, as authorized by Congress. Legislation to establish the Texas LowLevel Radioactive Waste Disposal Compact was signed by the Governor of Texas on June 9, 1993. Legislation ratifying the compact was enacted in Maine on June 21, 1993, and approved by a majority of the voters in a referendum held on November 2, 1993.

Eight compacts (31 States) plan to develop nine disposal facilities, and two compacts (11 States) will be served by the existing facility in Richland, Wash. In addition, two States not affiliated with compacts, New York and Massachusetts, intend to develop their own disposal facilities. The States of New Hampshire and Rhode Island, the District of Columbia, and the Commonwealth of Puerto Rico are not planning to develop disposal facilities and believe they may be able to fulfill their responsibilities either through the contracting process or compacting. The remaining State, Michigan, was expelled from the Midwest Compact on July 24, 1991, and has not announced plans to manage or dispose of its low-level radioactive waste.

On October 9, 1992, the Illinois Low-Level Radioactive Waste Disposal Facility Siting Commission voted unanimously to reject the disposal site proposed by the Illinois Department of Nuclear Safety (IDNS), as reported in the *1992 NRC Annual Report*, p. 138. The Governor of Illinois directed the IDNS to stop all efforts to locate a facility at the Martinsville site which was scheduled for operation in early 1995. On December 24, 1992, legislation was enacted that abolished the Siting Commission, repealed the statutory siting criteria, and directed IDNS to recommend a new process for timely and cost-effective establishment of a disposal facility. Legislation changing the procedure for siting was signed into law on March 3, 1993, and amended in July 1993. The forecast date for operation is now April 1999.

Only two new facilities, in California and in North Carolina, are now scheduled to be operational by January 1996; the latter will replace the existing Barnwell, S.C., facility. The host States of Texas, Nebraska, New Jersey, Massachusetts, Pennsylvania, Connecticut, Ohio and New York are forecast to be operational between 1996 and 2001.

On September 16, 1993, the California Department of Health Services issued a conditional license for a disposal site in Ward Valley, Cal. The Bureau of Land Management, U.S. Department of Interior, has control of the site, and, before construction of the facility can begin, the site must be transferred from Federal ownership to the State. Transfer of the land has been delayed because in mid-1991 the State Lands Commission, an independent State agency, refused to proceed with the transfer process because of concern about safety issues related to the facility. They also believed the taxpayers might be left with responsibility for any site remediation.

In a letter to the Governor of California, dated August 11, 1993, the Secretary of the Interior recommended that a formal hearing be held to focus on the issue of migration of radionuclides from the site so that Governor Wilson could carry out his responsibilities. On September 15, 1993, the Governor agreed to hold the hearing. A final decision on land transfer is also contingent on the designation of a critical habitat which could include Ward Valley for the desert tortoise, a threatened species, under the Endangered Species Act. On August 30, 1993, the U.S. Fish and Wildlife Service published its proposed rule on the critical habitat designation. A final decision on the land transfer may be forthcoming by the end of 1994 so that the facility could be operational in late 1995.

There are two lawsuits germane to these matters, as noted in the *1992 NRC Annual Report*, p. 139, that affect the NRC. The U.S. Court of Appeals for the Sixth Circuit, on June 2, 1993, affirmed the District Court judgment in favor of the Federal Government for *State of Michigan v. U.S.* The suit challenged as unconstitutional the Low-Level Waste Policy Amendments Act of 1985 and also demanded that NRC prepare a fresh National Environmental Policy Act analysis of the agency's 10 CFR Part 61 regulations on low-level radioactive waste disposal. Michigan did not appeal the decision to the U.S. Supreme Court. For *Diane Burton v. NRC*, the U.S. District Court of Nebraska on February 24, 1993, issued a decision dismissing the lawsuit in its entirety. The plaintiffs sought a declaration that the 10 CFR Part 61 site-ownership regulations are invalid, sought an order requiring NRC to issue regulations implementing the Nuclear Waste Policy Act provision that authorizes the Secretary of Energy to assume title and custody of disposal sites, and sought an order requiring the NRC to establish technical requirements for methods of disposal in addition to those now covered by Subpart D of 10 CFR Part 61.

The South Carolina legislature authorized the Barnwell disposal facility to remain open until January 1, 1996, subject to various conditions, although it originally had been scheduled to close on December 31, 1992. However, under both State law and a Southeast Compact Commission agreement, the Barnwell facility will close permanently to non-compact waste on July 1, 1994. The facility is accepting non-compact waste before that date, if the importation of such waste is approved by the Southeast Compact Commission for States making adequate progress toward providing for disposal. Michigan, New Hampshire, Rhode Island, and Puerto Rico are not currently eligible for access to the South Carolina facility.

Although the facility in Richland, Wash., will remain open indefinitely, compact action stopped the importation of waste from other than the Northwest and Rocky Mountain Compacts, effective January 1, 1993. Pursuant to a decision by the Rocky Mountain Compact Board, the Beatty, Nev., facility was closed December 31, 1992.

Because no new disposal facilities have been developed, and the compact commissions that control the existing disposal sites have either closed their sites or set conditions on receiving waste from outside their regional compacts, some licensees will be forced to store on-site. The disposal facility at Barnwell, S.C., is expected to close to non-compact generators on July 1, 1994, which will result in more widespread storage. Staff estimates that several thousand generators (including 68 power reactors), will be faced with on-site storage after that date.

Recognizing that interim storage may be required, the NRC has developed guidance and licensing procedures for storage. The NRC has also amended its regulations to permit power reactor licensees to receive back waste after processing off-site. The NRC is continuing to assess the need for additional guidance or licensing requirements, to supplement the existing regulatory framework for storage.

**Emergency Planning.** NRC staff from the Regions and the Office for Analysis and Evaluation of Operational Data met with emergency response officials from various States as part of a continuing "outreach program." The outreach program is intended to brief State officials on the NRC emergency response program, the Federal Radiological Emergency Response Plan, the Emergency Response Data System (ERDS), NRC/State liaison during an emergency and financial assistance.

ERDS Memoranda of Understanding (MOUs) were negotiated with the States of Arizona, Tennessee, Massachusetts, Maryland, New Jersey and New York during 1993. ERDS is a real-time data system designed to provide direct transmission of selected plant information from licensee on-site computers to the NRC Operations Center. The States can have the capability to receive ERDS data during events at power plants through an MOU with the NRC, and other States have also requested an MOU on ERDS.

**Liaison with American Indian Tribes.** The NRC continues to maintain communications with those American Indian Tribes, including their national organizations, potentially affected by, or otherwise interested in, NRC regulatory activities. While no Tribes have been formally accorded "affected" status under the 1987 Nuclear Waste Policy Amendments Act (as amended), those Tribes potentially affected by the Department of Energy's siting of a high-level waste repository at Yucca Mountain, Nev., continue to receive NRC reports and are advised in advance of any meetings relevant to the Commission's high-level waste program.

During the past year, NRC staff met with a number of tribal representatives to hear their concerns and provide information concerning nuclear activities on or near tribal land. In the area of area of high-level waste storage, disposal and transportation, meetings were held with representatives of the Mescalero Apache (N.M.) and the Fort McDermitt PaiuteShoshone Tribe (Nev.). Both Tribes are pursuing Monitored Retrievable Storage (MRS) facility studies and were interested in the Commission's role in protecting the public health and safety with respect to spent fuel transportation and licensing of an MRS facility. The NRC staff also participated in a meeting of the National Congress of American Indians' National Indian Nuclear Waste Policy Committee in Washington, D.C., as well as with the National Conference of State Legislatures' Legislative MRS Working Group meeting held in Williamsburg, Va. These meetings addressed in part the subject of State-Tribal relations in the area of MRS facility siting. Tribal interests are also represented by NCAI's membership in and participation at the NRC's Licensing Support System Advisory Review Panel meeting held in Las Vegas.

Tribal interest in nuclear-related activities has increased over the years and has helped bring about NRC staff interactions with the Navajo Nation regarding the Churchrock and Crownpoint, N.M. reclamation sites; the Cherokee Nation's interest in the Sequoyah Fuels, Okla., activities; and the Santee Tribe's interest in the Butte, Neb., proposed low-level waste site.

The NRC staff also continues to participate in the EPA-sponsored quarterly interagency meetings in an effort to keep up-to-date on American Indian issues. The meetings afford the opportunity to exchange new information of potential relevance and importance to Federal and tribal activities. And the NRC maintains liaison with the Department of the Interior/Bureau of Indian Affairs, in an effort to keep their constituency abreast of nuclear-related issues affecting Indian interests.

## Federal Liaison

The NRC's Federal Liaison is responsible for establishing and maintaining effective communications at the policy level between NRC and other pertinent Federal agencies. Liaison tasks include keeping appropriate NRC officials apprised of activities at other Federal agencies that may affect the NRC, and conveying to NRC management the salient views of other agencies regarding NRC policies, plans and activities.

The Federal Liaison is the NRC's contact with the Council on Environmental Quality (CEQ), as the contact prescribed by the National Environmental Policy Act (NEPA). In this capacity, the Federal Liaison communicates NRC analysis and comment on matters related to NEPA procedures and implementation to the CEQ and provides coordination with the NRC on those matters.

The Federal Liaison also serves as the NRC's point of contact with the Federal Coordinating Council for Science, Engineering and Technology (FCCSET). The Council—established to consider issues and developments in science and technology which affect multiple Federal agencies—provides a forum for coordinating those agencies' programs, sharing information, resolving conflicts, developing expertise, making policy recommendations, and identifying research needs, as well as promoting international cooperation, in science, engineering and technology. By statute, the Council is chaired by the Director of the Office of Science and Technology Policy and is composed of representatives from most of the major departments of the Executive Branch and from other elements of the Federal Government, including the NRC, whose representative is Chairman Ivan Selin. The Directors of the NRC's Offices of International Programs, of Nuclear Materials Safety and Safeguards, and of Nuclear Regulatory Research also serve on various FCCSET committees. The Federal Liaison participates in activities of FCCSET committees and subcommittees as either primary contact, staff contact, member or alternate, when requested. The Federal Liaison reviews and provides input to proposed legislation, rulemakings and correspondence affecting the NRC's policy relations with other Federal agencies, and reviews proposed Memoranda of Understanding with them.

The Federal Liaison also serves as the contact for identifying rulemakings significant to the States and ensuring that Agreement States are afforded the opportunity for substantive consultation during the development of rulemakings or other efforts of importance to the States.

# International Cooperation

Chapter 8

Recognizing that efforts to assure the peaceful, safe and environmentally acceptable uses of nuclear power necessarily involve international cooperation, the NRC has long maintained extensive contacts and regular exchanges of information with other nations. These cooperative programs are carried out through bilateral relationships, as well as through a number of multilateral institutions. As regulator of the world's largest civil nuclear program and long term sponsor of nuclear safety research, the NRC has the capability to contribute substantially to international nuclear programs—while benefiting from the experience of and experimentation by foreign nuclear operations—in such areas as nuclear power plant safety, radiation protection, the safeguarding of nuclear materials and their physical protection, waste management, and the decommissioning of nuclear facilities.

The NRC's international program has three broad objectives:

(1) Supporting U.S. foreign policy objectives.

- Helping to enhance nuclear power plant safety in countries with Sovietdesigned reactors.
- Helping to establish agreed nuclear safety principles world-wide.
- Assisting countries with developing nuclear power programs using U.S. nuclear technology—and those countries considering such technology—to build a solid safety/regulatory infrastructure through direct bilateral aid.
- Supporting efforts by multilateral organizations in the nuclear field—especially the International Atomic Energy Agency (IAEA)—to enhance nuclear safety in countries throughout the world.

(2) Helping to enhance U.S. national security.

- Providing the NRC's expertise and perspectives in the formulation and implementation of U.S. nuclear non-proliferation policies.
- Executing the NRC's export licensing activities in accordance with U.S. laws and policies.
- Supporting efforts to review and revise U.S. and multilateral export control systems relevant to the NRC's responsibilities.
- Participating in U.S. Government efforts to assist countries of the Former Soviet Union (FSU) in enhancement or establishment of systems for safeguarding nuclear materials.
- Assisting the Executive Branch to strengthen IAEA safeguards and physical protection, particularly where U.S. nuclear exports are involved.

(3) Improving the safety of NRC-licensed facilities in the United States.

- Exchanging information with other countries on the safe operation of nuclear facilities and the safe use of nuclear materials, especially those with advanced nuclear programs and plants similar to those in the United States.
- Conducting international research on high priority safety areas to complement and expand the NRC's research programs.
- Participating in key reactor and materials safety program activities of the Nuclear Energy Agency (NEA) and the IAEA relevant to NRC interests.

## FISCAL YEAR 1993 ACTIVITIES

During the report period, the NRC's activities in the international sphere expanded significantly.

Most noteworthy among them were the following:

- NRC support for two meetings of the U.S.-Russia Joint Commission on Technological Cooperation in Energy and Space, in which Vice President Gore and Russian Prime Minister Chernomyrdin led the two governments in defining significant new opportunities for cooperation in nuclear safety.
- Continued NRC activity in support of cooperation with the New Independent States of the former Soviet Union and countries of Central and Eastern Europe, including the strengthening of their regulatory organizations, training of foreign inspectors and joint undertakings in the areas of operational safety and risk reduction.

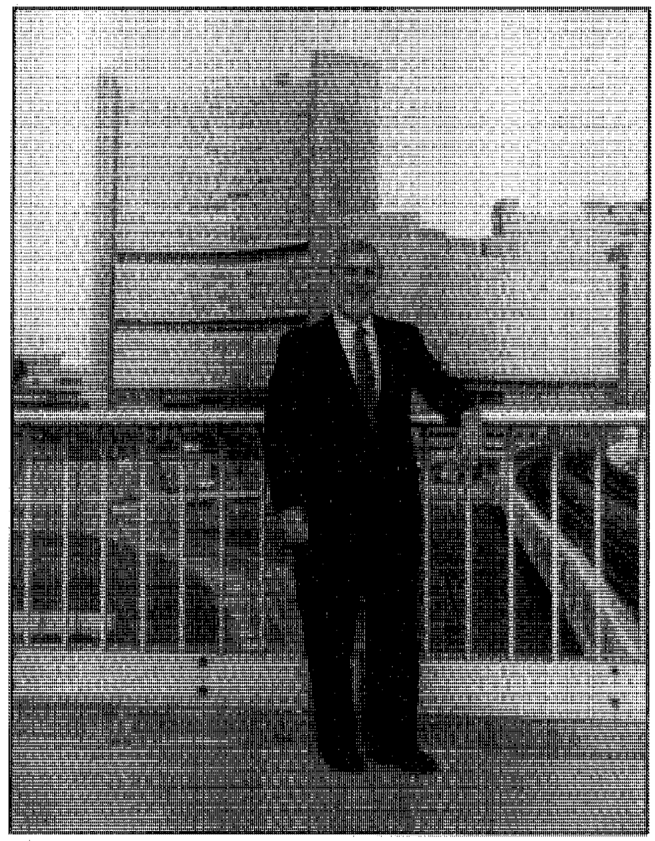
- Establishing the framework for cooperative programs to help countries of the Former Soviet Union—particularly Russia, Ukraine and Kazakhstan, to implement and improve systems for accounting and control of nuclear materials.

- Expanded regulatory cooperation with several Pacific Rim nations (Indonesia, China, Thailand, Korea and Taiwan) which have embarked on, or are considering, new or expanded nuclear power programs.

- Playing a leading role in development of an International Nuclear Safety Convention under the aegis of the IAEA.

- Extending the NRC's role in nuclear safety and technical assistance activities at the IAEA through the assignment of a senior expert as Nuclear Safety Attach at the U.S. Mission to U.N. System Organizations in Vienna, Austria.

- Continuing an active program of cooperative nuclear safety research with other nations, including Japan, the Russian Federation and France.



**James Richardson, former Director of the Division of Engineering in the NRC's Office of Nuclear Reactor Regulation, became the first Nuclear Safety Attache at the U.S. Mission to the United Nations Systems Organizations, in Vienna. The assignment involves daily visits to the International Atomic Energy Agency, which occupies the tallest of the buildings at the Vienna International Center, visible in the background.**

## BILATERAL SAFETY INFORMATION EXCHANGE

The NRC participates in a wide range of mutually beneficial programs of information exchange and cooperative safety research with counterparts in the international community. This section discusses the NRC's arrangements for exchange of information related to nuclear regulatory and licensing responsibilities.

**Safety Cooperation Arrangements.** Since 1974, when it formalized the information exchange arrangement program, the NRC has conducted most of its technical regulatory exchanges under the umbrella of a growing number of general safety cooperation arrangements that have been signed and renewed over the years. These now total 28: Argentina, Belgium, Brazil, Canada, China, The Czech Republic, Egypt, Finland, France, Germany, Greece, Hungary, Indonesia, Israel, Italy, Japan, the Republic of Korea, Mexico, The Netherlands, the Philippines, Slovakia, Spain, Sweden, Switzerland, the FSU, the United Kingdom, Slovenia and Taiwan.

These arrangements serve as communications channels with foreign nuclear regulatory organizations, ensuring prompt reciprocal notification of reactor safety problems that could affect either U.S. or foreign nuclear facilities and assisting in the identification of possible precursor events meriting further investigation. The arrangements also provide a framework for bilateral cooperation on nuclear safety, safeguards, waste management and environmental protection, as well as serving as the vehicle for NRC assistance to other countries in improving health and safety practices. They are typically of five years' duration, and may be renewed by mutual written agreement of the parties.

During fiscal year 1993, the NRC concluded its first information exchange and cooperation arrangement on nuclear safety matters with Indonesia, which has announced an intention to develop a nuclear power program. During this same period, the NRC renewed its information exchange and cooperation arrangements on nuclear safety matters with China, Israel and Greece, and continued active negotiations on the renewal of its arrangements with Germany, Japan and the United Kingdom. In December, the agency concluded a safety agreement directly with Slovenia as a successor to the former Yugoslavia.

As a key part of the agency's bilateral nuclear safety cooperation program, NRC Commissioners undertook a number of foreign visits in fiscal year 1993—to Argentina, Brazil, Japan, China, Korea, Taiwan, Philippines, Indonesia, Germany, France, Austria, Finland, Sweden, the Czech Republic, Slovakia and Mexico. These visits are an important means for encouraging exchange of information and experience on nuclear safety, gaining first hand knowledge of specific programs through selected site visits, and evaluating assistance that the NRC might provide. During the year, the NRC also received high-level visitors from France, Germany, Spain, the United Kingdom, South Africa, the Czech Republic, Brazil, China, Indonesia, Japan, Sweden, Taiwan, Israel and Thailand to discuss nuclear safety matters of mutual interest.

**Foreign Assignees Working at the NRC.** The NRC has an extensive on-the-job training program for assignees from other countries (usually from their regulatory organizations) that operates under the aegis of the bilateral information exchange arrangements. During fiscal year 1993, nine countries—Japan, Finland, Ukraine, Italy, Korea, The Czech Republic, Slovenia, Hungary and Bulgaria—sent 23 staff members to participate in the program. The participants completed assignments, which ranged generally from a few months to a year or more, working in the following areas: development of regulatory guidance for advanced reactors and evaluation of computer codes, instrumentation and control room design, seismic analysis of steel structures, review and evaluation of issues pertaining to core physics and fuel behavior, diagnostic analysis and incident investigation, review and assessment of operational experience, review of regulatory program issues, design certification reviews for the AP 600 and ABWR advanced-passive light water reactors, review of regulatory applications issues, including decommissioning rulemaking activities, inspection and enforcement, and all aspects of the development of a regulatory program.

**Nuclear Safety Advisory Committees Meeting in France.** Besides regular bilateral exchange meetings and discussions with the NRC's regulatory counterparts, the Nuclear Safety Advisory Committees for the governments of France, Germany, Japan and the United States met in Luynes, France, in October to exchange information on safety options for future reactors. The three day meeting covered topics such as the safety approach for future PWRs, general safety objectives and principles, integrity of mechanical components, severe accident analysis and research, and future containment design requirements.

## BILATERAL NUCLEAR SAFETY COOPERATION

During fiscal year 1993, the NRC carried on active nuclear safety cooperation programs with a large number of countries. Each of the geographical areas in which the NRC was active reflects somewhat different needs and interests.

### Former Soviet Union

Since the Chernobyl reactor accident in 1986, the United States has recognized a need to cooperate with the Former Soviet Union to improve reactor safety in plants which have been judged less safe in comparison with western designs and practices. Because the Soviet Union had not developed a nuclear safety culture based on a strong, independent regulatory organization, assistance to new regulatory authorities was a high priority at both the Munich (1992) and Tokyo (1993) Summits of the seven industrialized western democracies. Expanded nuclear safety activities were charted at these summit meetings, reflecting the fact that the end of the Cold War opened opportunities for cooperation in areas previously restricted.

**Russia: The Gore-Chernomyrdin Commission.** At the Vancouver Summit between President Clinton and Russian President Yeltsin in April 1993, the United States pledged significantly increased funding to assist Russia in nuclear safety. Additional money is being made available for a similar nuclear safety assistance program in Ukraine. The Presidents also set up a Joint Commission on Technological Cooperation chaired by Vice President Gore and Russian Prime Minister Chernomyrdin. The Gore-Chernomyrdin Commission (GCC) was created to establish a dialogue between the two governments at the political level for expanding cooperation in energy (including nuclear safety) and space technology, and to serve as a forum for jointly resolving practical problems in this important relationship. Later the Commission's mandate was extended to other areas, namely the environment, defense conversion, business development and scientific cooperation.

The United States sought to use the first Gore/Chernomyrdin Commission meeting in September 1993 to emphasize two messages in the nuclear area: (1) U.S. and Western concerns about the continued operation of the least-safe reactors; and (2) the need for a strengthened nuclear regulatory authority. The NRC helped with preparations for the meetings, which emphasized U.S. objectives to:

- Encourage the Russians to introduce risk reduction measures in their least safe plants.

- Enhance the independence and authority of the Russian regulatory body Gosatomnadzor.

- Improve operational training at Russian nuclear power plants through the use of U.S.-designed simulators.

- Help Russia develop emergency operating procedures.

- Complete arrangements for liability protection to permit U.S. industry participation in providing safety assistance.

- Close less-safe Chernobyl-type RBMK and VVER 440/230 reactors.

- Provide a sound economic basis for nuclear safety through market pricing, efficiency measures, conservation, and demand management in the energy sector.

The NRC also took the lead in arranging a highly successful visit to the St. Lucie (Fla.) nuclear power plant in for Prime Minister Chernomyrdin, just prior to his meeting with Vice President Gore. NRC Chairman Ivan Selin hosted the visit. Productive discussions were held with Prime Minister Chernomyrdin and other high-level Russian officials, both in Washington and Florida. The Prime Minister said he was impressed with his St. Lucie visit and believed cooperation with the United States could provide a valuable contribution to advancing nuclear safety in Russia.

**Chairman Selin, at left, with Russian Prime Minister Chernomyrdin, right, and Russian Minister of Atomic Energy Mikhailov, center, are shown visiting the St. Lucie (Fla.) nuclear power plant, prior to the first meeting in Washington, D.C., of the Joint Commission on Technological Cooperation, chaired by the Prime Minister and Vice President Gore.**

In the Washington meetings, the Vice President emphasized the importance of an independent nuclear regulator. He noted that, while the ultimate responsibility for nuclear safety resides with the operators of the power plants rather than the regulator, an independent, legally constituted, well funded safety regulator can assure that the operators achieve the proper degree of vigilance and devote proper attention to safety. The Vice President was able to elicit a commitment from the Prime Minister to a much strengthened nuclear regulatory body and agreement that this issue should be followed up in further Commission activities. On October 16, President Yeltsin issued a decree significantly expanding the Russian regulatory authority's sphere of influence to cover all nuclear facilities.

A successful second round of GCC discussions was held in Moscow in December. A high-level U.S. delegation, with Energy Secretary O'Leary and NRC Chairman Selin representing the U.S. side on the Nuclear Energy Sub-Committee, accompanied Vice President Gore. Important agreements on nuclear safety cooperation and joint principles of nuclear reactor safety were signed and substantial progress was made on resolving a number of other nuclear issues, such as developing a U.S.-Russian Agreement on Radiation Effects Research to create a framework for U.S.-Russian scientific cooperation in the study of health and environmental effects of ionizing radiation. This agreement was signed in January 1994.

**NRC Activities with Russia and Ukraine Under the JCCCNRS and the Lisbon Initiative.** The Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS), established by a U.S.-USSR Memorandum of Understanding in 1988, provides the framework for cooperation between the U.S. and the Former Soviet Union in nuclear safety. The insights gained through exchanges of information, discussions among specialists, and visits to each other's facilities through the Working Groups of the JCCCNRS, as well as certain internationally agreed principles of regulation and reactor safety, provide a solid foundation for the development of recommended improvements in regulatory infrastructures in countries of the Former Soviet Union and Central and Eastern Europe. Elements of a nuclear regulatory program should include:

- Development and acceptance of a legal basis for a strong and independent regulator.

- Provision of adequate resources, both material and personnel, to fund and staff an organization with the ability to monitor plant safety and operations.

- The authority to intervene in operations to insist on safety and, when and if necessary, to shut down a plant in the face of danger to the public.

- The adoption of internationally agreed safety principles.

- Openness and public accountability in reporting incidents which have safety implications, including a public voice in reactor licensing.

For the NRC, the year saw major efforts to implement projects with Russia and Ukraine under the U.S.-Lisbon Initiative of May 1992, which committed the U.S. to move beyond safety information exchanges by spending $25 million for specific projects to enhance nuclear power safety in the FSU. During the latter part of 1992, the JCCCNRS emerged as the coordinating mechanism for providing this assistance and the Chairmen of the Russian and Ukrainian regulatory agencies met with top NRC officials to agree on a program. The U.S. Agency for International Development (AID) completed arrangements to provide funding for nuclear safety activities in both countries. The NRC subsequently negotiated specific implementation plans with the Russian and Ukrainian regulatory representatives, beginning in fiscal year 1993.

A major part of the NRC's program involves technical training, covering all facets of regulatory activity, including licensing and inspection of nuclear power plants, management and funding practices, and accident response. The impact of the programs in both countries is already being felt. For example, using NRC training and documents, both regulatory agencies have drafted licensing and inspection procedures, increased their ability to perform safety analyses, and developed emergency response plans.

The technical program includes delivery and installation of computing and communications equipment which is difficult to obtain in Russia or Ukraine but is readily available in the U.S. After a period of determining specifications and letting contracts, most of the approved equipment deliveries have been made.

Recent examples of NRC assistance to Russia and Ukraine in nuclear safety include the following:

- The NRC developed a program to transfer NRC expertise and methodologies in reviewing U.S. emergency operating procedures to Russian regulatory personnel.

- NRC Technical Training Center personnel conducted a five day seminar on nuclear training methodology, fundamentals of inspection and operator licensing in September 1993 at the Ukrainian facilities in Kiev. The seminar was attended by 18 people. Ukraine has established a Training Methodology Council which will serve a purpose similar to that of the NRC's Training Advisory Group.

- Equipment to support a pilot incident response system arrived in Moscow during the civil unrest in the fall of 1993. The equipment was safely retrieved from the airport and is scheduled for installation in Moscow and at the Kalininskaya and Leningradskya nuclear power plants.

- During fiscal year 1993 there were approximately 30 visits by regulatory personnel, which totalled over 100 Russian and Ukrainian representatives. This amounted to approximately 1,300 person-days of training provided by the NRC.

**Annual JCCCNRS Meeting.** The fifth annual meeting of the JCCCNRS was held in March 1993 in Kiev, Ukraine. This was the first meeting held on a trilateral basis with the independent countries of Russia and Ukraine. It was the successful culmination of discussions, both within the U.S. Government (NRC/Department of Energy (DOE)/Department of State) and with the nuclear power leadership of Russia and Ukraine. The issues were sufficiently complex that pre-meetings at the working level were needed in Moscow and Kiev. The issues included restructuring the JCCCNRS in light of the new Lisbon Initiative activities, the role of the regulator in managing the JCCCNRS, the future of the cooperative exchange program, and the renewal of the 1988 Memorandum of Cooperation which established the JCCCNRS.

A significant outgrowth of the relationship between the United States, Russia and Ukraine is a recognition that each of their nuclear regulatory bodies must establish a position of authority as well as independence from the regulated industry. The NRC helped achieve this by:

- Assisting the Chairmen of the regulatory agencies of the two countries to become Co-Chairmen of the Joint Coordinating Committees for Civilian Nuclear Reactor Safety, which the United States has established separately with Russia and Ukraine.

- Encouraging the regulator and the nuclear power plant operating organization of each country to agree on a process for review and approval of regulatory procedures. This was particularly applicable in Russia with respect to operating a reactor during an emergency.

**Armenia.** In July, the NRC participated in the U.S. Government review of the reported plans by the Armenian Government to re-start the two reactors at Medzamor. The U.S. continues to be concerned about the operation of these early model Soviet-designed VVER-440 reactors, the adequacy of the seismic engineering aspects of the design and the availability of experienced and qualified operators. Regardless of the status of the reactors, the U.S. feels that there is a need in Armenia for an independent and competent regulatory authority. The NRC will continue to monitor the nuclear safety situation in Armenia, and will work with other concerned U.S. agencies and other donor nations in developing an appropriate response.

**Kazakhstan.** During fiscal year 1993, the NRC helped the U.S. Government formulate and implement nuclear policies regarding the new nation of Kazakhstan, which resulted from the breakup of the Soviet Union. Kazakhstan has a sodium-cooled fast breeder reactor, known as the BN-350, located at Aktau (formerly Shevchenko), and a number of other nuclear facilities, including research reactors, fuel-cycle facilities and the Semipalatinsk nuclear test site. During the report period, planning took place for the first visit by an NRC Commissioner to Kazakhstan, which occurred when Chairman Selin visited in early October 1993. The NRC staff is reviewing information related to nuclear safety that has been obtained in technical exchanges in Kazakhstan. The NRC is also monitoring the U.S. Government efforts to obtain from Kazakhstan information on the health and environmental effects of radiation from the nuclear explosions conducted over the years at the Semipalatinsk nuclear test site to see if useful information supplementing that from the Chernobyl accident and the operational problems at the Kyshtym plutonium processing facility will be obtained.

The NRC provided advisory information to Kazakhstan and Canada concerning the discovery of radioactive material in ferrophosphorous imported from Kazakhstan for distribution to steel production plants in the United States and Canada. The ferrophosphorous, intended for use in specialty steel manufacture, was slightly contaminated with cobalt-60, possibly from a radioactive marker in the refractory brick lining at a phosphorous plant in Dzhambul, Kazakhstan. Such markers are commonly used in furnaces throughout the world to measure wear of the brick lining. If such a marker dislodges completely, the entire source combines in the molten metal, making it slightly radioactive. The imported ferrophosphorous was found to present a minimal risk to public health and safety. Any of the material which is determined to be unsuitable for use in steel manufacture will probably be returned to Kazakhstan or sent to a low-level waste disposal site.

In June, an NRC representative described the NRC's programs for nuclear materials control and accounting and physical protection regulation at a seminar in Alma Ata, Kazakhstan, sponsored by the IAEA and the AEA, on the subject of "Organization of State Systems of Accounting and Control." This is part of an effort to help Kazakhstan improve its safeguards system. As noted in a later section of this chapter, an agreement between the United States and Kazakhstan in this area is under development.

## Central and Eastern Europe

Recent dramatic moves in this region toward political democratization and establishment of market economies have also opened greater opportunities for broader cooperation, including in the area of nuclear safety. With a legacy of Soviet-designed reactors, Central and Eastern Europe (CEE) countries also need assistance in improving safety practices, including support for newly established regulatory bodies.

**Training for CEE Specialists.** During the report period, the Agency for International Development (AID) made arrangements to provide funding for an NRC program under which numerous representatives of CEE countries are brought to the United States to attend seminars and training sessions on a range of safety-related subjects.

The NRC prepared and carried out a special training program to familiarize nuclear safety inspectors of four Central and Eastern Europe countries (Bulgaria, Czech Republic, Slovakia and Hungary) with NRC inspection procedures and techniques which could be adapted to their own needs and conditions. Region I was selected to host the training, which was also funded by AID. The purpose of the training was to encourage the countries involved to introduce more formal, explicit, written inspection procedures to aid the work of their inspectors, and to draw on their experience at the NRC to establish closer intraregional professional ties between inspectorates.

In June, an NRC team of inspectors spent over two weeks visiting the regulatory authorities in these countries and touring specific reactor facilities to observe inspection approaches and assess the training needs of each of the nuclear inspectorates. Based on information collected, a two-week training course was developed for the chief inspectors of each country to provide an overview of NRC inspection philosophy, principles, and rationale. Following this, a resident inspector from each country worked alongside NRC regional onsite inspectors for two months, receiving detailed on-the-job training and lectures on key aspects of inspection, participating in team inspections at several plant sites, and attending relevant courses at the TTC and in Region III.

**Bilateral Cooperation with the Czech Republic and Slovakia.** In late September 1993, Chairman Selin visited the Czech Republic to discuss nuclear safety regulation with Czech officials and to visit the Temelin nuclear power plant. The following day he visited Slovakia and the Bohunice nuclear power plant and held discussions on nuclear safety with senior officials.

Mr. Jan Stuller, Director of the new regulatory organization, the Czech Republic State Office for Nuclear Safety (SONS), and Dr. Miroslav Hrehor, Director of Administration, SONS, visited the NRC in the first week of April to discuss organizational and staffing priorities for carrying out their responsibilities with an NRC-like approach.

Mr. Stuller has a past association with the NRC, having spent six months (June-December 1992) as an assignee at AEOD, where he worked in the Reactor Operations Analysis Branch on "Primary System Integrity" issues.

Mr. Stuller and Dr. Hrehor met with Commissioners and key technical managers, explained the recent developments since the split of the Czech and Slovak Federal Republic, and noted possible implications for the role and function of SONS. They gathered pertinent information about the NRC's approach to estimating required staffing levels, based on assigned areas of responsibility and anticipated workload. The visitors used this information on their return to make a persuasive case concerning resources needed to ensure the safety of their nuclear program.

## Western Europe and Canada

The NRC has maintained traditionally strong ties with countries in this region, many of which have active and advanced nuclear programs. The NRC's relationships with these countries enables the U.S. regulatory authority to increase its knowledge of important new technical developments and to harmonize its regulatory approaches with those of other nations to the extent possible.

**France.** Because of the importance of their respective programs and activities, the NRC and the nuclear establishment of France actively continued their regular cooperative exchange activities. During the year, Commissioner de Planque made two official visits to France to participate in international conferences, to exchange information with key officials on nuclear safety and radiation protection matters and to visit a number of nuclear facilities. While there, she was the first Commissioner to meet Andre-Claude Lacoste, who was appointed in March as head of the Directorate for the Safety of Nuclear Installations, the French counterpart to the NRC. Commissioner Curtiss also visited France to discuss U.S. and French approaches to advanced reactor designs, plant standardization and safety requirements for future reactors. He also discussed the development by France and Germany of a European pressurized water reactor, known as the EPR, to be licensed in France and Germany.

The General Administrator of the Commissariat a l'Energie Atomique, the Inspector General for Nuclear Safety of Electricite de France (EdF), and a senior French Parliamentarian also visited the NRC for discussions with the Commissioners about nuclear safety topics of mutual concern. The Executive Vice-President for Engineering and Construction of EdF made a visit to communicate his views on the NRC's proposed revisions to the nuclear power plant siting regulations. Other French organizations also provided written comments on the proposed rule. There was also a regular exchange of visits at the staff level to discuss current reactor operational issues, licensing of advanced reactor designs and waste management plans and activities.

**Germany.** In October 1992, Commissioner Curtiss visited Germany for discussions which included U.S. and German safety philosophies for advanced reactors, standardization and safety requirements for future plants, efforts to develop a European pressurized water reactor design to be licensed in Germany and France, safety assistance to the FSU and CEE, waste management, the International Nuclear Safety Convention and other topics. The Commissioner also visited (1) the BMU (Federal Ministry for Environmental Protection and Reactor Safety), where he met with Director General Walter Hohlefelder and Director of Nuclear Safety Dr. Gast; (2) the GRS (the Company for Reactor Safety, which provides technical support to the BMU); (3) TUV Bayern, Munich (a state-level organization which provides quality assurance, inspection programs and technical advisory services on the safety of nuclear plants); and (4) Siemens Headquarters, where he met with the Director General of Siemens/Kraftwerk Union.

In November, GRS General Manager Dr. Adolf Birkhofer visited the NRC to discuss the status of German nuclear safety assistance to the FSU, and to get an update on U.S. initiatives and ideas in this area. The discussion included (1) an overview of key organizational changes in countries of the FSU and their potential implications for future cooperation; and (2) the possible costs and benefits of distributing limited safety upgrades to many Soviet-designed nuclear plants versus concentrating major upgrades on a limited number of carefully selected plants. In April Dr. Birkhofer revisited the NRC to convey his impressions of recent visits to Russia, Ukraine and Bulgaria and to discuss the membership of the new Nuclear Safety Advisory Committee of the EBRD, which has been established to help make judgments about nuclear safety assistance provided to the FSU and CEE by the multilateral Nuclear Safety Account that is administered through the EBRD.

In September 1993, Chairman Selin visited the Isar nuclear power plant, a late model design of the Konvoi type. The Chairman was especially interested in viewing the substantial physical security measures utilized at German reactors. He also had discussions on nuclear safety with Dr. Birkhofer.

Germany is dealing with the legacy of poor safety practices associated with uranium mill tailings in the former East Germany. In June, two German Parliamentarians visited the NRC for discussions about the NRC's approach to this problem in the United States. They also visited several U.S. mill tailings sites.

**United Kingdom.** In June 1993, Dr. Sam Harbison, the Chief Inspector of the British Nuclear Installations Inspectorate, visited the NRC to meet with the Chairman and Commissioners to discuss nuclear reactor licensing and related issues, new plant designs and the certification process, the fuel cycle and enrichment plant regulation, and regulatory aid to the Former Soviet Union. In addition to NRC Headquarters, he visited the NRC Region V office, the Diablo Canyon (Cal.) nuclear power plant, and the Electric Power Research Institute.

**Finland.** In October 1992 Commissioner Curtiss met with U.S. Embassy and Finnish Government industry and utility officials to discuss approaches to the safety of advanced reactor designs, plant standardization and development of the European pressurized water reactor. Also discussed were Finnish bilateral assistance for Central and Eastern Europe and waste management. Commissioner Curtiss also visited the Olkiluoto BWR, built by ASEA-ATOM, and their low/intermediate waste repository.

**Spain.** In July 1993, the Director General of the Spanish Waste Management Company, ENRESA, visited the NRC to meet with the Chairman and Commissioners to discuss the status of waste management programs in the U.S and Spain, activities at the low-level waste facility near El Cabril, the dual-purpose cask licensing process, inclusion of waste in an International Nuclear Safety Convention, public acceptance of waste disposal, and assistance to Central and Eastern Europe.

**Sweden.** The NRC staff is performing technical studies related to an event that occurred on July 28, 1992, at a Swedish BWR, Barsebaeck-2. While the reactor was operating at low power during restart, a safety valve for the reactor coolant system opened and resultant coolant flow stripped fibrous thermal insulation from piping near the valve. This debris was transported to the suppression pool by the flow of water from the reactor coolant and containment spray systems, and strainers in the containment cooling system were clogged within an hour, causing pump cavitation; this was ten times faster than anticipated by the Swedish regulatory authorities.

In January 1993, Director General Lars Hogberg of the Swedish Nuclear Power Inspectorate (SKI) visited the United States to attend the Probabilistic Safety Assessment (PSA) '93 meeting in Florida and to meet with the Chairman, Commissioners and staff to discuss nuclear safety topics of mutual interest. Discussions focused on the Barsebaeck-2 clogged strainer incident, which had led to the shutting down of the five oldest Swedish plants, on waste management, and on safety assistance to the former Soviet Union and Central and Eastern Europe.

During a visit to Sweden in October 1992, Commissioner Curtiss met with the Minister of Commerce and Industry, the Minister of the Environment, the Director General of SKI, and the President of the Waste Management Company. He also visited the Ringhals reactor site, where he toured the Unit 1 BWR, built by ASEA Atom, and compared it with the Unit 2 Westinghouse-built PWR.

Discussions in Sweden focused on ABB Atom's future reactor designs, including the BWR-90 and PIUS PWR; the future of nuclear power in Sweden; reactor safety issues raised by the Barsebaeck 2 incident and resulting SKI regulatory actions; Swedish assistance to the Baltics and Central and Eastern Europe; progress on the International Nuclear Safety Convention; and status of the Swedish waste program.

**Italy.** The NRC staff is evaluating the Westinghouse test program for the AP 600 advanced reactor design, including an integral systems test facility under construction in Piacenza, Italy, to examine the behavior of passive safety systems during the high-pressure phase of accidents.

**Canada.** During fiscal year 1993 over forty personnel exchanges between the NRC and Canadian nuclear organizations took place. Active cooperation continued on the NRC's review of the preapplication submitted by Atomic Energy of Canada Limited for eventual certification of the CANDU-3 design. There were also active exchanges in such areas as waste disposal, digital instrumentation and control, isotope production, hydrogen combustion research, Thermo-lag for control of plant fires, and emergency response and planning. A high-level delegation from the NRC's counterpart organization, the AECB, also visited the NRC's Technical Training Center in Chattanooga to discuss U.S.-Canadian training for representatives from the Former Soviet Union and Eastern Europe.

Officials from Canada also attended training courses and workshops sponsored by the NRC for radiation control personnel to help them maintain high quality regulatory programs.

## Pacific Rim Countries

This region includes a number of countries with well established nuclear programs (Japan, Korea, Taiwan). Moreover, it has assumed greater importance for the NRC because a number of other countries (Indonesia, China, Thailand, Malaysia) have either embarked on nuclear power programs recently, or have announced an intention to do so, and have requested assistance in developing regulatory programs.

**Japan.** As active partners in nuclear safety cooperation, Japan and the NRC conduct cooperative research as well as information exchanges on regulatory programs.

Commissioner Kenneth Rogers attended two days of the International Conference on Design and Safety of Advanced Nuclear Power Plants in Tokyo the last week in October. The meeting provided an overview of U.S., European, Japanese, and Korean progress in advanced light water reactor designs. While in Japan, Commissioner Rogers also met with the panoply of agencies and organizations directly involved in the development and control of nuclear energy in that country. Commissioner Rogers also visited the Tokai Works to tour reprocessing operations, mixed oxide fuel fabrication and atomic vapor laser isotopic uranium enrichment facilities, and the Large Scale Test facility to discuss its capabilities for conducting thermal hydraulic experiments.

In October 1992, Commissioner Forrest Remick visited Japan to participate in the International Conference on Design and Safety of Advanced Nuclear Power Plants in Tokyo as an invited speaker, and to meet with representatives of the Ministry of International Trade and Industry (MITI) for discussions on current energy and nuclear safety issues. In April 1993 Commissioner de Planque attended the Japan Atomic Industrial Forum's annual international meeting in Yokohama, Japan. Besides addressing the Forum as a keynote speaker, the Commissioner visited the Tsuruga nuclear power plant, the Monju fast breeder reactor facility, the Orai and Tokai research facilities, the Rokkashomura enrichment and reprocessing facilities, and the Kashiwazaki-Kariwa BWR and ABWR nuclear power units and held discussions on nuclear safety with nuclear officials.

In October 1992, the 7th MITI–NRC Regulatory Information Exchange meeting was held in Washington to discuss ongoing NRC–MITI cooperation. The MITI delegation subsequently toured the Millstone nuclear power plant, which was of particular interest to the Japanese because of its ongoing steam generator replacement work. In the same month, a delegation of Japanese mayors, assemblymen, and other local government officials from Japanese cities and towns with nuclear energy facilities was given a presentation on how the U.S. nuclear regulatory system works, with particular attention to such issues as steam generator replacement, plant life extension, and low-level waste.

The NRC is reviewing an application from GE Nuclear Energy for final design approval and design certification of its simplified boiling water reactor (SBWR) design, which relies on passive systems for reactor safety. As part of this work, the NRC is monitoring the vendor's test program to support the design. For example, in-service testing programs have been carried out at the Toshiba facility in Japan for this purpose.

**Republic of Korea.** During the year, the NRC provided continuing support to the Korean nuclear safety community, with greater emphasis on close cooperation in the area of operating reactor inspections.

In March, two inspectors from the Korean Institute for Nuclear Safety (KINS) began nine weeks of classroom instruction at the Technical Training Center in Chattanooga. They were enrolled in the CE Technology/Simulator series, following which they were assigned for two months in Region V to work with NRC inspectors learning how the NRC prepares for, conducts, writes up and analyzes results of inspections. The final phase of training was a three-month, "hands-on" assignment in the Resident Inspector's Office at Palo Verde, the nearest reference plant to Korea's Yonggwang Units 3 and 4, on which the Koreans have based their standardized design.

As part of the NRC's short term inspection assignments, two of which took place in July when two KINS inspectors arrived in Region I to prepare for, and then accompany, a motor-operated valve inspection at the Diablo Canyon (Cal.) nuclear power plant. The two inspectors in the longer term NRC assignments described above accompanied a radiation protection (chemistry team/mobile lab) inspection at Palo Verde (Ariz.) plant during this same period.

The NRC hosted the visits of several other Korean nuclear officials for technical discussions in such areas as advanced reactors, station blackout, emergency planning and response, the technical specifications improvement program, and NRC audit and investigation activities. Korea invited Commissioner Rogers, in October 1992, to deliver the keynote address to the annual meeting of the Korean Physics Society and to meet on the sidelines with selected nuclear officials. Commissioner Remick in November 1992 made visits to, and held discussions with, the broad Korean nuclear community. The Director of the Office of Nuclear Reactor Regulation (NRR), Thomas Murley, in July 1993, held discussions with several Korean nuclear organizations on their experience with construction of advanced reactors and how that experience may apply to the NRC's construction program under 10 CFR Part 52.

**China.** In recent years, the NRC has limited its cooperation with China to providing publicly available safety documents. However, with increased U.S. interest in China's expanding nuclear power program, which includes second and third nuclear power reactors at the Guangdong nuclear power plant (due to become operational in late 1993 and mid1994, respectively) and planning for a third power station, the NRC has broadened contacts on issues of nuclear safety.

In January 1993 Chairman Selin visited China and renewed the NRC–NNSA Protocol between the NRC and the Chinese National Nuclear Safety Administration (NNSA), which provides for the exchange of nuclear safety information in activities related to nuclear power generation, radiation protection, and nuclear material safety. While there, the Chairman encouraged the Chinese nuclear authorities to strengthen their emergency preparedness techniques and procedures and visited China's two nuclear power reactor sites.

In part as a result of this visit, two delegations from the PRC visited the NRC and selected nuclear power plants in the United States. In February, the Vice Governor of the Guangdong Province toured the Susquehanna nuclear power plant and observed its emergency exercise from the Pennsylvania Emergency Management Agency center in Harrisburg. In August, the NRC hosted twelve representatives from various governmental and utility organizations sponsored by the Emergency Response Office of the China National Nuclear Corporation. Besides taking a tour of the NRC Emergency Operations Center in Bethesda and presentations by staff of the Emergency Preparedness Branch, Division of Radiation Safety and Safeguards, NRR, the group attended an emergency exercise at the Zion (Ill.) nuclear power plant.

Commissioner Remick visited China in the spring of 1993 for an in-depth review of China's nuclear power program. He met with government, industry and utility representatives to discuss their current energy program and explore possible areas of safety cooperation. Now that China has two operating power reactors, the focus of the NRC's cooperation will be expanded to include areas of plant operation, maintenance, and inspection. Commissioner Remick suggested to the NNSA that they consider the temporary assignment of some of their resident inspectors to NRC Regional Offices to work with NRC inspectors.

**Taiwan.** In addition to routine exchanges of documents and regulatory personnel pursuant to the cooperation program on civil nuclear matters under the agreement between the Coordination Council for North American Affairs (CCNAA) and the American Institute in Taiwan (AIT), various high-level NRC visitors traveled to Taiwan for discussions with Taiwan regulatory authorities and for tours of their nuclear facilities.

In November 1992, Commissioner Rogers met with Taiwan nuclear authorities and representatives of the utility, Taiwan Power Company (Taipower), and toured the Kuosheng nuclear power plant. In January 1993, Chairman Selin, the highest ranking official U.S. visitor to Taiwan since the 1979 change in Taiwan's diplomatic status with the United States, held meetings with nuclear regulatory and utility authorities and toured the Chinshan nuclear power plant. NRR Director Murley visited Taiwan in July 1993 for an update on programs and issues and to learn about Taiwan construction practices.

The NRC hosted several high-level Taiwan visitors during the year to discuss cooperative activities and the status of the proposed new two-unit Lungmen nuclear power plant site as well as to discuss Taiwan's views on the proposed changes to 10 CFR Part 100, life extension of nuclear power plants, and low-level waste management.

**Indonesia.** The NRC continued to cooperate with Indonesia in its effort to develop a nuclear regulatory program. Current and planned efforts are keyed to 1995, when Indonesia intends to request tenders for the first of two 600-megawatt power reactors to be built near Mount Muria in north-central Java (about 450 kilometers east of Jakarta). Chairman Selin visited Indonesia in January 1993 to discuss with senior officials what the NRC could do to enhance the safety of the planned Indonesian nuclear power program. One possibility was the NRC's offer of on-the-job training assignments in safety and

**Commissioner Forrest J. Remick visited the Qinshan nuclear power plant in China during the report period.**

regulation for selected National Atomic Energy Agency (BATAN) personnel. The NRC has agreed to accept four BATAN staff-members-per-year for the next three years for on-the-job training and classroom instruction at the Technical Training Center in Chattanooga, beginning in January 1994. The head of the BATAN Atomic Energy Control Board paid a return visit to the NRC in April to discuss the proposed training assignments as well as NRC views on effective organization of their regulatory staff.

An NRC representative also participated in the U.S.-Indonesia Energy Bilaterals in Jakarta in July. The U.S. team was led by DOE and included participants from the Department of State as well as the NRC. The NRC presentation covered the NRC's role and responsibilities within the U.S. nuclear power regime, its international activities, principles of good regulation, and current and emerging safety issues.

**Philippines.** Chairman Selin visited the Philippines in January 1993 to meet with senior government policy makers and nuclear safety officials and to visit the Bataan nuclear power plant, which had been mothballed in 1986. The government of the Philippines had by then decided not to operate the plant and it was decided that there was no need for the NRC and Philippine nuclear safety officials to renew their regulatory Information Exchange Arrangement. However, as an outcome of the meeting, a Letter of Intent to maintain contact between the two organizations and to exchange nuclear safety information routinely was signed in July 1993.

**Thailand.** In July 1993 representatives of the Thai Senate Committee on Science, Technology and Energy visited the NRC to get an overview of the work done by an independent nuclear regulatory institution. Because of the rapid growth in consumption of electrical energy in recent years, the Thai Government has been investigating the possibility of investing in a nuclear power plant, with a decision likely to be made in the next three or four years.

OIP Director Carlton Stoiber described the agency's mission, and introduced presentations on low-level waste management regulation, reactor health and safety regulation, and the legal basis for a regulatory program. The NRC provided the Senate Committee with a set of basic regulatory documents with which to begin a regulatory program.

## Latin America

The three largest countries of Latin America—Argentina, Brazil, and Mexico—all have long-standing nuclear programs. Recent initiatives by Argentina and Brazil in the nonproliferation area have increased opportunities for nuclear cooperation with the United States, giving an impetus to NRC relationships with counterpart organizations.

**Argentina.** In November 1992 the U.S. Government and the Federal Republic of Argentina concluded negotiations on a new Peaceful Nuclear Cooperation Agreement. Although this agreement has not yet been sent to the President by the Executive Branch for subsequent transmittal to Congress for approval, the successful negotiation marks a change in the relationship between the two countries. A harbinger of that change was the signing in 1990 by the NRC and the Argentine Comision Nacional de Energia Atomica (CNEA) of an Information Exchange Arrangement, which has provided a channel for communication on safety issues. However, until Argentina implements full-scope safeguards, the United States cannot authorize significant nuclear exports.

Chairman Selin visited Argentina at the end of November 1992 for meetings with nuclear and governmental authorities, and to tour the Atucha I nuclear power plant. The Chairman encouraged the CNEA to make formal proposals of areas of possible collaboration under the NRC-CNEA agreement. A representative of the Argentine Ministry of Foreign Affairs visited the NRC in May 1993 to discuss possible cooperative activities between the CNEA and the NRC. Of particular interest to Argentina are training opportunities and involvement in computer code development and maintenance programs.

**Brazil.** In November 1992, Chairman Selin visited nuclear regulatory and governmental authorities in Brasilia and Rio de Janeiro and toured the Angra I nuclear power plant in Angra dos Reis, where he encouraged officials at the Comissao Nacional de Energia Nuclear (CNEN) to make formal proposals for cooperative activities under the NRC-CNEN Information Exchange Arrangement. However, because of the extensive governmental reorganization in Brazil in early 1993, including the CNEN, there have been few exchanges. An exception was the visit in May of the Special Assistant to the President of Brazil's agency for environmental protection (FEEMA) and newly appointed head of FEEMA's Radioactive Control Program. He conducted discussions with NMSS and OIP staff to get background on the identification and regulation of a variety of radiation sources and also discussed low-level waste, training for inspectors, and the development of an equivalent to the NRC's Agreement States program.

**Mexico.** In January 1993, Commissioner Curtiss visited the site of Mexico's Laguna Verde nuclear power plant and met with key Mexican nuclear officials in Mexico City to discuss work being done by the Mexican utility CFE in the area of risk-based regulation. There were also several other interactions during the year between NRC staff and Mexican technical personnel on this issue. The implementation of risk-based regulation in Mexico is comparable to that in the United States. The Laguna Verde plant has undertaken a major effort to use probabilistic risk assessment for such purposes as setting priorities for the resolution of safety issues, supporting rulemaking, and continued plant operation, and for making operational and maintenance decisions.

During fiscal year 1993 the CNSNS provided the NRC with a significant amount of information on operating events at Laguna Verde. Construction of a second unit continued, although at a slower place, and six operators were licensed for this unit during the year by the CNSNS.

Mexican officials also attended training sessions sponsored by the NRC for radiation control personnel to help enhance their regulatory program. The NRC's cooperative work with Mexico on nuclear waste disposal is discussed at the end of the section of this chapter on Cooperative Nuclear Safety Research.

## Africa and Middle East

The NRC has had only modest involvement with countries in these two regions, not only because of a lack of active nuclear power programs there, but also for broader policy reasons. Recent developments in Southern Africa and elsewhere may warrant an expanded role in regulatory cooperation with certain countries in these regions.

**South Africa.** With South Africa's public acknowledgement of a past nuclear weapons program, their decision to terminate this program, adherence and submission of all nuclear facilities to IAEA safeguards, as well as their active negotiations with the United States on a revised Agreement for Cooperation, the United States is restoring its peaceful nuclear cooperation with South Africa.

In July, Drs. J.W. de Villiers, Chairman, and Waldo Stumpf, Chief Executive Officer, of South Africa's Atomic Energy Corporation met with the Commissioners and senior staff to discuss a wide range of safety and regulatory topics. The visitors expressed an interest in developing closer ties in the safety area with the NRC. A few members of their regulatory counterpart to the NRC, South Africa's Council for Nuclear Safety, visited the NRC for discussions on licensing of reactor operators and environmental monitoring requirements and practices.

**Israel.** In July 1993 the NRC and the Israeli Atomic Energy Commission (IAEC) renewed the NRC–IAEC Information Exchange Arrangement for five years, following which the Commissioners held discussions with Israeli officials. During the visit, the Israeli IAEC Director General noted that Israel is technically capable of operating nuclear power plants and the country has few alternate energy sources, but its security situation has made it difficult to identify a suitable site for a plant. The NRC and the IAEC continue to exchange documents and information on operating conditions under the arrangement.

## MULTILATERAL NUCLEAR SAFETY COOPERATION

Besides its extensive program of bilateral cooperation with other countries, the NRC works closely in the area of nuclear safety with international organizations such as the International Atomic Energy Agency in Vienna, and the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD) in Paris. For example, the NRC employs data received from other countries through both agencies on events at their nuclear power plants in comparative studies of reactor operational experiences that may produce information applicable to the safety of U.S. reactors. Reports of operational events received from the NEA's Incident Reporting System, from the IAEA and from bilateral exchange programs with over 20 countries, are used by the NRC to supplement domestic data, and the NRC provides U.S. incident reports to the international community as well using these mechanisms. Chapter 3 provides further information on this program.

### IAEA Activities

**International Atomic Energy Agency (IAEA) General Conference and Board of Governors Meetings.** Concern about North Korea's nuclear activities and its announced withdrawal from the Nuclear Non-Proliferation Treaty (NPT) was the paramount issue at the IAEA's General Conference, held in Vienna in September 1993. The U.S. delegation was led by Energy Secretary Hazel O'Leary, with Chairman Ivan Selin serving as an alternate U.S. delegate. Nuclear safety agenda items focussed on IAEA activities related to the safety of nuclear power plants in Central and Eastern Europe and countries of the Former Soviet Union, strengthening radiation protection and nuclear safety infrastructures in the countries of the Former Soviet Union, the IAEA's safety services, safety principles for future nuclear power plants, preparation of an International Nuclear Safety Convention and the implementation and status of the Conventions on Early Notification of a Nuclear Accident and on Assistance in the Case of a Nuclear Accident or Radiological Emergency. A resolution endorsing the early completion of an International Nuclear Safety Convention was also adopted.

During the General Conference, Chairman Selin participated in bilateral meetings held by Secretary O'Leary with several delegations, including Russia, Japan, Germany, China and the United Kingdom. Chairman Selin, accompanied by the NRC Executive Director for Operations, James Taylor, and OIP Director Carlton Stoiber, also met with delegation heads from Armenia, Lithuania, the Czech Republic, Slovakia, Bulgaria, Slovenia, Hungary, Ukraine, Indonesia and Sweden. Mr. Taylor presented two papers at the Senior Regulators' session held during the General Conference, where regulators from twenty-five countries discussed enhancing safety culture, periodic safety reviews and relicensing, and licensing of a foreign-designed plant. Mr. Taylor received briefings by several senior IAEA officials on their programs.

Mr. Taylor was invited by the newly formed VVER regulators' group to participate as an observer at their organizational meeting held during the week of the General Conference. The United States and Germany were both invited to attend the first regular meeting of the group in Sofia, Bulgaria in December.

NRC officials were also members of the U.S. delegation to the February and June sessions of the IAEA Board of Governors.

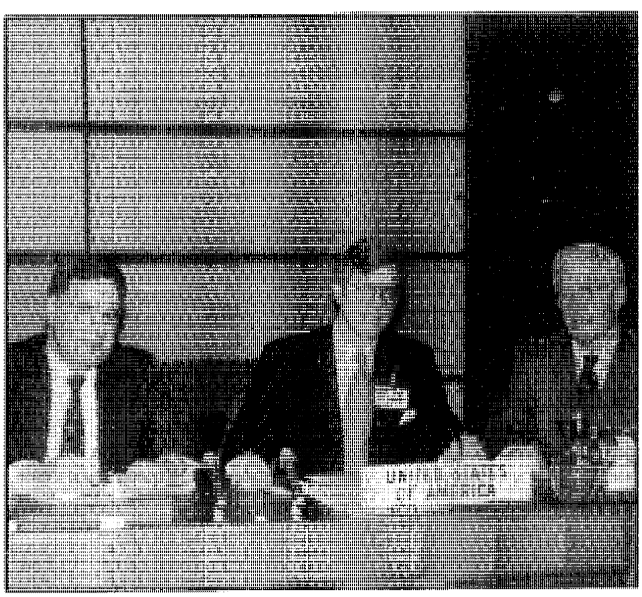
In May the Director General of the IAEA, Hans Blix, met with Commissioners in the United States to discuss a variety of current nuclear safety and safeguards issues.

**International Nuclear Safety Convention.** The NRC has had a leading role in the expert working group which is elaborating an International Nuclear Safety Convention. The October working group meeting produced a schedule which could lead to the Convention being opened for signature by the time of the IAEA General Conference in 1994. Two of the issues still to be resolved are the peer review process and the financing of meetings of the parties, which will be the primary implementation mechanism for the Convention.

**IAEA Meeting Participation.** During fiscal year 1993, NRC staff participated in 40 IAEA meetings on a wide range of nuclear safety issues, including: qualification and competence of nuclear power plant personnel, operator training, emergency planning, accident management, incident reporting, performance indicators, regulation of radiation sources, industrial applications, physical protection of nuclear materials, transportation of radioactive material, seismic probabilistic safety, safety assessment and siting of near-surface radioactive waste disposal facilities, long term storage of spent fuel, safety of shut down nuclear installations, fire protection, reactor aging management programs, safe operation of research reactors, and assistance efforts in the FSU and CEE. The NRC also nominates numerous participants from outside the NRC (usually the non-Federal sector) for attendance at other IAEA meetings related to the NRC's interests. (See Chapter 3 for discussion of NRC involvement in IAEA meetings on the development of performance indicators.)

## OECD/Nuclear Energy Agency

**Nuclear Energy Agency (NEA) Activities.** The NRC maintained an active involvement in OECD/NEA activities by serving on Standing Committees and Working Groups and participating as Deputy Head of the U.S. Delegation to two Steering Committee meetings in fiscal year 1993. With such a large share of the NEA's work directly related to the NRC safety priorities and U.S. nuclear interests, the NRC benefits greatly from representation on committees and working groups and regularly attending meetings. As an essential element of the Commission's agenda of international activities, 30 senior NRC staff members actively participated in various Standing Committees and Working Groups, with activities focusing on nuclear safety matters. This included participation on: (1) the Committee on Nuclear Regulatory Activities (CNRA); (2) the Committee on the Safety of Nuclear Installations (CSNI); (3) the Committee on Radiation Protection and Public Health (CRPPH); (4) the Committee on Radioactive Waste Management; and (5) the Group of Governmental Experts on Third Party Liability in the Field of Nuclear Energy.




**The NRC has participated actively in efforts to develop an International Nuclear Safety Convention and continued to do so during fiscal year 1993. Taking part in a negotiating session at the International Atomic Energy Agency in Vienna are, left-toright, Jack Heltemes, Deputy Director of the NRC Office of Nuclear Regulatory Research,; Richard Stratford, from the U.S. State Department; and Carlton Stoiber, Director of the NRC Office of International Programs.**

In fiscal year 1993, the Director of the Office of Nuclear Reactor Regulation was elected Vice Chairman of the CNRA, the Director of the Office of Nuclear Regulatory Research became Vice Chairman of the CSNI, and the Director of the Division of Industrial and Medical Nuclear Safety became Vice Chairman of the CRPPH.

The NRC also participated in five cooperative international research projects: the Halden Reactor Project, the International Alligator Rivers Project, the INTRAVAL Study, the Program for Inspection of Steel Components, and the TMI Pressure Vessel Examination Project. These projects enabled the NRC to save substantial amounts of money and time in developing answers to key safety questions.

During the year, NEA Director-General Dr. Kunihiko Uematsu, Deputy Director for Safety and Regulation of Nuclear Activities Klaus Stadie, and Deputy Director for Safety and Regulation and Head of the Division of Nuclear Development Geoffrey Stevens, visited the United States to discuss nuclear safety matters of mutual interest. During the period Commissioner de Planque chaired an NEA workshop in France on Radiation Protection Toward the Turn of the Century. The purpose of this workshop was to formulate a collective opinion on radiation protection, expected to be published in 1994. She also met in Paris with the U.S. Ambassador to the OECD and with Dr. Uematsu of NEA. Commissioner Curtiss also visited the NEA for nuclear safety discussions.

The 25th anniversary of the NEA Steering Committee was held in October in Tokyo, Japan, to commemorate the anniversary of the accession of Japan to the NEA, while also marking the NEA's opening of its membership to countries with nuclear power outside of the West. The Steering Committee took the following actions: (1) approved the revised mandate for the Radioactive Waste Management Committee, (2) approved forwarding the proposed 1993 Main Lines of the Program of Work and Budget to the OECD Council for final approval, (3) supported, after a very lengthy discussion, a cautious step-by-step, case-bycase consideration of non-OECD countries for membership, and (4) directed that a letter be sent to the OECD Secretary General supporting Korea (a non-OECD country) as a member of the Steering Committee. The meeting was followed by a seminar on "International Nuclear Energy Development and the New Trends of International Cooperation," as well as a site visit to the JAERI and PNC facilities at Tokai-Mura.

The 26th meeting of the NEA Steering Committee was held in April in Paris. The committee approved in general the Main Lines of the Program of Work and estimates of expenditures for 1994. It also approved in principle an NEA-sponsored international project (RASPLAV) involving OECD members and the Russian Research Center (I.V. Kurchatov Institute), designed to ascertain under what conditions a degraded, molten core can be retained within the reactor pressure vessel by external cooling. This is of great interest to regulators in a number of countries, including the NRC.

The Steering Committee also deferred until the fall meeting a decision on a proposed recommendation by the OECD council on informing the population about radiological emergencies, noted the nearly completed procedures for the admission of Korea to the NEA (the first non-OECD member country to join the NEA), and directed the Director General to respond negatively to a joint letter from Argentina and Brazil (both non-OECD member countries) requesting exploratory contacts with the NEA, pending progress in moving toward full-scope safeguards.

**European Community (EC).** In October, Commissioner de Planque made the first Commission-level visit to the EC in Brussels, reflecting the increasing importance of the NRC's cooperation with the EC. During the visit, Commissioner de Planque had discussions with the Directors General of the Directorates for Energy, Environment and Nuclear Safety, and the Joint Research Center.

Topics discussed included EC assistance activities involving Eastern Europe, Russia and Ukraine, the activities of the G-24, safeguards cooperation between Euratom and the IAEA, future trends in EC nuclear safety research, and the International Nuclear Safety Convention. A particular highlight of the meetings was an expression of mutual concern for the reactor situation in Eastern Europe and the FSU and an indication of each organization's willingness to cooperate on safety assistance activities.

## G–7, EBRD/Nuclear Safety Account and G–24

**G–7 Nuclear Safety Working Group.** During preparations for the July 1992 Munich Summit of Western industrial democracies, the G–7 nations decided to establish a Nuclear Safety Working Group (NSWG) to develop a program to address safety problems with reactors of Soviet design. The NSWG developed a near term program of nuclear safety assistance which included three elements: operational safety improvements, near term risk-reduction measures (fire protection, instrumentation and control upgrades, etc.) and regulatory enhancement. The NRC has led U.S. efforts in the regulatory enhancement area, which are described above in the section on Russian Federation and Ukraine. Another initiative launched by the G–7 NSWG was establishment of a special multilateral fund for nuclear safety assistance to be administered through the European Bank for Reconstruction and Development (EBRD). The G–7 also requested that the Group of 24 coordinate bilateral nuclear safety assistance.

The July 1993, Tokyo Summit requested the NSWG to develop a long-range strategy for energy development to support closure of the highest risk plants in the New Independent States (NIS) of the former Soviet Union. During the report period, OIP Office Director Carlton Stoiber, who originally led U.S. delegations to the G–7 NSWG as an official of the Department of State, continued to participate in meetings of the group as a representative of the NRC. Several meetings were held (Tokyo in May, Vienna in September, London in November) to discuss how energy sector studies prepared at the G–7's request by the World Bank, International Energy Agency and EBRD could be used to develop options for the closure of less safe plants in the NIS and Eastern Europe. At a January 1994 meeting in Washington with Executive Directors of the World Bank, the NSWG further developed issues regarding long term energy planning for eventual discussion and action at the next G–7 Summit, scheduled for July 1994 in Naples, Italy.

**EBRD Nuclear Safety Account.** As part of the G–7 nuclear safety initiative discussed above, it was decided in the autumn of 1992 to establish a multilateral fund to provide grant financing to countries utilizing Soviet-designed reactors for near term technical safety improvements. In March 1993 the fund was formally established as the Nuclear Safety Account (NSA), to be administered by an Assembly of Donors (and eventually a Steering Committee) at the European Bank for Reconstruction and Development (EBRD) in London.

The first two projects funded under the NSA were in Bulgaria (for upgrades to the Kozloduy facility) and Lithuania (for upgrades to the Ignalina facility). The NSA was initially established for a three-year period, with the possibility of extension for another three years. Current pledges to the fund total in excess of 100 million European Currency Units (or $115 million).

**G–24 Nuclear Safety Assistance Coordination Activities.** The G–24 is a group of nations which have joined together to coordinate their economic assistance programs for the countries of Central and Eastern Europe. Assistance in the area of nuclear safety is coordinated by a special G–24 group which has a broader charter that includes safety assistance to the Former Soviet Union as well. The group has established an organizational structure, including a Plenary, Steering Committee, and Technical Working Groups, which meet periodically to discuss coordination of various safety assistance efforts. A Nuclear Safety Assistance Coordination Center was established in Brussels, Belgium under the auspices of the G–24 to develop a data base of information related to nuclear safety assistance activities, to develop recommendations to minimize the likelihood of duplication of efforts, and to identify any potential assistance gaps. The NRC, as part of U.S. Government efforts to support the G-24 coordination process, participated actively in G-24 meetings and activities related to nuclear safety, including providing an NRC staff engineer for a five-month temporary assignment to Brussels to assist the Coordination Center.

## COOPERATIVE NUCLEAR SAFETY RESEARCH

The NRC conducts confirmatory regulatory research in partnership with nuclear safety agencies and institutes in more than twenty countries. Much of this activity is concentrated in three major subject areas: Severe Accident Research; Thermal/Hydraulic Code Maintenance and Assessment; and Piping Integrity and Material Research. Over fifty agreements are currently in force covering the NRC's international research work. Such agreements provide for shared use of research facilities, joint funding arrangements, prompt exchange of experimental results, coordinated analyses, and other forms of cooperation to produce confirmatory safety data of mutual benefit in a timely and cost effective manner.

Examples of activities conducted in fiscal year 1993 under the NRC's international nuclear safety research program are the following (see Chapter 9 for more information regarding these activities):

- Using the ROSA Large-Scale Test Facility in Japan to do confirmatory safety system testing to help provide technical bases for NRC licensing decisions on the AP 600 advanced reactor design. Modifications to provide cost-effective simulation of the AP 600 design were scheduled to be completed, and a series of tests performed, during 1994.

- Cooperating internationally to develop practical advanced analytic methods to improve predictions of pressure vessel fracture and assess integrity of pressure vessels under various operating conditions. This includes collaboration with a European Community program to simulate closely pressure vessels subjected to accident loading.

- Reviewing data from researchers in Russia, the Czech Republic and elsewhere in Eastern Europe, the United Kingdom and other European countries related to reactor pressure vessel (PV) embrittlement under intensive neutron bombardment, and thermal annealing of the vessel to mitigate embrittlement effects. This work includes studies sponsored by the NRC at the Russian Research Center (I.V. Kurchatov Institute) involving irradiation of samples of a U.S. pressure vessel at a Russian power reactor, followed by annealing and re-irradiation in the same reactor to simulate a way of continuing plant operations as PV embrittlement approaches. The information obtained will be a very valuable addition to the U.S. data base. Based on all the international efforts and other domestic research work, the NRC staff drafted a regulatory guide on thermal annealing for U.S. plant designs.

- Irradiating various stainless steel samples in the Halden reactor in Norway as part of an investigation of irradiation-assisted stress corrosion cracking of reactor core internal components, which becomes greater as reactors age and core materials absorb greater neutron flux.

## EXPORT AND IMPORT LICENSING

**NRC Export/Import Role**. Under the Atomic Energy Act of 1954, as amended, the NRC is responsible for licensing the export and import of nuclear-related materials and equipment to ensure these items are used only for peaceful purposes. This authority extends to production and utilization facilities, to special nuclear and source material, to byproduct materials, to certain nuclear-related components, and to other materials. In carrying out these responsibilities, the NRC obtains the views and recommendations of other governmental agencies and departments as needed or required. The NRC also is consulted by the Executive Branch on nuclear-related, dual-use exports licensed by the Department of Commerce (DOC), as well as nuclear technology transfers and nuclear material retransfers (subsequent arrangements) licensed by DOE. The NRC is also consulted by the Department of State (DOS) regarding agreements for nuclear cooperation between the United States and other countries. In fiscal year 1993, 113 technical international safeguards reviews were performed regarding export applications, agreements for nuclear cooperation, subsequent arrangements, and technology transfers.

**NRC Export Licensing Summary**. In fiscal year 1993, the NRC completed 125 export licensing actions. Of these, 60 involved export to EURATOM, Japan, Korea, Switzerland, and Taiwan of fuel for power reactors using low-enriched uranium. The NRC also issued five licenses authorizing the export of low-enriched uranium for use in foreign research and test reactors. No such licenses were issued for high-enriched uranium (HEU) exports, but two export shipments, greater than five kilograms, were made under authorizations from earlier years. The Energy

Policy Act of 1992 placed further restrictions on exports of HEU for research and test reactors, allowing new authorizations only if (1) there is no lower enriched fuel or target material available that can be used in the research or test reactor; (2) the recipient has provided assurances that it will use lower enriched material when it becomes available; and (3) the U.S. Government is actively developing such alternative material that can be used in the reactor.

**Report on U.S. Exports of High-Enriched Uranium.** In January the Commission submitted to Congress a report on the current disposition of previous U.S. exports of highenriched uranium. The report, completed within a prescribed 90-day period, was mandated by the Energy Policy Act of 1992.

**Transportation Safeguards and Safety for Nuclear Materials.** During this period, the NRC received about 200 notifications by its licensees of international shipments of special nuclear material and natural uranium for forwarding, when appropriate, to international authorities.

During fiscal year 1993, the NRC completed feasibility studies related to the testing of packages used for air transport of plutonium from one country to another through U.S. air space. The feasibility studies and testing were requested and funded by the Power Reactor and Nuclear Fuel Development Corporation (PNC) on behalf of the Japanese Government. (See Chapter 5 for discussion of transportation safeguards and transportation activities in fiscal year 1993.)

Also during fiscal year 1993, the NRC worked closely with DOE and other U.S. Government agencies to prepare a report to Congress on the safety of plutonium shipments by sea that is required by the Energy Policy Act of 1992.

During this report period, NRC staff assisted the Ukrainian nuclear regulatory body to establish a regulatory program for the transportation of radioactive materials, including standards and regulations, emergency response, and inspection and enforcement. Funding was provided by the U.S. Agency for International Development.

**International NPT Exporters Committee (Zangger Committee).** The NRC continued active support of the multilateral Zangger Committee, a group formed to help assure consistency in the application of nuclear export controls pursuant to the Treaty on the NonProliferation of Nuclear Weapons (NPT). The NRC provided regulatory guidance and implemented the obligations undertaken by the United States in the framework of the Committee. The NRC added a new appendix to its licensing regulations in 10 CFR Part 110 for clarification of the coverage of specially designed or prepared equipment for use in a plant for the production of heavy water, deuterium and deuterium compounds. This action conformed the NRC's regulations with those of other Zangger Committee countries.

**Nuclear Suppliers Group.** The NRC supported U.S. efforts to enhance multilateral export controls of the international Nuclear Suppliers Group, an entity similar in form, function and membership to the Zangger Committee, but without the direct linkage to the provisions of the NPT. It is thus able to address export controls with a broader perspective. Efforts continue by the Executive Branch agencies and the NRC to conform U.S. export regulations to the Nuclear Suppliers Group Guidelines for the export of dualuse commodities.

**Subgroup on Nuclear Export Coordination (SNEC).** The NRC continues to participate in this interagency body, which meets regularly to reach consensus decisions on export license applications which raise nuclear proliferation concerns. SNEC serves as a forum for exchanging and coordinating views among member Federal agencies on nuclear export licensing activities of the Department of Commerce, nuclear technology transfers authorized by the Department of Energy, and exports licensed by the Nuclear Regulatory Commission. Cases are referred to SNEC because of country destination, concern about end user/commodity, precedent setting nature of the proposed export, and agency request. SNEC reviews about 300-400 export cases annually. A large number of cases involve dualuse exports, but most involve computer exports. In 1993, 769 export cases were reviewed by SNEC, mostly involving computers. However, this number is expected to drop substantially in 1994 as a result of the Department of Commerce loosening its specific licensing controls over computer exports.

**Department of Energy Technology Transfers.** The NRC worked with DOE by providing comments and support for expedited procedures to process safety-related transfers of nuclear technology (training, advice, licenses and other assistance separate from exports of nuclear materials and equipment). Several cases involved power reactor training or engineering support to countries such as Russia, Ukraine, the Czech Republic, Argentina, Brazil and South Africa.

**Revisions to Department of Commerce Export Regulations.** This past year, the NRC worked with DOC on revising DOC's export regulations to cross reference those nuclear items licensed by DOC with related items licensed by the NRC, to emphasize to exporters that other U.S. export requirements may be applicable.

**Amendments to the NRC's Export-Import Regulations.** Three rules were published pertaining to the export-import of nuclear equipment and material. A final rule was published on March 9, 1993, to provide exporters with a better understanding of the scope of Part 110. Sections were restructured and simplified, and a new appendix was added listing byproduct materials under the NRC's licensing authority. Also, embargoed and restricted destinations were revised to reflect changes in proliferation concerns regarding some countries. On March 17, 1993, the NRC published a proposed rule to conform U.S. export controls to recently-agreed international guidelines. On October 28, 1993, a final rule was published regarding additional licensing criteria for the export of high-enriched uranium to implement Section 903(a) of the Energy Policy Act of 1992.

Work continued on proposed amendments to the NRC's import and export regulations to reflect the recommendations of the 1990 IAEA Code of Practice on International Transboundary Movement of Radioactive Waste. The amendments would tighten the NRC's controls over the import and export of low-level radioactive waste by requiring specific licenses for such material to enter or leave the United States.

**Strengthening Export Controls in the FSU.** A U.S. Government effort is in progress to assist republics of the Former Soviet Union in strengthening their export control systems to prevent further proliferation of weapons of mass destruction and associated technologies. In this connection, in June an NRC representative spoke at a conference in Virginia in which U.S. agencies involved in export control described their roles and responsibilities in the export control process, and the associated staffing and resources, for representatives of several of the New Independent States of the Former Soviet Union. There were also presentations on the global non-proliferation/export control regime, the role of export control in U.N. Security Council sanctions and the role of U.S. industry in the export process. Discussions included controls on nuclear-related and dual-use items and effective export enforcement. The representatives of the States of the Former Soviet Union also described their current approaches to export control and plans to strengthen these regimes.

## INTERNATIONAL SAFEGUARDS AND PHYSICAL PROTECTION ACTIVITIES

The NRC staff reviews pending export cases to confirm that appropriate IAEA safeguards and physical security arrangements will be applied to exports by the receiving country. Reviews are performed in conformance with U.S. non-proliferation laws, which are intended to ensure that U.S. exports will be protected and safeguarded during transit and use in the importing country and that exports will be used only for peaceful purposes.

The NRC staff also participates with other agencies in U.S. Government efforts to assist the IAEA in improving its safeguards system and maintaining the effectiveness of existing safeguards. The U.S. Program of Technical Assistance to IAEA Safeguards (POTAS) provides the largest share of voluntary technical support of any IAEA member state. The focus of most POTAS activity during 1993 was extension of the IAEA safeguards procedures by application of new methods and techniques to complement traditional safeguards methods. Through its participation in the Technical Support Coordination Committee, the interagency group which administers the POTAS program, the NRC applies its safeguards expertise in addressing international safeguards problems and enhancing the overall effectiveness of the IAEA safeguards program.

The NRC also participates in the U.S. Action Plan Working Group, which is mainly concerned with the bilateral exchange of international safeguards information. During 1993, U.S. representatives met with their counterparts from Germany, France, Japan, the United Kingdom, and the European Community to consider ways to strengthen safeguards efforts world-wide.

The NRC is also participating in the U.S. program to assist the States of the FSU to implement or enhance their national nuclear material control and accounting systems. The NRC is working with DOE in this effort, including the implementation of a U.S.-Russia MC&A Agreement signed on September 2, 1993. Similar agreements with Kazakhstan and Ukraine were also under development during the report period, and an MC&A Implementing Agreement was signed by the U.S. and Ukraine on December 18.

See Chapter 5 for further discussion of NRC safeguards activities.

## NUCLEAR NON-PROLIFERATION ACTIVITIES

**U.S. Non-proliferation Policy.** In a speech to the United Nations General Assembly in September, President Clinton announced the main features of his Administration's nonproliferation and export control policy. The policy provides continuing strong support for the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), for the International Atomic Energy Agency (IAEA) and its safeguards role, and for multilateral export controls.

The President outlined a new comprehensive approach to the growing accumulation of fissile material from dismantled nuclear weapons and within civilian programs. Proposals include: limiting stockpiles of high-enriched uranium (HEU) and subjecting them to high safety, security and accountability standards; a multilateral convention to prohibit production of HEU or plutonium for nuclear explosives purposes or outside of international safeguards; regional arrangements to constrain the production of fissile materials in regions of instability and high proliferation risk; placement of nuclear material which is surplus to defense needs under IAEA inspection; purchase of HEU from the Former Soviet Union and other countries for conversion to power reactor fuel; exploration of means to limit stockpiling of plutonium from nuclear power programs, and to minimize the civilian use of HEU; and initiation of a comprehensive review of long term options for plutonium disposition. The policy confirms that the United States does not encourage use of plutonium in nuclear power programs and does not engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. However, the policy states that the United States will maintain its existing commitments regarding the use of plutonium in nuclear power programs in Western Europe and Japan.

The NRC will be closely involved in implementation of a number of these policy initiatives. The NRC has provided expert advice to the Russian Republic on the creation of a materials control and accountancy system for their HEU stockpiles, and will continue to work with other U.S. Government agencies in coordinating U.S. assistance in safely dismantling their nuclear weapons. NRC staff has been working with the IAEA to create an effective and efficient system to inspect surplus nuclear material which will be placed under IAEA safeguards. The Executive Branch will also continue to consult with the NRC on nuclear supply and control issues and new bilateral cooperation agreements being negotiated with nuclear trade partners in Western Europe. Such consultations will consider the detailed meaning of the new Administration non-proliferation policies.

**Nuclear Non-Proliferation Treaty Extension.** The Nuclear Non-Proliferation Treaty (NPT) is the cornerstone of the international nuclear non-proliferation regime. The NPT, with more than 160 parties, supports fundamental U.S. national security and foreign policy objectives. In 1995 a conference of the parties will be held to decide whether to extend the treaty indefinitely, or for a fixed period, or fixed periods. In May 1993, the first of four Preparatory Conferences of the parties was held to decide procedural issues and establish a timetable for the conference itself (April 17–May 12, 1995). NRC staff participated in the Executive Branch process to define U.S. Government negotiating positions on the procedural issues. The NRC expects to attend the third (September 1994) and fourth (January 1995) Preparatory Conferences and the 1995 NPT Conference, during which information on U.S. Government compliance with its commitments to reduce its nuclear arsenal and provide nuclear safety and technical assistance will be key in defending the NPT regime as durable, predictable, and in the best interest of all nations.

**Negotiation of a New Nuclear Cooperation Agreement with EURATOM.** The Agreement Between the Government of the United States and the European Atomic Energy Community (EURATOM) Concerning Peaceful Uses of Atomic Energy provides the legal basis for the NRC to authorize exports of nuclear fuel and major nuclear reactor components to the EURATOM member states, including Belgium, France, Germany, Italy, the Netherlands, Spain, and the United Kingdom. The agreement expires in 1995 and is currently being renegotiated. The Department of State has the lead role for the United States in these negotiations. DOE and other U.S. agencies, including the NRC, provide technical and policy support to the Department.

# Nuclear Regulatory Research

Chapter 9

Activities of the NRC's Office of Nuclear Regulatory Research (RES) constitute an essential service to the regulatory process and are vital to the implementation of a substantial number of the agency's programs. The goal of the office is to ensure the availability of sound technical bases for timely rulemaking and related decisions in support of NRC licensing and inspection activities. RES also has responsibilities related to the implementation of Commission policies on safety goals and severe accident regulation, to the resolution of generic safety issues, and to the review of licensee submittals regarding individual plant examinations. It is also the responsibility of RES to conduct the NRC's rulemaking process, including the issuance of regulatory guides and rules that govern NRC licensed activities.

Regulations issued by the NRC in 1993 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which includes a listing of those guides issued, revised, or withdrawn during fiscal year 1993.

Pursuant to the Small Business Research and Development Enhancement Act of 1992, Public Law 102-564, the NRC supports the Small Business Innovation Research (SBIR) program, which stimulates technological innovation by small businesses, strengthens the role of small business in meeting Federal research and development needs, increases the commercial application of NRC-supported research results, and improves the return on investment from Federally funded research for economic and social benefits to the nation. The NRC has participated in the program since its inception in fiscal year 1982, promoting high quality, "cutting-edge" research of relevance and potential importance to the NRC mission. One goal of the program is to couple this research with follow-on private funding, pursuant to possible commercial application. As of fiscal year 1993, the NRC was supporting 20 SBIR projects-in-progress.

In 1993, the NRC staff continued its active participation in the national standards program, particularly with respect to setting priorities. NRC participation derives from a need for national standards to define acceptable ways of implementing the NRC's basic safety regulations. Approximately 187 NRC staff members serve on working groups organized by technical and professional societies.

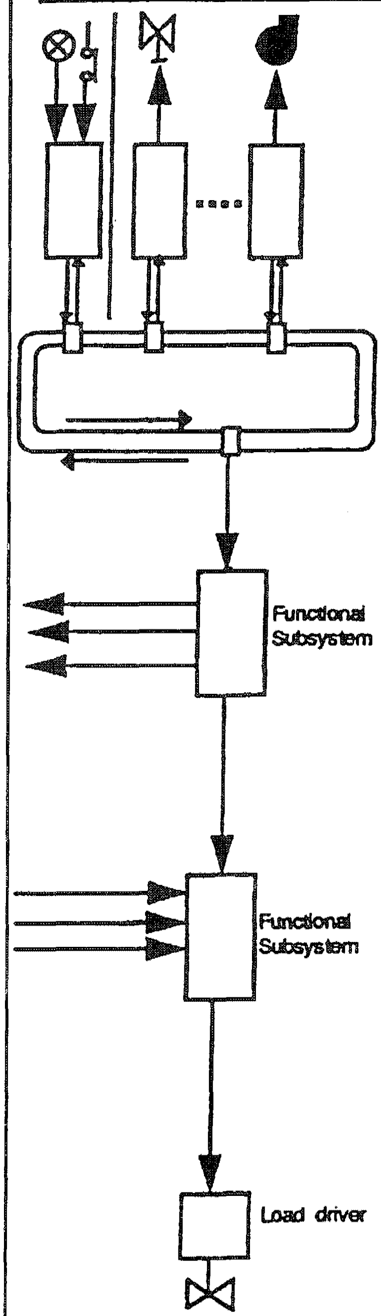
This chapter summarizes RES activities during fiscal year 1993 under the following major headings: Reactor Safety Research—Licensing Support; Reactor Safety Research—Regulation Support; the Nuclear Material and Low-Level Waste Regulation Program; and Assessing the Safety of High-Level Waste Disposal.

## Reactor Safety Research—Licensing Support

### STANDARD REACTOR DESIGNS

#### Engineering Issues for Advanced Reactor Designs

**Qualification of Advanced Instrumentation and Control System Hardware.** The Oak Ridge National Laboratory (ORNL) is conducting a study to identify functional and environmental issues arising from the application of new technologies to the instrumentation in both current and next-generation nuclear power plants. Specifically, the program seeks an understanding of the technical issues involved in evaluating long term properties and performance of "advanced" instrumentation and control (I&C) systems intended for plant upgrades and/or proposed for use in advanced light-water reactors (ALWRs). Special emphasis will be given to identifying vulnerabilities and environmental limitations that could be found in microprocessor-based systems in nuclear plant environments. Initial studies have focused on protection systems and the I&C systems employed in engineered safety systems. The environmental and functional issues studied thus far are discussed in a draft report (NUREG/CR-5904). In this document, evaluation templates are presented which were developed by assembling reasonably complete configurations of safety channel components for the major ALWR designs. The templates permit an evaluation of the impact of environmental stressors affecting these components and interfaces. Functional issues considered in the evaluation include distribution of functions, sources, supply and distribution of electrical power, calibration and testing capabilities, and attributes of failure predictions. The application and acceptance of




| Simplified Design | Subsystem Identification and Location | Effect of Environmental Stressors |
|---|---|---|
| | Protection system sensors and actuators; located inside containment. | Aging mechanisms in electro-mechanical components and cables are likely to be the same as in existing plants. |
| | Remote Multiplexing Units (RMUs) use time division multiplexed techniques; located outside containment. | Aging mechanisms and failure modes of micro-electronic packages differ from analog components. Fiber optic transmitters, cables, and receivers are also subject to failure modes and degradation mechanisms different from conventional (copper) cabling. New qualification methodologies may be needed. |
| | Fiber-optic, bi-directional, dual-redundant network using token passing techniques. | |
| | Comparison of measured variables to setpoints are performed here. Functions are performed by microprocessors under program control. Information is sent to other 3 divisions via fiber serial datalinks. | Microprocessor-based technology has failure modes different from analog technology. Need to ascertain how lengthy exposure to levels of smoke and chemical contaminants below the detectable threshold affect system reliability. |
| | Subsystem performs voting on all trip information from other three divisions to decide if a full reactor trip initiation is required. All functional subsystems for reactor trip (shown here) and engineered safety features (not shown) are in control room cabinets. | All subsystems are subject to the same environmental stressors, since they are in the same location. Effects of both normal stressors (e.g., EMI/RFI) and abnormal stressors (elevated temperature, smoke, chemical contaminants) need to be ascertained. |
| | Output of load driver sends trip signal to pilot valve solenoids to initiate a scram. | |

**Shown here is a template of an advanced light-water reactor (ALWR) protection system, with a description of the impact of environmental stresses on new instrumentation and control technologies.**

digital computers in reactor protection systems are also reviewed, in light of current industry standards.

**Technical Basis for Regulatory Guidance on Electromagnetic Interference Issues.** ORNL is developing the technical basis for regulatory guidance to address problems and malfunctions in I&C systems caused by electromagnetic and radio-frequency interference (EMI/RFI) and power surges. The concern stems from EMI/RFI and power surges affecting digital I&C systems at nuclear power plants and other process industries. The technical basis is grounded in good engineering practices, to ensure that sufficient levels of electromagnetic compatibility (EMC) are maintained between the nuclear power plant's electronic and electro-mechanical systems. Sound EMC design and installation practices are recommended to control interference sources and their impact on nearby circuits and systems, in accordance with IEEE Std. 1050–1989, "Guide for Instrumentation and Control Equipment Grounding in Generating Stations." The EMC "good practices" encompass circuit layouts, terminations, filtering, grounding, bonding, shielding, and physical separation. A test and validation program is in development to outline the test for EMI/RFI and surge-withstand capability (SWC) and the associated test methods to be followed, in order to ensure that I&C systems are capable of performing their intended functions. The test and validation program is being developed around the EMI/RFI test criteria from Military Standard (MIL–STD)—461, "Requirements for the Control of Electromagnetic Interference Emission and Susceptibility"; the associated EMI/RFI test methods extracted from MIL–STD–462, "Measurement of Electromagnetic Interference Characteristics"; and the SWC guidelines contained in IEEE Std. C62.41-1991, "Recommended Practice on Surge Voltages in Low-Voltage AC Power Circuits." Currently, the electromagnetic environment in a typical commercial nuclear power plant is undefined; for that reason, electric and magnetic spectral receivers need to be placed at various nuclear power plants. In the long term, unattended measurement data will be collected with these spectral receivers and used to provide the NRC with a realistic assessment of the probable ambient electromagnetic environment in nuclear power plants.

**National Codes & Standards.** Work initiated in fiscal year 1992 on assessing current national codes and standards—such as the American Society of Mechanical Engineers' Boiler and Pressure Vessel Code—is continuing through fiscal year 1993. The information derived from the study will be integrated into the advanced reactor design approval/licensing process. Canadian and other international quality assurance (QA) standards are also being evaluated for compliance with Federal QA standards.

**Evaluation of Low-Pressure Piping for Intersystem LOCA.** Development of evaluation criteria for a new design goal for ALWRs was completed in fiscal year 1993. The new design goal is that low-pressure piping attached to the reactor coolant loop be able to withstand reactor coolant pressures and temperatures. The condition in question, which could follow from multiple valve failures, is important because, for certain postulated events, it can lead to rapid core damage and the release of radioactivity outside the containment. The potential event is called an intersystem loss-of-coolant accident (ISLOCA) and is being treated as a severe accident. Because of the low frequency of occurrence of this potential event, the performance goal is to achieve a failure probability in the low-pressure piping of about 10 percent. A probabilistic methodology to estimate failure was applied to both carbon and stainless steel piping and other piping components—including flanges, valves, pumps, and heat exchanger tubes. The results will be used in the Design Certification for the advanced boiling water reactor (ABWR) and the System 80+ and are expected to be used for the passive designs as well.

**Experience Based Seismic Qualification.** In its Utility Requirements Document for ALWRs, the Electric Power Research Institute (EPRI) has proposed the use of experience as a method of seismic qualification, as an applicable substitute, on a case-by-case basis, for more traditional tests and evaluations. An expert panel established to assess the viability of the experience-based method has recommended a graded approach, in which equipment is assigned to one of three categories. In the first two groups, the use of experiential data is possible, but seismic capacities are different for the two groups, and the attributes for membership in the groups are also different. In the third group, the use of experiential data is not allowable. Equipment categories likely to be placed in the first two groups are horizontal and vertical pumps; motor-operated, manual, and check valves; thermal element assemblies; diesel generators; transformers; and batteries on racks.

**Containment Performance Goals.** In support of the NRC Severe Accident Policy Statement, as it applies to ALWRs, work has begun on development of performance requirements for containments under severe accident conditions. Deterministic criteria will be established for both steel and concrete containments, emphasizing standards for local and global strains and deformations. For these deterministic criteria, probabilistic models will be constructed to allow comparison with the conditional containment failure probability of 0.1, proposed by the Commission for implementation of its safety goal policy statement. Particular attention will be given to evaluating the severe accident performance requirements for containments mentioned in the EPRI Utility Requirements Document.

## Systems Performance of Advanced Reactors

**NRC Confirmatory Safety System Testing in Support of AP600 Design Review.** Westinghouse Electric Corporation has submitted the Advanced Passive 600-megawatt (AP600) nuclear power plant design to the NRC for design certification. RES is proceeding to conduct confirmatory testing of AP600 safety systems to help the NRC staff evaluate the safety of the AP600 reactor systems. Confirmatory safety system testing is not required for design certification but will help provide more technical bases for the NRC licensing decisions.

To carry out the testing, it was determined that the most cost-effective route was to modify an existing full-height, full-pressure test facility, rather than build a new one. All the existing integral effects test facilities, both in the United States and abroad, were screened to select the best candidate. The criteria for the initial screening included size, facility configuration similarity, availability schedule, the facility management's willingness to share the cost, and management's ability to enter into a confidential agreement with Westinghouse for handling proprietary information. The screening identified the best candidate as the Rig of Safety Assessment (ROSA) Large-Scale Test Facility of the Japan Atomic Energy Research Institute (JAERI). To confirm the initial results and to determine the extent of modification necessary to simulate the AP600, the Idaho National Engineering Laboratory (INEL) was engaged to perform a comparative study between ROSA and AP600, using the RELAP5 code.

A comparison between the existing ROSA facility and the AP600 design showed that ROSA did not contain certain key components important for safety response of the AP600. It was not obvious how much hardware modification to the ROSA facility would be needed to simulate the AP600. The fidelity of simulation must be balanced against the associated cost—with the fidelity high enough to result in a facility capable of producing data for code assessment covering the major AP600 phenomena in the correct sequence. At the same time, the cost has to be affordable and the schedule reasonably compatible with licensing needs. Facility modification was agreed upon and is being implemented by Sumitomo Heavy Industries, which constructed the ROSA facility and has been maintaining and operating it for the past several years, as a contractor to JAERI. The modification was expected to be complete by January 1994, with a series of tests to be performed in 1994.

As a confirmatory program, the ROSA/AP600 testing will cover not only design basis accidents but also "beyond-design basis" accidents, which vendors may not be required to address. There will be some counterpart tests among ROSA and two other scale facilities (SPES-2, and OSU) to help predict what will occur in a full-scale AP600 reactor.

**SBWR Test Facility.** Purdue University was awarded a three-year contract, on July 26, 1993, to build a facility to test the design for an advanced reactor, the Simplified Boiling Water Reactor (SBWR). The objective is to provide data by which to assess the capabilities of the NRC's computer codes to analyze the SBWR. The Purdue test facility will have all the necessary components and systems scaled from the SBWR design, including a vessel with electrically heated fuel rods, upper and lower drywells, suppression pool, gravitydriven cooling system (GDCS), passive containment cooling system (PCCS), isolation condenser System (ICS), drywell and wetwell sprays, piping and valves, and instrumentation.

The facility is a low-pressure, reduced-height facility, 1/4 of the SBWR height, with a volume 1/400 of the SBWR volume. The aspect ratio of the facility (i.e., diameter scale/height scale) is 1/2.5, which is close to the aspect ratio of the SBWR at 1. The facility is scheduled to be completed by December 1994. Approximately 50 performance tests are planned, to be completed by April 1996, covering a broad spectrum of loss-of-coolant accidents and transients. At least 10 of the tests will be completed prior to submission of the final safety evaluation report on SBWR, scheduled for July 1995.

**Human Reliability.** A study has been undertaken to develop methods for assessing the risk implications of changes in human performance, prompted by the introduction of advanced digital displays and controls. Research to establish a technical basis for minimum shift staffing in advanced control rooms will be initiated in fiscal year 1994. The research will be based on workload and task allocation studies.

## Advanced Reactor Risk Analysis

**Passive System Reliability Project.** The advanced passive reactors have engineered safeguard systems that maximize the use of passive devices—such as nitrogen-powered accumulators, natural circulation flow, and gravity-driven safety injection. These do not rely on active systems, such as a.c.-powered equipment, although certain valves may require stored energy (e.g., battery power) to change state. The passive designs are expected by the designers both to increase safety and decrease costs, as a result of their simplified design. However, because of the lack of actual working experience with the design and uncertainties in the modeling of processes such as natural circulation, there are uncertainties regarding the performance of the engineered safeguard systems.

The passive reliability project, presently focused on the Westinghouse AP-600 design, seeks to develop a candidate methodology for quantifying the uncertainty distribution in the core damage frequency, arising from uncertainty in the modeling of the natural processes. The project is scheduled to be completed in fiscal year 1994.

## Regulatory Application of New Source Terms

The Commission's reactor site criteria (10 CFR Part 100) require that an accidental release of fission products from the reactor core into the containment should be an assumed event and that its radiological consequences should be evaluated. The criteria for gauging the release into the containment are derived from the 1962 report, TID-14844, based on an assumed instantaneous release of fission products. Although this source term has long been included in the Commission's regulations for siting, it has traditionally had a greater effect on plant design than on siting.

Since 1962, a better understanding of the timing and nature of the fission product release has been obtained, and a number of areas subject to regulation have been identified that could benefit from changes derived from source term and severe accident research. In fiscal year 1993, work continued on a replacement to TID-14844. The comment period for a draft report, "Accident Source Terms for Light-Water Nuclear Power Plants" (NUREG-1465), which was issued in July 1992, expired in December 1992. Comments were solicited and received from an internationally recognized group of experts, as well as from the public. A final version of NUREG-1465 is in preparation. In connection with this effort, the following documents have been issued:

- Draft NUREG/CR-5901, "A Simplified Model of Aerosol Scrubbing by a Water Pool Overlying Core Debris Interacting With Concrete," dated October 1992.
- NUREG/CR-5950, "Iodine Evolution and pH Control," dated December 1992.
- NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," dated June 1993.

**Update of Siting Regulations.** In fiscal year 1993, staff efforts continued on the updating of 10 CFR Part 100, "Reactor Site Criteria." A proposed rule revising Part 100 was issued in the *Federal Register* (57 FR 47802), on October 20, 1992, for a 120-day comment period. The rule proposed to decouple the calculation of the exclusion area distance from calculations of the source term and dose, by specifying a minimum acceptable exclusion area distance and by setting forth population density criteria as well; an update of seismic considerations was also proposed which would incorporate probabilistic, as well as deterministic, methods. The comment period, extended twice, expired on June 1, 1993. Extensive comments, from both domestic and foreign sources, have been received. The staff is analyzing the comments and examining the various options proposed for updating reactor site criteria.

**Emergency Planning Regulations.** In fiscal year 1993, a proposed rule was published in the *Federal Register* (58 FR 29795) for public comment on the emergency planning licensing requirements for independent spent fuel storage facilities (ISFSIs) and monitored retrievable storage facilities (MRS). In June 1993, a proposed rule was published in the *Federal Register* (58 FR 34539) on revised emergency planning that would update and clarify some ambiguities that have surfaced in the implementation of the Commission's emergency planning exercise requirements.

# REACTOR AGING AND LICENSE RENEWAL

## Pressure Vessel Safety

This area of NRC research focuses on ensuring the structural integrity of the reactor system pressure boundary, i.e., keeping it free from damage and leaktight. Ensuring the structural integrity of the pressure boundary has been at the center of several recent wellpublicized regulatory issues—for example, the 1984 decision to require an accelerated schedule of five boiling water reactor (BWR) inspections because of cracking in the coolant pipes; the 1991 review of the Yankee Rowe (Mass.) plant; and the 1992 review of the Trojan (Ore.) plant's steam generators. The underlying concern in ensuring the integrity of the pressure boundary is that failure to do so could compromise the operator's ability to cool the reactor core and possibly bring about a loss-of-coolant accident (LOCA) that could be accompanied by a release of hazardous fission products.

Research in this area is a broad-based program, initiated in 1967. The original program was focused solely on the properties and fracture behavior of the reactor pressure vessel—the large, thick-walled steel cylinder that houses and supports the reactor core. As the full challenge of ensuring the integrity of this critical component was realized, the scope of the research program was expanded to include irradiation damage, service-induced cracking mechanisms, and methods for periodically inspecting the pressure vessel. Incidents of cracking and leaking in pipes and steam generator tubes have accentuated the need for materials data, analysis methods, and inspection techniques relevant to these components.

**Fracture Evaluation.** Addressing fracture analysis methods assumed a particularly large role in the overall program during fiscal year 1993. Fracture analysis work involves an ongoing program to develop and reduce to practice advanced analysis methods that will improve the ability to predict the allowable pressures and temperatures for the pressure vessel, and the ability to evaluate the integrity of the pressure vessel under design basis and hypothetical accident conditions. Basic work is being performed by researchers at the Oak Ridge National Laboratory (ORNL), augmented by research being performed at Brown University, the University of Illinois, Texas A&M University, and the U.S. Navy's Naval Surface Warfare Center (NSWC). The researchers are developing state-of-the-art analysis methods, and evaluating them against test data developed by ORNL, the National Institute for Standards and Technology (NIST), and the NSWC. The initial work has been very promising, and the program has been continued to permit evaluation of test geometries that are more typical of reactor pressure vessels. The researchers are coordinating their work with international efforts in this area. Collaborative efforts with a European Community program are expected to provide results that will closely simulate a reactor pressure vessel subjected to accident loads. These, in turn, will lead to a more realistic validation of the revised analysis methods.

During fiscal year 1993, the results of several such efforts were put to use in performing generic analyses of reactor pressure vessels fabricated from materials with a low resistance to a "ductile tearing" failure mode. In the early 1970's, the NRC recognized that some pressure vessels had been fabricated using steel plates and some weld types that did not provide the high resistance to this failure mode exhibited by most of the plates, forgings and welds used in reactor pressure vessels. The NRC issued Appendix G to 10 CFR Part 50 in 1973 to provide explicit requirements on the "Charpy upper-shelf" energy—a measure of the ductile tearing resistance of these materials—for both new construction and for operating plants. But it was recognized that some of the early vessels did not meet these requirements. Therefore, Unresolved Safety Issue A–11, "Reactor Vessel Materials Toughness," was established to develop methods for evaluating the integrity of pressure vessels that did not satisfy the Appendix G requirements. The issue was resolved in 1982 with the publication of "Resolution of the Task A–11 Reactor Vessel Materials Toughness Safety Issue" (NUREG–0744). A key aspect of the overall resolution was a request to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to recommend criteria that would satisfy the Appendix G requirement to demonstrate margins of safety equivalent to those in the ASME code. The code committee responded in February 1991 with a set of evaluation criteria that were reviewed and accepted by the NRC. The code also developed an analysis method which was similar to the NUREG–0744 recommended method but which benefited from several years of review and experience by the code committee members and by the NRC staff. The Code Case N–512 (Section XI, Division 1, February 1993) did not, however, address complete details of all the potential loading conditions for reactor pressure vessels, nor did it include guidance on determining appropriate material properties for use in the evaluation method.

The research staff undertook, in fiscal year 1993, to develop a draft regulatory guide expanding the guidance to include evaluation methods pertinent to all service loading conditions and providing specific guidance for estimating material properties. In developing this guidance, the staff drew on results from past NRC funded research efforts at ORNL and at the Pacific Northwest Laboratories, to provide a comprehensive fracture analysis methodology. Results from a Phase II Small Business Innovative Research (SBIR) program were also used as an acceptable method for estimating the material properties. The draft regulatory guide was published at the end of September 1993.

**Radiation Embrittlement.** Of special concern in ensuring the integrity of the reactor pressure vessel is the embrittlement of the pressure vessel steel caused by neutrons escaping from the reactor core during normal operation. These neutrons impinge on the pressure vessel wall and, through a complex process, reduce the ability of the steel to resist fracture. The embrittlement increases with continued operation. To ensure the continued safe operation of pressure vessels, the research program includes a significant effort to quantify the effects of neutron radiation embrittlement, to understand the mechanisms that control this process, and to find methods to mitigate the embrittlement and restore the original fracture toughness.

During fiscal year 1993, radiation embrittlement research efforts moved forward on several fronts. Test reactor irradiations were initiated by ORNL, using the University of Michigan test reactor, to evaluate the effects of neutron radiation on weld materials removed from the canceled Midland Unit 1 (Mich.) reactor pressure vessel. The materials being irradiated are representative of the so-called "limiting" material in several operating nuclear power plants. The materials are also being irradiated in the surveillance programs of an operating power plant, as part of an NRC-industry coordinated research effort. When the results from each of these programs are available in the late 1990s, they will provide important information about the embrittlement trends for these materials and equally important information about the differences between test reactor and power reactor irradiation conditions, as well as the mechanisms controlling embrittlement of these materials.

During fiscal year 1993, ORNL processed the updated Evaluated Nuclear Data File, Version B–VI, to develop neutron cross-section libraries that can be used in evaluating the neutron fluence for power reactors. These cross-section libraries are needed to predict the neutron fluence, which is an essential input in estimating the level of radiation embrittlement for reactor pressure vessels. The work will be completed in 1994, and the updated cross-section libraries will be available for use shortly thereafter. Besides the cross-section library work, researchers at ORNL have been working with researchers in the Czech Republic, and with other East European researchers, in performing calculations to predict the results of carefully controlled "benchmark" experiments, conducted by the Czech researchers. This continuing work is generating important data relevant to the NRC's program to validate neutron fluence calculation methods, and it is providing for technology transfer and validation of the methods used by the different laboratories. This work contributed to the staff's preparation of a draft regulatory guide on "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." The draft guide was published for public comment at the end of September 1993.

Research to better understand the mechanisms of radiation embrittlement continued in fiscal year 1993, with significant advances being made by ORNL, in conjunction with researchers in the United Kingdom, in modeling the complex interactions among the impinging neutrons and the atoms in the pressure vessel steel. This work is closely integrated with the experimental work going on at the University of California at Santa Barbara, at ORNL, and in Europe. Understanding the controlling mechanisms is essential to confidently extrapolating empirical models of radiation embrittlement to unique operating circumstances. Progress in mechanisms research is providing assurance that the empirical models are conservative and is helping to define the limits of extrapolation for those models.

Interactions with researchers in Russia, under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Group 3, continued in fiscal year 1993. The scope of the continuing effort has been expanded from radiation embrittlement to include the general subject of pressure vessel integrity analysis methods. During fiscal year 1993, coordinated test reactor irradiations were continued in the United States and in Russia, and ORNL was host to a visiting Russian scientist on a oneyear assignment to investigate radiation embrittlement and thermal annealing effects. The scientist's work at ORNL has produced valuable data concerning the ability of thermal annealing in mitigation of the effects of radiation embrittlement. Work in this area is continuing.

The interactions with the Russians have entailed first-hand examination of their procedures for conducting thermal annealing treatments of their reactor pressure vessels. The NRC began research into the metallurgical effects of thermal annealing many years ago and, in the mid-1980's, funded work to examine the engineering feasibility of this process for U.S. plant designs and conditions. During fiscal year 1993, the research staff made use of this experience and research and drafted a regulation and regulatory guide addressing both the engineering and metallurgical aspects of thermal annealing for U.S. plant designs. The performance-based regulation and guidance were undergoing final internal reviews at the close of the report period.

**Steam Generator Tube Integrity.** Steam generator tube degradation continues to be an important subject of continuing research. The thin-walled steam generator tubes are an integral part of the reactor system pressure boundary, and tube failures could lead to a LOCA that could result in containment bypass and the escape of radioactive fission products directly to the environment. During fiscal year 1993, the research staff worked closely with the regulatory staff in the review and evaluation of industry-proposed interim (or alternative) criteria for plugging those tubes where stress corrosion cracking degradation has been detected at the tube-to-tube-support-plate intersections. The research staff developed a mechanistically based model for predicting tube failure and cumulative leak rate of degraded tubes, both under normal and under postulated main steam line break conditions.

## Piping Integrity

The piping integrity research program has been an important part of the overall pressure boundary integrity research program for many years. It took on heightened importance in the early 1980's with the emphasis on intergranular stress corrosion cracking (IGSCC) in BWR piping systems, and the work on environmentally assisted cracking and piping fracture behavior became high priority activities. Over the last decade, the research on IGSCC has largely been completed and has contributed substantially to the NRC's understanding of this issue. The pipe fracture research is drawing to a close, as work during fiscal year 1993 focused on conducting large-scale pipe fracture tests to provide final validation of the methods used by the NRC and by the ASME code to ensure that piping will not fail under accident conditions. Further, work to evaluate the significance of the reactor operating temperature on the fracture behavior of cast stainless steel piping and components was completed during fiscal year 1993, and the results have been provided to, and are being used by, the regulatory staff in plant-specific evaluations.

While work in these areas is drawing to a close, it has become clear that further research is needed to address steam generator tube integrity. There is also a need for research to quantify the potential contribution of the water coolant to cracking in the reactor vessel internals, and in other penetrations and attachments; this need has been accentuated by the service cracking incidents. Finally, the research program on environmentally assisted cracking has been developing data on the effects of the water coolant on the fatigue life of pressure boundary components.

**Environmentally Assisted Cracking.** Irradiation-assisted stress corrosion cracking (IASCC) of core

- Recordkeeping (NUREG/CR-5848)
- HEPA Filters and Adsorbers (NUREG/CR-6029)
- Essential HVAC Chillers (NUREG/CR-6043)
- Fans and Blowers
- Impact of Aging on Accident Precursors
- Standard Technical Specifications Aging Evaluation.

**Technical Bases for License Renewal.** License renewal is a high-priority activity for the NRC and for the nuclear power industry. The NPAR program is assisting in the development of regulatory guidance addressing the technical bases and safety issues related to aging. This will help implement the license renewal rule and provide guidance regarding license renewal application requirements.

**Standard Technical Specifications Aging Assessment.** The NPAR program is evaluating the Standard Technical Specifications (STS) for selected nuclear power plant systems and components to determine the effectiveness of current surveillance requirements (SRs) in detecting age-related degradation effects. The purpose of these SRs is to ensure the operability and availability of safety-related systems and components by verifying and demonstrating that they are capable of performing their required functions. Aging effects have not always been recognized or addressed explicitly in the SRs; many significant forms of aging degradation may not be detected prior to failure, and, in some cases, the test methods and frequency of testing may even contribute to premature degradation.

Standard Technical Specifications aging assessments were completed during fiscal year 1993 for check valves, the auxiliary feedwater system, and the reactor protection system. These assessments reviewed the current surveillance and testing requirements to determine their effectiveness in detecting degraded conditions in systems, structures and components (SSCs) prior to failure; to determine if the current surveillance and test methods and frequency of testing contribute to premature aging degradation; to identify, based on the results of the in-depth aging assessments of specific systems and components conducted under the NPAR program, the parameters or indicators useful to monitor the degraded state of SSCs; and to develop recommendations for inspection, surveillance, trending, and condition monitoring methods to be incorporated as part of SRs in evaluating age-related degradation. Work was also initiated to combine the 10 STS aging assessments completed into a NUREG/CR report.

**Aging Management NUREG Update.** For several years the NPAR program has been developing technical understanding of the processes that—through time-dependent agerelated degradation of structures, systems, and components—might reduce operational safety margins in operating nuclear power plants below acceptable limits. The results from the NPAR program, and other complementary aging management programs, were compiled and critically reviewed in a draft report, "A Review of Information Useful for Managing Aging" (NUREG/CR-5562). The program results have proved a valuable resource for the NRC staff in such tasks as preparing the draft standard review plan for the review of license renewal applications for nuclear power plants and in the review of aging issues related to license renewal.

NUREG/CR-5562 was extensively revised and updated during fiscal year 1993, to provide greater insight and technical guidance for aging management including (1) identifying SSCs in which age-related degradation should be managed; (2) understanding aging mechanisms and identifying degradation sites in these SSCs; and (3) managing degradation through effective monitoring and maintenance programs, or by modifications to operating conditions.

**Information Management.** The aging assessments conducted by the NPAR program have generated an extensive volume of valuable information on aging processes and effective methods for detecting and mitigating aging degradation in safety significant systems and components. In order to make this information more readily available to the NRC regulatory staff, nuclear power plant licensees, other government agencies, and the public, a demonstration electronic document was developed during fiscal year 1993 by which technical and regulatory information can be rapidly and efficiently computer-searched for specific words and topics. The electronic document also includes a description and aging overview for systems and components of interest, a compilation of the aging assessment information, and colored line drawings that illustrate the various types and components where aging mechanisms are operative.

## Components, Systems, and Facilities

**Engineered Safety Features.** A nuclear plant's engineered safety feature systems are intended to control and mitigate certain specific occurrences in plant operations that might challenge the integrity of the reactor and/or jeopardize the safety of plant personnel or of the general public. Aging and service wear of these systems is clearly a safety concern. NPAR aging assessments have been conducted for air treatment and cooling system fans and for the high-efficiency particulate air (HEPA) filters and activated carbon beds that remove radioactive particulates and volatile radionuclides. The fan-aging assessment has revealed that aging degradation appears to be an important factor in the breakdown and failure of the cooling fans. Details compiled from surveys concerning aging and wear effects suggest that bearings are the component most frequently linked to fan-failure. The investigation also indicated that monitoring techniques that will detect irregularities arising from improper lubrication, cooling, alignment, and balance can help counteract many of the aging effects that could impair fan-performance. The aging assessment of the HEPA filters and activated carbon adsorbers identified heat, moisture, radiation, airborne particles, and contaminants as key stressors, with a resultant occurrence of aging agents and degradation, ranging from particle loading to degraded sealant and gasket properties. This work is documented in "Phase I Assessment of Nuclear Air Treatment System HEPA Filters and Adsorbers" (NUREG/CR-6029).

**Chillers.** Chillers are required in nuclear plants to cool rooms, such as the main reactor control room, which contain safety-related equipment essential to plant safety. Without proper cooling, control room temperature can rise rapidly, leading to operator stress and causing electronic equipment to give erroneous readings or spurious alarms, and even to begin to fail. The newer digital controls in the plants are even more sensitive to high temperatures than the older analog controls. An NPAR aging assessment of these essential chillers is in progress. Initial results of the study are documented in "Aging Assessment of Essential Chillers Used in Nuclear Power Plants" (NUREG/CR-6043). This work shows that chillers are affected by vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Aging is accelerated by moisture and noncondensible gases, as well as dirt and other contaminants within the refrigerant containment system; by excessive start/stop cycling; and by operating below the rated capacity. The primary cause of chiller failures seems to be a lack of condition monitoring and a failure to perform scheduled maintenance. A comprehensive assessment in progress seeks to identify actions which could help reduce chiller failures and premature aging through effective monitoring and preventive maintenance programs, to develop effective procedures for assuring the reliability of essential chillers, and to provide guidelines for a more effective transition to the new chiller refrigerants.

**Service Water System.** The NPAR service water system aging assessment was completed with the publication of "Nuclear Service Water System Aging Degradation Assessment" (NUREG/CR-5379). However, an advanced power plant monitoring and diagnostic system, identified in the course of the NPAR service water system aging assessment, has been successfully tested at a U.S. Marine Corps base. Deployment of this system is expected to result in approximately a 40 percent net reduction in facility life-cycle operation and maintenance costs at the Marine Corps bases.

**Emergency Diesel Generators.** Although the emergency diesel generator aging assessment was completed earlier, work continues on the incorporation of the findings and recommendations into relevant codes and standards, such as the Institute of Electrical and Electronic Engineers (IEEE). Guide document P-1205, IEEE "Guide For Assessing, Monitoring And Mitigating Aging Effects On Class 1E Equipment Used In Nuclear Power Generating Stations," documented the Working Group 3.4 effort. It included a seven-page appendix on diesel generators derived from NPAR information. From Working Group 4.2, IEEE Std. 387, "Standard Criteria for Diesel-Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations," has been approved for final balloting. This latest revision of the standard includes the aging results and information from the NPAR research.

**Aging Assessment and Mitigation of Major LWR Components.** Of intrinsic importance to reactor-aging research is the assessment and mitigation of aging damage to major components and structures. The objective of this aging assessment task, an element of the NPAR program, is to identify, develop and evaluate various aging management techniques for the major LWR components and structures. The approach is to gauge the degradation of the major LWR components and structures by the synergistic influences of radiation embrittlement, thermal fatigue, stress corrosion cracking, thermal embrittlement, erosion corrosion, and so forth.

Research completed in this area in 1992 focused on developing insights for aging management for selected LWR components and structures, in order to ensure continued safe operation. The studies also included an evaluation of advanced inspection and monitoring methods for characterizing the aging damage. The results should prove useful to the NRC in its task of resolving safety issues associated with LWR aging degradation and developing policies and guidelines for operating license renewal. The major components assessed in 1993 are the LWR metal containments and the LWR reinforced and pre-stressed concrete containments. PWR reactor pressure vessels and the PWR coolant piping research will continue. Results of these assessments are being documented in a multi-volume report, NUREG/CR-5314. A draft report (NUREG/CR-5824) discussing the identification of advanced monitoring methods for estimating stresses causing fatigue damage has also been completed.

**PRA-Based Methodology for Aging Assessments and Priority Assignments.** The riskbased methodology for assessment of aging in nuclear power plants and for defining priorities among risk contributions and maintenance activities (published in previous years as NUREG/CR-5587 and NUREG/CR-5510) is subject to uncertainties because of limited available aging data and also because of certain modeling assumptions. Research in 1993 focused on developing sensitivity and uncertainty analyses to address data and modeling uncertainties and to validate risk-based methods. This work was documented in draft NUREG/CR-6045 in 1993.

The application of age-dependent risk methodology requires age-dependent component failure rates. But age-dependent component failure rates are not generally available and need to be estimated from limited recorded plant failure data and plant maintenance logs. A major limitation of the age-dependent methodology has been the lack of recorded component aging data and approaches to develop aging failure rates based on the available information. To address this limitation, an approach was developed during fiscal year 1993 to incorporate age-dependence in probabilistic risk assessments (PRAs) that does not require absolute age-dependent component failure rates. Instead, the aging of a component is expressed in terms of relative aging rates that are found to be fairly constant across different components and different plants. A draft report (NUREG/CR-6067) was completed on the aging data assessment methodology.

Also in 1993, an important application of the risk-based methods resulted in the development of PRA-based approaches for identifying safety-related motor-operated valves (MOVs) having the most impact on plant risk covered under Generic Letter 89-10, "SafetyRelated MOV Testing and Surveillance." Dynamic tests and surveillance tests, in accordance with GL 89-10, could then be performed on those MOVs with the largest risk impact. Relative risk-importance of single MOVs and the interaction of multiple MOVs can be analyzed using this approach. A draft NUREG/CR documenting the results of this work is in preparation.

In addition to the above-described effort, work was initiated in 1993 to set priorities for environmental stressors associated with advanced digital instrumentation and control (I&C) systems in nuclear power plants, based on their risk-significance. Analog I&C systems in nuclear power plants are being replaced by digital systems. Digital I&C systems are vulnerable to common environmental stressors, such as moisture/humidity and temperature, and the effects of such stressors are being identified, and measures are being developed to rank them. The risk-based approaches are being tested for the I&C systems using plantspecific PRAs.

**Aging of Passive Components.** In earlier research efforts, a methodology was developed to include the effects of aging on passive components (pipes, structure, and supports) and the resultant impact on plant risk. The methodology is based on probabilistic structural analysis for calculating the failure probability of these components. The failure calculation can be incorporated into a plant-specific PRA, and two approaches for doing so were investigated during the report period. The first is a simple probabilistic structural analysis approach, and the second is an approach called failure attributes. The simple probabilistic structural analysis is an approximation of the large, complex structural probabilistic computer codes. The second uses those attributes that have been shown to most affect aging and failure. These approaches, including a screening for relative contribution to risk, are to be documented in a report in fiscal year 1994. Calculations were also completed to investigate the effects of passive components on the risk of containment failure. The draft report (NUREG/CR-5730) that describes the methodology for addressing the effects of aging on passive components is in preparation.

**Aging Effects on Motor-Operated Valve Performance.** In 1993, an effort was undertaken to assess the effects of aging on the operability of motor-operated valves (MOVs), and to identify those safety-related MOVs in typical PWR and BWR plants that are most susceptible to internal or external environmental aging effects. The investigation has drawn from the Nuclear Plant Aging Research reports of the Oak Ridge National Laboratory (ORNL) on valve aging, and from the Idaho National Engineering Laboratory reports for high-pressure injection systems. The current investigation includes a review of the inservice testing programs for four BWR and five PWR plants, in order to identify safety-related MOVs. From this review it has been determined that the majority of the safety-related valves are either butterfly, globe, or gate valves constructed of carbon steel. Internal and external environmental conditions are being combined with valve type and material of construction to identify those of greatest safety significance and most susceptible to the effects of aging. A draft NUREG/CR is being prepared to report the results of the investigations. Experiments are also under way to compare valve sliding surface friction factors for corroded and uncorroded valve material specimens and to assess measurement methods to detect the changes being evaluated.

**Check Valve Failure Data Characterization.** A detailed review of historical check valve failure data for operating nuclear plants in the United States was conducted by ORNL during the report period. The results are published as NUREG/CR-5944. The source of data for the review was the Nuclear Plant Reliability Data System (NPRDS), operated by the industry's Institute of Nuclear Power Operations (INPO). The purpose of the study was to better understand those failures that affect the internals of the valves. A total of 4,680 failure narratives were reviewed, from data covering the period 1984–1990; of this total, 1,227 were identified as involving internals degradation and were further characterized.

The characterization of the failed valves included the following parameters:

- Valve size
- Valve age at failure
- Plant age at failure
- System in which the valve was used
- General operating status of the system
- Valve manufacturer
- Failure mode
- Extent of degradation
- Detection method

- Affected areas.

The characterized data were analyzed for relative failure rates for each of the characterized parameters. Cross-tabulations of the parametric data were also made. Some notable observations were:

- There was not a strong relationship between valve age and failure rate.
- The largest valves ( > 10 inches) experienced about twice the failure rate of smaller valve sizes and such failures were generally more likely to be significant failures.
- The emergency service water, main feedwater, diesel starting air, and main steam systems experienced the highest failure rates.
- Just over half (54 percent) of the failures were detected by programmatic means, including inservice testing, surveillance testing, and other periodic inspection programs.
- Normally operating systems experienced only slightly greater failure rates than did the standby systems that are used only during shutdown or only when tested.

The results of the study are being used by ASME Code Working Group OM-22 in support of ASME code development activities. Follow-on studies for failures occurring during 1991 and 1992 are planned.

**Detection of Pump Degradation.** Pump degradation studies comprise an examination of the leading causes of pump degradation and a description of existing methods used in domestic and overseas nuclear facilities to diagnose pump problems. Research results are being published in the report, "Detection of Pump Degradation" (NUREG/CR-6089), which evaluates the criteria required for pump testing at U.S. nuclear power plants and compares them to features that are characteristic of state-of-the-art diagnostic programs and practices currently implemented by other major industries. Degradation caused by low-flow pump operation is also discussed, along with new analysis techniques that may be used to ascertain unstable flow. Since many pump operational problems can be attributed to the pump driver, motor current analysis methods are also presented that can assist in the determination of specific kinds of motor degradation.

Vibration spectral analysis is widely accepted as a powerful diagnostic tool for determining numerous types of pump degradation, such as misalignment, unbalance, looseness, and various bearing anomalies. Many nuclear plant maintenance departments use vibration spectral analysis to diagnose pump problems. Thermography and lubrication analysis are other important diagnostic technologies that have made significant improvements within the past decade, both in their ease of application and their diagnostic capabilities. Low-flow operation, which was often performed by using minimum flow loops to conduct required ASME code testing, has been observed to cause pump degradation through destructive low-flow phenomena. Motor power analysis techniques have also been developed that may assist in the determination of the onset of unstable flow conditions, as well as enable the pump analyst to determine the most efficient operational ranges of a particular pump system. The next major thrust of development in diagnostic methodologies is likely to be focused on the development and use of expert systems.

**Auxiliary Feedwater System.** In the area of auxiliary feedwater systems, a thorough review of system controls and functions was performed, and several limitations of current maintenance and surveillance practice were identified, such as the failure to verify many safety-related control functions by periodic testing and the degradation of auxiliary feedwater (AFW) pumps by testing at low flow. A follow-on study has categorized the limitations in current monitoring/operating practice and evaluated failure modes and component degradation caused by these practices. The findings have applicability to all plants in that they point out typical testing omissions or sources of degradation.

Significant conclusions of the study are reported in NUREG/CR-5404, Vol. 2. The study addresses the present testing of AFW pumps at the minimum flow condition which may lead to degradation of the pumps and does not give an indication of a degraded pump condition. The report discusses hydraulic instability at low-flow operation and provides examples from the industry of pump degradation. The report also provides head-capacity curves showing that mini-flow tests are inadequate for assessing pump capability or ability to operate at design basis conditions. An alternative method of testing is presented that consists of testing at normal operating pressure to minimize degradation and to verify flow at design conditions.

**Steam Turbine Drives.** Steam turbine drives for safety-related pumps are used in certain systems at most of the commercial nuclear power plants in the United States. Turbine-driven pumps, in combination with electric-driven pumps, are in use in most PWR-AFW systems. Turbine-driven pumps are used at most BWR plants in the reactor core isolation cooling and high-pressure coolant injection systems. The turbine-driven pumps provide a means of heat removal from the reactor coolant system and are potentially useful in the event of station blackout.

Evaluation of failure records indicates that the turbine governor is the component that is most often involved in reported turbine failures. There are multiple sources of governor problems, with dirty or water-contaminated oil and setpoint drift accounting for a large percentage of the failures (33 percent). Oil problems (involved in 18 percent of failures) can be mitigated by periodic chemical analysis of oil samples. Setpoint drift (15 percent) can be mitigated by periodic calibration of the governor. The evaluation indicated that current maintenance practices have not been consistent in these areas, although there are indications of improvement in recent years. The results of the study are being published as NUREG/CR-5857.

**Cables.** The research program conducted at the Sandia National Laboratories on the aging effects on electrical cables was completed in 1993, and the three-volume final report (NUREG/CR-5772, Vols. 1, 2, and 3) was issued. The program evaluated the capability of cable types commonly found in operating nuclear power plants to meet equipment qualification standards for periods of 20, 40 and 60 years of operation. (See the *1991 NRC Annual Report* for background on the research.) Several of the cables tested in this program failed at a fairly early age. NRC Information Notice 93-33 summarized the cable test results and alerted the utilities to possible deficiencies. Research on the aging degradation of cable connectors and penetrations is under way.

**Control Rod Drive Systems for CE and B&W PWR Plants.** The Babcock & Wilcox (B&W) and Combustion Engineering (CE) control rod drive (CRD) systems consist of mechanical and electrical components that position the control rod assemblies in the core in response to automatic or manual reactivity control signals. Both systems are designed to allow rapid gravity insertion of the control rods upon removal of the a.c. power that holds the rods. The study examined the design, materials, maintenance and operation of the system to assess the potential for age degradation.

A detailed operating experience review highlighted age-related component degradation and failures that significantly affected plant operation. These effects include power reductions, reactor shutdowns, and engineered safety feature actuation. While there have not been system failures, component failures and degradation resulted in increased component stresses and unnecessary thermal and pressure cycles which challenged other plant systems.

The majority of component failures in the CE control drive system were caused by the degradation of the control system (61 percent). Failures of the CRD mechanisms accounted for 60 percent of reported failures of the B&W control drive system. Aging was identified as the direct failure cause for 40 percent of the CE power and control system and 55 percent of the B&W CRD mechanism. The operating and environmental stresses for the system, and the aging effects resulting from continued exposure to these stresses, were evaluated for the major system components. Detailed failure modes and effects analyses were performed for the subsystems. In addition, a survey was made of the current surveillance, inspection, monitoring and maintenance practices of utilities.

**Effects of Solar-Geomagnetically Induced Currents on Plant Electrical Systems.** Transient disturbances in the earth's magnetic field caused by auroral currents from the sun can induce electrical potential gradients across the earth's surface. These gradients act like d.c. voltage sources impressed between the grounded neutrals of transformers at opposite ends of power transmission systems. To study the effects of these geomagnetically induced currents (GICs) on plant equipment, a plant-specific electrical distribution system for a nuclear power plant was modeled using the ElectroMagnetic Transient Program (EMTP). The model simulated on-line equipment and loads from the station transformer in the switchyard to the safety busses at 120 volts, to which all electronic devices are connected for plant monitoring.

The EMTP analysis used the half-cycle saturation of the station transformer (attributable to GIC) and studied the effects on the voltage harmonic levels noted at various electrical busses. The results indicate that the emergency circuits appear to be more susceptible to high harmonics, because of normally light load conditions. Protective relays (both electromagnetic and solid state relays) without an harmonic filter, which operate purely as a peak detector, are vulnerable to false-high readings with GICs as low as 50 amps present in the system. Based on these results, an input side harmonic filter can be used on undervoltage protective relays, which sense an undervoltage or degraded voltage condition for starting the on-site diesel generators. A report was prepared that includes other parametric studies on the subject and discusses potential harmonic effects on the uninterruptible power system.

## Engineering Standards Support

The national standards program is coordinated by the American National Standards Institute (ANSI). ANSI provides procedural guidelines to help ensure that participation in the private sector standards development process is sufficiently broad based and that input from individual interests are fairly considered. NRC participation in this process is compatible with Office of Management and Budget (OMB) Circular (OMB) A-119, dated October 26, 1982, which provides policies for Federal participation in the development and use of voluntary standards.

The NRC staff is particularly active on ASME codes and standards writing committees, because portions of the ASME Boiler and Pressure Vessel (B&PV) Code have, since 1971, been incorporated by reference into 10 CFR 50.55a, in order to establish requirements for the construction, inservice inspection, and inservice testing of nuclear power plant components. Section 50.55a is periodically amended to update the references to include more recent versions of the ASME B&PV Code. In 1993, work continued on rulemaking, begun in 1992, that not only would update the reference to the ASME B&PV Code, but would, for the first time, incorporate by reference the new ASME Operations & Maintenance (O&M) Code, which provides rules for inservice testing of pumps, valves, and snubbers. The proposed rulemaking would expedite implementation of certain new ASME B&PV Code requirements for qualification of personnel and equipment used to perform inservice nondestructive ultrasonic examinations on nuclear power plant components.

**Support of NRC Regulation.** As with all the structural integrity research, timely and effective transfer of the results into the NRC's regulatory structure is a high-priority activity. For the inspection procedures and technologies research program, an inter-office Technical Advisory Group has been established to provide input to the research plans and to facilitate transfer of the research results.

During fiscal year 1993, the research program provided both technical reports and direct support to the regulatory staff. Specifically, a document on the fundamentals of computer-based ultrasonic systems was prepared to serve as a resource on the technology and to describe the methods for characterizing computer-based ultrasonic systems. In addition, a review was completed that will be published as a supplement to the resource document to provide a systematic evaluation of the computer-based system called P-SCAN.

Direct support was provided to the regulatory staff by researchers at the Pacific Northwest Laboratories (PNL) who assisted in evaluating inspections to detect and characterize cracks that were discovered in the core shroud of a BWR. This application posed difficulties because of severe access limitations, and the ultrasonic system employed was a modification of a computer-based system for inservice inspection of piping. The PNL researchers were able to provide the benefit of their considerable experience in developing and applying unique inspection systems.

## Structural Integrity

Concrete structures are crucial to the safe operation of light-water reactor plants. In general, the performance of concrete structures in nuclear power plants has been good. However, there have been several instances where the capability of concrete structures to meet future functional and performance requirements has been challenged because of problems arising from either improper material selection, construction and design deficiencies, or environmental effects. Examples of some of the potentially more serious incidents include post-tensioning anchor head failures, leaching of concrete in tendon galleries, voids under vertical tendon bearing plates, containment dome delaminations, corrosion of steel tendons and rebars, water intrusion through basemat cracks, and leakage of corrosion inhibitor from tendon sheaths. Such incidents indicate that there is a need for improved surveillance, inspection and testing, and maintenance to enhance the technical bases for assurance of continued safe operation of nuclear power plants.

The structural aging (SAG) program is addressing the aging management of safety-related concrete structures in nuclear power plants for the purpose of providing improved technical bases for their continued service. The SAG program objective is to prepare documentation providing NRC staff reviewers with (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant continued-service reviews, as well as criteria, and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection or structural assessment programs; and (4) methodologies required to perform current assessments and life-predictions of safety-related concrete structures. To accomplish its objective, the SAG program has been active in three major technical task areas: (1) materials property data base, (2) structural component assessment/repair technologies, and (3) quantitative methodology for continued service determinations. Seventeen reports on these topics have been published since 1991. The final seven reports are expected to be completed in 1994.

Recent experience suggests the possibility that corrosion effects can significantly diminish the margin needed for containments to accommodate accidents beyond their design basis. Evidence of corrosion has been found in both Mark I BWR containments and in ice condenser PWR containments. The robustness of containments, verified in tests performed at Sandia National Laboratories, is manifested in their capacity to sustain loads well beyond design level. More assessments are planned, in order to gain a better understanding of the significant factors involved in corrosion, in the efficacy of inspection, and in capacity reduction.

## License Renewal Regulatory Standards

A final rule (10 CFR Part 51) concerning the environmental review for renewal of a nuclear power plant operating license is under development. The proposed rule was published for public comment in September 1991. Over 120 comments were received on the technical analyses and certain procedural aspects of the proposed rule. Concern was expressed that the proposed rule would constrain public comment on environmental issues at the time of license renewal review for an individual nuclear power plant. All comments are being considered in developing the final rule, the generic environmental impact statement (GEIS), and other supporting documents. The final rule and supporting documents are expected to be published in 1994.

# Reactor Safety Research—Regulation Support

## PLANT PERFORMANCE

### Reactor Safety Experiments

Experiments are being conducted on decay heat removal by natural circulation at North Carolina State University in a facility that is a scaled model of a Westinghouse pressurized water reactor. The scale is 1:9 in height and in diameter, for a volume scale of 1:700, and uses freon as the working fluid. The facility includes a secondary system to provide representative plant behavior. Parts of the facility are constructed with plexiglass to allow visual observation of internal conditions. Facility construction was underwritten by several utilities, and the model is used extensively for the training of utility personnel and university students. The NRC experiments involve conducting decay heat removal under conditions of reduced coolant inventory in the primary system, using the visual observation features of the experimental facility.

Experiments are also being performed at the University of Maryland in a scaled experimental facility that simulates a Babcock & Wilcox reactor and is 1:4 in height, with a 1:500 volume scale. This facility was originally constructed under NRC contract to study small-break, loss-of coolant accidents and, following successful completion of that program, its mission was shifted to the current study of the natural circulation of steam under severe accident conditions. For these tests, sulfur hexafluoride is being used as the working fluid. This current program will be completed in fiscal year 1994.

### Safety Code Development and Maintenance

The third semi-annual international thermal-hydraulic code applications and maintenance program (CAMP) meeting was held October 20-22, 1993, in Santa Fe, N.M. There are now 14 member countries in CAMP, each of which had representatives at this meeting, besides representatives from the NRC and its code development contractors.

Cash contributions of $390,000 annually are made by or on behalf of the 14 member countries to supplement the code development and assessment programs funded by the NRC. As part of their agreements, the members provide code assessment studies or other non-cash contributions to assist the NRC's assessment of the codes applicability, scalability, and uncertainty when applied to nuclear power plant safety.

The codes covered by the agreement are RELAP5/MOD3, TRAC/PWR, and TRAC/BWR. Papers presented at the meeting covered work on these programs funded by NRC and CAMP member contributions, CAMP member activities, code assessment performed by CAMP members, and code activities of U.S. code users.

## HUMAN RELIABILITY

About half of all safety-related events reported at nuclear power plants continue to involve human performance. Methods and data are needed to identify, systematically set priorities for, and suggest solutions to human performance issues during operation and maintenance activities at nuclear facilities.

The human factors and reliability assessment research program has three objectives: (1) to broaden the NRC's understanding of human performance and to identify causes of human error; (2) to accurately measure human performance for enhancing safer operations and precluding critical errors; and (3) to develop the technical basis for requirements, recommendations, and guidance related to human performance.

The bulk of the human factors regulatory research program is performed for human reliability and is divided into four inter-related program elements: (1) personnel performance, (2) human-system interfaces, (3) organizational factors, and (4) reliability assessment. Human factors research is also performed for systems performance of advanced reactors, generic safety issue resolution, and materials licensee performance. The human factors research related to these activities are reported under the appropriate sections of this chapter.

### Personnel Performance

Work continued on the development of a method to assess the effectiveness of training programs at nuclear power plants. Measures and supporting documentation for a training effectiveness evaluation method are the intended outcome of this effort. Data analyses are being incorporated into a final report on the factors that are considered when making decisions on operations staffing and on how staffing relates to safe startup, shutdown, and operation of nuclear power plants. A study to establish a technical basis for minimum shift staffing for operational crews at nuclear power plants, based on workload and task allocation, has been initiated. A handbook on the effects of environmental factors on human performance is currently being prepared for use by nuclear power plant inspectors. Two reports on training in effectively responding to accidents are being prepared. These reports describe decision-making and stress-coping skills that may be needed to respond to an accident situation, as well as potential training approaches for developing those skills.

An effort to identify measures for better characterizing the quality of personnel performance in an operational setting has been initiated. The measures and methods could be used for evaluating the effects on safety of changes in system interfaces, particularly changes from analog to digital interfaces.

## Human-System Interfaces

Human-system interface research includes NRC participation in the Organization for Economic Cooperation and Development (OECD) Halden Reactor Project, a multi-faceted program that includes verification and validation of digital systems, man-machine interaction, and surveillance and support systems for advanced control rooms. Specific NRC research needs were identified in the form of lessons-learned reports on (1) methods and tools for the development and verification and validation of safety-related software, (2) experience with development and quality assurance of software systems at the Halden Project, (3) manmachine interaction with computer-based systems, (4) test and evaluation activities on manmachine interaction with computer-based systems, (5) recommendations on advanced control rooms based upon the Halden Project's experience, (6) coordination and integration of computer-based operator support systems, and (7) reports containing information related to human reliability. These reports will serve as part of the technical bases for NRC guidelines.

Research continued to evaluate the positive and negative attributes of standards and computer-aided-software engineering tools for use in the certification of high-integrity software for nuclear power plant safety systems. And research continued in the uses of a computer-aided-software engineering tool for assessing the degree of functional diversity in software that performs safety functions. A project co-sponsored by the Electric Power Research Institute (EPRI) to develop guidelines for the verification and validation of expert systems is nearing completion. A research project co-sponsored by EPRI on verification and validation guidelines and quality metrics for digital high-integrity systems is also under way.

A project was initiated to independently evaluate, test and improve verification and validation guidelines for use in the audit of computer-based safety systems.

The NRC, assisted by the National Institute of Standards and Technology, sponsored a Digital Systems Reliability and Nuclear Safety Workshop during the report period. Goals to be achieved by the workshop were to (1) provide feedback to the NRC from outside experts regarding potential safety issues, proposed regulatory positions, and research associated with application of digital systems in nuclear power plants and (2) continue the in-depth exposure of the NRC staff to digital systems design issues related to nuclear safety by discussions with experts in the state of the art and practice of digital systems.

## Organizational Factors

Research on organizational factors has produced provisional data to be considered for use in such regulatory applications as risk-based inspection and diagnostic evaluation. Research also continued on alternative quantification methods for incorporating the influence of organizational factors into risk assessments. Studies show that an analysis of organizational factors can identify dependent failures across systems in a selected work process at a plant.

The NRC is reassessing the need for continuing research on organizational factors, taking into consideration the remaining obstacles to quantitatively incorporating organizational factors into risk assessments.

## Reliability Assessment

Work is nearly complete on the collecting, cataloguing and storing, in a computerized library, estimates of probabilities of operator error and hardware failure. Because one of the largest contributors to risk is operator cognitive error, research continued to gather data on cognitive performance, in an effort to validate a computer simulation model of human cognitive tasks during accident sequences. The data were gathered from operating crews responding to two simulated accident scenarios on training simulators.

Work has begun on the analysis of information from the simulator portion of the NRC-administered operator requalification examinations. Estimates from this source may provide valid error rates for use in a nuclear power plant PRA. Research continued on the risk impact of replacing existing nuclear analog systems with digital systems.

For several years the NRC has been developing reliability analysis tools to be used with PRA to analyze and improve the technical bases of selected requirements in technical specifications. These tools can evaluate risk technical specification changes, such as (1) surveillance test intervals, (2) allowed outage times, both during operation and during shutdown, (3) action statements that require shutdown, (4) technical specification defenses against dependent failures, and (5) scheduling preventive maintenance. Development of the methods is almost require shutdown, (4) technical specification defenses against dependent failures, and (5) scheduling preventive maintenance. Development of the methods is almost completed, and both detailed technical reports and a handbook to guide NRC reviewers in the use of these methods are in preparation.

## REACTOR ACCIDENT ANALYSIS

### Reactor Risk Analysis

Probabilistic risk analysis (PRA) is used by the NRC staff to support the resolution of a wide spectrum of reactor regulatory issues. In fiscal year 1993, these applications encompassed both specific issue-oriented projects and more general work, including development and demonstration of risk analysis methods and development of risk-related training and guidance for the NRC staff.

Issue-oriented projects continuing in 1993 included:

(1) *Analysis of Low-Power and Shutdown Accident Risks:* As a result of the Chernobyl accident and other precursor events around the world, an extensive two-phased project was initiated in 1989 to examine the potential risks of accidents initiated during low-power and shutdown modes of operation. Phase 1, completed at the end of 1991, was a coarse screening analysis of all operational modes (other than full power) for one BWR and one PWR, in order to provide support for the Office of Nuclear Reactor Regulation's (NRR) regulatory analysis and to guide the Phase 2 effort. A significant finding at that stage was that the traditional concept of technical specification modes of operation does not adequately delineate the plant operating boundary conditions needed for risk analyses. The Phase 2 effort has concentrated on a specific operating state for each of the two plants, selecting the potentially highest risk operating state, based on the Phase 1 results. In addition, a simplified analysis of potential in-plant and off-site accident progression and health consequences of such accidents has been performed and provided to NRR, in support of their regulatory activities, as documented in NUREG-1449. The results of Phase 2 will be published as NUREG/CR reports in fiscal year 1994.

(2) *South Texas Project Risk Analysis:* In 1992, the staff completed a review of the South Texas Project risk analysis and documented the results and findings (NUREG/CR-5606). The licensee has estimated the overall mean core damage frequency to be 2E-4-per-reactor-year, which is found to be within the range of core damage frequency estimates provided for similar Westinghouse PWR facilities. The licensee has subsequently requested modifications to its plant technical specifications based, in part, on its risk analysis. The RES staff is now working with NRR on the acceptability of the requested modifications and expects to complete this work in fiscal year 1994.

Methods development projects performed in 1993 included:

(1) *SAPHIRE Computer Tools:* A suite of computer codes called SAPHIRE (System Analysis Programs for Hands-on Integrated Reliability Evaluation) has been updated to version 5.0. This set of codes is to be used in performing probabilistic risk analysis and to permit an analyst to perform many of the functions necessary to create, quantify and evaluate the accident risks of nuclear power plants. The codes are being used extensively to perform the low-power and shutdown risk analyses described above, to perform analysis for resolving generic safety issues, and to set priorities for inspection activities. During 1993, PRA data from three more licensed nuclear power plants were added to the SAPHIRE data base, and most of the data from previous plant loads were updated to version 5.0. This brings the data base total to 13 plants, two of which are advanced concept plants that have been added to support the agency's design certification reviews. Courses continued to be provided to both the NRC staff and contractors on the use of these codes. When the documentation for version 5.0, scheduled for January 1994, is complete the new codes and user manuals will be sent to the Energy Science and Technology Software Center at Oak Ridge National Laboratory for general distribution.

(2) *Consequence Code Benchmark:* The NRC is working with the Commission of the European Communities and the Organization for Economic Cooperation and Development to carry out an inter-comparison exercise on probabilistic accident consequence codes. The six codes being evaluated are MACCS (United States), COSYMA (Germany), CONDOR (UK), OSCAAR (Japan), LENA (Finland), and ARANO (Sweden). The comparison exercise uses six radioactive accident source terms and calculates dose consequences for such measures as whole body dose and fatal cancers. The results of these comparisons will be obtained in fiscal year 1994 and will constitute a data base by which to judge the performance of new codes to predict reactor accident consequences.

(3) *Survey and Evaluation of Aging Risk Assessment Methods:* A survey and evaluation of aging risk assessment methods and applications is being performed. A draft NUREG/CR on the work has been received from the contractor and the final NUREG/CR will be published in fiscal year 1994. Major findings of the review include:

- The issue of aging in nuclear power plants cannot be addressed by models that are based solely on the current PRA structure and failure rates. Structures, systems and components (SSCs) neglected in the basic PRA model may become important because of aging;

the basic degradation mechanisms, such as fatigue, embrittlement, and erosioncorrosion, must be considered.

- Probabilistic models for degradation mechanisms would allow the effective use of information regarding the aging of SSCs. Failure-rate-based models cannot accommodate this type of information, which typically does not include a significant number of failures. The use of these probabilistic models for the degradation mechanisms would reduce the uncertainties present in models that are exclusively failure-rate based.

- Probabilistic models for the degradation mechanisms would also allow the development of effective risk management strategies.

- The development of a methodology that includes aging mechanisms can build on existing PRA models, appropriately modified, as current external event and analyses do.

Risk-related training and guidance development in 1993 included:

(1) *Guidance for Staff Use of Risk Analysis*: In a July 1991 letter, the NRC's Advisory Committee on Reactor Safeguards (ACRS) identified a number of concerns with the staff's uses of risk analysis. In response, the NRC's Executive Director for Operations formed a working group of staff management to "consider what improvements in methods and data analysis are possible and needed, the role of uncertainty analysis in different staff uses of PSA, ..." This working group was organized in early 1992 with the following objectives:

- To develop guidance on consistent and appropriate uses of PRA within the NRC.

- To identify skills and experience necessary for each category of staff use.

- To identify improvements in PRA techniques and associated data necessary for each category of staff use.

In October 1993, the group completed a draft final report that included initial guidance to the staff on the use of PRA in screening and analyzing reactor operational events and on basic terms and methods used in PRA. The report also contains a number of recommendations for supplemental guidance development, improvements to the NRC's PRA training program, and improvements in PRA tools and data bases used by the staff. It was expected that the final report would be published in December 1993.

(2) *Reactor Safety Training Course:* In response to a request from the Office of Analysis and Evaluation of Operational Data (AEOD), RES has developed a new course dealing with reactor safety in a broad context. Course components include a historical overview, "design-basis" accidents, "beyond-design-basis" accidents, accident progression in the reactor vessel, accident progression in the containment, radiological releases and consequences, and emergency response. The intended audience includes new agency employees and other NRC staff not generally familiar with these topics. Two presentations of the course were made at the NRC's Technical Training Center in Chattanooga in 1993. With developmental work on the course completed, responsibility for presenting it will be turned over to AEOD in fiscal year 1994.

## Containment Performance

In order to ensure that existing regulations adequately protect the public from the consequences of severe accidents, the NRC conducts research in many areas, among them source term release and transport, core-melt progression, fuel-coolant interactions, meltconcrete interactions, direct containment heating, and hydrogen combustion. The overall goals of the research are to develop technical bases for assessing containment performance over the range of risk-significant core-melt events, to develop an improved understanding of the range of phenomena expected during severe reactor accidents, and to develop improved methods for assessing fission product behavior. With these kinds of data, the NRC is better able to confirm the adequacy of its requirements for the design and reliability of the systems that may be needed to mitigate the effects of severe accidents.

**Melt-Concrete Interactions and Debris Coolability.** In those severe accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity where it can thermally and chemically interact with structural concrete. The consequences of these melt-concrete interactions can have a significant effect on containment loading, the potential modes of containment failure, and the radiological source terms. The major areas of concern associated with melt-concrete interactions during a severe accident are the penetration of the basemat and failure of the liner, the generation of radioactive aerosols and gases, including combustible gases, and the over-pressurization of the containment.

The NRC has conducted an extensive program of analytical and experimental research to obtain improved understanding of melt-concrete interactions. The experimental research takes the form of experiments simulating a wide variety of concretes used in nuclear power plants in the United States, taking account of the diverse accident scenarios that may lead to melt-concrete interactions. The analytical research is focused on the development of models for studying phenomenological aspects of melt-concrete interactions and includes a reassessment of models used to predict aerosol generation and radionuclide release.

Early experiments on melt-concrete interactions were conducted without the presence of an overlying water pool. The experimental data base generated from these experiments is extensive, spanning a broad range of melt release conditions, as well as concrete types. No further melt-concrete interaction experiments are currently planned. More recent experiments on melt-concrete interactions were conducted in the presence of an overlying water pool. In fiscal year 1991, the NRC initiated the WETCOR program, also called the debris coolability program, to address two specific issues: (1) the comparative coolability of oxidic and metallic debris and (2) the effects of boundary conditions on coolability, i.e., crust formation and stability. A report describing the WETCOR-1 test (NUREG/CR-5907), the only integral test conducted under this program, was published in fiscal year 1993.

The second experimental program on debris coolability, called the Melt Attack and Coolability Experiments (MACE) program, was developed as an extension of the Advanced Containment Experiments (ACE) program under the sponsorship of NRC, EPRI, and other largely governmental agencies in several countries. The MACE program is intended to determine the ability of water to cool prototypic core debris (urania-zirconia composition). Three tests were conducted under the MACE program in fiscal years 1992 and 1993. Data from these tests were analyzed in fiscal year 1993. Except for the last test that was terminated prematurely, the results from the MACE tests generally support the concept of crust formation at the melt-coolant interface with periodic access of water to the melt and partial melt cooling. A fourth MACE test at a scale larger than the previous tests is planned in fiscal year 1994 to study the effect of scale on crust formation, stability, and debris coolability.

A topic related to melt-concrete interactions, particularly in connection with the BWR Mark I containments, is that of melt-structure interactions, leading to early containment failure attributable to liner (shell) melt-through. The NRC research over the past few years has addressed key phenomena associated with the liner melt-through issue. Integration of the research into the assessment of the conditional probability of liner failure—both with and without an overlying water pool in the drywell—in the event of a core-melt accident that proceeds to vessel failure, was completed in fiscal year 1991 and documented in NUREG/CR-5423. The overall conclusion from the assessment was that the presence of water on top of the core debris would prevent containment shell failure. The report was peer reviewed, and, as a result of the review process, four areas were identified that needed further resolution—liner failure criteria, melt spreading, melt-concrete interactions, and melt release conditions. More work was performed in these areas in fiscal years 1992 and 1993. The results of the research generally confirm the conclusions of the original study, NUREG/CR-6025, which included updates from the additional research efforts, and was issued in fiscal year 1993.

**High-Pressure Melt Ejection—Direct Containment Heating.** In certain postulated reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. A molten core left unmitigated will slump and collect at the bottom of the reactor vessel. If a breach occurs, the core material will be ejected under pressure, and if the material should be ejected from the reactor cavity into surrounding containment volumes in the form of fine particles, thermal energy could be quickly transferred to the containment atmosphere. The metallic components of the ejected core debris could further oxidize in air or in steam and could generate a large quantity of hydrogen and chemical energy that would further pressurize the containment. This projected process is referred to as direct containment heating (DCH).

To help develop a data base by which to estimate the risk associated with high-pressure core-melt accidents, the NRC has, in fiscal year 1993, completed four DCH integral effects tests for a containment configuration simulating that of the Surry (Va.) nuclear power plant (a PWR plant), three in the 1/6th-scale Containment Technology Testing facility and one in the 1/10th-scale Surtsey facility at the Sandia National Laboratories. Analysis of the test results is near completion. Also, in the 1/40th-scale COREXIT facility at the Argonne National Laboratory, three integral effects tests were conducted using reactor materials (urania based melt) instead of an iron-alumina melt simulant. Separate effects testing is being performed at Purdue University in a 1/10th-scale model for a containment configuration simulating that of the Zion (Ill.) nuclear power plant (a PWR plant) to study the DCH phenomena in detail and will be completed in fiscal year 1994.

As part of the DCH issue resolution plan for PWRs, a study was completed and documented in "Integrated Report on DCH Issue Resolution for PWRs" (NUREG/CR-6109). The report outlines the DCH issue resolution process, demonstrates the process for two specific plants, Zion and Surry, and describes the approach for resolution for the remaining PWRs. Supporting documentation and studies related to this report have also been completed. One of these supporting documents is a study of DCH for the Zion plant, "The Probability of Containment Failure by Direct Containment Heating in Zion" (NUREG/CR-6075). Both reports are undergoing peer review and are expected to be completed in fiscal year 1994.

**Hydrogen Combustion.** Significant information exists on hydrogen combustion to assess the possible threat to containment and safety-related equipment. Some ancillary issues remain related to a better understanding of the likelihood of various modes of combustion at high temperature and in the presence of large quantities of steam, i.e., deflagrations, diffusion flames, accelerated flames, transition from deflagration-to-detonations (DDT), and detonations.

The largest current program in this topical area comes out of a joint agreement between the NRC and the Ministry of International Trade and Industry (MITI) of Japan (managed by the Nuclear Power Engineering Corporation). Under the agreement, a hightemperature hydrogen combustion program related to high-speed combustion modes, i.e., detonations and DDT, has been developed and is under way at Brookhaven National Laboratory. A small-scale developmental apparatus (SSDA) was constructed and has provided a preliminary set of experimental data to characterize the effect of high temperatures on the ability of hydrogen-air-steam mixtures to undergo detonation. Equally important, the SSDA was used to support the design of the larger-scale high-temperature combustion facility (HTCF) by providing solutions to a number of design and operational problems at high temperatures. The construction of the HTCF is complete, and hightemperature experiments will begin during fiscal year 1994. As a result of the cooperative agreement with Japan, the NRC has access to ongoing hydrogen mixing and distribution testing in the Tadotsu facility and the combustion testing in the Takasoga facility. This research provides a greatly expanded and improved data base for the validation of analytical tools.

A hydrogen research program is also under way to investigate diffusion flame behavior in low-speed hydrogen combustion. A small-scale facility has been designed and construction is close to completion. Experiments were performed to examine the influence of ignition source strength on flammability limits of the hydrogen-air mixtures at a temperature of 300K° and pressure of one bar. The results will be used to help resolve outstanding issues in severe accidents, i.e., hydrogen combustion aspects of DCH, hightemperature combustion phenomena, and detonation initiation by high-temperature steamhydrogen-particle jets.

**Severe Accident Codes.** Because of the difficulty in performing prototypic experiments for a variety of severe accident scenarios, substantial reliance must be placed on the development, verification, and validation of system-level computer codes for analyzing severe accident phenomena. A number of codes (e.g., MELCOR, CONTAIN, SCDAP/RELAP5) have been developed for various stages in severe accidents, both invessel and ex-vessel, for both BWRs and PWRs. Additional codes such as CORCON, VICTORIA, COMMIX, HMS, and IFCI are being developed and maintained to perform specific functions that require detailed modeling and will be used to benchmark the systemlevel codes discussed above.

***MELCOR*** is an integrated computer code that models the progression of severe accidents in light-water reactor (LWR) power plants. The code can be used to evaluate the progression of severe reactor accidents from initiation through containment failure and to estimate severe accident source terms and their sensitivities and uncertainties in a variety of applications. The entire spectrum of severe accident phenomena—including reactor coolant system and containment thermal-hydraulic response, core heatup, degradation and relocation, and fission product release and transport—is treated in MELCOR in a unified framework for both BWRs and PWRs.

MELCOR has been applied to the analyses of various plant accident transients, and assessment efforts have been completed by a number of United States and international user organizations. The focus of development efforts in fiscal year 1993 has been to improve capabilities to handle the phenomena of natural circulation, external heat transfer, and lower head failure and to model a few specific features of the advanced light-water reactor (ALWR). The efforts have also addressed a number of suggestions for improvement of the code made by an independent peer review group convened at the NRC's request. Further, a MELCOR Cooperative Assessment Program effort, started last year, has continued in fiscal year 1993. The goal of this effort is to create an international forum for information exchange on the applicability, limitations, and operational experience of MELCOR.

***CONTAIN*** is a detailed code for the integrated analysis of containment phenomena. The code provides the capability to predict the physical, chemical and radiological conditions inside a nuclear reactor containment in the event of a severe accident. The code also provides the capability to predict fission product releases to the environment in the event of containment failure. Among the models included in CONTAIN are heat and mass transfer, aerosols and fission product behavior, flammable gas combustion, core-concrete interactions, and direct containment heating. The code has the capability to analyze a wide variety of LWR plants, including their engineered safety systems and many kinds of accident scenarios.

One issue currently under investigation is direct containment heating (DCH) and pressurization of the reactor containment atmosphere by molten core materials ejected following the lower head failure of the vessel under pressure. A program to incorporate selected DCH models into the CONTAIN code was initiated last year and completed in fiscal year 1993, including the assessment against available experimental data. Also, plant cases were run with the updated CONTAIN code to determine the impact of DCH on the containment. Another development effort is related to containment analyses for ALWR designs. The industry is developing containment designs for ALWRs that incorporate passive cooling and decay heat removal features for protection against long term containment overpressure in accident situations. The CONTAIN code was modified in selected areas in this regard; it is planned to use the code to evaluate the experimental data generated by the industry's research.

*SCDAP/RELAP5* is a computer code that has the capability to perform detailed analyses of in-vessel core-melt progression phenomena during various severe accident conditions. The code has been used for severe accident analyses, including natural circulation studies and the analysis of lower plenum debris and lower head heatup. The systematic assessment of SCDAP/RELAP5 that began in 1991 identified several areas of modeling improvements. Work on these improvements brought to completion in fiscal year 1993 included: (1) improvement of code modeling and elimination of numerical errors (such as converting SCDAP fixed input format to free format, expanding input error checking and diagnostic printout, identifying and resolving programming errors causing the code to fail, and correcting the cladding strain model to eliminate over-prediction of cladding strain); (2) enhancement of code reliability (such as implementing time smoothing of the interface conditions between RELAP5 and SCDAP in flow area volume changes and heat transfer and resolving code failures during restart); and (3) improvement in late-phase molten pool formation and melt slumping modeling (such as modifying changes in junction and volume flow area to account for actual debris-bed height, independent of node size, and adding user input to allow radial spreading of liquefied material into adjacent channels). These modeling improvements will significantly reduce uncertainties in the code calculation of coremelt progression.

Other SCDAP/RELAP5 research activities accomplished in fiscal year 1993 include the completion of an independent peer review of the code and initiation of model extensions to the code, in order to address ALWR issues. Further model improvement and code documentation based upon the recommendations of the peer review committee are being considered. To ensure that SCDAP/RELAP5 meets design objectives and targeted applications, model assessment and validation efforts will continue.

*COMMIX* is a three-dimensional transient single-phase computer code for thermalhydraulic analysis of single and multi-component engineering systems. The code solves a system of time-dependent and multi-dimensional conservation of mass, momentum, energy, and transport equations. A number of phenomena encountered in postulated severe accidents in ALWRs are inherently multi-dimensional in nature. The COMMIX code is being developed to address issues such as natural circulation, flow stratification, and the effect of non-condensible gas distribution on local condensation and evaporation for the AP600 plant. Code upgrades that were completed in fiscal years 1992 and 1993 include implementation of multi-component capability, the development of the liquid film tracking model, and incorporation of heat and mass transfer models. In fiscal year 1994, a code validation effort will be initiated using the small-scale and 1/8th-scale test results for the Westinghouse AP600 passive containment cooling system. After the validation has been completed, COMMIX can serve as a benchmarking tool for portions of the CONTAIN code.

*CORCON* is a code developed as a best-estimate computational tool to calculate the thermal hydraulics and chemistry involving the progression of high-temperature core debris as it erodes concrete in the reactor cavity. A significant update of the code, designated **CORCON-MOD3**, was completed in fiscal year 1993. This update involves improved axial and radial heat transfer models; inclusion of a condensed phase chemistry model for oxide-metal reactions; improved coolant heat transfer models, including the effects of subcooling and gas injection on film boiling; addition of models for interphase mixing and stratification, improvement of models for bubble behavior (e.g., bubble size, bubble rise velocity, and void fraction); and incorporation of the VANESA model for aerosol generation and radionuclide release. A topical report has been prepared to describe the phenomenological models and correlations incorporated in the code and to identify acceptable limits of validity for the models and correlations. Extensive validation of the code was performed in fiscal year 1993 to determine its capabilities and limitations. The CORCON–MOD3 is now targeted for incorporation into CONTAIN and MELCOR.

*VICTORIA* is a computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe accident. The code provides detailed predictions of the fission product release from the fuel and transport in the RCS of radionuclides and non-radioactive materials during core degradation. During fiscal year 1993, assessment and validation of models used in the VICTORIA computer code against existing data bases and against new data from various experimental test facilities (e.g., FALCON VI, ST) were carried out.

Battelle Columbus Laboratory is performing, under NRC sponsorship, additional experiments and analyses on the revaporization of certain radionuclides. The purpose of these experiments is to identify thermodynamic properties of the radionuclides in the primary circuit during a severe accident. Boric acid is known to react with both cesium hydroxide and with cesium iodide to form the less volatile species cesium borate. The chemistry of boric acid is complex because it decomposes to a variety of different species, depending on the environmental conditions. The reactions under investigation will enhance the models already in the VICTORIA code.

*HMS* is a best-estimate, transient, three-dimensional code for analyzing the transport, mixing, and burning of hydrogen. The code can model geometrically complex structures with multiple compartments and can simulate the effects of condensation, heat transfer to walls and internal structures, chemical kinetics and fluid turbulence. During fiscal year 1993, the assessment work of HMS against three test problems was completed, and a report was issued documenting the governing physical equations and computational model of HMS. During fiscal year 1994, the HMS user's manual will be developed to provide the basic information for setting up and running problems with the code. Also, the HMS will be converted from a main frame computer code to a workstation environment code.

## Severe Accident Phenomenology

Severe Accident Phenomenology research seeks an improved understanding of the more serious of possible reactor accidents and explores such phenomena as source terms, core-melt progression, primary system failure from severe accidents, and fuel-coolant interactions, discussed below.

**Source Terms.** "Source Terms" refers to the magnitudes of the radioactive materials released from a nuclear reactor core to the containment atmosphere, taking into account the timing of the postulated releases and other information needed to calculate off-site consequences following a hypothetical severe accident. NRC research in this area is reflected in the updated version of TID-14844, which has been in use for three decades, in connection with plant siting assessments; the latest version of TID-14844 was published in draft "Accident Source Terms for Light Water Nuclear Power Plants" (NUREG-1465).

The NRC has also entered into an agreement with the Commissariat L'Energie Atomique of France (CEA) to participate in the PHEBUS-FP program. The program, sponsored by the CEA and the Commission of the European Communities, is an ongoing study of those phenomena that govern the transport, retention, and chemistry of fission products during LWR severe accident conditions, in an in-pile facility providing prototypical conditions.

The experimental data from PHEBUS-FP are confirmatory in nature and will be used to assess the revised source term assumptions used in NUREG-1465.

The PHEBUS-FP facility has received a license to refuel and start up. Final preparation for the first test, FPT0, is near completion, and the test is scheduled for the first quarter of fiscal year 1994. The test matrix consists of six tests, with testing at a rate of oneper-year. The test matrix has been revised to include fission product release tests under shutdown conditions for a degrading core in an air environment and for rubble beds.

**Core-Melt Progression.** "In-vessel core-melt progression" describes the state of an LWR reactor core from core uncovery up to reactor vessel melt-through, in unrecovered accidents or through temperature stabilization in accidents recovered by core reflooding. Melt progression provides the initial conditions for assessing the loads that may threaten the integrity of the reactor containment. Significant results of melt progression are the melt mass, composition, temperature (superheat), and the rate of release of the melt from the core, and later from the reactor vessel if vessel failure does occur. Melt progression research provides information about the in-vessel hydrogen generation, the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system, and the core conditions for assessing accident management strategies.

Current NRC research on melt progression is focused on two major issues. The first issue is determining whether there are any accident conditions for BWRs (and possibly PWRs) in which a metallic core blockage similar to that at Three Mile Island Unit 2 (Pa.; TMI-2) would not be formed. The second issue concerns the conditions of melt-through for the growing pool of ceramic melt above the metallic blockage.

On the issue of blockage of the core by metallic melt, TMI-2 and the results of the experiments cited above have indicated that, for "wet core" conditions (with water in the bottom of the core), the relocating molten metallic Zircaloy in the core freezes to block the lower core. All but one of the previous experiments for both PWRs and BWRs were performed for these wet core conditions, and this one experiment did not address the blockage or drainage question. The emergency operating procedures for U.S. BWRs, however, call for reactor depressurization, which lowers the water level below the reactor core so that core heatup occurs with very low steam flow through a "dry core." Analysis of this case indicates that the molten core metal (and later molten ceramic fuel) might drain from the core, rather than form a blocked core as at TMI-2. Drainage would produce a major difference in the mass and other characteristics of the melt released from the core and later from the vessel at melt-through.

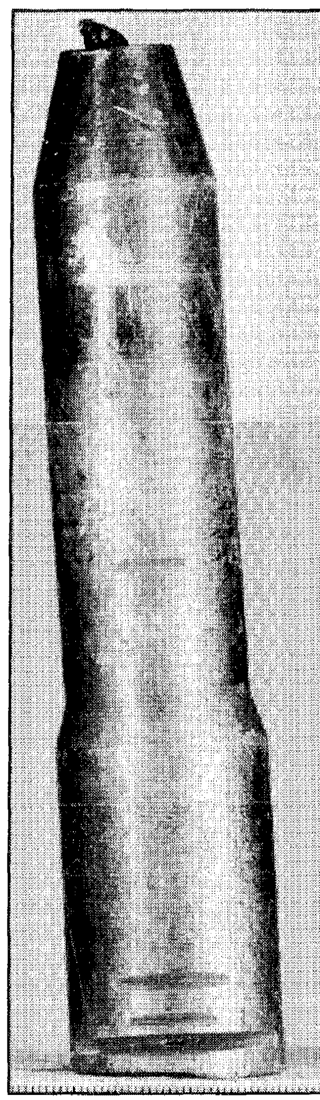
In fiscal year 1993, a series of new experiments was prepared to determine whether metallic melt drainage, core blockage, or core plate blockage occurs under BWR dry core accident conditions. The initial test in a simplified system was performed successfully. The experiment test assemblies are a mockup at full radial scale of a cross section of the lower quarter of a BWR core (and core plate region) where blockages might occur, and they have prototypic reactor materials, heat capacities, geometries, and temperature distributions. Melts of metallic Zircaloy that also contain control-blade materials are poured into a test assembly at prototypic rates, and the melt relocation and blockage behavior are observed.

In fiscal year 1993, late-phase melt progression experiment MP-2 was performed in the Annular Core Research Reactor (ACRR) to complete the currently planned experimental program on late-phase melt progression, which involves ceramic (fuel) melting and relocation. Along with the earlier MP-1 experiment, MP-2 has provided unique information on the governing processes in the growth and melt-through of a ceramic melt pool in a particulate ceramic debris bed in blocked core accidents. These were the conditions underlying the TMI-2 accident. With the results and interpretation of MP-1 and MP-2 in hand, an expert peer review group will be convened in fiscal year 1994 to review the status of the current knowledge of late-phase melt progression, the significance of the remaining late-phase melt progression uncertainties, and the need for, efficacy of, and nature of any further research in this area.

**Reactor Vessel Integrity.** In 1988, the NRC—in cooperation with 10 foreign countries, under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency (NEA)—undertook a follow-on program to the TMI-2 core examination conducted by the U.S. Department of Energy. At the outset of the program, called the TMI-2 Vessel Investigation Project (VIP), test specimens from the lower head of the TMI-2 reactor vessel were removed and examinations of the specimens were conducted to obtain information on the melt attack on the lower head during the accident. The United States and the foreign countries participating in the OECD/NEA project have performed metallurgical and mechanical examinations of the TMI-2 test specimens. Results of metallurgical examinations of the vessel steel samples allow for estimates of temperature histories of the lower head samples. These specimens indicated that some regions of the lower head reached temperatures during the accident that exceeded the transformation temperature of the steel (727°C). A hot spot was found in an elliptical region on the lower head (about one meter by 0.8 meter) in which the inner surface of the reactor vessel steel reached temperatures as high as 1,100°C, and remained at that temperature for approximately 30 minutes. Results of examinations of instrument tube nozzles extracted from the lower head showed that some nozzles were melted off by interaction with molten core debris, while other nozzles were not affected at all. In general, it was found that the nozzles with the greatest damage were located in the vicinity of the hot spot.

A more general study of reactor lower head failure for both PWRs and BWRs was also completed in fiscal year 1993. It was found that the mode and timing of lower head failure resulting from in-vessel melt progression have controlling effects on the subsequent containment loads during a postulated severe accident. A final report, "Light Water Reactor Lower Head Failure Analysis" (NUREG/CR-5642, October 1993), documents the results of this research. The report presents the results of potential failure mode analyses for a range of debris conditions, lower head designs, and accident scenarios. The failure modes include global creep-rupture of lower head, penetration tube melt-through, tube ejection, and ablation by jet impingement of molten core material. In addition, an analysis of a limited vessel wall area that may be heated to a high temperature, as occurred in the TMI-2 accident, was developed.

Results of the lower head failure analysis are presented in NUREG/CR-5642, in terms of key dimensionless parameters to provide "failure maps" that indicate the relative potential for failure of the lower head in various failure modes. Creep-rupture data and high-temperature material property data were required that were not previously available. Material data in the literature apply to design conditions, whereas these failure analyses require data in the vicinity of failure conditions, for example, creep-rupture times of 1-to-100 hours. Thus, the required data were obtained as part of the lower head failure program for both pressure vessel steel and penetration materials (inconel, stainless steel, and SA105 steel).



**Shown here are two nozzles that were removed from the reactor vessel of the Three Mile Island Unit 2 (TMI-2; Pa.) facility, as part of the TMI-2 Vessel Investigation Project (VIP). One nozzle did not undergo any damage and one was severely damaged, with most of its original length melted away. Data from reactor vessel examinations performed under the VIP, and other data derived from studies of the accident, were used in making calculations on potential vessel failure modes. All of the technical work for the TMI-2 VIP was completed during fiscal year 1993, and final reports issued in October 1993.**

**Fuel-Coolant Interactions.** Since the quantification of a steam explosion-induced missile as a possible mode of containment failure (alpha mode) in the reactor safety study called WASH-1400, significant progress has been made in understanding the limitations of damage from such potential missiles. For example, in NUREG-1150 alpha-mode failure is not a dominant contributor to early containment failure. The emphasis prior to NUREG-1150 on fuel-coolant interactions (FCIs) was given to conditions for in-vessel molten fuel pouring into a coolant pool and the likelihood of its causing containment failure by energetic interactions. The current emphasis of FCI research is to provide the appropriate phenomenological and analytical tools to address those aspects of FCI that are relevant to three specific issues: (1) FCI energetics, (2) fuel melt quenching in water pools, and (3) water added to a degraded core.

Complementary to the experimental programs on FCI, an Integrated Fuel-Coolant Interactions (IFCI) code was developed by the Sandia National Laboratories. Work was completed in fiscal year 1993 to modify the IFCI modules and produce a "stand-alone" code for workstation computers. An operational report with examples of runs using the standalone version and a code manual were completed in fiscal year 1993. Validation of the code will be the major effort in the next year.

The NRC and the Safety Technology Institute of the Joint Research Center (STI-JRC) of the Commission of the European Communities at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of FCI experiments at the FARO facility located in Ispra. At the STI-JRC FARO facility, large masses of prototypic reactor core materials can be melted and can interact with different depths of coolant at different temperatures and pressures. At least five molten FCI experiments will be conducted to obtain data prototypical of reactor conditions in the United States.

**In-House Severe Accident Analysis Capability.** Growth in the capability of workstation level computers provides the opportunity to run severe accident codes on other than main-frame computers. In fiscal year 1993, RES purchased workstations to enhance the in-house analysis capability at the NRC. Reactor plant descriptions, or decks, for analyses using the MELCOR, SCDAP/RELAP, CORCON, CONTAIN, and VICTORIA codes, have been installed on the workstations. Typical uses of this new in-house capability have been to review input decks developed by NRC contractors and to use these decks to extend previous analyses. In-house analyses have also been used to check new models in the codes and to do bounding calculations to determine the appropriateness of the new models. The following describes examples of some of the work done in this area in 1993:

(1) A CORCON ABWR deck was developed, and sensitivity analyses were performed to investigate the effects of melt temperature, melt mass, melt composition, concrete composition, and other code parameters that influence chemistry and heat transfer modeling. Results of this work were provided to NRR.

(2) An AP600 MELCOR input deck, developed by the Brookhaven National Laboratory, was run and evaluated. Several problems regarding control function activation levels were discovered, and NRR users were notified of these problems.

(3) The in-house effort has included an application of CONTAIN to the AP600. Sandia National Laboratories developed the input deck and studied heat transfer modeling uncertainties and the uncertainties in passive containment coolant system (PCCS) shell wetting. The in-house effort extended these studies to look at uncertainties in thermal radiation modeling in the PCCS and aerosol modeling in the containment. Future in-house code analysis effort will involve MELCOR 1.8.2 to review the new bottom head model developed at the Oak Ridge National Laboratory.

## Reactor Containment Structural Integrity

The major undertaking in this program for the next few years will be a cooperative one with the Ministry of International Trade and Industry (MITI) of Japan. Two areas of cooperation have been identified—one dealing with steel containments used in both the United States and Japan for BWR designs, the other related to pre-stressed concrete containments. The current generation of Japanese PWR containments are of a pre-stressed concrete design.

A reinforced concrete model was chosen for the NRC-sponsored testing at Sandia National Laboratories (SNL) that was performed in 1987. Subsequent analyses of the results of that model test have shed light on how potential failure modes develop in concrete containments. Some of the results are felt to be applicable to pre-stressed concrete containments as well. However, there are two main reasons for performing an additional pre-stressed containment model test:

- Pre-stressed designs are the most common concrete PWR containment type in the United States. There are 41 pre-stressed containments, as compared with 20 reinforced containments.

- The margin between the ultimate capacity and the design pressure for pre-stressed containments is now thought to be somewhat lower than that for reinforced concrete or steel containments; hence, it is important to have accurate predictions of the ultimate behavior of pre-stressed containments.

A test-to-failure of a model of a steel BWR containment vessel will also be included in the cooperative research program. The vessel would be fabricated in Japan and shipped to SNL. The test would complement the test-to-failure of a steel containment model performed by SNL in 1984, under NRC sponsorship. That model was cylindrical in cross section and was representative of PWR ice condenser and BWR Mark III containments. The proposed Japanese model would include the "knuckle regions" that are present in the U.S. BWR designs. It is currently presumed that state-of-the-art analytical methods can be relied upon to provide adequate predictions for the response of those designs to severe accident conditions. However, there are no experimental data against which the predictive methods can be checked. The proposed model test would fill that gap in the data base.

## Severe Accident Policy Implementation

In the 14 years since the Three-Mile Island accident, the NRC has sponsored an active program in research on severe nuclear power plant accidents, as part of a multifaceted approach to the assurance of safety in this context. Other elements of this approach include improved plant operations, human factor considerations, and probabilistic risk assessments. In August 1985, the Commission issued a Severe Accident Policy Statement (50 FR 32138), which concluded that existing plants posed no undue risk to public health and safety. However, the Commission recognized that systematic examinations of existing plants could identify plant-specific vulnerabilities to severe accidents for which further safety improvements could be justified.

**Individual Plant Examinations.** In connection with the implementation of the Commission's Severe Accident Policy Statement, the staff has required individual plant examinations (IPEs) of all existing plants to identify any plant-specific vulnerabilities to severe accidents. The task has involved development of guidance for performance of the IPE, preparing a Generic Letter to plant operators requesting the IPE, and developing review plans and eventually reviewing the results of the IPE submittals. Imposition of any requirement to correct any identified plant-specific vulnerabilities not voluntarily corrected will be governed by the backfit rule. Accident management is not part of the IPE process but will make use of the results derived from the process. The IPE process involves two different efforts. The first is an examination of existing plants for vulnerabilities to severe accidents resulting from events occurring within the plant (e.g., equipment failures, pipe breaks). The second effort is to consider severe accident vulnerabilities from external hazards (e.g., earthquakes, floods, winds). This activity is referred to as the individual plant examination for external events (IPEEE).

Twenty-six new submittals for internal events were received from licensees in fiscal year 1993, making an overall total of 63 submittals received to date. Staff evaluations were issued for Turkey Point Units 3 and 4 (Fla.), Oconee Units 1, 2, and 3 (S.C.), Beaver Valley Unit 2 (Pa.), and Diablo Canyon Units 1 and 2 (Cal.) and draft staff evaluations were completed for FitzPatrick (N.Y.), Surry Units 1 and 2 (Va.), Millstone Unit 1 (Conn.), and Monticello (Minn.). All IPE submittals are expected to be received and reviewed by the end of calendar year 1995.

The approach for review of the IPEEE will follow closely that developed for review of the internal-event IPE submittals. The staff initiated the procurement process to obtain contractual assistance for the IPEEE reviews. Four IPEEE submittals have been received, with two currently in the review process.

# SAFETY ISSUE RESOLUTION AND REGULATION IMPROVEMENTS

## Earth Sciences

The objective of NRC research in earth sciences, as related to reactor regulation, is to define potential earthquake ground motions at nuclear power plant sites and in the regions surrounding them. This information provides a basis for evaluating the effects of earthquakes on the plants and their safety systems.

Seismic hazards contribute a sizable proportion to its overall plant hazards and, because of inherent difficulties in defining them, they form an even more significant portion of the overall uncertainty in estimating plant hazards. In order to reduce these uncertainties, RES is continuing research into the causes and distribution of seismicity. Research is also progressing on improved methods of applying earth science information to estimates of ground motion levels for use in plant design.

**Seismographic Networks.** For about a decade and a half, the NRC has supported regional seismographic networks, primarily in the central and eastern United States where most of the nuclear plants are located and where seismicity is less well defined than in the western United States. These networks have provided essential earthquake data to better describe the seismicity in this region and to compare the seismicity with geologic and tectonic information, in order to gain insight into structures in the earth's crust that may create a potential for earthquakes. The NRC discontinued most of these networks in September 1992. Three networks, those operated by Weston Observatory, Massachusetts Institute of Technology, and Virginia Polytechnic Institute and State University, were continued through the report period until September 1993. The function formerly served by the regional networks has been taken over by the new National Seismographic Network (NSN), established through a cooperative agreement with the U.S. Geological Survey (USGS).

The NSN was officially dedicated in April 1991. At present, the network consists of 16 network stations and 13 cooperating stations operated by the IRIS consortium and various universities. A number of regional seismographic network stations have also been integrated into the network, forming a national seismic system and providing more detailed coverage for special regions, such as the New Madrid, Mo., area. The NSN operates with high-quality, three-component stations and satellite telemetry, thus providing data on significant earthquakes within minutes.

Near the end of the report period, a broad agency announcement was made public with the purpose of establishing research contracts for analyzing NSN data and other available seismological, geological, and geophysical data. This research will continue the type of investigations previously carried out by the universities operating regional networks. It is anticipated that the high-quality, broadband, and three-component data of the NSN will lead to new insights into the causes and distribution of seismicity and on the ground motion propagation characteristics of the earth's crust, particularly in the central and eastern United States.

**Northeastern Neotectonics.** During fiscal year 1993, investigations were conducted at Wonalancet/Ferncroft in central-east New Hampshire, near the epicenter of the 1940 Ossipee earthquake; and in the areas of northeastern Massachusetts and southeastern New Hampshire that were affected by the 1727 Newburyport and the 1755 Cape Ann earthquakes. Both areas are sites of current seismic activity.

In the Wonalancet/Ferncroft area, the investigations performed consisted of trenching, a ground-penetrating radar survey, geotechnical engineering tests, landslide/rockfall reconnaissance, and geophysics. No evidence was found that indicated the occurrence of prehistoric earthquakes larger than the 1940 event. Samples were taken for radiocarbon age-dating to determine the length of time that a large earthquake has not occurred in this area.

In the Newburyport and Cape Ann region, many thousands of linear feet of exposure were observed along marshes, estuaries, and rivers. Although liquefaction-susceptible soils were found, no seismically induced paleoliquefaction features were identified along these exposures. Samples for carbon-14 age-dating were obtained to constrain the time period during which a large earthquake has not occurred.

During fiscal year 1993, an investigation was begun in the epicentral area of the 1944 Cornwall-Messena earthquake (magnitude-5.5) to determine whether there was evidence there for prehistoric moderate-to-large earthquakes. The 1944 event induced numerous liquefaction features, and the strategy is to identify similar features that predate the 1944 features and suggest earlier occurrences of similar earthquakes in the late Holocene.

Another paleoseismic investigation that is under way along the Atlantic coast of North America is a study of tsunami deposits left by the 1927 Grand Banks earthquake (magnitude-7), the development of criteria to distinguish between these deposits and stormgenerated sediment, and a preliminary search for earlier tsunami deposits that were the result of prehistoric Grand Banks-sized earthquakes. (A tsunami is a large wave caused by an earthquake under the sea.) Like the previously described studies, this one is an attempt to extend the relatively short historic seismic record back into time.

**Faulting in Giles County Seismic Zone in Virginia.** In June 1992, two faults were discovered at a barrow pit site in Pembroke, Va., near the epicenter of the 1897 Giles County Modified Mercalli Intensity VIII (magnitude-5.8) earthquake. The faults displace high-level terrace soils of the New River, which consist of bedded silts and gravels estimated to be of Quaternary age (less than two million years old). One fault strikes north 64° east, dips 60° to the northwest, and displaces the terrace strata more than three meters. The second fault strikes north 70° east, dips 80° to the northwest, and offsets the soils about one meter. Minor tension cracks and slip surfaces are also present in the outcrop. Further examination revealed that these faults formed the eastern margin of a small graben (a narrow depression bounded by faults on either side) within the eastern limb of a small northerly trending anticline. Three possible origins for the faults are being considered—landslide, karst, and tectonic.

Additional investigations during fiscal year 1993 revealed that the faults bounding the graben (the two faults described above on the east boundary and another fault forming the west boundary) experienced approximately 11 meters of normal displacement. Other faults were mapped in the excavation, which eventually exposed up to 50 feet of vertical face. The other faults consisted of small normal faults with 30 centimeters or less of offset and reverse faults with apparent displacements up to one meter. Preliminary geophysical investigations south of the exposure have been inconclusive as to the origin of the faults, but some methods show promise of helping to resolve this issue. Pending the availability of funding, core borings, trenching, age-dating of soils, and geophysical profiling are planned for fiscal year 1994.

**Paleoseismicity of Southern Illinois and Indiana.** An investigation began in fiscal year 1991 to identify and analyze paleoseismic evidence along the Wabash River valley and valleys of its major tributaries. To date, hundreds of planar, nearly vertical, sand- and gravel-filled dikes—caused by earthquake-induced liquefaction—have been discovered in these valleys in Indiana and Illinois. The dikes range in width from a few centimeters to as much as 2.5 meters; the largest of them were found around Vincennes, Ind., and they decrease in size and abundance to the north and south of this area. Studies indicate that most of these features were caused by a large earthquake (estimated magnitude of about 7.5) that occurred in the Vincennes area between 2,500 and 7,500 years ago.

Investigations during fiscal year 1993 were carried out in southern Indiana and Illinois. Results thus far confirm the occurrence of a very large earthquake (magnitude-7.5) about 6,000–7,000 years ago, centered near Vincennes, and indicate that there was a liquefaction-producing, moderate earthquake about 4,000 years ago, and a strong earthquake centered in south-central Indiana 4,000-6,000 years ago.

**New Madrid Seismic Zone.** Several sites are being investigated in the New Madrid seismic zone to define faults associated with the seismicity there. The sites are at the intersection of Crowely Ridge and the west side of the Reelfoot Rift, Marston, Mo., where waterfalls formed in the Mississippi River during the 1811–1812 New Madrid earthquakes; at the Crittenden County fault zone, which is the local east margin of the Reelfoot Rift; and the Bootheel fault zone, which is located in the middle of the rift zone. Trenching will be carried out at specific sites pending the results of the geophysical studies.

Based on limited evidence, it is hypothesized that a recurrence interval for the 1811–1812 New Madrid earthquakes (magnitude-8) ranges from 550-to-1000 years. However, this idea was brought into question following recent studies of the banks of numerous Corps of Engineers drainage ditches that exposed Holocene soils, which were strongly affected by the 1811–1812 earthquakes but showed no indication of deformation by prehistoric events. The focus of another research project in this region is to determine whether geologic evidence supports a recurrence of earthquakes like the 1811–1812 magnitude-8 events and to attempt to determine the ages of those events and the regional extent, if they exist, and to develop criteria for identifying them. Preliminary results indicate that there is paleoliquefaction evidence for at least one such prehistoric event.

**Pacific Northwest.** The Pacific Northwest, from southwestern British Columbia to northern California, is underlain by the Cascadia subduction zone, into which three minor oceanic plates—the Explorer, Juan de Fuca, and Gorda plates—are being subducted beneath the North American plate. Although geological and geophysical evidence indicates active subduction, there have been no historic large-thrust earthquakes along the plate interface—the type of earthquake that characterizes other active subduction zones around the rim of the Pacific Ocean.

The USGS has completed a major five-year study of the geology and tectonics of the Pacific Northwest and continues to sponsor more limited research in this area. The NRC is partially funding several projects under this program in western Washington and Oregon. These efforts are continuations of investigations that revealed geological evidence suggesting the occurrence of several large prehistoric earthquakes during the past several thousand years. This evidence consists of several cycles of normal stratigraphic deposition of shallow marine sediments overlain by marsh deposits, each of which has been abruptly terminated by a catastrophic subsidence event and a new cycle has begun. These events are interpreted to be related to the occurrence of large subduction zone earthquakes. Along the coast, geologic and radiocarbon data indicate that the most recent of these events occurred about 300 years ago, affecting lowland soils at the Copalis River and at Willapa Bay about 65 kilometers apart. A 300-year-old event is also represented in northern California about 610 kilometers to the south. Two of the research projects have been concentrating on determining whether these widespread deformations were caused by a single magnitude-9 earthquake or by several smaller events of magnitude-8 or less. The data available so far support either hypothesis.

Investigations along the coast of north-central Oregon confirmed the regional subduction zone subsidence events, but also identified geological evidence for local prehistoric earthquakes and subsidence-like evidence that may have been related to nonseismic phenomena such as storm surges or flooding from the damming of estuaries by sand barriers.

In conjunction with these studies, a study is under way to identify and define seismically induced paleoliquefaction features in the region to determine whether strong shaking occurred during these subsidence events. Reconnaissance investigations in the Chehalis River valley and other drainages in southwestern Washington did not identify such features, even though there are long stretches of exposures of liquefaction-susceptible soils along the river banks.

The first positive evidence for seismic shaking that can be attributed to a subduction zone earthquake in the Pacific Northwest was found in the Columbia River estuary. The evidence consisted of seismically induced paleoliquefaction features (sand dikes and sills) on islands within the estuary. The features range from up to 0.3 meter in size, which are very numerous in the vicinity of Astoria, Ore., to fewer in number and smaller in size upstream ranging from 7.5-to-10 centimeters about 30-to-0 kilometers away, and 2.5-to-5 centimeters wide about 60 kilometers inland. The dikes and sills are estimated to be about 300 years old, based on the estimated age of soils cut by the dikes (specifically, a 1,482-year-old layer of pumice), younger undisturbed soils, and the age of the oldest living trees (240 years) unaffected by the event. The evidence for shaking is correlated with the 300-year-old subsidence event in southwestern Washington.

Field studies in fiscal year 1993 showed that paleoliquefaction features extend at least an additional 30 kilometers upstream in the Columbia River for a total distance of 90 kilometers from the coast. Preliminary geotechnical investigations of the liquefaction susceptibility of soils on Wallace Island in the Columbia River estuary suggest that the shaking that accompanied the 300-year-old event was probably less than that which would be expected from a great subduction zone earthquake.

Geological evidence from excavations at West Point, Wash., 10 kilometers northwest of downtown Seattle, indicates that tsunami-like surges of sandy water from Puget Sound covered a tidal marsh that subsided at least 1/2 meter about 1,100 years ago. Estuarian mud about 1/2-meter-thick overlies the sand and marsh deposits. Radiocarbon age dates of plants buried beneath the mud range from 900-to-1,300 years. These data—along with other geological evidence gathered by other researchers in the Puget Sound region (such as submarine slides in Lake Washington, uplift at Restoration Point on Bainbridge Island, geophysical and stratigraphic evidence for a large east-west striking fault in south Seattle-Seattle fault)—suggest the occurrence of a large (magnitude-7) earthquake on the Seattle fault about 1,100 years ago.

Field studies in fiscal year 1993 found additional evidence of a tsunami generated by the 1,100-year-ago event in a cove on Whidby Island in Puget Sound in the form of a buried sheet of sand that underlies the cove and laps up on its flanks. The cove is considered to be favorably oriented to receive a seismically generated sea wave from an earthquake on the Seattle fault.

**Fault Segmentation Studies.** It is well known that faults do not usually rupture over their entire length during a single earthquake. Numerous structural and paleoseismic studies and investigations of historical earthquakes indicate that there are physical controls within a fault zone that define the extent of rupture and divide a fault into segments and that these segments can persist through many earthquake cycles. The project is being carried out to establish a basis for recognizing and identifying geometrical and structural features that constrain or control rupture propagation within a fault zone.

Evaluation of the segmentation for selected faults was begun in fiscal year 1991 using paleoseismic recurrence data and information on slip-per-event and slip rate. Studies in fiscal year 1992 continued on these faults, including the Rodgers Creek-Hayward fault zone, the segment of the San Andreas fault that ruptured during the 1989 Loma Prieta earthquake, the Wasatch fault zone, and the Calaveras, Superstition Hills, Imperial, White Wolf, Lost River, Red Canyon-Hebgen, Dixie Valley-Stillwater, Pleasant Valley, North Anatolia (Turkey), Pitagcachi (Mexico), Oued Fodda (Algeria), Marriot Creek, Tennant Creek (Australia), and Landers faults.

Work during fiscal year 1992 on the Rodgers Creek fault provided the first estimates of the timing of individual paleoearthquakes with events at about 1000 A.D., between 1200 and 1400 A.D., and between 1650 and 1808. Additional evidence supporting a six-kilometer-wide step between this fault and the Hayward fault was found. Studies at Grizzly Flat on the San Andreas fault revealed evidence for the last two large surface faulting events, one after 1800 A.D. (probably 1906), and the other before 1636–1660 A.D. Along with evidence gathered by other researchers farther north along the fault, these findings indicate a recurrence interval along this part of the San Andreas fault of about 250 years. Initial data on fault geometry, lithology, and rupture direction, collected for the Coyote Lake, Morgan Hill, and Alum Rock earthquakes on the Calaveras fault, indicate a south-to-north progression of events. However, a north-to-south rupture propagation during each event was indicated.

Studies were begun late in fiscal year 1992 on the complex 85-kilometer-long surface rupture of the 1992 Landers earthquake (magnitude-7.5) to determine its implications for segmentation modeling. The rupture was characterized by strike-slip faulting containing at least three major geometric segments with echelon steps up to 2.5 kilometers across.

In fiscal year 1993 six trenches were dug across the 1992 rupture along the Homestead Valley fault where it truncated an alluvial fan, and farther to the south where the rupture cuts across a playa. Four events have been identified: the 1992 event, an event about 4,000 years ago, one 8,000 years ago but with a very large error band, and an event about 14,000 years ago with an even larger error band. The next step is to excavate trenches across a segment of the Emerson fault that did not rupture in 1992 and continue to try to correlate events from fault segment to fault segment to test the fault segmentation model and the characteristic earthquake model.

**Strong Ground Motion Studies.** In 1989, in cooperation with the French Commissariat a l'Energie Atomique, a seismic experiment was undertaken at Garner Valley, Cal., to measure *in-situ* amplification and attenuation of seismic waves that propagate through a soil column from bedrock to ground surface. The original contract was for the design, construction and deployment of five downhole accelerometers and a field operable data-acquisition system. In 1990, EPRI funded the installation of another downhole accelerometer and four surface accelerometers, along with added data-acquisition capability for the extra accelerometers. As presently deployed, the system comprises five surface accelerometers in a linear array spanning 310 meters and five accelerometers at depths from six meters to 220 meters. The network is located seven kilometers from the San Jacinto fault, at the northern end of the Anza seismic gap on the fault, where a magnitude-6.5 or greater earthquake can be expected, and 35 kilometers from the Indio segment of the San Andreas fault.

Since its operation began, the downhole seismic array has recorded numerous earthquakes ranging in magnitudes from 6.1 to approximately 1.0. Analyses of the data through fiscal year 1991 indicated that the spectral amplitudes recorded at ground surface are amplified on average by a factor of 10 over the spectral amplitudes at 220 meters depth. Resonance peaks have spectral ratios (surface spectral amplitudes divided by those at 220 meters) of about 40 for frequencies near 1.7, 3.0, and 12.0 Hz.

In fiscal year 1992, more than 250 earthquakes were recorded, the largest of which was the April 23 Joshua Tree earthquake at a distance of 45 kilometers from the array and a depth of 13 kilometers. Maximum acceleration recorded from the event was 89 cm/s2 at ground surface. Recordings were also obtained from the foreshock and the aftershock. Unfortunately, the data-acquisition system was not working on June 25 during the Landers earthquake. Amplification characteristics for ground motion of fiscal year 1992 earthquakes are being analyzed.

Because of the relative lack of near-field recordings of large intra-plate earthquakes, such as those in the eastern and central United States, the prediction of strong ground motions radiated by these types of earthquakes is severely hindered. To compensate for this lack of near-field recordings, an analytic method was developed by the USGS to correct teleseismic recordings of the Global Digital Seismic Network for focal mechanisms, interference of the depth phases, and the teleseismic attenuation, in order to estimate the acceleration source spectrum of the earthquake in the frequency band from 50 seconds-to-2 Hz. Many large intra-plate earthquakes have been analyzed to estimate the acceleration spectral level expected for near-field strong ground motion in northeastern North America. In fiscal year 1992 the extensive near-field and regional accelerograph recordings from the 1989 Loma Prieta earthquake were analyzed with a view to applying the results to predicting strong ground motions in eastern North America.

During fiscal year 1993, by studying the "S-wave trains" from 97 earthquakes recorded by the Eastern Canada Network, including the Saguenay, Mt. Laurier, Miramichi, Goodnow, Gaza, and Painesville earthquakes, a model for attenuation of ground motions was developed, and information was obtained about propagation and source characteristics in the eastern United States.

One of the objectives in the USGS strong ground motions program is to use the stochastic model to predict ground motions from earthquakes in eastern North America. In fiscal year 1992, an extension of the Boore and Atkinson (1987) ground motion predictions to deep soil sites was completed, representing an initial step in generalizing the prediction methodology to account for local variations in site geology. During fiscal year 1992, much of the initial development of a strong-motion data base, including selection of those earthquake records that meet established quality control criteria, was completed.

The updated work in fiscal year 1993 resulted in a division of sites into four building code-like classes based on shear wave velocities: soil A— >750 m/s, soil B—750-to-360 m/s, soil C—360-to-180 m/s, and soil D— <180 m/s.

Another ground motion study that was made during fiscal year 1993 was in regard to rupture histories of eastern North American earthquakes. Sophisticated methods were used in the western United States (inverting teleseismic and strong motion recordings for space-time slip distributions) to analyze large eastern U.S. earthquakes such as Miramichi, two of the Nahanni events, Ungava and Saguenay. The Saguenay event exhibited a concentrated source rupture pattern with an initial high-stress drop that spread over a broader area. The slip concentrations of the two Nahanni earthquakes were spatially complementary. The Ungava rupture took place within the upper three kilometers of the crust.

At the Savannah River site, a seismic array has been installed in a borehole. Four events in South Carolina were recorded, the largest of which was a magnitude-4 at Summerville. The data are still being analyzed, but initial results indicate that stress drops increase strongly with increasing moment.

Digital aftershock data from the 1992 Petrolia, Cal., earthquake sequence were analyzed in an attempt to determine the reasons for the high accelerations recorded at several of the stations. The results indicated that the high ground motions at the Cape Mendicino Station and the Petrolia General Store sites were most likely caused by site responses. The results of the analytic technique applied to other sites with anomalous readings indicated that the causes were attributable to either poor instrument calibration or to wave propagation characteristics.

**Crustal Strain Measurements.** During fiscal year 1993, the crustal strain network for the central and eastern United States was measured for the third time since 1987. After this strain network was established, it became the backbone of a new geodetic network for the United States based on Global Positioning System (GPS) measurements. In addition, high precision GPS networks have been established for many states and, within the next few years, all of the United States will be covered with detailed high precision GPS networks for surveying purposes. Because of this, many stations are now available for strain determinations, in addition to the original 45 stations of the crustal strain network. These additional stations will also be periodically resurveyed and, in many locations, permanent GPS stations have been established that will provide a continuous record of measurements.

Because the intra-plate strain rates in this region are expected to be low, many years may be needed to arrive at meaningful strain determinations. However, with the large number of high precision GPS stations now available, it should eventually be possible to get a very detailed picture of strain distribution. Detailed information on deformations in the crust and their temporal rates will then provide a basis for refinements in seismic hazard determinations.

**Probabilistic Seismic Hazard Assessments.** Probabilistic seismic hazard assessments (PSHAs) were instituted about a decade ago, and they have become an increasingly important aspect of site evaluations for nuclear power plants and other facilities. The revision to Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," still in progress, will put substantial emphasis on PSHAs as part of the investigation required for proposed nuclear power plant sites. PSHAs are of particular interest in the central and eastern United States where uncertainties created by a lack of detailed knowledge of the seismicity make it difficult, by a deterministic evaluation, to arrive at a dependable estimate of seismic hazards.

Two large-scale PSHA studies are available for the central and eastern United States. One was performed by Lawrence Livermore National Laboratory (LLNL) and sponsored by the NRC (NUREG/CR-5250); the other was performed by EPRI and sponsored by utilities in the Seismicity Owners Group. The two studies used similar methodologies and produced hazard curves with similar characteristics; they also produced consistent relative hazard rankings for plant sites in the region. A serious problem arises, however, from the fact that, at certain sites, absolute hazard levels may differ significantly.

Results from both studies are used by NRC staff for regulatory decisions, but, for future nuclear plant design and licensing, more consistent hazard values will be needed. At the end of fiscal year 1992, an effort was begun to analyze differences between the LLNL and EPRI seismic hazard methodologies and to arrive at a more unified methodology that will produce more reliable absolute hazard levels. From previous analyses, it was known that methods of eliciting expert opinions and certain other factors—such as seismic parameters and ground motion models—cause some of the observed differences. The computer programs used for the LLNL and EPRI methods, although different, are designed to solve the same basic equation and do not seem to be a cause of discrepancy.

The study will be conducted cooperatively by the NRC and the Department of Energy (DOE), both because its cost will be relatively high and because the DOE also has an interest in PSHA methods for assessing the numerous critical facilities it operates. EPRI will also make a significant contribution to the research through the DOE. The NRC is sponsoring a peer review by a panel formed by the Committee on Seismology of the National Academy of Sciences/National Research Council. The peer review panel will provide an independent, scientific review of the project and thus ensure the impartiality and objectivity of the study. It is expected that the study will be completed in 1994.

## Plant Responses to Seismic And Other External Events

Besides the earth science research discussed above, the NRC seismic research program includes several engineering-oriented programs to determine the effect of earthquakes on nuclear plant structures and safety systems.

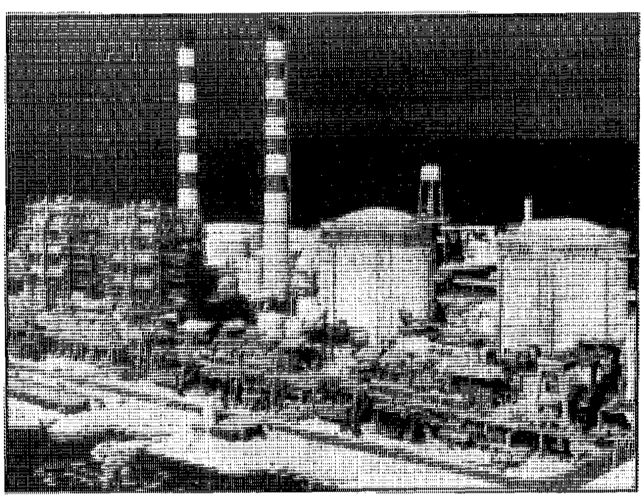
**Revision of Appendix A to 10 CFR Part 100.** On October 20, 1992, the NRC published for public comment (57 FR 47802) the proposed revision of Appendix A to 10 CFR Part 100 (see the *1991 NRC Annual Report*, p. 185). The public comment period was extended twice—the first time (58 FR 271) so that the expiration date would be consistent with the expiration date of the supporting regulatory guides (57 FR 55601); the second time (58 FR 16377) in response to a public request. The comment period expired on June 1, 1993.

Responses were received from approximately 47 domestic and foreign commenters. The domestic organizations providing comments included state geological surveys, the U.S. Geological Survey, the Association of Engineering Geologists, and industry representatives. Nine foreign countries either individually or as a group provided comments. The staff is reviewing all the comments and will revise the regulations and guidance documents as appropriate during fiscal year 1994. Revisions of the geologic, seismic, and earthquake engineering criteria are being performed in conjunction with the revision of the reactor site criteria, 10 CFR Part 100.

**Seismic Testing of Relays.** Seismic testing of relays to determine the influence of relay chatter on circuit breaker tripping among other things was completed in fiscal year 1993. The research initially was intended to support the resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants," but will also serve the needs of IPEEE and seismic PRAs for advanced light-water reactors (ALWRs). Results obtained in fiscal year 1993 indicate that relay chatter may or may not be acceptable in specific circuits depending on the circuit parameters. Thus, the two millisecond chatter criterion in IEEE codes may not be appropriate in all cases. Further evaluations are needed to decide on the appropriate course of action.

**Hurricane Andrew.** On August 24, 1992, Hurricane Andrew, a Category-4 hurricane, struck the Turkey Point (Fla.) nuclear power plant with sustained winds of 145 mph (233 km/h). During fiscal year 1993, a combined NRC and Institute of Nuclear Power Operations (INPO) team investigated the impact of the hurricane on the Turkey Point Units 3 and 4 nuclear power plants. The emphasis of the investigation was on identifying those areas, events or conditions that were problematic for the facility and Florida Power and Light (FPL) staff as well as those special preparations or actions that had a positive effect on the course and consequences of events relevant to plant safety. A representative from RES interviewed FPL staff about the performance of structures associated with the nuclear units, the chimneys and Bunker C oil tank associated with the fossil units, and an earlier systematic evaluation they had performed—the individual plant examination to identify severe accident vulnerabilities because it included the results of the wind and fire externalevent analyses. The team report, "Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20–30, 1992" (NUREG–1474), was published in March 1993.

Earthquake Response. Following 1992 earthquakes in the Cape Mendocino and Landers/Big Bear areas of California, investigations were conducted to document the impact that the seismic events had on several selected non-nuclear industrial facilities. The facilities selected included a fossil-fueled power plant, cogeneration power plant, lumber mill, and a cement plant because they had steel framing, piping, and some equipment similar to that used in the nuclear power plant facilities. The effects of an earthquake on facilities of this type is perceived to be more severe than at a nuclear power plant because, in general, they do not have the stringent design requirements associated with nuclear facilities. The findings supplemented the existing experience data base and provided additional insights into the performance of nuclear power plant structures and equipment during an earthquake. Proposals have been made to use an experienced-based approach for the seismic qualification of selected equipment in ALWR designs.



**In August 1992, Hurricane Andrew struck the Turkey Point (Fla.) nuclear power plant site with sustained winds of 145 mph. During fiscal year 1993, a team combining personnel from the NRC and the industry's Institute of Nuclear Power Operations investigated the impact of the storm on operations at Turkey Point and issued its report in March 1993. Turkey Point is a two-unit pressurized water reactor facility in Dade County, Fla., about 25 miles south of Miami, on the western shore of Biscayne Bay.**

**Shear Wall Ultimate Drift Limits.** The ultimate drift limit is defined as the lateral displacement at the top of the wall relative to its base normalized by the height of the wall. When performing seismic probabilistic risk assessments (PRAs) and seismic margin assessments (SMAs), the ultimate drift limit is necessary to estimate the seismic capacity of concrete nuclear power plant structures. In many cases, loss of equipment function has been considered to occur when the ultimate drift limits are reached; hence, the ultimate drift limit is a failure parameter in these studies. Seismic PRAs and SMAs have been identified as acceptable methods for performing the seismic portion of the individual plant examination of external events (IPEEE) for severe accident vulnerabilities. A research program was started this fiscal year with the objectives of establishing appropriate values of ultimate drift limit and obtaining the statistics to define this parameter in a probabilistic sense. It is anticipated that the technical report will be published in the second quarter of fiscal year 1994.

**Seismic Analysis of Piping.** The ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I provides rules for the design of piping systems in nuclear power plants. In general, the design rules have been proved over the years to result in a design that affords reasonably certain protection of life and property and provides a margin for deterioration in service, so as to give a reasonably long safe period of usefulness.

Recent developments in the nuclear industry have resulted in proposed changes in the assumptions underlying design of piping systems for the ALWRs. Both the industry and the NRC staff have gained knowledge from piping tests conducted in the mid-1980's under NRC and industry sponsorship and from the data base obtained from actual seismic events at various facilities. The inherent ability of welded piping systems to withstand extremely large seismic inertia loadings is now recognized, and there is a clearer understanding of the likely failure modes of piping systems under earthquake loadings. Design criteria need to address actual seismic failure modes in piping and need to be revised to eliminate excessive conservatisms that do not add to safety and may hinder plant operation in the long term.

Therefore, the NRC initiated a program during the report period whose objectives are: (1) to assist the NRC staff in developing regulatory changes on the subject of seismic analysis of piping systems and perform supporting research activities as needed; and (2) to evaluate the cumulative impact of proposed changes on the overall safety margins of the piping systems.

This program will be completed in 1995, allowing the staff to develop its position on the piping design requirements.

**Cooperative International Seismic Programs.** The NRC's participation in international seismic test programs is beneficial both for the sharing of research resources and for gaining different perspectives on seismic design issues. The pooling of resources allows the development of larger-scale tests, an important element in the validation of methods for predicting the seismic response behavior of nuclear plant systems.

The Large-Scale Seismic Test (LSST) facility is one of the largest in the world for soil-structure interaction (SSI) research. The construction of a 1/4-scale model of a reinforced concrete containment—10.5 meters in diameter and 16.5 meters high (11.1 meters above the ground)—was completed in March 1993. All instrumentation was completed by April 1993, and a formal dedication ceremony was held in Hualien, Taiwan.

The LSST program was initiated in January 1990 and is expected to continue for five years. The goal of the program is to collect real earthquake-induced SSI data, in order to evaluate computer codes used in SSI analysis of nuclear power plant structures. In the program, observations will be made on the motions of the reactor building model and the surrounding ground during large-scale earthquakes. The expectation is that the test model will be shaken by numerous earthquakes in this seismically active area of Taiwan. Instrumentation located on the scale model and in the field along a three-dimensional strong ground motion array will record any observed data. The LSST program at Hualien, Taiwan, is a follow-on to the SSI experiments at Lotung, Taiwan.

EPRI has organized the Hualien LSST experiment and coordinated participation with the Taiwan Power Company (Taipower), the NRC, the Central Research Institute of Electric Power Industry (CRIEPI), the Tokyo Electric Power Company (TEPCO), the Commissariat a l'Energie Atomique (CEA), Electricite de France (EdF), Framatome, the Korea Power Engineering Co. (KOPEC), and Korea Electric Power Corporation.

During the report period, a collaborative effort involving exchange of technical information was established with the Ministry of International Trade and Industry (MITI) and Nuclear Power Engineering Corporation (NUPEC) of Japan. In this effort, NUPEC is carrying out a seismic proving test program for a main steamline typical of the PWR plants and a feedwater system typical of the BWR plants. These tests will be conducted at the shake table of Tadotsu Engineering Laboratory and will begin in late 1994 and continue in 1995. Tests will be conducted for several levels of seismic excitation and also using energy absorber supports for the piping systems. The NRC in this collaborative effort will carry out pre- and post-test analyses to assess the applicability of currently available analytical models. In addition, data will also be obtained from NUPEC for seismic proving tests of a computer system and a reactor shutdown cooling system.

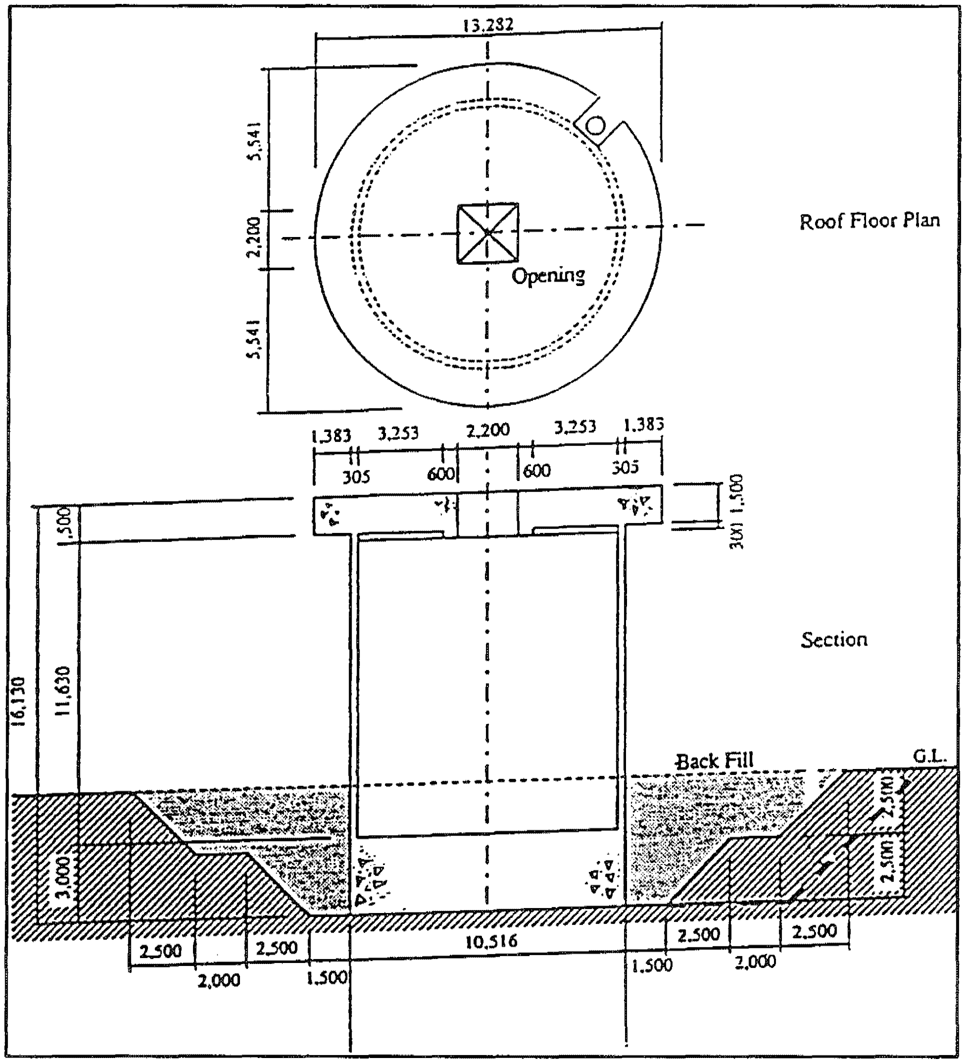
## Generic Safety Issue Resolution

In December 1983, the Commission approved a priority listing, prepared by the staff at the request of the Commission, of all generic safety issues (GSIs), including TMI-related issues, based on the potential safety significance and cost of implementation of each issue. Information and guidance on GSIs are reflected in the NRC's Five-Year Plan.

**Priorities of Generic Safety Issues.** The NRC has continued to employ the methodology set out in the *1982 NRC Annual Report* for determining the priority of GSIs. In December 1983, a comprehensive list of the issues was published in "A Prioritization of Generic Safety Issues" (NUREG–0933), and the list has been updated semi-annually with supplements in June and December. The list of issues includes TMI Action Plan (NUREG–0660) items. The results of the NRC's continuing effort to identify significant unresolved GSIs will be included in future supplements to NUREG–0933.

During fiscal year 1993, the NRC identified five new generic issues, established priorities for 12 issues (Table 1), and resolved 10 GSIs (Table 2). Table 3 contains the schedules for resolution of all unresolved GSIs.

**Resolution of Human Factor Generic Safety Issues.** Generic Issue HF4.4 on procedures other than emergency operating procedures was resolved while an effort continued to develop a report describing work that relates to the use of procedures for low-power and shutdown operations. Generic Issue HF5.1, on local control stations, was resolved and a lessons learned report on existing industry practices will be produced. Activity continued on the development of an advanced control room design review guide. A developmental, interactive version of the guides was demonstrated to the NRC staff. The final version is being developed for an application through Windows software. Generic Issue HF5.2 on annunciators was resolved and guidelines for the review of annunciators and alarm systems is being prepared for incorporation in the advanced control room design review.




The plans for the test model of the Large-Scale Seismic Test (LSST) facility are shown. The model, 10.5 meters by 16.5 meters high, was finished in March 1993 and installed in Taiwan. The expectation is that the model will be shaken by numerous earthquakes in the seismically active area of Taiwan and the effects recorded in detail. The international project was organized by the Electric Power Research Institute and involves, besides the NRC, governmental and industrial entities from Taiwan, Japan, France and Korea.

## Elimination of Requirements Marginal to Safety

The NRC has instituted a program to eliminate requirements that are marginal to safety. The basic objective is to avoid dilution of safety efforts by reducing resource application to marginal safety issues. This improvement in efficiency is expected to result in a net beneficial effect on safety.

As part of the program (57 FR 55156) to eliminate requirements that are marginal to safety and yet impose a regulatory burden that more directly enhances safety, the NRC conducted a public workshop on April 27 and 28, 1993, in Bethesda, Md. The purpose of the workshop was to provide information on the NRC program, solicit comments from the public and regulated industry on the program, and discuss a number of specific initiatives being considered. The NRC encouraged the public and the regulated industry to attend the workshop and provide input to the NRC in the early stages of the program. In order to facilitate discussions at the workshop, advanced material on a framework for a performance-based regulatory approach and applications to three specific rulemakings were published (58 FR 6196) prior to the workshop.

Over 320 people attended the two-day workshop, including representatives from 44 utilities, 5 industry groups, 8 vendors, 34 engineering and consulting firms,

## Table 1. Issues Prioritized in FY 1993

| *Number* | *Title* | *Priority* |
|---|---|---|
| 146 | Support Flexibility of Equipment and Components | RESOLVED |
| 149 | Adequacy of Fire Barriers | LOW |
| 152 | Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads | LOW |
| 155.3 | Improve Design Requirements for Nuclear Facilities | DROP |
| 156.3.6.2 | Emergency DC Power | LOW |
| 159 | Qualification of Safety-Related Pumps While Running on Minimum Flow | DROP |
| 160 | Spurious Actions of Instrumentation Upon Restoration of Power | DROP |
| 161 | Use of Non-Safety-Related Power Supplies in Safety-Related Circuits | DROP |
| 162 | Inadequate Technical Specifications for Shared Systems at Multi-plant Sites When One Unit Is Shut Down | DROP |
| 164 | Neutron Fluence in Reactor Vessel | DROP |
| 166 | Adequacy of Fatigue Life of Metal Components | NEARLY RESOLVED |
| 168 | Environmental Qualification of Electrical Equipment | NEARLY RESOLVED |

## Table 2. Generic Safety Issues Resolved in FY 1993

| *Number* | *Title* |
|---|---|
| 105 | Interfacing Systems LOCA at LWRs |
| 120 | On-Line Testability of Protection Systems |
| 142 | Leakage Through Electrical Isolators |
| 143 | Availability of Chilled Water Systems and Room Cooling |
| 153 | Loss of Essential Service Water in LWRs |
| B-56 | Diesel Reliability |
| HF4.4 | Guidelines for Upgrading Other Procedures |
| HF5.1 | Local Control Stations |
| HF5.2 | Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation |
| I.D.3 | Safety System Status Monitoring |

# Table 3. Generic Safety Issues Scheduled for Resolution

| *Issue Number* | *Title* | *Priority* | *Scheduled Resolution Date* |
|---|---|---|---|
| 15 | Radiation Effects on Reactor Vessel Supports | HIGH | 03/96 |
| 23 | Reactor Coolant Pump Seal Failures | HIGH | 12/94 |
| II.H.2 | Obtain Technical Data on the Conditions Inside the TMI–2 Containment Structure | HIGH | TBD |
| 24 | Automatic Emergency Core Cooling System Switch to Recirculation | MEDIUM | 08/94 |
| 57 | Effects of Fire Protection System Actuation on Safety-Related Equipment | MEDIUM | 12/93 |
| 78 | Monitoring of Fatigue Transient Limits for Reactor Coolant System | MEDIUM | TBD |
| 106 | Piping and Use of Highly Combustible Gases in Vital Areas | MEDIUM | 10/93 |
| B–17 | Criteria for Safety-Related Operator Actions | MEDIUM | 09/94 |
| B–55 | Improve Reliability of Target Rock Safety Relief Valves | MEDIUM | TBD |
| B–61 | Allowable ECCS Equipment Outage Periods | MEDIUM | 12/94 |
| 83 | Control Room Habitability | NEARLY RESOLVED | 12/93 |
| 145 | Improve Surveillance and Startup Testing Programs | NEARLY RESOLVED | 01/94 |
| 155.1 | More Realistic Source Term Assumptions | NEARLY RESOLVED | 01/94 |
| 166 | Adequacy of Fatigue Life of Metal Components | NEARLY RESOLVED | TBD |
| 168 | Environmental Qualification of Electrical Equipment | NEARLY RESOLVED | TBD |
| B–64 | Decommissioning of Nuclear Reactors | NEARLY RESOLVED | 10/93 |
| I.D.5(3) | On-Line Reactor Surveillance Systems | NEARLY RESOLVED | 10/93 |

4 public interest groups, and 6 State, Federal, and international government agencies. Representatives from an international union, law firms, and academia were also present. The discussions at the workshop have been documented in "Proceedings of the Workshop on Program for Elimination of Requirements Marginal to Safety," (NUREG/CP-0129) dated September 1993.

To implement the program, the NRC is currently taking action on requirements related to containment testing, fire protection, and quality assurance programs. The requirements in these areas will be modified to be less prescriptive and more performance-based to allow cost-effective implementation of regulatory safety objectives with marginal impact on safety. The NRC plans to use probabilistic risk analysis technology and its safety goals in reformulating requirements in these areas.

## Reactor Regulatory Standards

The Commission issued a final rulemaking on April 26, 1993 (58 FR 21904), 10 CFR Part 50, on training and qualification of nuclear power plant personnel. The final rule amends the Commission's regulations to require each applicant and holder of a license to operate a nuclear power plant to establish, implement and maintain programs for the training of nuclear power plant personnel that consider all modes of operation. The rule requires that the training programs be derived from a systems approach to training, as defined in 10 CFR Part 55. The objectives of the rule are to codify existing industry practices related to personnel training and qualification and to meet the directives contained in Section 306 of the Nuclear Waste Policy Act of 1982 (Public Law 97-425).

The Commission issued a final rulemaking on July 22, 1993 (58 FR 39130), 10 CFR 50.65, on monitoring the effectiveness of maintenance at nuclear power plants. The proposed rule was published on March 22, 1993 (58 FR 26938). The rule requires that the licensee conduct maintenance activities once every refueling cycle but not exceeding a period of 24 months. Because of the quality and quantity of data, this will provide a greater assurance that the nuclear power plant will operate safely.

The Commission issued a proposed rule on May 20, 1993 (58 FR 20336), 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses. The proposed amendment would delete the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's six-year license as a prerequisite for license renewal. Forty-two comments were received, the majority of which supported the proposed amendments. The final rule was expected to be published in December 1993.

**Regulatory Analysis.** The Commission issued the proposed regulatory analysis guidelines for public comment on September 7, 1993 (58 FR 47159). The proposed guidelines represent the NRC's policy-setting document with respect to regulatory impact analyses (RIAs). The document contains a number of policy decisions for the preparation of an RIA performed to support NRC actions affecting reactor and nonreactor licensees.

Along with the guidelines, the NRC issued a draft report, "Regulatory Analysis Technical Evaluation Handbook" (NUREG/BR-0184). The purpose of the handbook is to provide guidance to regulatory analysts, to promote preparation of highquality RIAs, and to implement the policies of the guidelines. The handbook expands upon the policy concepts included in the guidelines and translates the six steps in preparing an RIA into implementable methodologies for the analysts. The guidelines and handbook establish the guidance and structure of the existing operating procedures, the better to integrate backfit analysis requirements and safety goal policy considerations.

Also to aid NRC analysts in preparing RIAs, the NRC published "Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States: 1992-1996" (NUREG/CR-4012, Volume 3), which updates replacement energy costs associated with short term outages. These estimates can be useful in quantifying the overall impact of proposed regulatory actions when these requirements would necessitate retrofitting and short term outages at nuclear power reactors. The NRC will continue to develop these methodologies in an effort to facilitate NRC decision-making in evaluating the need and effectiveness of the regulatory actions.

During the report period, about 15 safety-related RIAs were completed or initiated to justify specific regulatory actions for reactor and non-reactor licensees.

**Maintenance Rule and Regulatory Guide.** The purpose of the maintenance rule is to require commercial nuclear power plant licensees to monitor the effectiveness of maintenance activities for safety-related and certain non-safety-related plant equipment, as defined in 10 CFR 50.65, in order to minimize the likelihood of failures and events caused by the lack of effective maintenance. The rule requires that licensees monitor the performance or condition of certain structures, systems, and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that those SSCs will be capable of performing their intended functions. Such monitoring would take into account industry-wide operating experience. Where monitoring proves unnecessary, licensees would be permitted the option of relying upon an appropriate preventive maintenance program.

The following chronology outlines the completion of the process to issue regulatory guidance to implement the maintenance rule.

In November 1992 the draft regulatory guide and regulatory analysis for endorsement of the industry guidance document NUMARC 93-01 for implementation of the maintenance rule was issued for public comment (FR 57 55286). Eleven responses to the request for public comments were received. By the end of January 1993, the NRC staff had reviewed and resolved all public comments.

In June 1993, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which endorses NUMARC 93-01, dated May 1993, was issued.

**Summary of Rulemaking Actions.** During fiscal year 1993, 94 rulemaking actions were processed, of which 22 rules were formally published, 10 were terminated/withdrawn, and 62 are ongoing (see Table 4). Besides the 62 ongoing rulemaking actions, there are 30 potential rulemaking actions, and it is estimated that in fiscal year 1994 there will be approximately 15-to-20 new rulemaking requests requiring RES review and approval by the Executive Director of Operations.

## Reactor Radiation Protection And Health Effects

The NRC maintains a program of research and standards development in radiation protection and health effects intended to ensure continued protection of workers and members of the public from radiation and radioactive materials in connection with reactor licensed activities. The program is currently focused on improvements in health physics measurements, identification and dissemination of cost-effective dose reduction techniques, assessing health effects consequences of postulated reactor accidents, and monitoring health effects research.

**Revision of Part 20 Radiation Protection Standards.** Staff efforts in support of the implementation of the new 10 CFR Part 20 rule continued in fiscal year 1993. These efforts included development of training courses, publication of questions and answers on Part 20, and publication of regulatory guidance. Also, several minor corrective rulemakings were completed.

Three new regulatory guides needed to implement the revised 10 CFR Part 20 were published. These guides are:

(1) Regulatory Guide 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," was published in July 1993. The guide describes practical methods acceptable to the NRC staff for estimating intake of radionuclides using bioassay measurement techniques.

(2) Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," was published in June 1993. The guide describes a framework of graded radiation protection procedures recommended to ensure that control for access to high and very high radiation areas are appropriate to the radiation hazard present in those areas.

(3) Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities," was published in July 1993. The guide provides guidance for materials licensees only and is addressed later in this chapter.

**Brookhaven National Laboratory ALARA Center.** The Brookhaven National Laboratory (BNL) ALARA Center, funded by the NRC, continued its surveillance and dissemination of DOE and industry dose reduction and ALARA research. BNL continued publication of the series that abstracts national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination (NUREG/CR-3469, Volume 7, July 1993). BNL also continued publication of the newsletter, "ALARA Notes," on about a quarterly schedule. In 1993, BNL focused on making the data base more easily accessible, adding information from overseas contacts, making final plans for an international conference on dose reduction, and continuing development of an ALARA handbook. The center provided information and advice on dose reduction to NRC staff and licensees.

**New Skin Dose Computer Code.** A revised computer code (VARSKIN II) for calculating dose to the skin from radioactive materials on the skin was published (NUREG/CR-5873, December 1992). The revised code is more flexible than earlier versions, allowing consideration of factors such as self-absorption, particle shape, and particles on clothing.

**Occupational Exposure Data Systems.** The NRC continued to collect and process data in the computerized data system called the Radiation Exposure Information Reporting System (REIRS). REIRS provides a permanent record of worker exposures for reactors and several other categories of licensees. A report on 1991 exposures, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities—1991" (NUREG-0713, Volume 13, July 1993), was issued. Compilation of the statistical reports indicated that approximately 200,000 individuals were monitored and half received a measurable dose. The average measurable dose dropped from 0.36 rem in 1990 to 0.31 in 1991. The collective dose obtained from summing all the individual doses dropped from the 1990 value by 20 percent to about 32,000 person-rems. The data base also includes exposure data on individuals who have terminated employment with certain licensees. Data on some 687,000 persons are in the

## Table 4. Rulemaking Actions Processed During FY 1993

| *Rulemaking Activities* | *Number* |
|---|---|
| Final Rulemakings Published | 22 |
| Rulemakings Terminated/Withdrawn | 10 |
| Ongoing Final Rulemaking Actions | 22 |
| Ongoing Proposed Rulemaking Actions | 37 |
| Rulemakings on Hold | 3 |
| Total Rulemakings | 94 |

system, most of whom worked in nuclear power plants. NRC continued to respond to requests for individual exposure data from the system. The data also assist in the examination of the doses incurred by transient workers as they move from plant to plant (about 2,900 in 1991).

**Water Chemistry and Decontamination.** Advanced Process Technology, funded by the NRC, determined the effects of hydrogen water chemistry on radiation buildup in BWRs and identified the most promising mitigating techniques.

The Idaho National Engineering Laboratory, funded by the NRC, obtained information on out-of-core PWR power plants that will be useful to NRR in evaluating system contamination (radionuclide surface concentrations and exposure rates) in advanced reactor designs.

OMNI Tech International, funded by the NRC, has developed an on-line UV–Vis spectrometer that will be used to determine the concentration of the vanadous ion during chemical decontamination of nuclear power plants.

The NRC published "Enhanced Removal of Radioactive Particles by Fluorocarbon Surfactant Solutions" (NUREG/CR-6081, August 1993). The report provides test results for the radiation stability and the application of environmentally compatible liquids to the nondestructive decontamination of nuclear equipment using ultrasonics.

**Performance Testing of Extremity Dosimeters—Pilot Test.** The NRC published "Performance Testing of Extremity Dosimeters—Pilot Test" (NUREG/CR-5989, July 1993). The report is the third of a series of tests run against the draft performance standard for personnel extremity dosimeters, ANSI N13.32, in order to establish the appropriateness of the standard for use in dosimeter processing certification. The NRC presently requires licensees to become accredited or to use dosimeter processors accredited under the National Voluntary Laboratory Accreditation Program (NVLAP) operated at the National Institute of Standards and Technology (NIST). At present NVLAP accredits whole body dosimeter processors and will add accreditation of extremity dosimeter processors as soon as the standard is jointly approved for use by the NRC and NIST.

**National Institute of Standards Technology.** Interagency Agreement, RES-93-01, between the NRC and NIST involves an ongoing study aimed at establishing traceability between NIST and Pacific Northwest Laboratories (PNL) for neutron irradiations. PNL provides the neutron irradiation to NIST/NVLAP as part of its duties as the testing laboratory for dosimeter processor accreditation run under the NVLAP.

**Electronic Personnel Dosimeters.** PNL is presently involved in developing a set of performance tests and implementing procedures that would permit electronic personnel dosimeters (EPDs) to be used in place of film or thermoluminescent dosimeters (TLDs) to establish radiation doses for radiation workers. The product of this effort is to be a report that could be used by the NRC to evaluate EPDs until such time as an appropriate ANSI standard for EPDs becomes available. The report would be used as the basis for a possible future certification program to qualify EPDs for use in radiation measurements.

**Gamma Dose Spectrometer.** Work is being carried out under a Small Business Innovative Research Phase II contract that involves the development of a gamma-ray dosimeter/spectrometer that will measure the gamma-ray spectrum over a wide range of energies. From this information and the electronic signal retrieved from the dosimeter, it will be possible to calculate, through the use of appropriate algorithms, the dose delivered to the skin, the eye and the whole body. To date, an Active Differential Absorption Spectrometer has been designed, developed, and tested.

**Spent Fuel Heat Removal.** The Oak Ridge National Laboratory, funded by the NRC, is also continuing to improve the data base in the guide for BWR and PWR fuel decay heat generation by including analysis of recent data to provide a basis for evaluating the adequacy of the storage system heat removal capability to limit fuel rod temperature.

**Radiation Exposure Monitoring and Information Transmittal (REMIT) System.** A new software package, REMIT, for electronically reporting radiation exposure measurements to the NRC was made available (58 FR 41526; August 4, 1993). REMIT is designed to assist NRC licensees in meeting the reporting requirements of 10 CFR 20.1001 through 20.2401, as outlined in Regulatory Guide 8.7, Revision 1, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." REMIT is a personal computer (PC)-based, menu-driven system. It includes provisions for various dose calculations and can produce NRC Forms 4 and 5 in paper and electronic format. In addition, REMIT can import and export data from ASCII and data base files.

# Nuclear Material And Low-level Waste Regulation Program

## NUCLEAR MATERIALS

### Nuclear Materials Research

**Materials Licensee Performance.** Through its human factors regulatory research program, the NRC seeks to improve its understanding and to maintain its requirements concerning the effect of human performance on the safety procedures involving the medical and industrial use of nuclear materials.

Function and task analyses of the systems involved in teletherapy and remote after-loading brachytherapy were completed as a first step in better understanding the root causes of human error associated with these systems. In-depth studies of procedures, training, human-system interface, and organizational policies and practices were also completed. Reports are being prepared on setting priorities of function and task performance problems related to human errors in terms of their safety significance and an evaluation of alternative approaches for resolving safety-significant problems. Specific work on the human factors evaluation of the industrial radiography system was discontinued and is being reconsidered in view of the rule change to 10 CFR Part 34 established in 1993.

### Materials Regulatory Standards

The Commission issued a proposed rule (10 CFR Parts 30, 40, 50, 70, and 72) on January 11, 1993 (58 FR 3515) that would allow self-guarantee as an additional mechanism for financial assurance. The proposed rule is in response to a petition for rulemaking (PRM-30-59) submitted by the General Electric Company and Westinghouse Electric Corporation. The rule would allow certain financially strong, non-electric utility licensees to use self-guarantee as financial assurance for decommissioning funding. It would not apply to electric utility licensees. The final rule is expected to be completed in fiscal year 1994.

A final rule (10 CFR Parts 31 and 32) is under preparation on requirements for the possession of industrial devices containing byproduct material. The rule would require licensees to provide the NRC with specific information about radioactive material used under a general license. The proposed action would improve the public health and safety by reducing the likelihood of unnecessary radiation exposures from radioactive materials by ensuring that generally licensed devices are accounted for and disposed of properly. It is expected that the final rulemaking will be completed in fiscal year 1994.

A final rule (Appendix H to 10 CFR Part 73) on day-firing (of firearms) qualifications for security personnel at Category I fuel cycle facilities was published on August 31, 1993 (58 FR 45781). The rule was needed to provide assurance that security force personnel maintain required weapon-handling and marksmanship skills by annual performance testing. The rule is applicable to the specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.

Final rules (10 CFR 72.214) adding casks VSC-24 and TN–24 to the list of approved spent fuel storage casks were published on April 7, 1993 (58 FR 17948) and on October 5, 1993 (58 FR 51762) respectively. The rules would increase the number of spent fuel storage casks from which the holders of power reactor operating licenses can choose to store spent fuel under a general license.

A final rule (10 CFR Parts 26, 70, and 73) on fitness for duty for category I facilities and shipments was published in June 1993 (58 FR 31467). The rule amends the regulations for the possession, use, or transport of strategic special nuclear material (SSNM). The action was necessary to ensure that specific employees of licensees who possess, use, or transport SSNM do not have a drug or alcohol problem. The rulemaking will become effective in December 1993.

A final rule (10 CFR Part 73) on physical protection requirements at fixed sites was published in March 1993 (58

CFR 13699). The rule clarifies the Commission's regulatory intent that protection against both radiological sabotage and theft of special nuclear material is not required at all facilities. The final rule also adds a requirement that protection be provided against radiological sabotage at non-power reactor licensees who operate at or above two megawatts thermal, where deemed necessary.

A proposed rule (10 CFR Part 73) that would require a physical fitness program for security personnel at category I facilities was published for public comment on October 6, 1993 (58 FR 52035). The rule would add new requirements for a physical fitness program and annual performance testing or a quarterly site-specific content-based performance test. It is expected that the final rulemaking will be completed in fiscal year 1994.

A proposed rule (10 CFR Part 72) on reporting events at Independent Spent Fuel Storage Installations (ISFSIs) and the Monitored Retrievable Storage (MRS) installation was published on September 14, 1993 (58 FR 48004). The rule would ensure that significant events such as contamination events, personal injuries, fires, and explosions at these facilities were promptly reported so that the Commission could evaluate whether the licensee has taken appropriate actions and whether prompt NRC action is necessary. The proposed rule would improve public health and safety by reducing the likelihood of unnecessary radiation exposures from these events. It is expected that the final rulemaking will be completed in fiscal year 1994.

A proposed rule (10 CFR Parts 30, 32, and 35) on the medical use of byproduct material was published in July 1993 (58 FR 33396). This action, taken in response to a petition for rulemaking (PRM-35-9), is intended to provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians greater discretion to prepare radioactive drugs containing byproduct material for medical use. The proposed rule would also allow research involving human subjects using byproduct material and the medical use of radio-labeled biologics. It is expected that the final rulemaking will be completed in fiscal year 1994.

A proposed rule (10 CFR Parts 30 and 35) extending the expiration date of the Interim Final Rule related to the preparation and therapeutic use of radiopharmaceuticals was published in May 1993 (58 FR 26938), and a final rule was subsequently published in July 1993 (58 FR 39130). This action allows licensees to continue to use byproduct material under the provisions of the Interim Final Rule until the NRC completes a related rulemaking to address broader issues for the medical use of byproduct material (including those issues addressed by the Interim Final Rule). It is expected that the broader rule will be issued as a final rule in fiscal year 1994. This extension of the expiration date was necessary to maintain the relief provided by the Interim Final Rule until the broader rule is issued.

A petition for rulemaking from the States of Washington and Oregon (PRM-60-4) was denied on March 4, 1993 (58 FR 12342). The petition requested the Commission to change the definition of high-level waste in its regulations so that some of the radioactive waste materials being processed at the DOE Hanford site could be classified as highlevel waste. The petition was denied because the existing NRC regulations on waste classification are well established and can be applied on a case-by-case basis without revising the definition of high-level waste.

The petitioner, Amersham Corporation, requested that petition PRM-35-8 (add iridium-192 wire for the Intestinal Treatment of Cancer) be withdrawn. The withdrawal notice was published on August 23, 1993 (58 FR 44466).

## Materials Radiation Protection And Health Effects

**Irradiator Rulemaking.** On February 9, 1993, the NRC published (58 FR 7715) a final rule on licenses and radiation safety requirements for irradiators. The rule established a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators. Irradiators use gamma radiation to irradiate products to change their characteristics in some way. The safety requirements apply to panoramic irradiators (those in which the material being irradiated is in air in a room that is accessible to personnel when the source is shielded) and underwater irradiators in which the source always remains shielded under water and the product is irradiated under water. The rule does not cover self-contained dry-sourcestorage irradiator devices, medical uses of sealed sources (such as teletherapy), or nondestructive testing (such as industrial radiography). The effective date of the rule was July 1, 1993.

The NRC is now in the process of publishing for comment a draft regulatory guide, "Guide for the Preparation of Applications for Licenses for Non-Self-Contained Irradiators," which is related to the irradiator rulemaking. The guide describes the information that an applicant should submit for a new or renewed license application. Issuance was scheduled for November 1993.

**Air Sampling.** In September 1993, the NRC published "Air Sampling in the Workplace" (NUREG-1400). The report provides technical information on air sampling that will be useful for facilities following the recommendations in the NRC's Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace." That guide addresses air sampling to meet the requirements in the NRC's regulations (10 CFR Part 20) on radiation protection. The report describes how to determine the need for air sampling based on the amount of material in process modified by the type of material, release potential, and confinement of the material. The purposes of air sampling and how the purposes affect the types of air sampling provided are discussed. The report discusses how to locate air samplers to accurately determine the concentrations of airborne radioactive materials to which workers will be exposed. The need for and the methods of performing airflow pattern studies to improve the accuracy of air sampling results are included. The report presents and gives examples of several techniques that can be used to evaluate whether the airborne concentrations of material are representative of the air inhaled by workers. Methods for adjusting derived air concentrations for particle size and methods for calibrating for volume of air sampled and estimating the uncertainty in the volume of air sampled are described. Statistical tests for determining minimum detectable concentrations are presented. How to perform an annual evaluation of the adequacy of the air sampling is also discussed.

In April 1993, the NRC published "DEPOSITION: Software to Calculate Particle Penetration through Aerosol Transport Systems" (NUREG/GR-0006). DEPOSITION is user friendly software to calculate particle losses in aerosol transport systems. Revision 1 to Regulatory Guide 8.25 states that use of the DEPOSITION software is an acceptable method to calculate particle loss in aerosol transport systems. The software was developed at Texas A & M University under an NRC grant. Research will continue there in fiscal year 1994 on the design of sampling probes to minimize loss of particles at inlets and on where to place sampling probes in ducts relative to bends and contractions in duct diameter, in order to obtain a representative sample.

**Solubility of Particles in the Lung.** In August 1993, a research contract was awarded to the Inhalation Toxicology Research Institute on "Methods for Determining the Solubility of Radioactive Materials in Order to Implement 10 CFR Part 20." The new 10 CFR Part 20 lists derived air concentrations (DACs) and annual limits on intakes (ALIs) for inhalation of radioactive materials according to their solubility in the lung. Compounds are classified as "D" (soluble), "W" (moderately insoluble), or "Y" (highly insoluble) based on their clearance half times. This roughly translates to the time needed to dissolve in lung fluid, i.e., days (clearance half-time less than 10 days) for D class compounds, weeks (clearance half-time between 10 and 100 days) for W class compounds, and years (clearance half-time greater than 100 days) for Y class compounds. The objective of the research is to identify methods that the NRC can accept for determining the solubility in the lung of radioactive materials that may be inhaled so that licensees can determine internal radiation doses to meet the requirements in 10 CFR Part 20.

**ALARA Levels for Effluents from Materials Facilities.** Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities," was issued in July 1993. Section 20.1302(b) of 10 CFR Part 20 requires licensees to demonstrate compliance with the annual dose limit for members of the public in 20.1301. In addition, 10 CFR 20.1101(b) requires that licensees use procedures and controls to achieve doses to members of the public that are as low as is reasonably achievable (ALARA). The document provides guidance for materials licensees, such as medical and academic institutions, on acceptable methods of demonstrating compliance with this new mandatory ALARA requirement.

**Patient Release Criteria.** A proposed rule to amend 10 CFR Parts 20 and 35 concerning the criteria for the release of patients administered radioactive material, as well as a regulatory guide and a comprehensive regulatory analysis to be published as a NUREG, was drafted and is expected to be published for comment in December 1993. The rulemaking action addresses the requests of two petitions for rulemaking: PRM-20-20 from Dr. Carol S. Marcus and PRM-35-10 from the American College of Nuclear Medicine (ACNM). The petitioners requested that the Commission adopt a dose limit of 0.5 rem for individuals exposed to patients who have been administered radioactive material. It is expected that the final rulemaking, as well as the regulatory guide and NUREG, will be completed in fiscal year 1994.

**Improvement of Health Effects Models.** "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis" (Revision 1 to NUREG/CR-4214) contains health effects models and risk coefficients intended for use in severe accident analyses, probabilistic risk assessments, emergency response planning, and safety goal and cost/benefit analyses. An addendum, "Modification of Models Resulting from Recent Reports of Health Effects of Ionizing Radiation," was published in 1991. The reports that led to modification of the models presented in NUREG/CR-4214 are the reports of the United States Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 1988), the National Academy of Sciences/National Research Council BEIR V Committee (NAS/NRC, 1990), and the revised recommendations of ICRP-60 (ICRP, 1991). A second addendum, "Modification of Models Resulting from Addition of Effects of Exposure to Alpha Emitters," was published in fiscal year 1993. Revision 2 of NUREG/CR-4214, Part 1, "Introduction, Integration and Summary," which incorporates the new information presented in the two addenda was also completed in fiscal year 1993 and will be published in fiscal year 1994. This project is complete.

**Embryo/Fetal Dose from Maternal Intake.** A study to improve understanding of the contribution of maternal radionuclide burdens to prenatal radiation exposure was continued in fiscal year 1993, with significant progress. Revision 1 to NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Dose," (March 1992) provides a method for calculating internal doses to the embryo/fetus with an expanded data base of radionuclides. Research that will permit inclusion of additional radionuclides, such as technetium, molybdenum, and certain transuranic elements began in fiscal year

1993 and will continue in fiscal year 1994. An addendum, "Relationships Between Annual Limits on Intake and Prenatal Doses" to NUREG/CR-5631 was prepared in fiscal year 1993 and will be published in fiscal year 1994. The methods and data developed under this project have been used by the NRC in preparing Regulatory Guide 8.36, "Radiation Dose to Embryo/Fetus," which describes acceptable methods of compliance with 20.1208 of 10 CFR Part 20. The guide might be revised to incorporate the information presented in the addendum. The methods developed under this project are also useful in calculations of doses in cases of accidental releases of radioactive materials.

**Criticality and Fuel Cycle Safety.** A draft regulatory guide for criticality safety was published for comment on January 25, 1993 (58 FR 6022). The draft guide was developed to provide guidance to licensees on an appropriate nuclear criticality safety training program for the use of special nuclear material, especially the prevention of criticality accidents.

The Los Alamos National Laboratory continued its examination and revision of TID-7016, "Nuclear Safety Guide," for simplification of use, evaluation against new experimental data, and use of current computational codes. The document is a standard guide and reference used by industry and the NMSS staff for initial criticality safety evaluations.

The Oak Ridge National Laboratory (ORNL), funded by the NRC, continued its methods validation of the criticality analytical sequences in SCALE-4 using ENDF/B-V cross-section data. The validation effort will qualify the applicability of SCALE-4 to criticality safety problems covering the range of interest within the Fuel Cycle Safety Branch of NRC/NMSS. The SCALE code system was developed at ORNL for criticality, shielding, and thermal analysis of nuclear facility and package designs. The system is currently used at ORNL in support of several tasks funded by NMSS. In particular, SCALE-4 is used by ORNL and the NRC staff for criticality safety analyses relevant to licensing issues. Valid criticality safety analyses require validation of both methods applied and the user who applies them. The goal of the project is to validate the Criticality Safety Analyses Sequences (CSAS) within the SCALE-4 system by analyzing a large number of benchmark critical experiments whose parameters (enrichment, geometry, fissile fuel/moderator ratio, etc.) cover the range of interest within the NMSS Fuel Cycle Safety Branch.

The availability of a draft regulatory guide for the fire protection of fuel facilities was published on April 22, 1993 (58 FR 21606). The regulatory guide was developed to provide guidance to applicants and licensees with respect to the information needed for the preparation of the fire protection sections (or chapters) of an application for a new license or for renewal of or amendments to an existing license for a fuel cycle facility. The guide also presents a standard format for submitting this information.

### Uranium Enrichment

The Commission is considering issuing a proposed rulemaking to amend 10 CFR Part 76, "Regulation Governing the Operation of Gaseous Diffusion Facilities." The rulemaking is required by the Energy Policy Act of 1992 and will establish both the procedural and technical requirements for certification of the operation of the gaseous diffusion facilities by U.S. Enrichment Corporation. It is expected that a proposed rulemaking will be published in fiscal year 1994.

## LOW-LEVEL WASTE

NRC research in support of licensing activities for low-level radioactive waste (LLW) disposal facilities centers on (1) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial for LLW disposal and (2) evaluation of the overall performance of disposal systems. The NRC LLW research program is described in NUREG-1380, published in 1989. That document identifies issues and regulatory needs with a strategy and a schedule for resolving them. NRC- funded LLW research is useful not only for the NRC licensing staff but also for the States regulating LLW disposal (see Chapter 7). In order to make their research results available to the States, NRC research contractors, besides publishing their work, gave presentations at meetings well attended by State representatives—such as "Waste Management '93" and the Annual DOE LLW Management Conference. In addition, the NRC and the U.S. Geological Survey conducted a three-day meeting at Reston, Va., with State participation, on hydrogeology and geochemistry research addressing LLW concerns.

### Materials and Engineering

**Engineered Enhancements and Alternatives to Shallow Land Burial.** Many States and State compacts are considering alternatives to shallow land burial for the disposal of LLW. Several concepts have been proposed as alternatives, and, of these, the most popular is the use of concrete as the principal construction material for engineered barriers to contain LLW. The National Institute of Standards and Technology (NIST) has continued investigating, on behalf of the NRC, the durability of concrete as an engineered alternative to shallow land burial, while the Idaho National Engineering Laboratory (INEL) completed an evaluation of concrete barriers in limiting radionuclide transport (NUREG/CR-6070). The NIST studies include concrete sulfate resistance research, the determination of diffusion coefficients for sulfate and chloride ions, the modeling of stresses caused by sulfate attack in concrete, investigation of cracking in concrete, and the durability of super-plasticizers that may be used in concrete to reduce its transport properties and improve its strength. Three reports are being prepared as NUREG documents that will be available in fiscal year 1994—(1) a new method to determine chloride diffusion coefficients, (2) the determination of sulfate diffusion coefficients, and (3) the modeling of stresses caused by sulfate attack in concrete. A new effort was started at NIST to develop computer models on the degradation of concrete. It is expected that a peer review panel report assessing these models and their application to LLW performance assessment will be issued in the first quarter of fiscal year 1994.

**Anion Retention in Soil.** The University of California at Berkeley investigated the use of natural soil materials to retard migration of anions at radioactive waste disposal facilities. Most soils are much more effective in retarding the migration of cations than anions. Certain long-lived radionuclides, such as I-129 and Tc-99, may be in an anionic form at LLW disposal facilities. The anticipated application of this work is to identify materials that could be used to condition the near field either in or around LLW disposal facilities to retard migration and attenuate activities of radionuclides in anionic form. A literature review (NUREG/CR-5464) indicated that a group of soils called andisols, which are derived from the weathering from volcanic parent material, have significant potential for retardation of anionic forms of I-129 and Tc-99. Field work is under way in the western United States to determine if exploitable deposits of andisols with anion exchange capacity are available. Preliminary results were published (NUREG/CR-5974) and work is continuing.

**LLW Waste Forms.** The stability of decontamination waste obtained from nuclear reactors using commercial decontamination processes and solidified in cement is being studied. Decontaminated LLW (collected from the Peach Bottom (Pa.) nuclear power plant) is being tested at INEL. The studies are aimed at ensuring that radionuclide and chelating agent leaching characteristics, as well as the compressive strength of the cement solidified waste, are consistent with NRC technical positions and the requirements of 10 CFR Part 61 for waste form stability. Field lysimeter studies containing radioactive ion-exchange resins solidified in cement and vinyl ester-styrene are being conducted at the Oak Ridge and Argonne National Laboratories to determine radionuclide release rates under certain environmental conditions. Studies continued at INEL to investigate biodegradation of LLW by micro-organisms to ensure stability requirements, as required by 10 CFR Part 61. Studies continued at Pacific Northwest Laboratories (PNL) to investigate activated metal and radioactive waste streams for radionuclides not included in the listing of long-lived radionuclides in 10 CFR Part 61, to determine scaling factors for assessing hard-to-measure radionuclides in LLW, and to obtain activated metals from operating reactors for leaching and field lysimeter research studies. Studies were started at PNL to determine the presence of radionuclide-chelating complexes in leachates and behavior in soils.

**Infiltration of Water.** The University of California at Berkeley, in cooperation with the University of Maryland, is continuing to field test a variety of covers for LLW disposal units at the Maryland Agricultural Experiment Station in Beltsville, Md. (Results are reported in NUREG/CR-4918, Volume 6.) Two designs are proving to be particularly effective. One, called bioengineering water management, not only reduced water infiltration to a negligible amount but also dewatered two experimental cells. Since this is a surface cover, it lends itself to use as a remedial action cover for sites susceptible to subsidence. The New York State Energy Research and Development Administration finished construction in 1993 of a bioengineering water management cover over such a trench at the West Valley LLW disposal facility, and the monitoring of performance has just begun. A second cover consists of a resistant layer barrier (compacted clay) over a conductive layer barrier. This second system has functioned perfectly since its completion in January 1990. However, its long term performance needs to be assessed.

PNL has developed an "infiltration evaluation methodology" (NUREG/CR-5523) that is being tested on the Las Cruces Trench field data set (NUREG/CR-5998) in the international INTRAVAL project. The incorporation of various one- and twodimensional analyses has also been applied and tested for conducting infiltration analyses for performance assessment and engineering design analysis (NUREG/CR-6114).

## Hydrology and Geochemistry

**Radionuclide Migration in Soil.** A significant area of uncertainty in predicting site performance is the degree to which soils can retard radionuclide migration. To reduce this uncertainty, the NRC is investigating mechanisms controlling radionuclide movement through soils. The Sandia National Laboratories (SNL) are working on characterizing retardation mechanisms. The University of California at Davis is investigating the mechanisms and rates of dissolution for a variety of silicate minerals. This will be useful for understanding and modeling processes occurring on mineral surfaces that affect both sorption and leaching. PNL is examining the role in radionuclide transport played by microparticulates and naturally produced organic complexants.

**Hydrology and Contaminant Transport.** PNL has evaluated and developed a data set from an earlier field study involving subsurface injection of radioactive tracers in heterogeneous unsaturated porous media at the Hanford site. The data sets reported in NUREG/CR-5996 cover a period of 10 years and will allow confirmatory analyses of existing flow and transport models that are to be used in LLW performance assessment. Work is being completed by the Massachusetts Institute of Technology (MIT) and Princeton University on the application of stochastic methods for simulating flow and transport in heterogeneous soils. In particular, Princeton University has completed and is now testing ground-water ventilation models for simulating vapor phase transport associated with LLW facilities, and MIT is applying their stochastic approach to field data sets.

## Compliance, Assessment, and Modeling

**Performance Assessment.** Research is continuing on a performance assessment methodology. Emphasis is being given to engineering enhancements to shallow land burial, specifically the incorporation of concrete degradation models and the performance of cover materials. SNL has evaluated the current status of models used in performance assessment and published the results (NUREG/CR-5927, Volume 1). SNL is currently making improvements to the performance assessment methodology and assessing the validation approaches for the performance assessment models. MIT has been investigating the use of stochastic methods for dealing with large-scale nonuniformity of site hydrologic characteristics. The University of Arizona and New Mexico State University are working cooperatively with MIT by providing a field test at Las Cruces, N.M., of MIT's theoretical work.

**LLW Source Term Modeling.** During fiscal year 1993, the existing LLW source term code, BLT (breach, leach and transport), was benchmarked against field data and verified. Extensions to incorporate additional geochemistry and gaseous release are currently being investigated and planned for inclusion in the code during the next fiscal year.

**Modeling of Tritium Migration at Arid Sites.** The University of California at Berkeley, working cooperatively with CSIRO, developed a three-dimensional deterministic model based on soil physics to predict tritium migration at arid disposal sites (published as NUREG/CR-5980). The model awaits confirmation through the use of a controlled release from a known source at an arid site. Planning for such an experiment is under way.

## Low-Level Waste Regulatory Standards

A proposed rule to amend 10 CFR Parts 20 and 61 to revise low-level waste shipment manifest information and reporting was published in April 1992. The rule would improve the quality and uniformity of disposal information from low-level radioactive waste by requiring the use of standardized NRC forms when the waste is shipped. The forms would incorporate Department of Transportation regulations that are currently being made final. It is expected that the final rule will be published in 1994.

A final rule to amend 10 CFR Part 61 to clarify that requirements related to the performance of land disposal facilities for LLW are applicable to aboveground disposal (i.e., built on the ground without an earthen cover) was published in June 1993.

A petition for rulemaking (PRM-61-2) from the New England Coalition on Nuclear Pollution was published in the *Federal Register* on July 23, 1992. The petitioner requests that the Commission amend its regulations regarding waste classification of lowlevel radioactive waste to restrict the number and types of waste streams that can be disposed of in near-surface disposal facilities. Recommendations on the need for rulemaking were expected to be developed by December 1993.

## Environmental Policy and Decommissioning

**Timeliness.** A proposed rulemaking (10 CFR Parts 30, 40, 70, and 72) on timeliness in decommissioning a materials facility was published on January 13, 1993 (58 FR 4099). The proposed rule would amend the Commission's regulations to establish timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities. The principal effect of these amendments is to formalize and codify the NRC's requirements for timeliness in decommissioning of materials facilities. Seventeen comment letters were received and the final rulemaking is expected to be completed in fiscal year 1994.

**Documentation.** A final rule (10 CFR Parts 30, 40, 70, and 72) was published on July 26, 1993 (58 FR 39628). The rule amends the NRC's decommissioning regulations to require holders of a specific license for possession of byproduct material, source material, special nuclear material, and independent storage of spent nuclear fuel and high-level waste to prepare and maintain additional documentation identifying areas where licensed materials and equipment were stored and used. The Commission's intent is to provide both the NRC and the licensee the necessary information to ensure complete decommissioning of licensed facilities. This action is consistent with similar requests made at the Congressional Committee hearing on decommissioning and an earlier GAO report.

**Contamination Monitor.** In September 1993, the NRC awarded a Small Business Innovative Research Phase II contract to Shonka Research Associates for a "Contamination Monitor Using Visual Identification." The monitor, using a positionsensitive proportional detector to detect contamination, is mounted on wheels and can be pushed over the floor like a lawn mower. The output from the detector goes to a computer, which creates a visual display in goggles worn by the operator. As the operator pushes the monitor over the floor, a "virtual reality" display of contamination can be seen on the floor. The monitor is faster and less expensive than conventional contamination detectors.

**Waste Oil Incineration.** A final rule (10 CFR Part 20) on disposal of waste oil by incineration at nuclear power plants was published on December 7, 1992 (57 FR 57649). The rulemaking action, responding to a petition for rulemaking originally filed by Edison Electric Institute and the Utility Nuclear Waste Management Group (PRM-20-15), allows reactor licensees to pursue the option of incineration of waste oils contaminated with small amounts of radioactivity without the need for specific authorization.

**Updating 10 CFR Part 40.** An advance notice of proposed rulemaking (10 CFR Part 40) concerning updating of requirements for licensing of source material was published on October 28, 1992 (57 FR 48749). The contemplated rulemaking would improve the control of source material through more specific regulation and update applicable requirements to conform with the revised standards for protection against radiation. A broad range of issues are being considered in the categories of exemptions, general licenses, specific licenses, and mills and mill tailings. One specific issue identified in the advance notice is being handled in a separate rulemaking. A proposed rule conforming the NRC's regulations governing uranium mill tailings to a proposed EPA rule of June 8, 1993, was prepared for Commission consideration and is expected to be published in fiscal year 1994. The rule would address the timeliness of completion of radon barriers on uranium mill tailings and verification of the effectiveness of those barriers.

**Enhanced Participatory Rulemaking.** An Enhanced Participatory Rulemaking on radiological criteria for the decommissioning process was initiated to actively solicit early input from a wide spectrum of interests. Seven public workshops were held across the United States (Chicago, San Francisco, Boston, Dallas, Philadelphia, Atlanta, and Washington, D.C.), and these were followed by eight generic environmental impact statement (GEIS) scoping meetings in four cities (Washington, D.C., San Francisco, Oklahoma City, and Cleveland). The EPA participated in these workshops and meetings and is a cooperating agency in the development of the GEIS. The staff began evaluating the input received and worked on preparing a staff draft of decommissioning criteria. The staff plans to complete the draft in fiscal year 1994 and continue seeking early input by releasing it to Agreement States and other interested parties in January 1994. The proposed rule is scheduled for publication in fiscal year 1994.

The staff will continue to build experience with the use of the methodology described in the draft "Manual for Conducting Radiological Surveys in Support of License Termination" (NUREG/CR-5849), in the context of the present and proposed criteria for unrestricted release.

The NRC issued "Residual Radioactive Contamination from Decommissioning: Technical Basis For Translating Contamination Levels to Annual Total Effective Dose Equivalent" (NUREG/CR-5512, Volume 1) in October 1992. The complete report will consist of three volumes and one supplement. This first volume is designed to provide screening models, mathematical formulations for the screening models, and referenced parameter values for estimating doses, above natural background, to individuals from residual radioactivity associated with lands and structures after decommissioning licensed facilities. The modeling structure permits the use of either generic or site-specific parameters to be used as screening estimates of radiation doses from multiple environmental pathways. It is expected that the software to implement the models, D&DSCREEN, will be released for testing late in fiscal year 1994 and will be accompanied by the user manual, NUREG/CR-5512, Volume 2.

**Safety Issues Related to Permanently Shutdown Reactors.** Brookhaven National Laboratory continued its determination of technical and safety criteria that should remain as part of decommissioning regulations under 10 CFR Part 50 when a licensee initiates action to permanently shut down the nuclear reactor in preparation for decommissioning activities. This project will develop a comparison of the safety requirements for a shutdown versus an operating nuclear power reactor after the reactor has permanently shut down. It will also perform financial assurance analysis for off-site liability requirements for shutdown reactors. It will examine the environmental impact of the potential increase in the spent fuel transport and radiological exposure to the public in the event the licensees prefer to ship and store their spent fuel. A draft report for public comment is expected in fiscal year 1994.

The Pacific Northwest Laboratories (PNL) continued providing support for nuclear facility decommissioning issues. An update of waste burial costs, "Report on Waste Burial Charges." was published in May 1993 as Revision 3 to NUREG-1307. "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station" (NUREG/CR-5884) was published for public comment in September 1993. The report re-evaluated decommissioning costs for an earlier decommissioning study of a reference PWR reactor. "Estimating Pressurized Water Reactor Decommissioning Costs" (NUREG/CR-6054) was issued in October 1993. The report describes the computer program used for estimating reactor decommissioning costs in the NUREG/CR-5884 re-evaluation study.

PNL continued developing an information base on the actual radioactive contamination expected to be encountered at LWRs at the time of decommissioning, using actual field sampling and theoretical analysis. PNL is also updating and extending an information base on the technology, safety, and costs for decommissioning fuel cycle and non-fuel cycle nuclear facilities, using actual decommissioning data and analysis.

Responding to the Energy Policy Act of 1992, the Commission's Below Regulatory Concern Policy Statement was withdrawn. The withdrawal notice was published on August 24, 1993 (58 FR 44620).

The petitioner, the University of Utah, requested that its petition, PRM-20-14 (disposal of biomedical waste containing small amounts of radioactivity) be withdrawn. The withdrawal notice was published on July 22, 1993 (58 FR 39173).

# Assessing The Safety Of High-Level Waste Disposal

## HIGH-LEVEL WASTE RESEARCH

The NRC maintains active research programs in rock mechanics and engineering, hydrology, geology, waste package performance, materials science, geochemistry, and several other disciplines to develop technical bases and expertise for its independent high-level waste (HLW) licensing decisions concerning the proposed repository in unsaturated volcanic tuff at the Yucca Mountain (Nev.) site, currently under consideration by the DOE as directed by the Congress in December 1987. The research combines theoretical study with laboratory and field experiments to provide an integrated approach to improve the NRC's capability of assessing the long term safety and compliance with regulatory standards of the proposed repository. Key technical issues being addressed include long term performance of waste package materials, unsaturated flow and transport mechanisms, assessment of the likelihood and potential consequences of volcanic and seismic events, geochemical processes, and the long term performance of engineered waste isolation systems.

### Engineered Systems Research

**Stability of Underground Opening.** When specifying suitable site conditions for an HLW repository, 10 CFR Part 60 specifically requires consideration of natural phenomena and site conditions that could adversely affect achievement of the prescribed performance objectives. An important phenomenon that could affect both the short and long term performance of a repository is ground motion resulting from seismic activity or motion caused by underground nuclear explosions at the Nevada Test Site. Ground motion from either source could cause rock displacement and pressure changes in ground-water levels that could violate repository performance objectives.

To investigate the effects of seismicity on the underground openings for an HLW repository, the NRC is sponsoring research at the Center for Nuclear Waste Regulatory Analyses (CNWRA). The research includes the laboratory characterization of jointed fractured rock expected at the HLW repository horizon, the assessment of computer codes to calculate rock response to earthquakes, and field studies at the Lucky Friday Mine, Idaho, to measure rock displacements and ground-water response to mine seismic events. Results from the study indicate that underground openings at high states of stress are more sensitive to seismic loads than previously thought and that repetitive earthquake loading causes the displacement of rock joints, making the rock mass less stable. Seismic events of even small magnitudes cause changes in ground-water pressures as a result of volume changes in the rock. Rock/ground-water response to seismic events greater than those at the Lucky Friday Mine will be studied at the Garner Valley Site in California. The California site can be subject to seismicity up to magnitude-6.5.

**Thermohydrological-Mechanical Coupled Interactions.** One important component of the safety analyses for HLW disposal is the coupling of the interactions between the rock mass, the ground water, and the thermal stresses induced by the high-temperature wastes. Coupling of the processes implies that one process affects the initiation and progress of the other, and independent consideration of each process is bound to be flawed. The NRC is a participant in an international multi-disciplinary and cooperative research effort to study the coupled thermo-hydrological-mechanical (THM) processes under the acronym DECOVALEX (DEvelopment of COupled Models and their VALidation against EXperiments). The objectives of the study are to increase the basic understanding of THM coupled processes, support the application of codes for THM modeling for jointed hard rocks, and design validation experiments by means of THM model studies. In fiscal year 1993, two benchmark problems simulating near-field conditions in an HLW repository were modeled, analyzed, and compared to results obtained by researchers from various countries using different computer codes. There was good agreement in the results. It is expected that the DECOVALEX study will be published in the International Society of Rock Mechanics Journal.

**Materials Science.** An understanding of the materials science aspects of the engineered barriers in HLW disposal systems is necessary to permit the NRC to judge whether test data and models offer reasonable assurance of compliance with regulatory requirements. During 1993, the CNWRA conducted research on localized corrosion rates for candidate HLW package materials in tuff ground waters and continued to evaluate potential regulatory problems arising from stress corrosion cracking and metallurgical instability. Research was also initiated on the effect of water refluxing (resulting from thermal gradients) on degradation of waste package materials.

The CNWRA continues investigations of contaminant transport and material corrosion on ancient Minoan copper, bronze, and lead artifacts that were buried under silicic tuff 3,600 years ago. New metallic artifacts were uncovered, and the tuff immediately adjacent to the artifact was sampled to determine the extent of elemental transport in the unsaturated tuff. Hydrologic tests were conducted at the site in an attempt to model the hydrologic conditions to which the metals have been exposed. This work is providing data related to ongoing modeling research in copper alloy corrosion on unsaturated environments.

## Geologic Systems Research

**Hydrogeology.** Since ground water is considered to be the primary agent of radionuclide transport from an HLW facility to the accessible environment, the NRC is actively studying ground-water infiltration, recharge, flow, and transport processes in partially saturated fractured rock. An experimental site in unsaturated fractured tuff, similar to Yucca Mountain, called the Apache Leap Tuff site, has been instrumented and characterized for testing instrumentation, methods, and analyses similar to those being used or proposed by DOE. In particular, field data sets are being collected and interpreted for testing mathematical models of flow and transport in partially saturated fractured rock. This work has been incorporated into the INTRAVAL project for model validation of ground-water transport models using field experiments. Key technical uncertainties dealing with determination and confirmation of ground-water travel times, presence and influence of perched-water systems, and preferential flow from persistent discontinuities are being addressed.

Scientists at the CNWRA in San Antonio, Tex., are testing approaches to largescale unsaturated flow in heterogeneous, stratified, and fractured geologic media. The BIGFLOW code has been developed, tested, and documented for use in their stochastic methodology (NUREG/CR-6028) for simulating flow in variably saturated, heterogeneous geologic media. To quantitatively account for spatial heterogeneity, CNWRA has applied the real space renormalization group method for parameter estimation. Work has begun on an analysis of the regional hydrogeologic processes in the vicinity of the Yucca Mountain focusing on the appropriate integration of hydrogeologic, geophysical, and geochemical information and methods for testing alternative mathematical models of the regional flow and transport system.

The validity of conceptual and numerical models used to describe ground-water flow and radionuclide transport for various hydrogeologic settings is being evaluated in the INTRAVAL project. The NRC staff and research contractors from CNWRA, the University of Arizona, Sandia National Laboratories, Massachusetts Institute of Technology, Princeton University, and the Pacific Northwest Laboratories are participating in this international effort involving 13 scientific parties from 10 countries.

Cooperative experiments and data analyses being done under a cooperative agreement between NAGRA (Switzerland) and the NRC, negotiated during fiscal year 1987, continue to augment the field-testing program cited above.

Research was completed on a grant at Johns Hopkins University to investigate numerical coupled thermohydrogeochemical modeling of large-scale transport processes that led to the formation of unconformity-type uranium ore bodies such as those found in the Alligator Rivers region of Australia and the Athabaska Basin in Canada. The model was able to simulate the detailed thermal, hydrological, hydrogeochemical, and mineralogical conditions and processes and the detailed final shape, structure, and mineralogy of the Cigar Lake, Canada, uranium ore body. This research model has the potential for providing a basis for major improvements in radionuclide transport modeling by performance assessment models.

**Geochemistry.** Geochemistry is involved in the assessment of all of the HLW performance objectives and many of the siting and design criteria. Because the chemistry of the proposed site is dominated by the chemical effects of an enormous mass of rock compared to a relatively small mass of waste and engineered components, knowledge of geochemical effects on the behavior of the repository system and its components is essential. Over the long term, the performance of engineered components will depend on their compatibility with the geochemistry of the site. Moreover, there are geochemical clues to the past history of the site, including former site temperatures, the histories and ages of ground-water samples, and the ages of prehistoric geologic events such as earthquakes and volcanic eruptions. The NRC has a geochemistry research program actively investigating key technical uncertainties related to HLW disposal safety.

In 1993 geochemistry research focused on improvement of the tools used for the analysis of HLW geochemical processes. Assessments of thermodynamic data for key minerals (zeolites) that strongly affect the ground-water chemistry of Yucca Mountain were completed by the CNWRA. Precise thermodynamic data and modeling showed that dissolution of the mineral analcime controls Yucca Mountain water chemistry. Research on the exchange of ions in solution with accessible sites in zeolite minerals continued to show the success of an ion-exchange model applied by the CNWRA to model interaction between zeolites and a number of dissolved fission product radionuclides. CNWRA research showed that the behavior of dissolved actinide elements could not be characterized well by simple linear models and research continues on improving assessment methods for these important radionuclides. An extensive evaluation of more complex adsorption models was completed and experiments were initiated to measure key sorption data.

From 1988-to-1993, the NRC was one of five countries participating in the International Alligator Rivers Analog Project (ARAP). This project investigated the Koongarra uranium ore body in Australia. This ore body is a research site relevant to nuclear waste disposal because it has been subjected to dissolution and transport over the past one million years by oxidizing ground water flowing through rock fractures. A zone of dispersed uranium and other radionuclides has formed along the ground-water flow path. Processes controlling the long term release and transport of uranium and other radionuclides can be intensively investigated. A two-day final project presentation was given in October 1993 to the OECD Natural Analogues Working Group in Toledo, Spain. Mineralogical and hydrogeochemical analysis, combined with numerical geochemical equilibrium and kinetic modeling, showed that many of the same minerals and geochemical processes as those found at Koongarra will likely control radionuclide release and transport at Yucca Mountain. The project showed that integration of multidisciplinary geophysical, geological, geochemical, and hydrological data is necessary to describe a complex site in which ground-water flow occurs by both matrix and fracture flow. Numerical geochemical modeling powerfully described and predicted long term evolution of the Koongarra site, but the need for better geochemical thermodynamic data for uranium silicate minerals (which will also tend to form at Yucca Mountain) was identified. Key uncertainties in making long term predictions at Koongarra are characterizing long term climatic and hydrogeologic conditions. Performance assessment models were the subject of considerable investigation and discussion. Some investigators concluded that available performance assessment models were more sensitive than the Koongarra site to hydrological processes and less sensitive to geochemical processes. It is clear that these models would benefit from further research and development.

The NRC is sponsoring work by the CNWRA to investigate contaminant transport in an unsaturated tuff at the Nopal I site in Pena Blanca uranium district, Chihuahua, Mexico. The site is a tuff-hosted brecciated uranium ore body, which is analogous in many respects to the proposed repository at Yucca Mountain. This site is being studied to better understand the nature of contaminant transport in a fractured, unsaturated tuff (i.e., the relative roles and interaction of the matrix and fractures in transport and the alteration of uraninite) and in an oxidizing environment. Detailed geologic, fracture, and gamma spectroscopy maps have been completed on the cleared, exposed surface of the ore body. Transport of uranium tends to be concentrated along iron-stained fractures supporting in general the findings from the ARAP project on the association of uranium sorption with iron hydroxide minerals. Migration is generally thought to be fracture controlled; however, samples collected both along fractures and in the tuff perpendicular to the fractures indicate the relative mobility of uranium in the fractures versus the matrix. Uranium series disequilibrium studies are being conducted to determine the extent and nature of uranium mobility. The site also is being studied as an analogue of spent fuel corrosion. Detailed mineralogic and petrologic studies of the primary uraninite indicate a reasonable approximation of an oxidized spent fuel, but there are distinct differences in trace element concentration and grain size compared to unoxidized irradiated spent fuel.

**Geology.** The NRC has an ongoing project in volcanism in the Basin and Range to evaluate the potential for disruption of the repository by igneous activity. This work at the CNWRA focuses on determining the extent and availability of volcanic, tectonic, and geophysical data from the region surrounding Yucca Mountain. The NRC began a project on Central Basin and Range tectonics at the CNWRA. An extensive literature survey was completed for both volcanism and tectonics and a computerized digital geologic data base for the Yucca Mountain region has been successfully initiated. A review of age determination techniques for young basaltic volcanic rocks was also completed this year.

The NRC began an additional volcanology project at the CNWRA to examine the disruptive scenarios of small-volume basaltic volcanism in the Basin and Range. Three active volcanic sites have been identified for analogue studies of eruption dynamics, extent of disruption to the hydrogeology, and host rock mechanical stability based on volcanic events. Ancient cinder cone fields have also been identified as areas for the study of magma emplacement dynamics and comparison with the data collected from active sites.

Research on contemporaneous deformation rates in the Death Valley region, using global positioning satellite interferometry, was started in fiscal year 1993. The investigation will provide the NRC with an understanding of the regional geologic forces that will affect volcanism, seismicity, and faulting in the Yucca Mountain area and will provide a basis for informed NRC assessment of the DOE's site characterization data concerning geologic stability. At present the possibility that measurable deformation was occurring prior to and following the Little Scull

Mountain earthquake near Yucca Mountain is being investigated.

## Performance Assessment Research

The NRC will assess the claims of compliance made by the HLW licensee, the DOE, with the NRC's quantitative requirements for HLW disposal given in 10 CFR Part 60. Included (by reference in 10 CFR 60.112) in these requirements is the overall HLW repository performance standard, 40 CFR Part 191, set by the Environmental Protection Agency. The development of a methodology to quantitatively evaluate repository performance and the evaluation of the conceptual models used in the methodology are necessary to assess compliance.

The NRC is sponsoring research at the CNWRA to evaluate conceptual models used in the performance assessment of a repository in unsaturated, fractured tuff. In fiscal year 1993, the CNWRA investigators completed development of a computer program (PORFLOW, NUREG/CR-5991) that is capable of simulating two-phase fluid flow conditions and radionuclide transport in geologic media. The program will be used to examine the representation of ground-water flow conditions during the thermal phase of the repository and simplifications used to represent the source term or releases from the engineered barrier system.

# Proceedings And Litigation

Chapter 10

This chapter covers significant activities, proceedings and decisions of the NRC's Atomic Safety and Licensing Boards (ASLBPs), as well as noteworthy decisions of the Commission in its appellate review of ASLBP decisions. The chapter includes a judicial survey of important litigation involving the NRC during the fiscal year.

**Office of the Secretary.** The Secretary of the Commission manages the official NRC adjudicatory and rulemaking dockets for the Commission. The adjudicatory dockets contain the filings of all parties to the Commission's licensing and enforcement proceedings; transcripts of the adjudicatory hearings held in each case; and all Orders and Decisions issued by the Commission, or the Commission's Atomic Safety and Licensing Boards. The rulemaking dockets contain the comments of members of the public on newly proposed agency rules and rule amendments, as well as comments on specific petitions for rulemaking and NRC/State Agreements on which the NRC seeks views before taking final action.

The Docketing and Service Branch also serves Orders of the Commission and the Atomic Safety and Licensing Boards on parties to proceedings and certifies indexes of the dockets to the courts.

## ATOMIC SAFETY AND LICENSING BOARDS

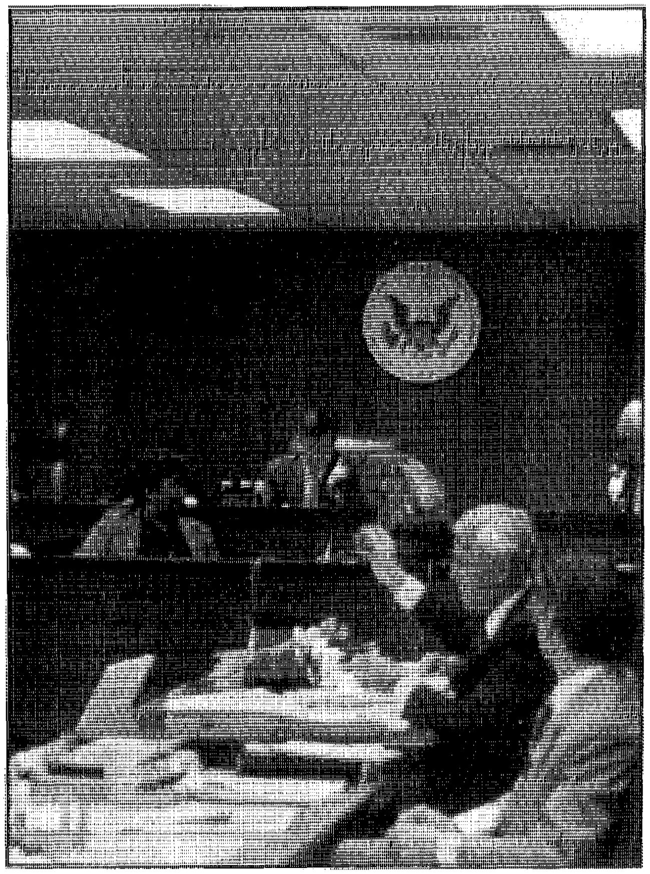
Adjudicatory hearings at the Nuclear Regulatory Commission are conducted by Licensing Boards or presiding officers drawn from the Atomic Safety and Licensing Board Panel. A total of 642 cases have been filed since the first Licensing Board case began on November 9, 1962.

The panel's judges are lawyers or technical members with expertise in a wide variety of disciplines. Their appointment to the panel is based upon recognized experience, achievement and independence in the appointee's field of expertise. During fiscal year 1993, the panel was made up of 39 administrative judges (16 full-time and 23 part-time). By profession, they included 11 lawyers, 10 public health and environmental scientists, 15 engineers or physicists, and three medical doctors. (See Appendix 2 for a listing of the names and the disciplines of fiscal year 1993 panel members.)

The panel's Licensing Boards consist of three administrative judges, usually one legal member and two technical members. The Chief Administrative Judge assigns individual judges to those hearings where their particular professional expertise will assist in resolving the kinds of technical and legal matters at issue in the proceeding. Some contested matters may be heard by a single administrative judge or administrative law judge from the panel. The panel's policy in one-judge proceedings is to assign a legal or technical administrative judge from the panel as an assistant to the presiding administrative judge, thereby insuring the requisite level of technical expertise associated with the traditional threemember Licensing Boards.

The agency's regulations provide the opportunity for numerous types of hearings. Hearings by panel judges include: *reactor licensing hearings* where, as provided by the Atomic Energy Act of 1954, as amended by the Energy Reorganization Act of 1974 and the Energy Policy Act of 1992, a hearing is required to be held on every application for a combined construction permit and operating license for a nuclear facility that produces electric power; *license amendment hearings* which allow affected parties to challenge proposed license amendments for nuclear reactors; *materials licensing hearings* which allow affected persons to contest NRC licensing actions involving the commercial use of nuclear materials; *enforcement hearings* which allow individuals, employees, licensees, contractors, subcontractors, and vendors to contest penalties assessed against them by the NRC staff for alleged infractions of NRC regulations; *antitrust hearings* which allow affected parties to challenge the licensing of nuclear reactors if the operation of such reactors would create or maintain a situation inconsistent with the antitrust laws; *special hearings* which can be ordered by the Commission for any nuclear-related matter; *personnel related hearings* in which NRC employees are allowed to bring grievance cases and Equal Employment Opportunity cases before panel judges; and *Program Fraud Civil Remedies hearings* which allow NRC employees and other individuals to contest NRC action against them for alleged fraudulent claims made to the NRC.

Hearings at the NRC may be either formal or informal. The formal proceedings consist of the traditional procedures used in non-jury Federal Court cases including pre-trial discovery between the parties and formal trial



A typical hearing of the ALSBP involves three panel members of differing expertise. The board shown in session above, in a hearing involving a civil penalty appeal, are Judge Peter A. Morris, a physicist; Judge G. Paul Bollwerk, the legal expert; and Judge James H. Carpenter, an environmental scientist.

procedures at the hearing. Formal procedures traditionally have been used at the NRC in cases involving the licensing of reactors and for enforcement proceedings brought by the agency against individuals and licensees. Informal hearing procedures are authorized in matters affecting one of the agency's more than 7,000 materials licensees. While the deliberative process for judges remains the same under either type of hearing, informal hearings involve significantly different procedures for developing the record upon which decisions must be based. The principal differences include the use of a presiding officer (a single administrative judge), written submittal by the parties instead of a hearing on the record, and, if the presiding officer determines it to be necessary after considering the written submittal, oral presentation by the parties subject to questioning by the presiding officer.

The panel employs a number of case management techniques to make the adjudicatory process as efficient as possible. Licensing Boards frequently structure their hearing schedules into distinct phases, each dealing with discrete groupings of related issues. In complex proceedings involving numerous topics and multiple issues, the panel sometimes creates separate Licensing Boards and assigns one or more discrete topics to each board. These parallel adjudications save time and enable panel members' expertise to be more precisely matched to the issues to be resolved. Panel judges also expedite their proceedings by consolidating admissible contentions and monitoring the discovery portion of proceedings to avoid undue delay.

Panel judges have also saved the parties and the government considerable litigation expenses by actively encouraging case settlement. To this end, they remind parties of settlement opportunities throughout the proceedings, they foster a free exchange of ideas among the parties conducive to the amicable resolution of differences, and, when appropriate, they utilize dispute resolution techniques to encourage negotiated settlements. Panel judges also stand ready, when warranted, to request that settlement judges, drawn from the panel, be assigned to their cases to facilitate settlement.

During fiscal year 1993, the panel actively managed its caseload to improve various statistical measures of efficiency. The vast majority of proposed contentions were resolved prior to hearing and a significant number of docketed cases were settled prior to final adjudication.

In 1993, the panel also continued its leadership role in automating the hearing process. In past years, important innovations have included the installation of computerized workstations for the judges and key panel personnel. To assist in decision writing, judges can now access full-text documents from their computers using in-house customized data base management systems while simultaneously doing legal research on the computer by utilizing external systems such as LEXIS and WESTLAW. In addition, judges and professional support staff can, from their desks, draft, share and comment on proposed decisions; access and quickly search either the panel's electronic docket or the Commission's document-retrieval system; and communicate with each other or other employees of the NRC through the Commission's electronic mail system. In selected complex cases, the full text of significant documents such as pre-filed testimony and hearing transcripts are electronically indexed and added to the judges computerized data base. In 1993, the panel's capabilities were enhanced by its inclusion on the NRC's LAN network system. Planned future upgrades for the panel include installation of a personal computer-based software system, using Personal Librarian Software, which will permit full-text inclusion of all case related documents into the panel's electronic data base.

## Panel Caseload

During fiscal year 1993, the panel's caseload comprised a total of 30 proceedings. Thirteen involved nuclear power plants or related facilities and 17 involved other Commission licensees. Ten cases were closed and 14 new cases were docketed.

The panel's 1993 caseload followed the trend, begun in the late 1980's, of cases primarily concerned with NRC enforcement actions, materials licensing actions, and actions pertaining to the regulation of nuclear reactors that have been licensed and operating. This caseload differed significantly from the three previous decades which were dominated by construction permit and operating license proceedings for licensing new reactors. In the near future, the panel expects an infusion of contested proceedings involving decommissioning of reactors and material licensee sites, license renewal of reactors, and design certification of new reactors.

Some of the panel's more significant decisions issued during fiscal year 1993 include the following.

## Antitrust Decision

A most significant decision in 1993 involved the Perry/Davis-Besse antitrust proceeding: *Ohio Edison Company* (Perry Unit 1 (Ohio) nuclear power plant); *Cleveland Electric Illuminating Company and Toledo Edison Company* (Davis-Besse Unit 1 (Ohio) nuclear power plant), LBP-92-32, 36 NRC 269 (1992). In this proceeding, the operating utilities requested that the antitrust license conditions be deleted for the Perry/Davis-Besse nuclear facilities. They contended that the conditions were no longer justified since these facilities had higher costs of generating electric power compared to competing resources. Thus, they reasoned, the facilities could not assist in the creation or maintenance of a situation inconsistent with the antitrust laws as set out in Section 105(c) of the Atomic Energy Act, as amended. The Licensing Board rejected licensees' argument by focusing on the purpose of the antitrust laws and analyzing the nature of market power. The board concluded that, in an electric utility case such as this, the test for determining a situation inconsistent with the antitrust laws is weighed in terms of the possession and use of market power. According to the board, market power is determined by numerous factors such as firm size, market concentration, barriers to entry into the market, pricing policy, profitability, and past competitive conduct. Because it is not limited, as argued by the licensees, to the comparative cost of doing business as measured by the cost of power generation, there was insufficient basis for suspending the Perry/Davis-Besse license conditions.

## Stays in NRC Proceedings

Several important panel decisions dealt with attempts to stay NRC proceedings. In *Oncology Services Corporation*, LBP-93-6, 37 NRC 207 (1993), the Licensing Board granted a 120-day stay of an enforcement proceeding sought by NRC staff to protect ongoing Federal and State criminal investigations concerning the licensee. Although the presiding officer found some prejudice to the licensee from delaying the NRC proceeding, he determined that, on balance, the greater harm could occur from premature disclosures in the criminal investigations. He, nevertheless, recognized a duty to monitor the delay to ensure that the good cause for delay continued, and he warned that the delay would be cancelled once the balance tilted in favor of going ahead with the hearing process. He also moved forward with aspects of the hearing which were unaffected by the investigations. To monitor the delay, he set timetables for submitting status reports on the ongoing investigations.

## Standing to Intervene in NRC Proceedings

Several 1993 decisions involved the issue of "standing to intervene" in NRC licensing proceedings. To demonstrate that a petitioner has sufficient standing to participate as a party in an NRC proceeding, the petitioner must show that the licensing action in question may cause it actual injury in fact and that the petitioner's interest is within the zone of interests protected by the NRC's governing statutes. The decision in *Babcock and Wilcox* (Apollo, Pennsylvania Fuel Fabrication Facility—Decommissioning Plan), LBP-93-4, 37 NRC 72 (1993) involved the issue of standing as it relates to the National Environmental Policy Act (NEPA). Because NEPA requires Federal agencies to undertake appropriate assessments of the environmental impacts of their actions, the petitioner claimed that it sustained injury in fact when the NRC staff filed a more limited Environmental Assessment rather than a full Environmental Impact Statement with respect to a proposed licensing activity. In deciding this claim, the presiding officer recognized that under NEPA a more lenient standard exists in determining injury in fact since the public has the right to be informed about the environmental consequences of an agency's actions. However, he concluded that petitioner had failed to show a concrete harm to a legitimate health, safety or environmental interest because its injury complaint was confined to economic interests (e.g., property values, local tax revenues) and it had framed its concerns in terms of undefined injury to the local community as a whole rather than to injury the petitioner itself would suffer.

A second decision involved standing to intervene in a license recapture proceeding. In *Pacific Gas and Electric Company* (Diablo Canyon Units 1 and 2 (Cal.) nuclear power plant), LBP-93-9, 37 NRC 433 (1993), a petitioner contended that the 13-to-15 additional years that would be added to a nuclear facility's operating license (recapture time for construction of the facility) was a potential accident threat sufficient to establish requisite injury in fact. The licensee argued that the extension of operating time sought by the recapture amendment was purely a ministerial or administrative change to the license which could not produce injury in fact. The Licensing Board granted standing on the basis that the risks associated with a potential accident are the same for the original operating period as they would be for the recapture period; therefore, residency within a 50-mile radius of the plant was sufficient in establishing standing just as it was in the original operating license proceeding.

Another decision involving a time-related standing question occurred in *Texas Utilities Electric Company* (Comanche Peak Unit 2 (Tex.) nuclear power plant), LBP-92-37, 36 NRC 370 (1992). There a board was asked to consider intervention in a construction permit extension proceeding in which a utility had requested a three year extension for completing construction of its nuclear facility. The board concluded that the same standing principles apply to an extension of an existing construction permit as they do for a new construction permit or operating license application. Thus, one of the petitioners was granted standing on the basis of his residence being located within 50 miles of the nuclear facility. Another ruling regarding standing in the *Texas Utilities* decision addressed a petitioner's claim that personal injury he had sustained, allegedly resulting from the utilities mismanagement, supported his standing in the proceeding. The board denied this claim since the alleged mismanagement was not related to the proposed extension of the construction permit completion date and the petitioner's grievances were in the area of employment rights and could not be redressed by any decision concerning license extension that would be issued in the proceeding.

In *Georgia Power Company*, et al. (Vogtle Units 1 and 2 (Ga.) nuclear power plant), LBP-92-38, 36 NRC 394 (1992), the standing decision turned on proof of residence. To meet the standing requirement, a petitioner claimed residence within 50 miles of the Vogtle Plant. The licensee raised a factual dispute with regard to petitioner's residence by alleging petitioner to have declared his only residence to be in another State and to have voted there. The board placed the burden of proof on the petitioner to establish residency by a preponderance of the evidence.

## Contentions in NRC Cases

Another line of 1993 panel decisions concerned the acceptability of contentions proffered by intervenors for litigation. In *Pacific Gas and Electric Company* (Diablo Canyon Units 1 and 2 (Cal.) nuclear power plant), LBP-93-9, 37 NRC 433 (1993), the utility claimed that the issue raised in one of the contentions was barred because it had been addressed in a prior Partial Director's Decision under 10 CFR 206. The Board ruled that the claim was not barred from litigation because a Director's Decision under 10 CFR 206 is not afforded appellate review, even for abuse of discretion, and thus does not constitute an adjudicatory decision under Section 189(b) of the Atomic Energy Act, as amended, 42 USC 2239(b). In this same decision, the Board also ruled procedurally that the validity and admissibility of late-filed contentions in the case should be considered before ruling on the timeliness aspects. Even though the contentions theoretically could have been summarily dismissed for their lack of timeliness without considering the contentions admissibility, the board reasoned that it was in the public interest to take this approach since the seriousness of the asserted safety and environmental problems alleged merited a closer look to avoid the possibility of not considering them for a purely procedural reason.

Another case involved the admission of contentions in a license recapture proceeding where the licensee requested that the years for constructing its nuclear facility not be included as part of the 40 year operating license period. In *Pacific Gas and Electric Company* (Diablo Canyon Units 1 and 2 (Cal.) nuclear power plant), LBP-93-01, 37 NRC 5 (1993), the utility sought to limit the scope of the petitioner's contentions claiming that the recapture proceeding was an administrative change equivalent to a proceeding for a license renewal where contentions are limited to issues of age-related degradation of structures, systems, and components. Because the Commission had not enacted regulations regarding the scope of contentions allowable in recapture proceedings, the Licensing Board ruled that the scope in those cases should be similar to that permitted in any license amendment involving a degree of risk to the public. However, as characteristic of the limited scope of most license amendment cases, the Licensing Board ruled that the scope of contentions in this case was limited to direct challenges to the permit holder's asserted reasons that show good cause justification for the delay for construction. *Texas Utilities Electric Company* (Comanche Peak Unit 2 (Tex.) nuclear power plant), LBP-92-37, 36 NRC 37 (1992).

## Injunctive Relief Based on Wasting of Assets

A Commission materials licensee, which had been ordered by the Commission to decommission and decontaminate its site, attempted to sell a significant portion of its corporate assets to a sister foreign corporation while an enforcement case against the licensee was pending. In an unpublished opinion, *Safety Light Corporation, et al.* (Bloomsburg Site) (January 22, 1993), the Licensing Board enjoined the licensee from disposing of its assets on authority of 10 CFR 2.718(m) which allows a presiding officer to "[t]ake any action consistent with the [Atomic Energy] Act [19 CFR Chapter 1], and sections 551–558 of Title 5 of the United States Code. The board concluded that the sale could inhibit the ability of the licensee to decommission the site by the dissipation of the licensee's assets and that not decommissioning the site could endanger the public health and safety.

### Attorney Client and Work-Product Privileges

In *Georgia Power Company,* et al. (Vogtle Units 1 and 2 (Ga.) nuclear power plant), LBP-93-11, 37 NRC 469 (1993), the intervenor claimed attorney client and attorney work-product privilege for six tape recordings sought by the licensee during the discovery phase of the proceeding. The intervenor had been instructed by his attorney to make excerpts of several tape recordings of conversations he had with various licensee employees in preparation for a hearing before a Department of Labor Administrative Law Judge. The intervenor previously had given these tapes to the NRC Office of Investigations and a Congressional Subcommittee. In concluding that the intervenor must produce these tapes, the Licensing Board found that the tapes were not privileged because the intervenor had not acted as his attorney's agent when preparing the tapes and the original tapes were not prepared in anticipation of the hearing. An attorney work-product privilege also did not apply since none of the attorney's thought processes was alleged to be directly disclosed in the tapes. The board further concluded that intervenor waived any privilege that may have attached to the tapes by presenting them to the NRC Office of Investigations and to the Congressional Subcommittee.

### Discovery Request for Protected Information

In *Pacific Gas and Electric Company* (Diablo Canyon Units 1 and 2 (Cal.) nuclear power plant), LBP-93-9, 38 NRC 433 (1993) and LBP 93-13, 38 NRC 11 (1993), the intervenor sought to discover information contained in certain reports prepared by the Institute for Nuclear Operations (INPO) concerning maintenance and surveillance programs at the licensee's nuclear plant. Although a Federal Court of Appeals had earlier determined that INPO reports furnished to the NRC need not be released under the Freedom of Information Act (FOIA), the board found that INPO reports are not privileged for litigation in the traditional sense, but rather only subject to non-disclosure under the FOIA. After due consideration for the need for the information by both parties and the board and the fact that the request was limited to a single INPO report, the Board ruled that the report should be made available to the intervenor subject to a protective order limiting access to the information to specified intervenor representatives, allowing no copying of the information, and allowing reference to the material to be made in litigation only through *in camera* sessions.

### Double Jeopardy

In the panel's first case involving 10 CFR Part 13, the NRC's implementation of the Program Fraud Civil Remedies Act, In the Matter of Lloyd P. Zerr, ALJ-93-1, 38 NRC ___ (September 20, 1993), the NRC staff sought to collect funds and civil penalties for alleged false claims the defendant made to the NRC. There had previously been a criminal case against this defendant for the same cause of action. A settlement was reached in the criminal case when the defendant made restitution of funds to the government and the U.S. Attorney dismissed the criminal indictment. Based on that settlement, the defendant sought dismissal of the NRC civil suit, arguing that, among other things, the NRC suit violated the Fifth Amendment by placing him in double jeopardy. The Chief Administrative Law Judge denied this motion on grounds that the criminal settlement would not result in double jeopardy in the NRC civil case since the pre-trial diversion agreement in the criminal action did not constitute jeopardy as contemplated by the double jeopardy clause. Under the pre-trial diversion agreement, defendant merely had obtained the benefit of not being prosecuted at the cost of not being placed in jeopardy. The judge also noted that Congress may impose both a criminal and civil sanction for the same act and that there was nothing in the pre-trial diversion agreement which prohibited the NRC from instituting an action against the defendant under the Program Fraud Civil Remedies Act.

## COMMISSION DECISIONS

The Commission exercises all authority for the appellate review of decisions of presiding officers and Atomic Safety and Licensing Boards in agency adjudications. The Office of Commission Appellate Adjudication assists the Commission in its adjudicatory role. Summarized below are the more significant Commission decisions in fiscal year 1993.

### Advanced Medical Systems, Inc.

Advanced Medical Systems, Inc. (AMS), petitioned the Commission for review of an Atomic Safety and Licensing

Board decision that dismissed as moot a long-pending proceeding in which AMS challenged the NRC staff's issuance of two immediately effective orders. Issued in 1987, the orders required AMS to decontaminate its facility. AMS satisfactorily completed the decontamination, but sought to litigate the legitimacy of staff's issuance of the immediately effective orders. Because decontamination was complete, the Licensing Board concluded that there was no remedy it could fashion, even if AMS proved that the immediately effective orders had been issued improperly. Accordingly, at the NRC staff's request, the Licensing Board dismissed the proceeding.

The Commission denied AMS's petition for review in *Advanced Medical Systems, Inc.*, CLI-93-8, 37 NRC 181 (1993). The Commission noted that both parties agreed that the decontamination actions required by the orders had been completed. AMS argued that a live controversy over the orders still existed because AMS continued to suffer adverse effects from the orders. Specifically, AMS claimed two lingering effects from the orders: (1) that it would be subject to escalated fines in the event of any further violations, and (2) that it continued to receive negative publicity.

Applying agency enforcement policy under 10 CFR Part 2, Appendix C, § IV.B.3 and Table 2, the Commission concluded that the orders in question would not be considered a basis for escalating any future enforcement action against AMS. The Commission also determined that none of the press reports cited by AMS made any reference to the immediately effective orders. Instead, the negative publicity stemmed from the NRC's inclusion of AMS in the agency's "Site Management Decommissioning Plan," which listed dozens of sites identified as in need of decontamination. AMS remains on the list because the agency's estimated cost for final decommissioning exceeds AMS's financial assurance statement. AMS also remains on the list because AMS's liquid waste hold-up tank room remains contaminated. Decontamination of the tank room had been a requirement of the 1987 decontamination orders, but at AMS's request the staff agreed to allow AMS to defer decontamination of the room until a future date. In sum, the Commission found no link between the challenged 1987 orders and the negative press reports cited by AMS. The orders also would not reasonably lead to future escalated enforcement sanctions.

Even when an agency order no longer has any effect, a case may not be moot if it is capable of repetition, yet evading review. AMS submitted that the proceeding challenging the two immediately effective orders was not moot because it fell within this exception to the mootness doctrine. To fall within this exception, AMS needed to demonstrate that the challenged action was too short in duration to be fully litigated *and* that there is a reasonable expectation that AMS will be subjected to the same action again. The Commission first determined that the challenged orders were not short-lived, fleeting actions which by their nature easily evade review. Indeed, AMS took two years to comply with the orders. Moreover, any such future immediately effective orders would not likely evade review because in 1992 the Commission adopted procedural rules that provide for a prompt review of the immediate effectiveness of staff enforcement orders. See *Revisions to Procedures to Issue Orders: Challenges to Orders That Are Made Immediately Effective*, 57 FR 20,194 (May 12, 1992; amending 10 CFR 202(c)).

The Commission further concluded that the orders did not reasonably appear capable of repetition. The NRC staff had acknowledged that AMS satisfied the decontamination requirements in the orders. Any future orders would be based on different future circumstances or matters beyond the scope of the instant orders. Thus, AMS had not shown that the proceeding satisfied the mootness exception criteria. The Commission did not identify any prejudicial error in the Licensing Board's dismissal of the proceeding, and accordingly denied AMS's petition for review.

## Comanche Peak Nuclear Power Plant

Requests for Late Intervention. In 1993, the Commission faced a number of challenges related to the licensing of the Comanche Peak Unit 2 (Tex.) nuclear power plant. The Atomic Safety and Licensing Board had dismissed the operating license proceedings for the Comanche Peak plant in July 1988, pursuant to a settlement agreement between the NRC's staff, the Texas Utilities Electric Company (TU), and the Citizens Association for Sound Energy, the only remaining intervenor at that time. In January 1993, the Citizens for Fair Utility Regulation (CFUR) requested the Commission to offer a new opportunity for a hearing on the operating license for Comanche Peak Unit 2. The Commission concluded that, in effect, CFUR's request represented a petition for late intervention and a request to reopen the hearing on the Comanche Peak Unit 2 operating license. These requests were denied. The Commission stressed that when a petitioner seeks late intervention and the record has been closed, the petitioner must satisfy the criteria both for late intervention and for reopening the record. A petitioner may not seek to reopen a proceeding before first becoming a *party* to the proceeding. CFUR had voluntarily withdrawn from the original Comanche Peak proceeding in 1982, and thus could not seek to reopen the record without first being granted late intervention. CFUR had not addressed the Commission criteria for either late intervention or reopening of the record.

Subsequently, CFUR filed a petition for late intervention in February 1993, citing new knowledge of deficiencies in the fire barrier called Thermo-Lag, installed in Unit 2. In *Texas Utilities Electric Co.*, CLI-93-04, 37 NRC 156 (1993), the Commission denied CFUR's petition for late intervention. CFUR captioned its petition as filed in both the Unit 1 operating license proceeding and a separate construction permit amendment proceeding, involving TU's proposal for an extension of the construction permit. The Commission noted that the issuance of a full-power license for Comanche Peak Unit 1 in April 1990, barred any further opportunity for hearing on the Unit 1 operating license. Thus, any challenge to the Unit 1 license could only be in the form of a petition filed under 10 CFR 2.206. Consequently, the Commission denied any portion of CFUR's request that related to the Unit 1 proceedings. The NRC staff's issuance of a low-power license for Unit 2 in February 1993 would not, though, preclude late intervention in the Unit 2 fullpower operating license proceeding. However, the Commission concluded that CFUR's petition did not establish good cause for late intervention. Information about the adequacy of and possible defects in Thermo-Lag had been public for years, and CFUR had been aware for at least six months that Thermo-Lag had been installed in Unit 2. After evaluating the factors for late intervention under 10 CFR 714, the Commission determined that the potential for delay in the licensing process for Unit 2 outweighed any apparent contribution that CFUR could provide to the development of the record.

## Construction Permit Amendment Proceeding

In January 1993, two appeals were filed of the Atomic Safety and Licensing Board's denial of two joint petitions for intervention and for hearing with respect to a construction permit extension request filed by TU in February 1992. The licensee sought an extension of the construction completion date from August 1, 1992, to August 1, 1995. On the ground of failure to submit a viable contention, the Licensing Board denied a petition filed by B. Irene Orr, D.I. Orr, Joseph J. Macktal, Jr., and S.M.A. Hasan. The other petition was filed by R. Micky Dow and Sandra Long Dow, "doing business as" the Disposable Workers of Comanche Peak, and dismissed for lack of standing to intervene. All parties but Messrs. Macktal and Hasan appealed. Based upon a determination that good cause had been shown and that no significant hazards considerations were involved, the NRC staff granted the construction permit amendment on July 28, 1992.

In *Texas Utilities Electric Co.* (Comanche Peak Unit 2 (Tex.) nuclear power plant), CLI-93-10, 37 NRC 192 (1993), the Commission denied both appeals as moot. The Commission denied the Dows' appeal on the additional ground that they failed to file a brief to perfect their appeal. The Commission found that the construction status of Unit 2 rendered the construction permit extension proceeding moot. One test for mootness is whether the relief sought would make a difference to the legal interests of the parties. The Commission concluded that the relief that the petitioners sought—a denial of the construction permit extension—would make no difference to their interests because the licensee no longer required a construction permit extension.

The Commission's analysis rested upon applicable provisions of the Atomic Energy Act (AEA), the NRC's regulations, and the Administrative Procedure Act (APA). The Commission first noted that the language of Section 185 of the AEA implies that if construction of a facility has been completed prior to expiration of the construction permit, then the permit will not expire and will remain in force, to be converted to an operating license following the necessary findings set out in the remainder of section 185. Thus, Section 185 establishes that no rights under a construction permit will be forfeited unless two circumstances are present: (1) the latest date for construction completion has passed, and (2) the facility is not completed. If construction is complete, no further extension of the completion date is required, the permit will not expire and, by implication, the permit holder retains its rights under the permit. The Commission further noted that under 10 CFR 50.56 and 50.57(a)(1), "substantial completion" of a facility satisfies the AEA's requirements regarding "completion" of a facility.

The Orrs stressed that Texas Utilities (TU) failed to complete construction of Unit 2 by August 2, 1992—the completion date specified in the construction permit—prior to TU's request for an extension. Although on July 28, 1992, staff found "good cause" for an extension and granted the extension, the Orrs contended that once they had filed a timely request for a hearing on the extension, the Commission should have prohibited TU from continuing construction of Unit 2. The Commission concluded that because of the "timely renewal doctrine," found in Section 9(b) of the APA, 5 USC 558(c) and adopted in NRC regulations at 10 CFR 109, a timely request for an extension of the completion date maintains the construction permit in force by operation of law. Accordingly, the licensee may lawfully continue construction activities pending a final determination of its construction extension application. TU had filed an application for an extension in February 1992, well before the August 2, 1992 completion date. Although the petitioners' challenge to the application for extension left unresolved--pending a final determination after a possible hearing-the validity of the permit extension issued by staff, the construction permit remained in force by virtue of both staff's issuance of the order extending the completion date, *and* TU's timely application for an amendment to extend the completion date. Whether in fact TU demonstrated "good cause" for an extension had become a moot issue. TU lawfully completed construction under the permit, and required no further extension of the completion date. The Commission also determined that this case did not fall under the mootness exception for those cases that are "capable of repetition, yet evading review" because there was no reasonable expectation that the controversy over a construction permit extension would recur between TU and the Orrs.

R. Micky Dow and Sandra Long Dow, representing the Disposable Workers of Comanche Peak Steam Electric Station filed another petition to intervene in March 1993. The Commission in CLI-93-11, 37 NRC 251 (1993), denied this petition because it failed to address the factors for a late-filed petition and the Commission's standing requirements.

## Requests for Stay of Low- and Full-Power Licenses

The Orrs also sought a stay of the issuance of the low-power license for Comanche Peak Unit 2, issued by the NRC staff on February 2, 1993. The Orrs sought a stay of the license pending the Commission's resolution of their appeal of the Atomic Safety and Licensing Board decision that denied their petition to intervene in the construction permit extension proceeding. The Commission denied the stay request on February 3, 1993. First, the Commission stressed that the stay request could not properly be considered in the operating license proceeding because the petitioners were not parties to the license proceeding, and their stay request did not address the five factors for late intervention found in 10 CFR 2.714(a)(1)(i)-(v). Moreover, the petitioners had not even addressed the Commission criteria under 10 CFR 2.788 for a stay, but instead merely made generalized allegations, unsupported by evidence. Secondly, the Commission found that § 2.788, upon which the petitioners had based their stay request, only provided for stays of decisions or actions in the proceeding under review, which in this case was the construction permit amendment proceeding. The stay request had not been tied to that proceeding.

The Orrs sought a stay again when they challenged issuance of the full-power license for Unit 2. The petitioners argued that TU did not have the character and competence to operate the plant because through restrictive settlement agreements with former coowners the licensee had kept safety-related information from the NRC. Again the Commission noted that the petitioners were not parties to the operating license proceeding, and had not properly sought late intervention in the operating license proceeding. In addition, issuance of the full-power license would have no effect on the petitioners' appeal of the construction permit amendment proceeding because that appeal had become moot and been dismissed. The Commission, however, addressed the character allegations made by the Orrs, finding that no aspect of the allegations justified a stay of the issuance of the full-power license. For instance, all former co-owners of the licensee informed the agency that they had never interpreted the settlement agreement as prohibiting them from informing the NRC of safety concerns.

## Oncology Services Corporation

On December 1, 1992, the Nuclear Regulatory Commission was notified of an incident involving the loss of a 3.7 curie iridium192 source from Oncology Services Corporation's (OSC's or licensee's) Indiana Regional Cancer Center in Indiana, Pa. The NRC staff investigated the incident and on January 20, 1993, issued an immediately effective order suspending OSC's license to provide brachytherapy treatment at the Pennsylvania cancer treatment facilities named in its license. Order Suspending License (Effective Immediately), 58 FR 6825 (February 2, 1993).

According to the staff's order, on November 16, 1992, an iridium-192 sealed-source, which was inserted into a catheter in the abdomen of a nursing home patient, broke off and remained in the patient when the patient was returned to the nursing home that same day. The staff alleged that as a result of this incident the patient received a significant amount of radiation exposure, and that numerous other individuals, including health care workers, visitors, sanitation workers, and other members of the general public, were exposed unnecessarily to radiation.

The staff also identified certain practices and procedures as indicative of a significant corporate management breakdown in the control of licensed activities. The staff concluded that this breakdown was of the utmost regulatory concern because it contributed to the occurrence of the overexposure incident. Consequently, the staff concluded that the public health, safety, and interest required an immediately effective suspension of the license.

Since the initial suspension was imposed, the staff relaxed the order and, on a case-by-case determination, allowed OSC to treat patients upon a good cause showing for the individual treatment. The staff further relaxed its suspension order to allow resumption of licensed activities at two of the facilities named in the OSC license. The suspension remains in effect for the other facilities named in the license.

On February 5, 1993, OSC filed a request for hearing on the staff's order, and an Atomic Safety and Licensing Board was established pursuant to OSC's request. On February 23, 1993, the NRC staff filed a motion pursuant to 10 CFR 2.202(c)(2)(ii) (1993), requesting that the Licensing Board delay the conduct of this proceeding for an initial period of 120 days. The Licensing Board granted, in part, the staff's motion by staying for 120 days (through and including June 23, 1993) discovery and any portion of the proceeding which necessarily must follow discovery. LBP-93-6, 37 NRC 207 (1993). The Licensing Board found that there was good cause for delaying the proceeding because the staff had demonstrated that the interests of the government outweighed the interests of the licensee. OSC filed before the Commission a petition for interlocutory review of this order.

In *Oncology Services Corp.*, CLI-93-13, 37 NRC 419 (1993), the Commission granted OSC's petition for interlocutory review. The Commission determined that when a licensee is subject to an immediately effective suspension order, the licensee's due process interest in a prompt hearing could be threatened by a 120-day stay of the proceeding. The Commission found, however, that as a practical matter, review of the final Licensing Board order would provide no relief from any conceivable harm that could be suffered as a result of the 120-day stay imposed by an allegedly erroneous Licensing Board order. No effective relief could be fashioned at that time because the stay would soon expire by its own terms. However, because the stay was due to expire on June 23, 1993, and because some of the same issues were raised again in the NRC staff's motion for an additional delay filed June 3, 1993 (after OSC filed its petition for review, but prior to the Commission's grant of the petition), the Commission took the unusual step of directing the Licensing Board to refer to the Commission for review any ruling granting the NRC staff's motion for an additional delay.

In its June 3, 1993 motion, the staff requested an additional 120-day delay in discovery. In LBP-93-10, 37 NRC 455 (1993), the Licensing Board granted a 90-day stay of discovery, through and including September 21, 1993. In accordance with the Commission's instruction in CLI-93-13, the Licensing Board referred this ruling to the Commission.

In CLI-93-17, the Commission vacated as moot portions of LBP-93-6, the Licensing Board order challenged by OSC in its original petition for review. When the original stay expired by its own terms on June 23, 1993, the portion of LBP-93-6 pertaining to the initial 120-day stay ceased to have any operative effect or purpose. Thus, the portion of the proceeding related to the Licensing Board's granting of an initial 120-day stay had become moot, and accordingly was vacated. Left before the Board was the portion of LBP-93-6 pertaining to procedural matters. The Commission left undisturbed this unchallenged portion of LBP-93-6.

In CLI-93-17 the Commission also affirmed the Licensing Board's ruling that the staff had demonstrated good cause for delaying the proceeding. To determine if the staff had good cause for the delay, the Commission weighed the interests of the licensee and the government. In weighing these interests, the Commission applied the tests that were applied in *United States v. Eight Thousand Eight Hundred and Fifty Dollars in United States Currency*, 461 U.S. 555 (1983) and *FDIC v. Mallen*, 486 U.S. 230, 242 (1988).

The Commission determined that the staff had shown a compelling interest by demonstrating (1) that the risk of wrongful deprivation of the licensee's interests had been reduced; (2) that discovery of certain matters would interfere with an ongoing investigation being conducted by the Office of Investigations; and (3) that the government has a strong interest in protecting the integrity of the investigation. The Commission acknowledged that the stay would adversely affect the licensee's interest to some extent. However, the licensee offered little, if any, specification of financial or other consequences from the suspension. The absence of any particular showing of financial burden or detriment to patient care, coupled with the staff's rescission of the order at two OSC facilities, led the Commission to conclude that OSC had suffered only moderate, if not minimal, harm to its interests. Without a more particularized showing of harm, OSC's argument that the stay affected its interests did not tip the scale in OSC's favor. The Commission also noted that the proceeding was not at a complete standstill. The Licensing Board was pursuing resolution of matters that do not depend on discovery.

## Doctor's Byproduct License Revocation

The NRC staff's issuance of an order revoking Dr. Randall C. Orem's byproduct material license gave rise to this proceeding. About a year after granting the byproduct material license, the NRC staff discovered that the facility described in Dr. Orem's license application as the location for the possession and use of radioactive material never had been constructed. Dr. Orem's application contained a drawing of a facility and stated that the facility "was being finished at this time." Dr. Orem requested a hearing on the order, but eventually reached a settlement with the NRC staff. The Atomic Safety and Licensing Board approved the settlement in August 1992. In the settlement agreement, the NRC staff agreed not to pursue any further action against Dr. Orem, and further agreed not to hold any of the facts associated with the proceeding against Dr. Orem in the event that he applied for a license or license amendment.

In its discretion, the Commission in *Randall C. Orem, D.O.*, CLI-93-14, 37 NRC 419 (1993), reviewed the Licensing Board's approval of the settlement agreement between Dr. Orem and the NRC staff, but chose not to overturn the agreement. The Commission, however, underscored the significance of the duty of an applicant or licensee to provide the Commission with accurate information. The Commission noted that a false statement is material if the information has a natural tendency or capability to influence a reasonable agency expert. Whether the NRC staff actually relied upon the statement has no bearing on the materiality of the false statement. The description and status of an applicant's proposed facility clearly are material matters that can influence the NRC staff's decision to grant a byproduct material license.

The Commission chose not to disturb the settlement agreement, despite reservations about the provision in the agreement not to hold against Dr. Orem any facts associated with the proceeding. The Commission concluded that, on balance, the agreement would not be contrary to public interest. The agreement achieved without further litigation the intent of the enforcement order—revocation of Dr. Orem's license. Moreover, Dr. Orem had followed NRC regulations and the terms of the license, apparently having realized that he could not procure licensed materials without having completed the facility described in his application. Although the Commission permitted the settlement agreement to stand as written, the Commission strongly cautioned all applicants and licensees of their obligation to provide the agency with accurate and complete information. Even if a consultant assists in the preparation of an application, as was the case with Dr. Orem, the applicant remains responsible for the contents of the application. Chairman Selin and Commissioner Curtiss dissented from the Commission's decision. Both would have rejected the settlement agreement's provision not to hold the facts of the proceeding against Dr. Orem in any future licensing action that he may request.

## Perry Nuclear Power Plant

Susan L. Hiatt and the Ohio Citizens for Responsible Energy, Inc. (OCRE) appealed a decision by the Atomic Safety and Licensing Board, LBP-92-4, 35 NRC 114 (1992), which denied their request for intervention status and for a hearing on an amendment to the Perry (Ohio) nuclear power plant's operating license. The license amendment, requested by the Cleveland Electric Illuminating Company (CEIC), would delete the reactor vessel material surveillance program withdrawal schedule from the Perry plant's technical specifications, and would transfer the schedule to the facility's updated safety analysis report. The proposed amendment resulted from an agency initiative to streamline and improve technical specifications. The NRC staff had encouraged licensees to propose license amendments that would delete unessential or duplicative provisions from technical specifications. Staff considered unnecessary the presence of the withdrawal schedule in the technical specifications because Appendix H to 10 CFR Part 50 already provides regulatory requirements governing the withdrawal schedule, and already requires the licensee to obtain prior staff approval of any changes to the schedule.

In their challenge to the proposed amendment, OCRE and Ms. Hiatt argued that the amendment violates section 189a of the Atomic Energy Act (AEA), 42 USC 2239(a). Under section 189a of the AEA, the Commission must provide notice of a proposed license amendment in the *Federal Register*, and must provide an opportunity for a hearing on the amendment. Section 189a also provides that any Commission denial of a request for a hearing is subject to judicial review. Licensees may seek changes to provisions included in plant technical specifications only through a license amendment application. Removal of the withdrawal schedule from the technical specifications would mean that any proposed changes to the schedule no longer would require a license amendment, and thus would not give rise to the rights to notice, opportunity for a hearing, and opportunity for judicial review.

Because of the requirements under Appendix H, however, the NRC staff would continue to exercise the same degree of oversight and control over the withdrawal schedule. As a result, the petitioners reasoned that the only significant effect of the removal of the withdrawal schedule from the technical specifications would be that any future changes to the schedule would no longer involve the opportunities under section 189a for public participation and intervention. Staff's retention of control over the withdrawal schedule indicated to the petitioners that the schedule has safety significance. Given this apparent safety significance, the petitioners alleged that any future staff approvals of proposed schedule changes would be *"de facto"* license amendments, issued without the procedural protections required under the AEA for license amendments.

The Licensing Board denied OCRE and Ms. Hiatt's petition for intervention on the ground that the petitioners lacked standing to intervene. The Licensing Board concluded that the petitioners had not shown how they would be injured or threatened by the proposed amendment. To intervene in a proceeding petitioners must establish that they have sufficient interest in the subject at issue. Petitioners must allege with particularity how they would be injured by the action they challenge, and how a favorable decision would redress their injury. The Licensing Board concluded that the petitioners had not presented how they would be injured or threatened by the proposed amendment. The Board found the petitioners' claim of procedural injury speculative since the licensee might never propose any change to the withdrawal schedule, and the amendment itself was merely an administrative transfer of the schedule, and would not result in any alteration to the schedule.

In *Cleveland Electric Illuminating Company* (Perry Unit 1 (Ohio) nuclear power plant), CLI-93-21 , 38 NRC ___ (1993), the Commission reversed the Licensing Board's decision and granted the petitioners' appeal. The Commission found that OCRE and Ms. Hiatt had standing to intervene because they sufficiently had alleged how the proposed amendment could injure them. Specifically, the Commission found that the loss of the rights to notice, opportunity for a hearing, and opportunity for judicial review, constitutes a real and not hypothetical injury. Deletion of the withdrawal schedule from the technical specifications would directly result in the petitioners no longer having opportunities under section 189a to challenge any proposed changes to the withdrawal schedule. The Commission did not view the possibility that the licensee would indeed seek to change the schedule as unduly speculative since an intent of the license amendment was to simplify the required procedural steps for withdrawal schedule changes.

The Commission noted that standing may be based upon the alleged loss of procedural rights as long as the procedures at issue are designed to protect a concrete interest of the petitioners. A fair reading of the petitioners' claims indicated that they ultimately feared a radiological injury. The petitioners had expressed the concern that if they were deprived of the opportunity to challenge future proposals to alter the withdrawal schedule, the surveillance of the Perry reactor vessel could become lax, which could then potentially result in vessel embrittlement, and ultimately lead to the release of radioactive fission products into the environment. The Commission further noted that Ms. Hiatt, a member of OCRE, lives within 15 miles of the Perry facility.

The Commission's decision did not reflect any view on the merits of the petitioners' claims, but only established that they had alleged a sufficient interest for standing to intervene. However, to actually be granted intervention status, Commission regulations require petitioners not only to have standing, but also to proffer at least one factual or legal contention that satisfies 10 CFR 2.714(b). Therefore, the Commission remanded to the Licensing Board the contention submitted by the petitioners, for the Board to evaluate the contention's admissibility.

## Rancho Seco Nuclear Power Plant

After a public referendum favoring cessation of operation of the Rancho Seco (Cal.) nuclear power plant, the Sacramento Utility District (SMUD) obtained a conversion of its operating license to a "possession only" license, which eliminated SMUD's authority to operate Rancho Seco. SMUD then filed before the Commission an application for termination of its license and a proposed decommissioning plan for the Rancho Seco nuclear power plant. The plan provides for 10 to 20 years of on-site storage of the facility followed by the removal of the residual radioactivity. The licensee also submitted to the NRC staff an Environmental Report related to decommissioning.

In response to a March 1992 notice of opportunity for hearing, the Environmental and Resources Conservation Organization (ECO) filed a petition to intervene and request for hearing on SMUD's decommissioning plan for Rancho Seco. The Atomic Safety and Licensing Board denied ECO intervention status. LBP-92-23, 36 NRC 120. The Licensing Board found that ECO had failed to establish standing either as a matter of right or of discretion, and had failed to set forth at least one viable contention. ECO filed before the Commission an appeal challenging the Licensing Board's denial of intervention.

In *Sacramento Municipal Utility District*, CLI-93-3, 37 NRC 135 (1993), the Commission reversed the Licensing Board and granted ECO discretionary intervention. 37 NRC 135 (1993). At the time that ECO filed its appeal, the Commission was examining the process for review and approval of decommissioning plans, including the timing and scope of public participation in the decommissioning process. ECO did not clearly demonstrate the requisite interest to participate in an NRC proceeding, but the arguments raised by ECO in support of its standing presented complex questions of law and fact which, if resolved in ECO's favor, would support standing. Because of the unusual circumstances of the case, the Commission chose to decide the appeal without resolving the petitioner's standing as matter of law, but instead rested its decision on the Commission's discretionary authority to hold hearings and permit participation in its proceedings. The Commission did not intend that this decision become precedent for any other matter that may come before the Commission.

In support of intervention ECO had submitted before the Licensing Board an environmental contention and safety contentions. ECO also had raised a number of procedural matters which it labelled as contentions, although these matters did not concern the substantive merits of the licensee's decommissioning plan. With respect to the environmental contention, the Commission declined to reconsider its prior determination that resumed operation of a facility is not to be considered an alternative to a proposal to decommission a facility except perhaps in extraordinary circumstances (e.g., a national emergency) not present here. See *Long Island Lighting Co.*, CLI-90-8, 32 NRC 201, 207 (1990); Sections 108, 186(c), 188 of the Atomic Energy Act, 42 USC 2138, 2236, 2238. The Commission also rejected ECO's attempt to reintroduce as "no action" the "resumed operation" alternative that the Commission had already declined to consider.

ECO's theory appeared to be that taking "no action" on decommissioning would preserve Rancho Seco for resumed operation. ECO argued that under NEPA, analysis of the "no-action" alternative required consideration of environmental impacts of possible forms of replacement energy for Rancho Seco. The Commission concluded that, under NEPA, consideration of the "no-action" alternative in the manner suggested by ECO was not required because it involved consideration of environmental impacts that could be avoided only by the highly speculative and not reasonably foreseeable resumed operation of Rancho Seco.

The Commission admitted and remanded back to the Licensing Board part of ECO's environmental contention that related to the probability of a loss of off-site power. And the Commission permitted ECO to amend its contention with respect to the adequacy of SMUD's decommissioning funding plan, without having to meet the criteria for late filings, a requirement normally applied to amended contentions at that stage in a proceeding.

ECO also appealed the Licensing Board's denial of ECO's motion requesting that the Licensing Board withhold an order wholly denying the petition for leave to intervene and the request for a hearing until ECO was given an opportunity to file contentions after issuance of the agency's environmental and safety review documents. The Commission left unresolved the question of whether a prior hearing is required in decommissioning proceedings, and determined that as a matter of discretion in this particular case the Commission would offer a prior hearing.

SMUD filed a motion for reconsideration of the Commission's Memorandum and Order in CLI-93-03. SMUD argued that the Commission should reconsider its determinations to grant discretionary intervention, to admit the contention regarding loss of off-site power, to allow ECO the opportunity to file an amended contention regarding SMUD's decommissioning funding plan, and to offer a prior hearing. In CLI-93-12, 37 NRC 355 (1993), the Commission denied SMUD's motion for reconsideration. The Commission concluded that SMUD failed to identify any error or abuse of discretion by the Commission in deciding CLI-93-03, and that although SMUD asserted that its interests were compelling and that CLI-93-13 was highly prejudicial, SMUD did not articulate any specific harm that it was suffering as a result of the order.

## Vogtle Nuclear Power Plant

Georgia Power Company (GPC) appealed an Atomic Safety and Licensing Board decision that granted Allen L. Mosbaugh's petition to intervene and for hearing on a proposed transfer of the license to operate Units 1 and 2 of the Vogtle (Ga.) nuclear power plant. *Georgia Power Company (Vogtle Electric Generating Plant, Units 1 and 2)*, LBP-93-5, 37 NRC 96 (1993). The proposed licensing action would transfer all operational control over Vogtle units 1 and 2 from GPC, the present licensee, to Southern Nuclear Operating Company, Inc. (Southern Nuclear). Mr. Mosbaugh alleged that Southern Nuclear's management lacks the requisite character and integrity to assure the safe operation of the plant, and that therefore Southern Nuclear should not become the Vogtle facility licensee.

On appeal, GPC argued that Mr. Mosbaugh's petition should have been denied by the Licensing Board because Mr. Mosbaugh lacks standing to intervene and failed to submit an admissible contention. For standing, a petitioner must allege an "injury in fact" from the licensing action being challenged. GPC asserted that the transfer of operating authority to Southern Nuclear would not result in any significant change in the personnel managing the plant, and therefore the transfer would not result in any *new* injury to Mr. Mosbaugh. GPC explained that the on-site organization responsible for operations at the facility would be transferred as a unit to Southern Nuclear. GPC further noted that as to off-site management, three of four GPC off-site managers currently are Southern Nuclear officers, and upon authorization of the transfer, these would continue their management roles, except they would do so as solely Southern Nuclear officers. Consequently, GPC concluded that Mr. Mosbaugh failed to allege an injury linked to the proposed transfer. GPC stated that the only anticipated change in personnel was that the executive vice president of GPC would no longer report to the president of GPC, but would instead report solely to the Board of Directors of Southern Nuclear, and that this reporting change would be insignificant because the president of GPC will remain a member of the Southern Nuclear Board of Directors.

The Commission in *Georgia Power Company* (Vogtle Units 1 and 2 (Ga.) nuclear power plant), CLI-93-16, ___ NRC ___ 1993) agreed with the Licensing Board that the transfer of control could pose an injury to Mr. Mosbaugh. The Commission stressed that although key management officers from Southern Nuclear may already be in managerial roles at Vogtle in their roles as GPC officers, their current presence at Vogtle did not obviate the need for Southern Nuclear to show—before it is granted licensee status—that its management is indeed willing to follow NRC regulatory requirements. The Commission was unwilling to conclude at such a threshold stage in the proceeding that no injury would result from transferring responsibility for safe operation to persons in Southern Nuclear's corporate management, alleged by Mr. Mosbaugh to have violated agency safety regulations, and to have submitted false information to the NRC.

Transfer of the license would signify that a different corporate entity, Southern Nuclear, would be responsible for activities at Vogtle. As a result, even if the same personnel operate the plant on the day following a transfer, those individuals would report to a different organization. This new organization would potentially have the capacity to replace plant management and affect plant operation in any number of ways. The Commission further noted that at such a preliminary stage, it was ill-equipped to evaluate to what extent the license transfer may result in changes in the relative influence of personnel at the Vogtle facility, and what the potential effects might be of any such shift in influence. Consequently, the Commission declined to find that the transfer represented merely a corporate name change, as GPC had argued.

The Commission, however, stressed that not every licensing action throws open an opportunity to inquire into the "character" of a licensee. This proceeding involved a direct and obvious relationship between the character allegations and the licensing action in dispute. The significance of a complete transfer of operational control over a nuclear power plant licensed to operate at full-power made relevant the licensee's integrity and willingness to abide by regulatory requirements.

Although particular Southern Nuclear officers alleged to be unfit managers were already involved in the operation of the Vogtle facility in their roles as GPC officers, the Commission noted that Mr. Mosbaugh's concerns could still be at least partially redressed by a favorable decision. For instance, if the transfer were granted subject to changes in the structure and personnel of Southern Nuclear, as in limiting or prohibiting particular activities by specific officers, the alleged potential harm from operation under the proposed transfer could be prevented. Moreover, any factual findings of poor management integrity made in this proceeding could bar re-litigation of character allegations in any later proceedings brought on the same facts.

The Commission also rejected GPC's claim that Mr. Mosbaugh failed to satisfy the Commission's requirements regarding the admission of contentions. Specifically, GPC argued that the contention admitted by the Licensing Board failed to identify any disputed portion or omission in GPC's transfer application, as prescribed by 10 CFR 2.714(b)(2)(iii). The Commission found that the regulation was never intended to preclude contentions that rest on relevant matters that are not required to be addressed in an application. Under the circumstances of this proceeding, no purpose would be furthered by requiring Mr. Mosbaugh to identify a *specific* portion of GPC's application, when Mr. Mosbaugh's contention related solely to character allegations, and the licensee had not addressed character issues in its application, nor been under any requirement to do so. The Commission thus concluded that section 2.714(b)(2)(iii) only requires the petitioner to identify an error or omission in an application if the application either actually contains a disputed provision or has omitted required information.

## JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1993 is summarized below.

### Pending Cases

*Kelley v. Selin*, Nos. 93–1646; 93–1710; 93–3613; 93–3749 (6th Cir.).

This lawsuit, now pending before the Sixth Circuit, has a complicated procedural history, reflected in the four separate case numbers listed above.

On May 4, 1993, plaintiffs, the Michigan Attorney General, several private citizens, and the Lake Michigan Federation, filed suit in Federal District Court for the Western District of Michigan asking for an injunction that would prevent Consumers Power Company from using an NRC-approved dry storage cask, the "VSC-24," for storing spent fuel from the Palisades (Mich.) nuclear power plant. *Kelly v. Selin*, No. 4:93–CV–67. Plaintiffs' central claim was that the NRC had failed to perform a site-specific NEPA analysis of the effects of using the VSC-24 cask at Palisades.

In reply, the NRC argued that it had met all its NEPA responsibilities, and that in any case, the actions complained of proceeded from a recently issued rule that added the VSC-24 cask to the list of NRC-approved spent fuel casks. Since the suit was in effect a challenge to the rule, the NRC asserted that only the Court of Appeals had jurisdiction over the case. The District Court agreed and at plaintiffs' request, transferred the case to the Sixth Circuit Court of Appeals, where it was docketed as No. 93–1646. The plaintiffs also appealed to the Sixth Circuit (which docketed the appeal as No. 93–1710) the District Court's ruling that it lacked jurisdiction over the case.

Once in the Sixth Circuit, the plaintiffs asked that court for an order halting use of the VSC-24. After considering the responses of the NRC and Consumers Power, the Court of Appeals denied the request on May 17, 1993, noting among other things that the plaintiffs had not sought administrative relief from the NRC.

On May 21, therefore, the plaintiffs filed a motion with the NRC's Secretary, asking that the Commission either stay the effectiveness of the new rule or rescind it. On May 26, the EDO issued a document in which he denied the requested stay, and with regard to the request for rescission of the rule, found the petition incomplete. He therefore declared that he would hold the matter in abeyance and allow the plaintiffs time in which to supplement their petition.

On June 4, the plaintiffs (now petitioners) filed a petition for review of the rule approving the VSC–24 cask. This was docketed as No. 93–3613 and consolidated by the court with the other two cases on June 9. The petitioners in this case included a new party, a citizens' group calling itself "Don't Waste Michigan."

On June 17, the NRC filed a motion for partial dismissal of the case, pointing out that two of the petitioners (the Michigan Attorney General and the Lake Michigan

Federation) had a request for administrative relief pending before the Commission, and that this rendered their petitions for review "incurably premature," under established case law. Consumers Power Company also filed a motion to dismiss.

On June 22, the Michigan AG and the Lake Michigan Federation wrote to the EDO, asking to withdraw their motion for rescission. The EDO granted that request on July 1, and so informed the court. On July 9, those petitioners filed a new petition for review of the April 7 rule, as a precaution in case their earlier petitions of review were dismissed. It was docketed as No. 93–3749 and consolidated with the other cases. The case is now being briefed before the Court of Appeals.

*Nuclear Information and Resource Service v. NRC*, No. 93–1164 (D.C. Cir.).

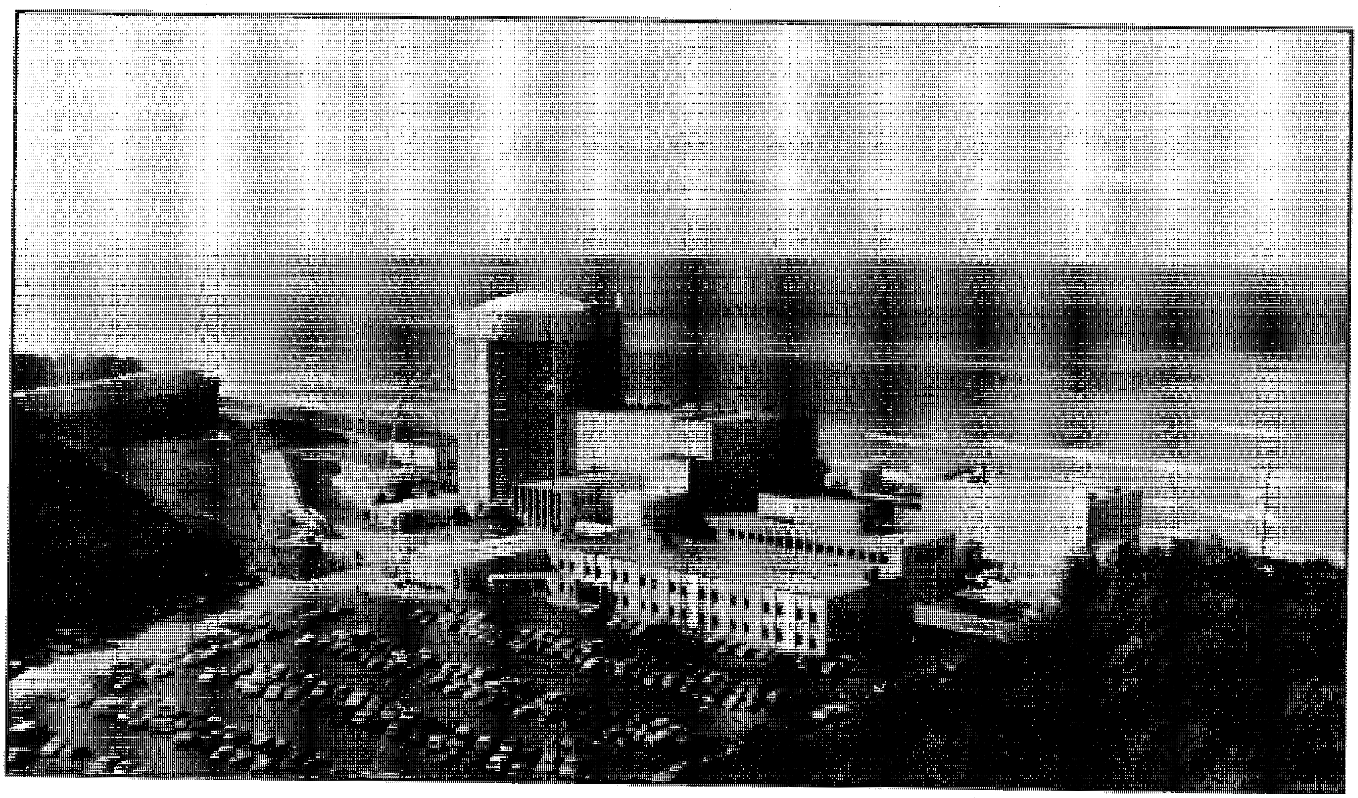
Petitioner seeks judicial review of the Commission's revisions of Part 52, issued in December 1992, that were designed to conform the Commission's regulations to the recently-enacted Energy Policy Act. See 57 FR 60975 (December 23, 1992). Petitioner apparently intends to argue both that the Commission should have followed a notice-and-comment process before issuance of a revised Part 52 and the revisions misconstrue congressional intent. No briefing schedule has yet been established.

*NRC v. FLRA*, No. 93–1704 (4th Cir.).

In April the Federal Labor Relations Authority ruled that the NRC must enter labor negotiations over union proposals regulating Inspector General investigatory interviews. In reaching this result the FLRA overruled a prior precedent insulating IG investigatory practices from labor management negotiations. At the strong urging of the NRC's Inspector General, the agency recommended to the Department of Justice that a petition for review be filed challenging the FLRA decision.

In early June, the DOJ filed a petition for review on the NRC's behalf in the United States Court of Appeals for the Fourth Circuit. The case is now being briefed.

*State of New Jersey v. Long Island Power Authority*, Civ. No. 93-4269 (GEB) (D.N.J., filed on September 21, 1993; decided on October 12, 1993; appeal pending).



**In May 1993, the Michigan Attorney General, several private citizens, and the Lake Michigan Federation sought an injunction in Federal Court preventing an NRC licensee from using an NRC-approved dry storage case for storing "spent" reactor fuel (see text). The licensee, the Consumers Power Company, planned to use the cask to store fuel from its Palisades (Mich.) plant, located on the eastern shore of Lake Michigan. The case was still in litigation at the close of the report period.**

The State of New Jersey brought this lawsuit on September 21 against the NRC, the Coast Guard, the Long Island Power Authority and the Pennsylvania Electric Company in Federal District Court in Trenton, N.J. The State sought to prevent barge shipments of slightly spent fuel from the defunct Shoreham nuclear power plant in New York to the Limerick plant near Philadelphia. New Jersey challenged the shipments as illegal under the National Environmental Policy Act and the Coastal Zone Management Act.

The District Court heard argument on the case within a day of its filing, and denied a temporary restraining order. The State then sought temporary relief unsuccessfully from the United States Court of Appeals for the Third Circuit and from Justice David Souter of the Supreme Court. In the meantime, Judge Brown scheduled an October 4 hearing on New Jersey's motion for a preliminary injunction, and on motions to dismiss that all defendants, including the NRC, had filed.

On October 12, Judge Brown denied the preliminary injunction and dismissed New Jersey's lawsuit. He agreed with the NRC's jurisdictional argument that it could be sued only in the Court of Appeals where, as here, its licensing decisions were at issue. He also agreed with the Coast Guard's argument that the Coastal Zone Management Act did not come into play because the Coast Guard had not approved the barge shipments within the meaning of the Act.

New Jersey has appealed the District Court's decision to the Third Circuit. The State has also sought relief from the Commission in a recently-filed petition that demands a halt in the barge shipments under 10 CFR 2.206 as well as a hearing under 10 CFR 2.714.

During the litigation, several shipments have proceeded successfully. The NRC is working in close collaboration with Department of Justice attorneys on the case.

## Significant Judicial Decisions

*Allied-Signal, Inc. v. NRC,* Nos. 91-1407, 91-1435, 92-1001 and 92-1019 (D.C. Cir.).

These consolidated lawsuits challenged the NRC's annual fee rule for fiscal year 1991. Fiscal year 1991 was the first year that the NRC was required by statute to collect 100 percent of its budget from its licensees through annual fees and users fees. See 1990 Omnibus Reconciliation Act, 42 USC 214. The Court of Appeals (Williams, Silberman and D. Ginsburg) issued a decision upholding the rule in part, remanding aspects of the rule for reconsideration and requiring the agency to grant an exemption to one of the petitioners.

In several respects, the court's decision is helpful to the NRC. First, the court found no Congressional directive that the NRC *must* spare from annual fees those licensees who cannot "pass through" NRC fees to customers. Second, the court approved the NRC's generic approach to fee-setting, finding unworkable petitioners' argument that the agency should adjust each licensee's annual fee according to the amount of regulatory attention it receives. Finally, the court upheld the NRC's "equal fee-perlicense" approach to allocating fees among licensees in each category (power reactors, fuel fabricators, uranium mills, etc.).

The court remanded the case to the NRC to reconsider two questions: (1) whether the agency inequitably exempted colleges and universities from fees because of an inability to "pass through" costs while declining to give the same treatment to private businesses; and (2) whether the agency unreasonably apportioned fees for low-level waste without regard to the actual waste generated by each licensee. The court pointedly declined to vacate the annual fee rule on these grounds, but left to the NRC the responsibility either to change its rule appropriately or to reaffirm it on the basis of a fresh set of reasons.

Lastly, the court ordered repayment of annual fees assessed against Combustion Engineering for one of its two licenses. The court accepted Combustion Engineering's argument that its special situation—it maintains two licenses for one uranium enrichment process—entitled it to an exemption under the NRC's "fairness and equity" standard for exemptions. Granting exemptions to CE for fiscal years 1991 and 1992 will result in refunds totalling more than one million dollars.

The NRC chose not to seek further review, and has complied with the court's instruction to refund a portion of Combustion Engineering's fiscal years 1991 and 1992 annual fees. The Commission also reconsidered the remanded issues of cost passthrough and low-level waste cost allocation, pursuant to the court's opinion in the proposed and final fiscal year 1993 fee rules.

*American Public Power Association v. NRC,* No. 92-1061 (D.C. Cir.).

Petitioners challenged the NRC's License Renewal Rule insofar as it does not require antitrust review at license renewal. Petitioners argued that the "plain meaning" of section 105(c) of the Atomic Energy Act—providing for antitrust review of license "applications"—calls for antitrust review of license renewal applications. On April 13 the Court of Appeals (Silberman, D. Ginsburg & Williams) issued an opinion rejecting petitioners' position.

The court found that the statutory language was not "all that clear." In the license renewal rulemaking the NRC had relied heavily on legislative history that appeared to rule out antitrust review in license renewal proceedings. The court ruled that the legislative history was not itself "dispositive," but concluded that the Commission's statutory construction nonetheless was "permissible," in view of "the imprecision in the statutory language and the Commission's plausible reliance" on the legislative history.

The court also rejected petitioner's complex statutory argument that at least commercial nuclear plants licensed under section 104b ("research and development" reactors) ought to be subject to antitrust review at license renewal. The court accepted the NRC's argument that the Atomic Energy Act contains a "grandfather clause" (section 102(b)) effectively immunizing section 104b plants from antitrust review absent unusual circumstances.

Finally, the court rejected petitioners' "policy arguments" as a ground for upsetting the NRC's approach. The court stated that it "suppose[d] the NRC could have accepted petitioners' arguments and determined to conduct antitrust review as a matter of discretion, but we cannot say that the Commission's construction of the statute is unreasonable."

*Critical Mass Energy Project v. NRC*, No. 90-5120 (D.C. Cir.). This is a long-running Freedom of Information Act lawsuit

seeking access to confidential "SEE IN" documents prepared by INPO and shared with the NRC. To protect the documents from disclosure, the NRC has invoked FOIA exemption 4 (protecting confidential commercial information received from private parties). In April 1991 a panel of the Court of Appeals fund the record insufficient to conclude that release of the SEE-IN documents would impede the NRC's ability to obtain full information. The panel remanded the case for further fact-finding. On September 6, however, the full Court of Appeals vacated the panel decision and granted rehearing *en banc*.

In reaching this conclusion the *en banc* court limited the reach of a longstanding D.C. Circuit precedent, *National Parks and Conversation Association v. Morton*, 498 F.2d 765 (D.C. Cir. 1974), where the court had required government agencies invoking Exemption 4 to demonstrate how disclosure would harm a government interest. The present case stops short of overruling *National Parks* outright, because of the doctrine of *stare decisis*, but limits the *National Parks* test to situations where (as in National Parks itself) the government has obtained information by compulsory process (e.g., by subpoena or regulatory requirement).

The dissenters (R. Ginsburg, J. Mikva, C. Wald and J.J. Edwards) would continue to follow *National Parks* in all cases. In their view "[t]he *National Parks* formulation fits the congressional design better than the virtual abandonment of Federal Court scrutiny approved by the court today for Government withholding of commercial or financial materials submitted voluntarily."

In March 1993 the Supreme Court denied plaintiff's petition for a writ of certiorari, bringing this lengthy litigation to a close.

*State of Michigan v. United States*, No. 91-2281 (6th Cir., June 2, 1993).

Three years ago Michigan brought suit in Federal District Court against the United States, the NRC and several other agencies. The suit challenged as unconstitutional the 1985 LowLevel Radioactive Waste Policy Act and also demanded that the NRC prepare a fresh NEPA analysis of the agency's Part 61 regulations on waste disposal. The Supreme Court resolved the constitutional question in *New York v. United States*, 112 S. Ct. 2408 (1992), where it approved the entire 1985 Act except for its "take-title" provision. The District Court threw out the Michigan's NEPA claims for lack of jurisdiction.

The United States Court of Appeals for the Sixth Circuit (Ryan, Milburn & Coffin) now has affirmed the District Court judgment. The court ruled that Michigan's challenge to the NRC's Part 61 regulations on NEPA grounds required Michigan first to ask the agency to change its regulations, followed by judicial review directly in the Court of Appeals under the exclusive jurisdiction provisions of the Hobbs Act. The court also ruled that Michigan lacked standing "to police the Nuclear Regulatory Commission" by disclaiming an attack on agency regulations and seeking NEPA relief in the abstract. The court reasoned that, for standing purposes, NEPA reviews "are only significant because of their effect on the regulations."

# Management And Administrative Services

Chapter 11

This chapter deals with internal events and activities of the NRC, such as changes on the Commission itself and in agency structure, consolidation of NRC offices in a single location, noteworthy aspects of and initiatives in personnel management, developments in the agency's information resources management program, license fees levied and collected, activities of the Office of the Inspector General, contracts awarded by the Office of Small Business and Civil Rights, and events conducted under the Federal Women's Program.

## Changes in the Commission and in NRC Structure

The term of Commissioner James R. Curtiss ended June 30, 1993, and, as of the end of calendar year 1993, the vacancy on the Commission had not been filled. (See Chapter 1 for changes in NRC senior staff.) Near the end of the report period, the Commission decided to reduce staff size and scope of activity at Region V (San Francisco), and to designate the installation a Field Office, consolidated with activities of Region IV (Dallas).

## NRC Consolidation Near Completion

At the start of fiscal year 1993, the installation of the exterior concrete pre-cast panels and windows had just commenced at the new NRC Headquarters building in Rockville, Md., called Two White Flint North (TWFN). The General Services Administration entered negotiations with the developer to lease the plaza level and garage for the NRC. A full-service cafeteria, credit union, fitness center, and employee store are planned for the plaza level. Occupancy of TWFN for more than 1,300 NRC staff was scheduled to commence in late spring of 1994.

# PERSONNEL MANAGEMENT

## 1993 NRC Staff-years Expended

During fiscal year 1993, the NRC expended a total of 3,374 staff-years in carrying out its mission. Total staff-years included permanent full-time staff, permanent part-time staff, temporary workers, and consultants. This figure excludes full-time-equivalent exempt employees, and consultants who did not work during the year.

## Recruitment

During the report period, the NRC hired 91 permanent full-time employees and lost 137 permanent full-time employees, the latter figure representing an attrition rate of 4.20 percent. During the report period, the agency participated in 90 recruitment trips, generating approximately 2,108 applications. Recruitment trips are one of three distinct modes by which recruitment is carried out, the others being advertisement and an applicant inventory/tracking system.

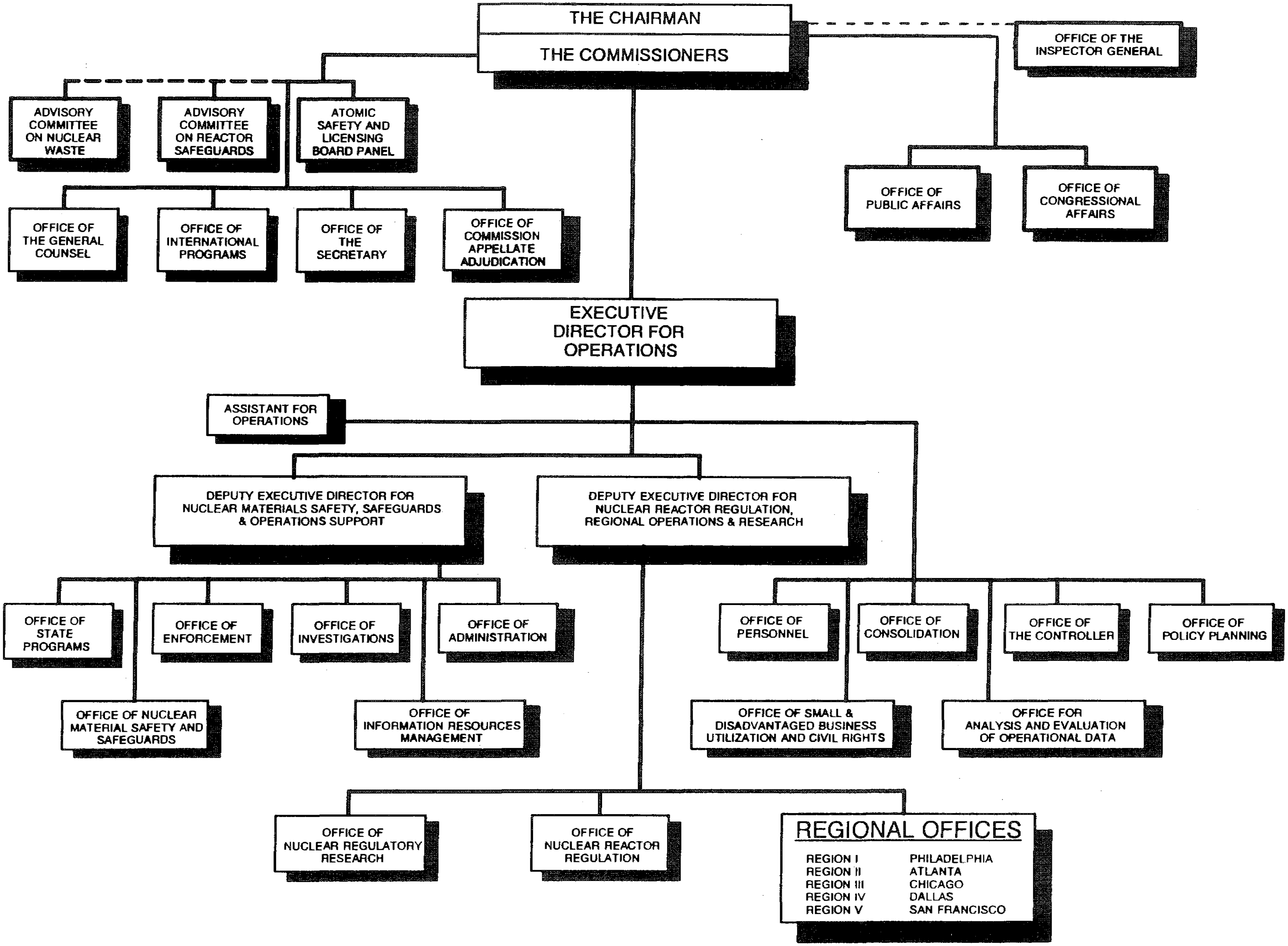
## Awards and Recognition

In fiscal year 1993, the NRC continued to give full recognition to and commendation for excellent performance on the part of agency staff. At its Annual Awards Ceremony in May, the NRC presented employees with four NRC Distinguished Service Awards and 43 Meritorious Service Awards. During fiscal year 1993, NRC employees also received 539 Performance Awards, 587 Special Act Awards, 446 High Quality Performance Salary Increases, 10 Suggestion Awards, and 375 Certificates of Appreciation. Twelve employees were nominated for awards sponsored by other Federal agencies and national organizations. Three employees were recipients of outside awards. Seven NRC executives received Presidential Meritorious Executive Rank Awards, 98 received Senior Executive Service (SES) bonuses, and 11 received SES pay level increases.

## Benefits

Employees' premiums for Federal Employees Group Life Insurance were reduced in January 1993. The

# UNITED STATES NUCLEAR REGULATORY COMMISSION



reduction resulted in the first open enrollment period in seven years, when employees could obtain or increase their insurance coverage without meeting health requirements. Some 614 employees (18 percent) used the open enrollment period.

Responding to the President's staff reduction goals, the NRC obtained approval from the Office of Personnel Management (OPM) to offer early retirement to eligible employees who retired between May 1, 1993, and February 28, 1994. The agency determined that 523 employees were eligible for early retirement and provided each an estimate of his/her annuity. The NRC also provided pre-retirement seminars and individual retirement counseling. By December 1, 1993, 17 employees had retired under the special authority.

The Voluntary Leave Transfer Program provides income protection to employees affected by a medical condition through the voluntary donation of annual leave by other employees. The experimental five-year program was to end on October 31, 1993; however, President Clinton signed the leave sharing program into law on October 8, 1993.

## Labor Relations

On October 1, 1993, the President signed Executive Order 12871 dealing with Labor-Management Partnerships in the Federal Government. The order expands the scope of bargaining and calls for a more cooperative and less confrontational relationship between labor and management. Pursuant to the order, the agency, together with the union, has set out to establish an agency partnership committee, to foster a cooperative relationship and to identify problems and propose solutions. The agency will also provide training in consensual methods of dispute-resolution, helping parties to a dispute to work together in framing possible resolutions.

## National Performance Review

The Office of Personnel (OP) has been carefully reviewing the human resources management recommendations in the National Performance Review (NPR) report, published in September 1993. While many NPR recommendations require changes in the law or in OPM regulations, others may be implemented without delay. OP has already begun to implement some of the suggested changes. Two of the changes which will have an impact on the agency are (1) the reduction of full-time equivalent resources and the ratio of supervisors and managers to employees, and (2) the elimination or reduction of personnel directives and processes. While the former change will affect the nature of supervisory relationships, the latter will provide managers with more flexibility and fewer procedural barriers in managing NRC's human resources.

## Succession Planning

The NRC reopened its Senior Executive Service (SES) Candidate Development Program for the first time in several years. Twenty-three selections were made from among more than 170 applicants. The candidates will undergo approximately one year of training, job rotations, and varied executive experiences to prepare them for potential appointment to the SES.

The NRC also announced its first Supervisory Development Program. Twenty-seven employees were selected to participate in 18 months of training and other activities that will prepare them for future positions as supervisors and managers at the NRC.

## Training and Development

During this fiscal year, OP provided more than 90 different on-site courses in the areas of probabilistic risk assessment; end-user computer applications; and executive, management, supervisory and administrative skills. The NRC also sponsored a wide variety of training and other developmental programs conducted at colleges and universities, at other government agencies, and in the private sector.

The Probabilistic Risk Assessment (PRA) Technology Transfer Program continued to offer highly specialized courses in risk assessment. During this fiscal year, 12 different courses were offered. Courses substantially revised or added to the curriculum include *Probability and Statistics for PRA* and *Integrated Reliability and Risk Analysis System* (IRRAS).

The end-user training curriculum was revised to provide instruction on the new computer resources available at the NRC. Many of these courses address how to use the AUTOS network, which links headquarters and regional employees electronically. Courses were also designed or revised for WordPerfect, Harvard Graphics, and Windows.

Two new EEO-related courses were also developed: *Age in the NRC Workforce* and, *Working with People with Disabilities*. The latter course discusses requirements of the new American with Disabilities Act.

Three major administrative training efforts during fiscal year 1993 involved financial management training, acquisition training, and ethics training. To assist Allotment Financial Managers and Funds Certifiers in understanding the new requirements of the Chief Financial Officer's Act, the NRC presented a *Financial Management Seminar*. To assist employees who procure information technology systems, a series of acquisition courses was devised. And to assist employees in complying with the new requirements for standards of ethical conduct for employees of the executive branch, a seminar on ethics was provided to more than 2,500 NRC employees.

The Individualized Learning Center continued to provide employees with convenient access to training through the latest in audio/video, computer-based, and multi-media programming. More than 180 programs were available to NRC employees in project management, communication, management and supervision, computer skills, secretarial skills, and employee assistance.

The NRC also sponsored a number of programs to help NRC employees develop the skills necessary to meet the NRC's future clerical, administrative, technical and management needs. Developmental programs sponsored by the agency include the Certified Professional Secretaries Program, the Administrative Skills Enhancement Program, the Computer Science Development Program, the Women's Executive Leadership Program, the Executive Potential Program for Mid-Level Employees, the Graduate Fellowship Program, the Intern Program, and the Senior Fellowship Program.

### Employee Assistance and Health Programs

During the fiscal year, the NRC Employee Assistance Program (EAP) staff continued to give individual counseling and referral assistance to NRC personnel with such problems as chemical dependency, job stress, chronic illness, sexual harassment, and family issues. The agency continued to make EAP services readily accessible to regional and field personnel through its interagency agreement with the Public Health Service. Supervisors were trained in recognizing and confronting troubled employees and referring them to the EAP. The agency conducted programs for employees on a variety of substance abuse and wellness topics.

Health Units operated by the Public Health Service provided a variety of health services to headquarters employees, including limited treatment and referral for on-the-job illness or injury; physical examinations for employees age 40 years and older; screening for diabetes, glaucoma, high blood pressure, and cancer; mammography testing; immunizations; and health awareness programs on topics such as AIDS, smoking cessation, carpel tunnel syndrome, and heart disease.

## NRC INFORMATION RESOURCES

### Strategic Planning

A team of senior managers and executives from across the NRC, working with the Office of Information Resources Management (IRM), has completed an NRC Information Technology Strategic Plan. In the spirit of the National Performance Review's commitment to a long term investment in change, this Strategic Plan focuses on clear goals and objectives. The Strategic Plan addresses three major areas: (1) Information Technology Program Management, (2) Information Technology Infrastructure, and (3) Information Technology Information and Applications Management.

Key recommendations of the Strategic Plan called for improving communication and coordination with the customer by establishing an Information Technology Council to advise the Director of IRM in the framing of the NRC's IT strategies; accelerating the replacement of agency workstations to ensure that the NRC's technology infrastructure is robust, reliable, and capable of supporting current and future applications needs; implementing a new document management system to meet current and anticipated programmatic needs and to enhance information availability and access; and reevaluating and improving selected agency work processes, including commercial contracting, materials licensing, and manpower tracking to support license-fee billing.

IRM fiscal year 1993 accomplishments in support of the Strategic Plan include: (1) establishment of the IT Council by Charter in May, (2) presentation of the Plan before the Commission in August (SECY-93-198), (3) progress on initiatives to support replacement of agency workstations and re-evaluate the agency's commercial contracting process, and (4) completion of an IRM skills assessment survey.

The Strategic Plan is the basis for Information Technology program guidance and activities in the NRC Five-Year Plan (FYP) and IRM's Operating Plan. The Strategic Plan satisfies IT planning requirements established in OMB Circular A-130, "Management of Federal Information Resources." In the future, the Executive Director for Operations (EDO) will give the Commission an annual update to the IT strategy in January or February of each year, so that it can be reviewed in conjunction with the FYP program guidance.

### Nuclear Documents System

The NRC employs central document processing and storage in its management of documents. The NRC's Nuclear Document Management System (NUDOCS) is the agency's centralized document data base; it provides a search and retrieval system for information associated with the licensing and inspection of nuclear reactors and materials, as well as for documentation related to nuclear regulatory, adjudicatory, and high-level and low-level waste issues.

During the report period, the NRC competed and awarded a replacement contract for document processing. Substantial increases have taken place in the volume of documents to be processed daily, in response to increased demand. As outlined in last year's report, a major planning objective was the redesign of the central processing dataentry system. The new data-entry system introduces more modern techniques for document capture and provides a more readily maintainable computer design. The new design also provides the vehicle needed to support the agency's steadily growing workload of document processing. The new data-entry system is scheduled to be brought on-line in 1994.

Throughout the report period, NRC continued to accommodate requests from members of the public for access to the publicly available portion of the NUDOCS data base. However, members of the public were advised that the NRC's Public Document Room (PDR), which is the NRC's primary source for publicly available documents, also provides an on-line Bibliographic Retrieval System for publicly available NRC documents and that this system provides a means of identifying agency documents and ordering copies of these documents. (See "Public Document Rooms," in Chapter 7.) A copy of the "Public Document Room Users' Guide," which gives information on gaining access to this system and other services available through the PDR, is provided to members of the public when they are granted access to NUDOCS. During the report period, a survey was taken among 280 members of the public who were registered users of the system at the time. Survey recipients were asked to identify their continuing need to access the system. About 9 percent of the 171 public users responding indicated that they no longer required access to the system.

## Agency Upgrade of Technology For Office Systems (AUTOS)

Work was completed during the report period on the final year of the three-year Agency Upgrade of Technology for Office Systems (AUTOS) program to improve office automation at the NRC. By fiscal year's end, more than 3,300 microcomputers had been successfully connected to the AUTOS network. In addition to office automation, AUTOS provides an integrated environment for access to agency automation resources and an important electronic link inter-connecting all NRC employees at Headquarters and in the Regions. Originally intended as a replacement for the outdated IBM 5520 and Displaywriter word-processing equipment, AUTOS constitutes an agency-wide network infrastructure supporting many of the routine administrative functions carried out daily by NRC offices. AUTOS also provides networking capability by integrating high performance engineering workstations that enable technical staff to share computer codes, data, and other resources, and to connect with public networks, National Laboratories, research institutes, and universities. To date, AUTOS has been extremely successful and gives promise of continuing to help increase individual productivity levels agency-wide.

## OFFICE OF THE INSPECTOR GENERAL

The Inspector General Act of 1978, as amended, created an independent and objective Office of the Inspector General (OIG) within the NRC. The OIG's primary mission is to assist the NRC in operating more effectively and efficiently by identifying ways to improve the agency's programs and operations through the prevention and detection of fraud, waste, and abuse. The OIG accomplishes its mission by performing audits, investigations and inspections.

The OIG audit staff conducts performance and financial audits. Performance audits focus on the NRC's administrative and program operations by evaluating how managerial responsibilities are carried out. OIG's financial audits review the NRC's internal control systems, transaction processing, and financial systems. The OIG investigative staff conducts investigations and inspections concerning the agency's programs and operations.

The NRC's OIG also shares some unique responsibilities with the agency. The NRC's primary mission is to provide adequate assurance that public health and safety is protected in the commercial use of nuclear materials and in the operation of nuclear facilities. The OIG, therefore, plays a critical role by assessing and reporting on the NRC's efforts to ensure that its safety-related programs are operating effectively.

Of particular importance is the NRC's responsibility for ensuring that individuals who identify nuclear safety concerns regarding the use of nuclear materials do not suffer adverse job actions resulting from such activities. The OIG continually assesses the NRC's efforts to combat this type of unlawful discrimination. OIG's initiatives in this area have led to congressional hearings and the formation of an agency task force.

During fiscal year 1993, the OIG (1) completed 21 audits of the NRC's operations and programs, (2) analyzed 76 contract audit reports, (3) performed 7 contract audits, and (4) closed out 100 investigations.

## OIG Fiscal Year 1993 Audits

**NRC's Management of Medical Misadministration Information.** On November 16, 1992, at a hospital in Pennsylvania, a wire containing highly radioactive material broke and was unknowingly left inside an 82-year-old patient receiving radiation therapy for cancer. This event contributed to the patient's death, which occurred five days later. The accident went undiscovered until December 1, 1992. Soon, thereafter, a newspaper published a series of articles detailing a 17-year history of other mistakes in radiation therapy that harmed patients. These events focused the attention of Congress and the public on the NRC's regulation of medical licensees.

The NRC is responsible for creating rules and programs to protect the public from undue radiation exposure. The agency also informs Congress and the public of its progress in meeting this objective. Inherent in these responsibilities is the need to analyze regulatory data, identify adverse trends, and ensure that resources are effectively managed and focused to address problem areas.

In a 1980 ruling, the NRC recognized the need to collect and analyze information on medical "misadministrations." The NRC's objective was to more accurately determine the frequency of these occurrences and to evaluate problem trends. The OIG conducted a review and found that NRC still needs to make important improvements in its management of misadministration information if it is to fully achieve this regulatory objective. For example, reported misadministrations increased nearly three-fold over the last three years compared to the average of the preceding nine years; NRC staff is unable to fully explain the increase.

The NRC recently made three significant changes to its reporting criteria, including one requiring licensees to report only the misadministrations of greatest magnitude. Even with these changes, the number of reported incidents is rising. The NRC relies on estimates (last published in 1987) of annual therapeutic procedures that use radiation. However, since the last publication, the estimates have not been revised or independently confirmed.

The OIG also found weaknesses in the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) Annual Reports, used to identify important emerging trends. Because of the manner in which AEOD prepares key information, the NRC did not detect significant inaccuracies in its 1989 and 1990 data. OIG also found that NRC staff base their regulatory decisions on case-by-case reviews and assessments of licensee events, not on misadministration trends.

These problems led OIG to conclude that the NRC has not fully met its 1980 objective. OIG believes that it is essential for the NRC to have accurate data to determine whether broad program adjustments are needed to better protect public health and safety. To correct these long-standing weaknesses, OIG recommended that the NRC independently obtain and verify the number and type of procedures involving the medical use of byproduct materials that licensees perform annually, and establish performance indicators to strengthen its regulatory oversight. NRC management agreed with OIG's recommendations.

**Audit of the NRC's Fiscal Year 1992 Financial Statements.** The OIG is required by the Chief Financial Officers Act to audit the Principal Financial Statements of the NRC at the end of each fiscal year. OIG used a contractor to perform the audit of the principal statements for the fiscal year ended September 30, 1992, including assessing the agency's internal control structure and compliance with applicable laws and regulations.

The audit findings of each major area are summarized below.

*Principal Financial Statements*

A qualified opinion was issued on the Statement of Financial Position as of September 30, 1992. The qualified opinion resulted from the incompleteness of the Property, Plant and Equipment Account, attributable to a lack of historical records. There was also a lack of assurance regarding the Department of Energy's (DOE's) compliance with laws and regulations related to NRC funds paid for work performed at DOE's national laboratories under an interagency agreement between the agencies.

*Internal Control Structure*

Four material internal control weaknesses were reported that had an effect on the financial statements: problems related to the general ledger; concern over funds spent at DOE's national laboratories; failure to bill licensees in a timely manner for services rendered; and lack of a policy for capitalizing supplies inventory, leasehold improvements, and automated data-processing software. The Chief Financial Officer (CFO) did not believe that untimely billing of fees should be characterized as a material weakness. The CFO felt that the agency had made great strides in reducing the billing cycle and would review other cost-effective methods to reduce the time required to bill for services rendered.

It was also reported that there was a need for NRC to present budgeted and actual expense information in its financial statements at the programmatic level. The NRC had elected to show budgeted and actual expenses at the appropriation level.

Six recommendations were made to improve the NRC's internal control structure. The NRC agreed with the intent of all of the recommendations.

*Financial and Administrative Accountability Improvements Needed for RES Work Funded at DOE Laboratories*

In December 1991, the OIG initiated a review of the NRC's project management practices for the services DOE provides. From fiscal year 1989 through fiscal year 1991, NRC payments to DOE totaled $187 million for laboratory work on approximately 500 annual projects managed by the NRC's Office of Nuclear Regulatory Research (RES). Of the total fiscal year 1992 RES budget of $120 million, $67 million was budgeted to pay for research projects conducted at the laboratories.

OIG found numerous deficiencies in financial and administrative accountability at RES. Taken together, these discrepancies constituted a serious management breakdown in the oversight of research projects at DOE laboratories and the stewardship of government funds. Specifically, OIG found that projects were not being closed upon completion, resulting in at least $1.4 million that was unnecessarily tied up and that could have been available for other agency uses. In addition, managers could not adequately account for NRC-funded property and equipment at DOE laboratories.

In March 1993, the acquisition value of all NRC-funded property and equipment at the laboratories was $76 million; RES funded a significant portion of this amount. OIG found that (1) funds from prior fiscal years were improperly transferred from project to project without the required approval of the Office of the Controller; (2) final DOE laboratory performance on projects was not evaluated by project managers as required; (3) RES did not use available management tools for tracking project status, leaving the agency unable to determine the completion status of 1,400 projects begun since 1975; (4) project managers did not review project costs and could not determine the financial status of their projects; (5) files were missing, incomplete, or disorganized; and (6) key personnel were not adequately trained in financial and administrative accountability.

The NRC's Executive Director for Operations (EDO) and RES management promptly and decisively acted to address and rectify many of these problems. However, the agency will remain financially vulnerable until corrective actions are completed and adequate internal controls are established. Therefore, the OIG made some further recommendations to further strengthen financial and administrative accountability. NRC agreed with these recommendations.

*Significant Weaknesses Hamper the NRC's Computer Security Program*

In fulfilling the agency's mission, NRC management and technical and administrative staff depend heavily on data obtained from a number of automated information systems maintained within the agency. Consequently, protecting these information systems and their data from theft, abuse and/or tampering is vitally important to the NRC.

During fiscal year 1991, NRC contracted with the Los Alamos National Laboratory to perform an independent compliance review of the NRC computer security program. Los Alamos provided NRC with a report in November 1991 that made 30 recommendations. The OIG examined the results of the Los Alamos review and the NRC's actions to implement the recommendations.

The OIG found that NRC had not implemented 15 of the 30 recommendations. As a result, some of the serious weaknesses in the NRC's computer security program noted by Los Alamos still existed. Important controls—such as system testing, certification, auditing, and configuration management—had not been developed; the NRC's computer security policy was outdated; and the NRC had not properly identified potential threats to its sensitive and classified information. There were also concerns about the staffing and operational placement of the NRC's computer security function.

The OIG recommended that NRC report the weaknesses in the computer security program as a material weakness under the Federal Managers' Financial Integrity Act and develop a detailed action plan to correct these weaknesses. NRC management agreed with these recommendations.

*IRM's Management of Its Contracts*

The NRC's Office of Information Resources Management (IRM) is responsible for managing the NRC's information resources in the areas of computer, telecommunications, and information services. IRM provides a wide range of services, such as information systems development and maintenance, and the acquisition, management, and support of information resources. In fiscal year 1992, IRM's budget for contractual program support was approximately $46 million; nearly $37 million of this amount was spent on contracts. The remainder was spent on purchase orders and interagency agreements with DOE.

The OIG reviewed IRM's processes, guidelines, and controls for managing contracts used to carry out its mission and assessed IRM's effectiveness in overseeing contracts for microcomputer support and systems development and maintenance.

The OIG found that IRM had not yet established office-wide policies and procedures covering the use and management of contracts; had not always adhered to prescribed procurement regulations and authority limits; and had not consistently performed requirements analyses. As a result, in some instances, IRM had exceeded its procurement authority, had made unauthorized commitments that required ratification by agency contracting officials, and, in one case, had attempted to make a major purchase without the approval of contracting officials.

Agency officials agreed with OIG's recommendations to strengthen IRM's overall contract management.

## OIG Fiscal Year 1993 Investigations

**Medical Misadministration.** An investigation was initiated based on an NRC Incident Investigation Team's (IIT's) examination of therapy misadministration and loss of an iridium-192 source at the Indiana Regional Cancer Center, Indiana, Pa. The investigation concentrated in five areas related to NRC operations.

The OIG found that existing NRC policy guidance for licensing of high dose rate (HDR) remote after-loading devices was not followed by the Regional Office in handling some of Oncology Services Corporation's (OSC's) licensing actions. And some license amendments were issued despite knowledge by some NRC staff of policy discrepancies or lack of guidance on various amendments. Furthermore, the investigation confirmed that an NRC section chief made an inappropriate remark concerning a licensee. The OIG also found deficiencies in Region I's handling of an allegation against OSC, which included a lack of documentation for allegation resolution, a lack of adequate issue identification, and inappropriate allegation disclosure to the licensee.

Questions about OSC's license and transportation of the HDR device were raised in 1991. The NRC staff conducted inadequate inquiries into these concerns and allowed the licensee to continue operating without restriction. The investigation determined that the HDR device had never been evaluated for portable use. Also, NRC management, despite assurances to the contrary, was unaware that the licensee's Radiation Safety Officer did not possess the required training for installing the device following transportation. The OIG also found deficiencies in the Headquarters' system for tracking regional requests for policy guidance and technical assistance.

Because of this incident (and others nationwide) involving the misadministration of nuclear medicine, the Senate Committee on Governmental Affairs, chaired by Senator John Glenn of Ohio, held hearings in May 1993. These hearings addressed the effectiveness of the NRC's regulatory efforts concerning radioactive pharmaceuticals.

**OIG Participation With Incident Investigation Team.** The NRC's EDO establishes Incident Investigation Teams (IITs) for the purpose of performing single-agency investigations of significant events and to determine whether NRC activities preceding and contributing to the event were timely and adequate. As part of an ongoing program, the OIG participates in agency IITs.

In February 1993, an intruder drove a vehicle through an open entry point at the north gate of the Three Mile Island (Pa.) nuclear power plant. The intruder continued past the entrance gate and headed toward the reactor containment building. The intruder's vehicle crashed into the protected area fence and then through a roll-up door into the turbine building. The Federal Bureau of Investigation, the State Police, an Army Explosive Ordnance Disposal Team, and the nuclear plant's site protection force responded to the subsequent alert. The intruder was apprehended within four hours of the incident.

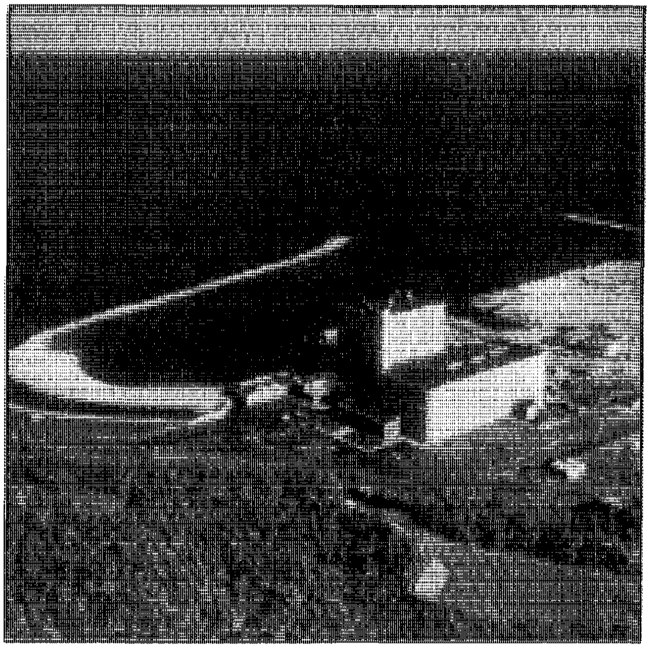
An observer from OIG joined the IIT at the plant for the on-site portion of the investigation. The final IIT report recommended that the design-basis threat, as defined by the NRC, be re-evaluated. At the time the event occurred, the utilities were not required to protect against the threat of an individual driving an explosives-laden vehicle into the protected area. Subsequent to this event, the Commission adopted a recommendation for such protection.

**Pilgrim Licensee Allegations.** The OIG initiated an investigation based on information from Citizens Urging Responsible Energy (CURE). CURE asserted that NRC did not ensure that the Boston Edison Company (BECo), licensee for the Pilgrim (Mass.) nuclear power plant, corrected identified problems. CURE supported its allegations by providing information about safety-related technical problems at PNPS that were not properly handled by the NRC.

The OIG investigation concentrated on how NRC staff conducted inspection activities and developed findings and conclusions; no re-inspections were made, nor were the technical assessments made by NRC staff questioned. The investigation did not establish wrongdoing concerning issues surrounding the PNPS restart in 1988, and Confirmatory Action Letter 86–10. The investigation did not reveal sufficient evidence to support the assertion that the NRC deliberately minimized specific problems at PNPS. The allegations concerned discrepancies between NRC reports and BECo documents related to certain events at PNPS.

The OIG could not fully investigate each allegation because the inspectors were unable to recall the events. The failure of inspectors to retain documentation of field activities showed the NRC's vulnerability when questioned about activities and findings regarding past events.

Nuclear Utility Terminates Employment of NRC Allegers. In a 1991 investigation, three security employees at a nuclear power plant reported allegations of misconduct by NRC employees to the OIG. At the same time, these security employees reported numerous safety concerns to the NRC. During an NRC inspection, a number of the safety concerns at the power plant were substantiated. Likewise, the OIG investigation validated certain allegations against NRC employees. Subsequent to the 1991 inspection and investigation, the utility terminated the security employees' positions, allegedly in retribution for reporting safety concerns to the NRC. Based on this allegation of retribution, the OIG initiated another investigation.



Among the NRC's Inspector General investigations during fiscal year 1993 was one alleging that the NRC had not ensured that its licensee had corrected previously identified problems at the Pilgrim (Mass.) nuclear power plant, shown above. The investigation did not disclose sufficient evidence to support the assertion that the NRC had deliberately minimized specific problems at the plant (see text), which is located on Plymouth Bay, 35 miles south of Boston.

During OIG interviews, utility officials maintained that the terminations resulted from a reorganization of the security department. OIG determined that the process used by the utility to justify the terminations was prejudicial to the allegers. OIG noted that several utility managers involved in the termination decision were aware that security employees had reported allegations to the NRC and to a utility investigative group.

OIG referred this investigation to the NRC for possible enforcement action. OIG also referred this case to the U.S. Department of Justice for prosecutive review.

## Other OIG Activities

**Thermo-Lag fire barrier.** One of the more significant issues over the past year was an allegation that Thermo-Lag, the fire barrier material used in 80 percent of the nation's nuclear power plants to protect safety and emergency shutdown electrical circuits from fire, is inadequate. An OIG inspection revealed that the agency relied on unverified nuclear industry test results when it originally approved the use of this firebarrier material. In 1992, NRC informed the industry that Thermo-Lag should be treated as inoperable.

Besides the inspection addressing the adequacy of NRC staff performance related to the acceptance and review of Thermo-Lag, the investigative staff—in coordination with the NRC's Office of Investigations—initiated a related investigation to determine if there was any criminal activity by the NRC, licensee employees, or the manufacturer in connection with the testing of this material. Although the investigation is still ongoing, the inspection has concluded that NRC staff did not adequately review the capability of this fire barrier material to meet NRC fire endurance requirements. Had the staff thoroughly reviewed test reports submitted by industry or verified test procedures and results, several problems with the test program and the material would have been discovered.

On March 3, 1993, the OIG testified at a hearing convened by the U.S. House of Representatives Subcommittee on Oversight and Investigations, Committee on Energy and Commerce. The purpose of the hearing was to review the NRC's actions to ensure that nuclear power plants employ adequate passive fire protection for backup emergency electrical systems designed to safely shut down a plant during a fire. This hearing was based on the August 1992 OIG inspection report of the staff's acceptance and review of Thermo-Lag. The OIG presented the Subcommittee with the findings of the inspection report and the NRC's subsequent efforts to address these problems.

**Whistleblower Protection Program.** The NRC has responsibility for regulating the operation of nuclear power plants and the activities of nuclear materials licensees through inspections and audits. The magnitude of licensed activities is so extensive that the NRC can inspect only a fraction of them; therefore, the NRC relies on licensee and contractor employees to report safety concerns to both the licensee and NRC. If employees are subject to retaliation for reporting these concerns, there can be serious safety consequences.

In July 1993, the NRC OIG completed an inspection of the adequacy of the NRC's response to whistleblower retaliation complaints. The inspection determined that the NRC's process for handling allegations of retaliation does not provide adequate protection for whistleblowers. A subsequent hearing on this issue occurred in July before the Senate Subcommittee on Clean Air and the Environment. The inspection and congressional hearing prompted the establishment of a Special Review Team for Reassessment of the NRC Program for Protecting Allegers Against Retaliation. Concurrent with this action, the OIG initiated a review of past complaints to further evaluate this process.

# OFFICE OF SMALL BUSINESS AND CIVIL RIGHTS

## Small and Disadvantaged Business Utilization Program

The Small and Disadvantaged Business Utilization Program annually establishes procurement preference goals, in compliance with provisions of Public Law 95–507, amending the Small Business Investment Act of 1957. The following is a summary of the agency's estimated and actual contract awards during fiscal year 1993.

- It was estimated that $70,000,000 in total prime contracts would be awarded during fiscal year 1993. The actual total for prime contract awards was $86,485,685.

- It was estimated that small business prime awards would be $34,000,000, or 48.57 percent of the total estimate. The actual achievement for small business prime awards was $40,361,012, or 46.67 percent, of the actual dollar awards, reflected in the previous item.

- The NRC estimated that awards to "8(a) firms" would be $15,000,000, or 21.43 percent, in fiscal year 1993. Awards to "8(a) firms" were actually $20,135,208, or 23.28 percent, of the actual dollar awards of all prime contracts, regardless of dollar value.

- The goal for prime contract awards to small disadvantaged business firms other than "8(a) firms" was $1,000,000, or 1.43 percent. The actual achievement was $181,422, or 0.21 percent, of the dollars reported in the first item, above.

- The estimate for prime contract awards to small business concerns owned and operated by women was $3,200,000, or 4.57 percent. Awards to such firms came to $1,503,305, or 1.74 percent, of the total dollar amount of all prime contacts, regardless of dollar value.

- The NRC's total subcontract goal in fiscal year 1993 was $3,150,000. The NRC's actual subcontract dollar awards were $3,025,170.The goal for subcontract awards to small business was $2,200,000, or 69.84 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $2,150,160, or 71.08 percent, of total subcontracts awarded.

- The goal for subcontract awards to small disadvantaged businesses was $410,000, or 13.02 percent. Subcontracting awards to small disadvantaged businesses actually totaled $376,000, or 12.43 percent of total subcontract dollars awarded.

During the report period, 125 interviews were conducted with firms wanting to do business with the NRC, and 30 follow-up meetings were arranged with NRC technical personnel. The staff of the NRC's Office of Small Business and Civil Rights (SBCR) also participated in five major small business conferences. Most noteworthy among these were the Small Business Development Week, in May 1993, and the Minority Enterprise Development Week, in September 1993.

## Civil Rights Program

Commission briefings were held on the status, problems and progress of the NRC EEO program, in accordance with the provisions of the Energy Reorganization Act of 1974, as amended. Discussions included issues of concern identified by the various EEO constituency group committees, and objectives and initiatives of the agency management to promote equal employment opportunity and affirmative action.

The Office of Small Business and Civil Rights sponsored a three-day advanced training course for headquarters and regional EEO Counselors at Hunt Valley, Md. The training was developed and conducted by the Boston, Mass., firm of Delaney, Zorn and Seigel, with presentations by NRC staff from SBCR, and the Offices of Personnel and the General Counsel. The primary focus of the course was the use of alternative dispute resolution techniques in the informal resolution of complaints of employment discrimination, and a review of new Equal Employment Opportunity Commission (EEOC) regulations for processing complaints of discrimination.

The staff of SBCR participated in periodic training courses for NRC employees concerning various aspects of the EEO program.

The annual accomplishment report for the NRC's Multiyear Affirmative Action Program was submitted to the EEOC. The EEOC noted that although there continues to be under-representation of certain minority groups in particular occupations, the NRC over the past five years has successfully increased the representation of minorities and women in most of the major occupations of general engineers, nuclear engineers, general physicists, health physicists, and management and program analysts.

During fiscal year 1993, the agency's EEO Counselors made 131 contacts for the purpose of counseling agency employees. Of these cases, formal complaints ensued in only 12, or 9 percent; this result speaks well for the effectiveness of the counseling process, and for the cooperation which exists between the managers, supervisors, counselors and complainants.

During the report period, the Office of Small Business and Civil Rights cosponsored with the EEO Advisory Committees several special programs in observance of specific EEO-related events. In fiscal year 1993, events were held celebrating National Disability Employment Awareness Month, Hispanic Heritage Month, Black History Month, Women's History Month, and Asian Pacific-American Heritage Month.

## Federal Women's Program

The agency continues to highlight the contributions women have made throughout history in every aspect of society. To commemorate National Women's History Month, programs were held in March 1993 throughout the agency depicting the contributions women have made to American history. A special program was held at Headquarters with the theme "Discover a New World: Women's History." As part of this program, guest speakers from The American Historical Theatre dramatized the lives of four outstanding leaders in the suffragist movement: Elizabeth Cady Stanton, Sojourner Truth, Margaret Brent and Alice Paul. The presentation was highly commended by all who attended.

Also in connection with this observance, dramatic posters were placed on display at the six headquarters buildings depicting contributions by women of all ethnic groups and in all areas of endeavor. These posters have been handsomely framed and are on display in the corridors of the Office of Small Business and Civil Rights to serve as a daily source of education and inspiration to the NRC staff. The regional Federal Women's Program representatives, as well as the headquarters Federal Women's Program Advisory Committee, sponsored numerous "lunch-time seminars" on career opportunities, and also on women's health issues.

During fiscal year 1993, one woman was selected for the SES and one for the Senior Level System (SLS). The women's SES "feeder" population increased by 13: two women were promoted to GG-13, nine were promoted to GG-14 and two were promoted to GG-15. Women constituted 37 percent of those taking advantage of training opportunities offered during the fiscal year and 43 percent of the rotational assignments. Women were very competitive in selections for the supervisory development programs: two women were selected for the Women's Executive Leadership Program; two of the five selectees for the Executive Potential Program were women; 11 of the 27 selectees for the NRC Supervisory Development Program were women; and five women were selected for the NRC Senior Executive Service Candidate Development Program.

The Federal Women's Program Annual Working Conference was held in Las Vegas, Nev., in conjunction with the 24th National Training Program for Federally Employed Women. The NRC chose as the theme for this working conference, "Helping to Keep the NRC Environment Conducive to the Growth and Development of All Employees." The Acting Federal Women's Program Manager and 12 employees from the Headquarters Federal Women's Program Advisory Committee, as well as regional Federal Women's Program representatives, participated in this working conference. The conference provided an opportunity for headquarters and regional Federal Women's Program representatives to get to know each other and develop agency-wide goals for the Federal Women's Program.

# Appendix 1

# NRC Organization

(As of December 31, 1993)

## COMMISSIONERS

Ivan Selin, Chairman
Kenneth C. Rogers
Forrest J. Remick
E. Gail de Planque

### The Commission Staff

Office of Commission Appellate Adjudication, Stephen G. Burns, Director
Office of Congressional Affairs, Dennis K. Rathbun, Director
General Counsel, William C. Parler
Office of the Inspector General, David C. Williams, Inspector General
Office of International Programs, Carlton R. Stoiber, Director
Office of Public Affairs, William M. Beecher, Director
Secretary of the Commission, Samuel J. Chilk

### Other Offices

Advisory Committee on Nuclear Waste, Dr. Martin J. Steindler, Chairman
Advisory Committee on Reactor Safeguards, Dr. J. Ernest Wilkins, Jr., Chairman
Atomic Safety & Licensing Board Panel, B. Paul Cotter, Jr., Chief Administrative Judge

## EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, James M. Taylor
Deputy Executive Director for Nuclear Reactor Regulation,
Regional Operations and Research, James H. Sniezek
Deputy Executive Director for Nuclear Materials Safety,
Safeguards and Operations Support, Hugh L. Thompson, Jr.
Assistant for Operations, James L. Blaha

### Program Offices

Office of Nuclear Material Safety and Safeguards, Robert M. Bernero, Director
Office of Nuclear Reactor Regulation, Thomas E. Murley, Director
Office of Nuclear Regulatory Research, Eric S. Beckjord, Director

### Staff Offices

Office of Administration, Patricia G. Norry, Director
Office for Analysis and Evaluation of Operational Data, Edward L. Jordan, Director
Office of Consolidation, Michael L. Springer, Director
Office of the Controller, Ronald M. Scroggins, Controller
Office of Enforcement, James Lieberman, Director
Office of Information Resources Management, Gerald F. Cranford, Director
Office of Investigations, Ben B. Hayes, Director
Office of Personnel, Paul E. Bird, Director
Office of Policy Planning, Richard H. Vollmer, Director
Office of Small Business and Civil Rights, Vandy L. Miller, Acting Director
Office of State Programs, Richard L. Bangart, Director

### Regional Offices

Region I—Philadelphia, Pa., Thomas T. Martin, Regional Administrator
Region II—Atlanta, Ga., Stewart D. Ebneter, Regional Administrator
Region III—Chicago, Ill., John B. Martin, Regional Administrator
Region IV—Dallas, Tex., James L. Milhoan, Regional Administrator
Region V—San Francisco, Cal., Bobby H. Faulkenberry, Regional Administrator

**The NRC** is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended; the Energy Reorganization Act of 1974, as amended; the Nuclear Nonproliferation Act of 1978; and in accordance with the National Environmental Policy Act of 1969, as amended, and other applicable statutes. These responsibilities include protecting public health and safety, protecting the environment, protecting and safeguarding materials and plants in the interest of national security, and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience; and confirmatory research. The Commission itself is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

**The Executive Director for Operations** is the chief operational, financial, and administrative officer of the Commission and is authorized and directed to discharge such licensing, regulatory, financial, and administrative functions of the NRC and to take actions as are necessary for day-to-day operations of the agency. The Executive Director for Operations (EDO) supervises and coordinates policy development and operational activities of EDO staff and program offices, and implements Commission policy directives pertaining to these offices.

**The Office of Nuclear Material Safety and Safeguards** licenses, inspects, and regulates facilities and materials associated with processing, transporting and handling nuclear materials, as well as the disposing of nuclear waste, and regulating uranium recovery facilities. The Office also regulates related facility decommissioning. The safeguards staff of the Office reviews and assesses protection against potential threats, thefts and sabotage for licensed facilities, working closely with other NRC offices in coordinating safety and safeguards programs and in recommending research, standards and policy options necessary for their successful operation.

**The Office of Nuclear Reactor Regulation** ensures the public health and safety through licensing and inspection activities at all nuclear power reactor facilities in the United States. The Office oversees all aspects of licensing and inspection of manufacturing, production, and utilization facilities (except for facilities reprocessing fuel and performing isotopic fuel enrichment), and receipt, possession and ownership of source, byproduct, and special nuclear material used or produced at facilities licensed under 10 CFR Part 50. The Office develops policy and inspection guidance for programs assigned to the Regional Offices, and assesses the effectiveness and uniformity of the Regions' implementation of those programs. The Office identifies and takes action in coordination with the Regional Offices regarding conditions and licensee performance at such facilities that may adversely affect public health and safety, the environment, or the safeguarding of nuclear facilities, and assesses and recommends or takes action in response to incidents or accidents. The Office is responsible for licensing issues and regulatory policy concerning reactor operators, including the initial licensing examination and requalification examinations; emergency preparedness, including participation in emergency drills with Federal, State, and local agencies; radiation protection; security and safeguards at such facilities, including fitness for duty; and the inspection of nuclear supplier facilities. The Office also conducts technical review, certification, and licensing of advanced nuclear reactor facilities and renews current power reactor operating licenses.

**The Office of Nuclear Regulatory Research** plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of safety issues for nuclear power plants and other facilities regulated by the NRC. It develops and promulgates all technical regulations; coordinates research activities within and outside the NRC, including appointment of staff to committees and conferences; and coordinates national volunteer standards efforts including appointment of staff to committees.

**The Regional Offices** are under the supervision and direction of the Executive Director for Operations and carry out NRC regulatory programs originating in the various Headquarters Offices.

## THE COMMISSION STAFF

**The Office of Commission Appellate Adjudication** is responsible for monitoring cases pending before presiding officers; for providing the Commission with an analysis of any adjudicatory matter requiring a Commission decision (e.g., petitions for review of Initial Licensing Board decisions, certified questions, interlocutory referrals, stay requests), including available options; for the drafting of any necessary decisions, pursuant to the Commission's guidance, after presentation of options; and for consulting with the Office of the General Counsel in identifying options to be presented to the Commission and in drafting the final decision to be presented to the Commission.

**The Office of Congressional Affairs** provides advice and assistance to the Chairman, Commission and NRC staff on all NRC relations with Congress and views of Congress toward NRC policies, plans and activities; maintains liaison with Congressional committees and members of Congress on matters of interest to the NRC; serves as primary contact for all NRC communications with Congress, reviewing and concurring in all outgoing correspondence to members of Congress; coordinates NRC internal activities with Congress; plans and develops NRC's legislative program; and monitors legislative proposals, bills and hearings.

**The Office of the General Counsel** directs matters of law and legal policy, providing opinions, advice, and assistance to the Commission and staff with respect to all activities of the agency.

**The Office of the Inspector General** conducts investigations and audits directed principally toward improving program management, ensuring the integrity of the NRC's regulatory programs, and preventing and identifying fraud, waste, and abuse in the agency's programs and operations.

**The Office of International Programs** provides advice and assistance to the Chairman, Commission and NRC staff on international issues. The office formulates and recommends policies concerning nuclear exports and imports, international safeguards, international physical security, non-proliferation matters, and international cooperation and assistance in nuclear safety and radiation protection. The office plans, develops and implements programs to carry out policies established in these areas; plans, develops and manages international nuclear safety information exchange programs; and coordinates international research agreements. The office obtains, evaluates and uses pertinent information from other NRC and U.S. Government offices in processing nuclear export and import license applications; establishes and maintains working relationships with individual countries and international nuclear organizations, as well as other U.S. Government agencies; and assures that all international activities carried out by the Commission and staff are properly coordinated internally and Government-wide and are consistent with NRC and U.S. policies.

**The Office of Public Affairs** develops policies, programs and procedures for informing the public of NRC activities; prepares, clears and disseminates information to the public and the news media concerning NRC policies, programs and activities; keeps NRC management informed on media coverage of activities of interest to the agency; plans, directs and coordinates the activities of public information staffs located at the Regional Offices; conducts a cooperative program with the schools; and carries out assigned activities in the area of consumer affairs.

**The Office of the Secretary** provides executive management services to support the Commission and to implement Commission decisions; advises and assists the Commission and staff on planning, scheduling, and conducting Commission business; prepares the Commission's meeting agenda; codifies Commission decisions in memoranda directing staff action; monitors staff compliance with pending actions, and tracks commitments through the automated tracking system; manages the staff paper and COMSECY systems; initiates and monitors the status of office automation initiatives into the Commission's administrative system; processes and controls Commission correspondence; maintains the Commission's official records and acts as Freedom of Information coordinator for Commission records; maintains the official Commission adjudicatory and rulemaking dockets and serves Commission and Atomic Safety and Licensing Board issuances in all adjudicatory matters and public proceedings; directs and administers the NRC Historical Program; operates and administers the NRC Public Document Room and its Bibliographic Retrieval System for providing access to members of the public and designated foreign countries to NRC's publicly available documents; and functions as the Federal Advisory Committee Management Officer.

## SUPPORT STAFF

**The Office of Administration** directs the agency's programs for contracting and procurement; document services, including preparation and publication of the NRC's annual report to the President and the Congress, and administration of the Freedom of Information Act and Privacy Act requests; transportation services; security of personnel, facilities and information; administration of local public document rooms; rulemaking support; management of space and equipment, and other administrative services.

**The Office for Analysis and Evaluation of Operational Data** provides agency coordination for the collection, storage, and retrieval of operational data associated with licensed activities, analyzes and evaluates such operational experience and feeds back the lessons of that experience to NRC licensing, standards and inspections activities. The Office is also responsible for the NRC incident response program and the technical training center, as well as the tracking of licensee performance indicators.

**The Office of Consolidation** was created to oversee realization of the agency's longterm objective of consolidating all of the NRC's Headquarters operations at a single location; consolidation will be completed by the end of fiscal year 1994, at which time the Office will be merged with the Office of Administration.

**The Office of the Controller** develops and maintains NRC's financial management programs, including policies, procedures and standards of accounting and financial systems—such as payroll and travel expenses—and preparation of the agency budget.

**The Office of Enforcement** develops policies and programs for the enforcement of NRC requirements, manages major enforcement actions, and assesses the effectiveness and uniformity of regional enforcement actions.

**The Office of Information Resources Management** develops, provides and administers information resources of the agency in the areas of computer, telecommunications, and information services. These include data base management, office automation, computer hardware and software, systems development, computer operations, timesharing, nation-wide telecommunications equipment, the Customer Support Center, user training, document control and management, central files, records management and services, library, graphics, and other information support services to the agency.

**The Office of Investigations** conducts, supervises and assures quality control of investigations of licensees, applicants, contractors or vendors, including the investigation of all allegations of wrongdoing by other than NRC employees and contractors. The Office develops policy, procedures and standards for these activities.

**The Office of Personnel** plans and implements NRC policies, programs, and services to provide for the effective organization, recruitment, placement, utilization and development of the agency's human resources.

**The Office of Policy Planning** serves as the principal advisor to the Executive Director for Operations (EDO) and the Commission for policy planning in support of the NRC mission. The office provides the lead in the agency's Strategic Planning Process. The Director, who serves as Chair of the Steering Committee for Strategic Planning, is responsible for developing and examining long-range policy issues relevant to NRC programs. The office assesses policy issues, operational environments, and alternatives, to provide recommendations to the EDO and the Commission.

**The Office of Small Business and Civil Rights** develops and implements the NRC's program in accordance with the Small Business Act, as amended, insuring that appropriate consideration is given to small business firms, including women-owned and minority businesses. The Office develops and recommends NRC policy providing for equal employment opportunity and develops, monitors and evaluates the affirmative action program to assure compliance with the policy. The Office also serves as contact with local and national public and private organizations with related interests, and administers the Historically Black Colleges and Universities Program.

**The Office of State Programs** is responsible for establishing and maintaining good community relations between the NRC, the States, local governments, other Federal agencies, and Indian Tribe organizations; serves as primary contact for policy matters between the NRC and these groups; keeps the agency apprised of activities of these groups, as they may affect NRC, and conveys to NRC management the groups' views on NRC policies, plans and activities; coordinates liaison with other Federal Agencies through the Federal Liaison Program; administers the State Agreements Program; provides training and technical assistance to Agreement States; integrates Federal regulatory activities with the States; and maintains cooperative and liaison activities with the States.

## NRC ADVISORY COMMITTEES AND LICENSING PANELS

**The Advisory Committee on Nuclear Waste,** established in 1988, advises the Commission on nuclear waste disposal facilities.

**Advisory Committee on Medical Uses of Isotopes,** established in 1958, is composed of qualified physicians and scientists, employed under yearly persona services contracts. The committee considers medical questions referred to it by the NRC staff and gives expert opinions on the medical uses of radioisotopes. The Committee also advises the NRC staff, as required, on matters of policy.

**The Advisory Committee on Reactor Safeguards** is a statutory committee of scientists and engineers advising the Commission on safety aspects of proposed and existing nuclear facilities and on the adequacy of proposed reactor safety standards and performing such other duties as the Commission may request. The Committee conducts a continuing study of reactor safety research and submits an annual report to the Congress. The Committee also administers a fellowship program.

**The Advisory Panel for the Decontamination of Three Mile Island Unit 2,** established in October 1980, provides the NRC with views and perspectives of residents of the Three Mile Island area near Harrisburg, Pa., and affords State officials the opportunity to participate in the Commission's decision-making process regarding the cleanup of the damaged nuclear facility. The panel consists of representatives of agencies of the Commonwealth of Pennsylvania, of local government, of the scientific community, and persons having their principal place of residence in the vicinity of the Three Mile Island nuclear power plant. The panel held its last meeting during fiscal year 1993 and has been disbanded.

**The Atomic Safety and Licensing Board Panel** is a panel of lawyers and others with expertise in various technical fields from which three-member Licensing Boards are drawn to conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, amend, suspend or revoke NRC licenses.

**The Licensing Support System Advisory Review Panel,** established in 1989, advises the NRC's Licensing Support System Administrator (LSSA) and the Department of Energy (DOE) on selected aspects of the design, development and operation of the support system.

**The Nuclear Safety Research Review Committee,** established in 1988 on the recommendation of the National Research Council, provides advice to the Director of the Office of Nuclear Regulatory Research regarding the direction of NRC's nuclear safety research programs.

# Appendix 2

# NRC Committees and Boards

**Advisory Committee on Reactor Safeguards**
(Membership as of December 1993.)

The Advisory Committee on Reactor Safeguards is a statutory committee established to advise the Commission on the safety aspects of proposed and existing nuclear facilities and the adequacy of proposed reactor safety standards, and to perform such other duties as the Commission may request.

CHAIRMAN: DR. J. ERNEST WILKINS, Jr., Distinguished Professor of Applied Mathematics and Mathematical Physics, Clark Atlanta University, Atlanta, Ga.

VICE-CHAIRMAN: MR. JAMES C. CARROLL, retired Manager, Nuclear Operations Support Department, Pacific Gas & Electric, San Francisco, Cal.DR.

DR. IVAN CATTON, Professor of Engineering, Department of Mechanical, Aerospace and Nuclear Engineering, School of Engineering and Applied Science, University of California, Los Angeles, Cal.

MR. PETER R. DAVIS, President, PRD Consulting, Idaho Falls, Idaho.

THOMAS S. KRESS, Head of Applied Systems Technology Section, Oak Ridge National Laboratory, Oak Ridge, Tenn.

DR. HAROLD W. LEWIS, Professor Emeritus of Physics, Department of Physics, University of California, Santa Barbara, Cal.

MR. WILLIAM J. LINDBLAD, retired President of Portland General Electric, Portland, Ore.

MR. CARLYLE MICHELSON, retired Principal Nuclear Engineer, Tennessee Valley Authority, Knoxville, Tenn., and retired Director, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.

DR. ROBERT L. SEALE, Professor of Nuclear and Energy Engineering, Department of Nuclear and Energy Engineering, College of Engineering and Mines, University of Arizona, Tucson, Ariz.

DR. WILLIAM J. SHACK, Associate Director, Energy Technology Division, Argonne National Laboratory, Argonne, Ill.

MR. CHARLES J. WYLIE, retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.

**Atomic Safety and Licensing Board Panel**
(Membership as of December 1993)

*FULL-TIME PANEL MEMBERS:*

CHIEF ADMINISTRATIVE JUDGE B. PAUL COTTER, JR., Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—EXECUTIVE ROBERT M. LAZO, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—TECHNICAL FREDERICK J. SHON, Engineer, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE CHARLES BECHHOEFER, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE PETER B. BLOCH, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE G. PAUL BOLLWERK, III, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE JAMES H. CARPENTER, Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE RICHARD F. COLE, Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE JAMES P. GLEASON, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE CHARLES N. KELBER, Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE JERRY R. KLINE, Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE PETER S. LAM, Nuclear Engineer, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE MORTON B. MARGULIES, Chief Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda Md.

JUDGE THOMAS S. MOORE, Legal, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE THOMAS D. MURPHY, Health Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE IVAN W. SMITH, Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.

*PART-TIME PANEL MEMBERS:*

JUDGE GEORGE C. ANDERSON, Marine Biologist (retired), University of Washington, Seattle, Wash.

JUDGE GLENN O. BRIGHT, Engineer (retired), U.S. Nuclear Regulatory Commission, Norman, Okla.

JUDGE A. DIXON CALLIHAN, Physicist (retired), Union Carbide Corporation, Davidson, N.C.

JUDGE THOMAS S. ELLEMAN, Nuclear Engineer, North Carolina State University, Raleigh, N.C.

JUDGE GEORGE A. FERGUSON, Nuclear Physicist (retired), Howard University, Shady Side, Md.

JUDGE HARRY FOREMAN, Medical Doctor (retired), University of Minnesota, St. Paul, Minn.

JUDGE RICHARD F. FOSTER, Environmental Scientist, Sunriver, Ore.

JUDGE DAVID L. HETRICK, Nuclear Engineer, University of Arizona, Tucson, Ariz.

JUDGE ERNEST E. HILL, Nuclear Engineer, Hill Associates, Danville, Cal.

JUDGE FRANK F. HOOPER, Marine Biologist (retired), University of Michigan, Ann Arbor, Mich.

JUDGE ELIZABETH B. JOHNSON, Nuclear Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.

JUDGE WALTER H. JORDAN, Physicist (retired), Oak Ridge National Laboratory, Oak Ridge, Tenn.

JUDGE JAMES C. LAMB, III, Environmental Engineer, George Washington University, Charlottesville, Va.

JUDGE EMMETH A. LUEBKE, Physicist (retired), U.S. Nuclear Regulatory Commission, Chevy Chase, Md.

JUDGE KENNETH A. McCOLLOM, Electrical Engineer (retired), Oklahoma State University, Stillwater, Okla.

JUDGE MARSHALL E. MILLER, Legal (retired), U.S. Nuclear Regulatory Commission, Daytona Beach, Fla.

JUDGE PETER A. MORRIS, Physicist (retired), U.S. Nuclear Regulatory Commission, Potomac, Md.

JUDGE RICHARD R. PARIZEK, Geologist, Pennsylvania State University, University Park, Pa.

JUDGE HARRY REIN, Medical Doctor, Longwood, Fla.

JUDGE LESTER S. RUBENSTEIN, Nuclear Engineer (retired), U.S. Nuclear Regulatory Commission, Oro Valley, Ariz.

JUDGE DAVID R. SCHINK, Oceanographer, Texas A&M University, College Station, Tex.

JUDGE GEORGE F. TIDEY, Medical Doctor, University of Texas, Houston, Tex.

JUDGE SHELDON J. WOLFE, Legal (retired), U.S. Nuclear Regulatory Commission, McLean, Va.

*PROFESSIONAL STAFF:*

LEE S. DEWEY, Chief Counsel and Director, Technical and Legal Support Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JACK G. WHETSTINE, Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.

**Licensing Support System Advisory Review Panel**
(Membership as of December 1993.)

The Licensing Support System Advisory Review Panel (LSSARP) was established in 1989 to advise the NRC and the Department of Energy (DOE) on selected aspects of the design, development and operation of the Licensing Support System, currently administered by the Deputy Director of the NRC Office of Information Resources. The panel consists of representatives of the NRC, DOE, the State of Nevada, the local government of Nye County (Nev.), the National Congress of American Indians, a coalition of nuclear industry organizations, and other Federal agencies with experience with large electronic document management systems.

CHAIRMAN JOHN C. HOYLE, U.S. Nuclear Regulatory Commission.

BOYD ALEXANDER, U.S. Patent and Trademarks Office.

KIRK BALCOM, State of Nevada.

MIKE BAUGHMAN, Las Vegas, Nev.

DENNIS BECHTEL, Clark County, Nev.

STEVE BRADHURST, Nye County, Nev.

LES BRADSHAW, Nye County, Nev.

WAYNE CAMERON, White Pine County, Nev.

BARBARA CERNY, U.S. Department of Energy.

DAVID COPENHAFER, U.S. Securities and Exchange Commission.

EVE CULVERWELL, Lincoln City, Nev.

PETER CUMMINGS, Las Vegas, Nev.

BILL ELQUIST, Lander County, Nev.

ARLO FUNK, Mineral County, Nev.

PETE GOICOECHEA, Eureka County, Nev.

DANIEL GRASER, U.S. Department of Energy.

CHRISTOPHER HENKEL, Edison Electric Institute.

JUANITA HOFFMAN, Esmeralda County, Nev.

ROBERT HOLDEN, National Congress of American Indians.

ELGIE HOLSTEIN, Nye County, Nev.

FELIX KILLAR, U.S. Council for Energy Awareness.

STEVEN KRAFT, Edison Electric Institute.

JOHN LAMPROS, White Pine County, Nev.

ANTHONY LESSARD, Mineral County, Nev.

CORINNE MACALUSO, U.S. Department of Energy.

LORETTA METOXEN, National Congress of American Indians.

BRAD METTAM, Inyo County, Cal.

MALACHY MURPHY, Nye County, Nev.

JASON PITTS, Lincoln County, Nev.

JAMES REGAN, Churchill County, Nev.

JAY SILBERG, Shaw, Pittman, Potts & Trowbridge.

LENARD SMITH, Lincoln County, Nev.

HARRY SWAINSTON, State of Nevada.

## OTHER NRC ADVISORY GROUPS

### Advisory Committee on the Medical Uses of Isotopes
(Membership as of December 1993)

The Advisory Committee on Medical Uses of Isotopes (ACMUI) was established in July 1958. The ACMUI, composed of qualified physicians and scientists, considers medical questions referred to it by the NRC staff and gives expert opinions on the medical uses of radioisotopes. The ACMUI also advises the NRC staff, as required, on matters of policy. Members are employed under yearly personal services contracts.

CHAIRMAN: DR. BARRY A. SIEGEL, Professor of Radiology, Mallinckrodt Institute of Radiology.

DR. PETER R. ALMOND, University of Louisville School of Medicine, Louisville, Ky.

DR. DANIEL S. BERMAN, Cedar Sinai Medical Center, Los Angeles, Cal.

JUDITH I. BROWN, Health Policy Consultant for American Association of Retired Persons, Washington, D.C.

DR. DANIEL F. FLYNN, Department of Radiation Medicine, Massachusetts General Hospital, Boston, Mass.

DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.

DR. A. ERIC JONES, U.S. Food and Drug Administration, Rockville, Md.

DR. CAROL S. MARCUS, Asst. Chief, Nuclear Medicine, Los Angeles County Harbor-UCLA Medical Center, Torrance, Cal.

MS. JOAN A. MCKEOWN (R.T.), Director of Radiation Safety, Presbyterian-University of Pennsylvania Medical Center, Philadelphia, Pa. DR.

DR. WIL B. NELP, University of Washington, University Hospital, Seattle Wash.

MR. ROBERT M. QUILLIN, State of Colorado, Denver, Colo.

DR. JUDITH ANNE STITT, University of Wisconsin Hospital, Department of Human Oncology, Madison, Wis.

MR. DENNIS P. SWANSON, University of Pittsburgh School of Pharmacy, Pittsburgh, Pa.

### Advisory Committee on Nuclear Waste
(Membership as of December 1993.)

The Advisory Committee on Nuclear Waste reports to and advises the Nuclear Regulatory Commission on nuclear waste disposal facilities, as directed by the Commission. This includes 10 CFR Patrs 60 and 61 and other applicable regulations and legislative mandates, such as the Nuclear Waste Policy Act, the Low-Level Radioactive Waste Policy Act, and the Uranium Mill Tailings Radiation Control Act, as amended. The primary emphasis is on disposal facilities.

CHAIRMAN: DR. MARTIN J. STEINDLER, Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Ill.

VICE-CHAIRMAN: DR. PAUL W. POMEROY, President, Rondout Associates, Inc., Stone Ridge, N.Y.

DR. WILLIAM J. HINZE, Professor, Department of Earth and Atmospheric Sciences, Purdue University, West Lafayette, Ind.

### Advisory Panel For The Decontamination Of Three Mile Island Unit 2

The 10-member panel held its last meeting during fiscal year 1993. The Advisory Panel was formed by the NRC in 1980 to provide input to the Commission on major cleanup issues. The last meeting (the 78th overall) was held in Harrisburg, Pa., on September 23, 1993. Commissioner Kenneth Rogers attended the final session to express the Commission's appreciation to the Advisory Panel for their dedication and service over the past 13 years.

CHAIRMAN: ARTHUR E. MORRIS, Resident and former Mayor of Lancaster, Pa.

VICE CHAIRMAN: JOEL ROTH, Resident of Harrisburg, Pa.

JOHN LEUTZELSCHWAB, Professor of Physics, Dickinson, College, Carlisle, Pa.

ELIZABETH MARSHALL, Resident of York, Pa.

KENNETH L. MILLER, Director of the Division of Health Physics and Professor of Radiology, Milton S. Hershey Medical Center, Hershey, Pa.

FREDERICK S. RICE, Resident of Harrisburg, Pa.

GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.

THOMAS SMITHGALL, Resident of Lancaster, Pa.

ANN TRUNK, Resident of Middletown, Pa.

NIEL WALD, Professor, Department of Environmental and Occupational Health, University of Pittsburgh, Pittsburgh, Pa.

MICHAEL T. MASNIK, Designated Federal Official, Non-Power Reactors and Decommissioning Project Directorate, NRC Office of Nuclear Reactor Regulation.

LEE H. THONUS, Alternate Designated Federal Official, Non-Power Reactos and Decommissioning Projects Directorate, NRC Office of Nuclear Reactor Regulation (Region I).

**Nuclear Safety Research Review Committee**
(Membership as of December 31, 1993.)

The Nuclear Safety Research Review Committee, established in 1988 on the recommendation of the National Research Council, provides advice to the Director of the Office of Nuclear Regulatory Research regarding the direction of NRC's nuclear safety research programs.

CHAIRMAN: DR. DAVID L. MORRISON, Technical Director, Energy, Resource and Environmental Systems Division, MITRE Corporation, McLean, Va.

DR. E. THOMAS BOULETTE, Sr. Vice President, Nuclear Operations, and Station Director, Pilgrim Station, Boston Edison Co., Plymouth, Mass.

MR. SOL BURSTEIN, retired Vice President and Director of Wisconsin Energy Corp.; Vice Chairman of the Board and Director of Wisconsin Electric Co. and Wisconsin Natural Gas Co., Milwaukee, Wis.

DR. SPENCER H. BUSH, Review & Synthesis Associates, Richland, Wash.

DR. ROBERT D. HATCHER, JR., Professor, Department of Geological Sciences, University of Tennessee, and Distinguished Scientist, Environmental Sciences Division, Oak Ridge National Laboratory, Oak Ridge, Tenn.

DR. HERBERT S. ISBIN, Professor Emeritus, Department of Chemical Engineering and Materials Science, University of Minnesota, Minneapolis, Minn.

MR. EDWIN E. KINTNER, retired Executive Vice President of GPU Nuclear Corp., Parsippany, N.J.

DR. FRED J. MOLZ III, Huff Professor of Civil Engineering, Auburn University, Auburn, Ala.

DR. NEIL E. TODREAS, Professor and Head, Department of Nuclear Engineering, Massachusetts Institute of Technology, Cambridge, Mass.

DR. ROBERT E. UHRIG, Distinguished Professor of Engineering, Nuclear Engineering Department, University of Tennessee, Knoxville, Tenn., Distinguished Scientist, Instrumentation and Control Division, Oak Ridge National Laboratory, Oak Ridge, Tenn.

DR. RICHARD C. VOGEL, retired Senior Scientific Advisor, Electric Power Research Institute, Palo Alto, Cal.

# Appendix 3

# Local Public Document Rooms

Copies of most documents originating in the NRC or submitted to it for review are placed in the Commission's Public Document Room (PDR) in the Gelman Building, 2120 L Street, N.W., Washington, D.C., for public inspection. Other PDRs are maintained in the five Regional Offices (for documents related to nuclear material licenses, i.e., most byproduct and source material licenses). In addition, documents related to licensing proceedings or licensed operation of specific facilities are made available in local PDRs established in the vicinity of the proposed or existing nuclear facility. The locations of the local PDRs, the names of the persons to contact, and the names of the facilities for which documents are retained are listed below. (N.B. Updated listings of local PDRs may be obtained by writing to: Freedom of Information Act/Local Public Document Room Branch, Division of Freedom of Information and Publications Services, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.)

## ALABAMA

- Ms. Susan Todd, Head Librarian
  Athens Public Library
  405 E. South Street
  Athens, Ala. 35611
  Browns Ferry nuclear plant
  Browns Ferry low-level waste storage

- Ms. Bettye Forbus, Director
  Houston Love Memorial Library
  212 W. Burdeshaw Street
  P.O. Box 1369
  Dothan, Ala. 36302
  Jospeh M. Farley nuclear plant

- Ms. Peggy McCutchen
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte nuclear plant

## ARIZONA

- Ms. Linda Risseeuw, Librarian II
  Business and Science Division
  Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde nuclear plant

## ARKANSAS

- Ms. Frances Hager
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One nuclear plant

## CALIFORNIA

- Ms. Margaret J. Nystrom
  Documents Librarian
  Humboldt County Library
  636 F Street
  Eureka, Cal. 95501
  Humboldt Bay nuclear plant

- Ms. Judy Horn, Department Head
  University of California
  Main Library
  P.O. Box 19557
  Irvine, Cal. 92713
  San Onofre nuclear plant

- Ms. Hanne Robinson
  Central Library
  828 I Street
  Sacramento, Cal. 95814
  Rancho Seco nuclear plant

- Mr. Johanna Brown, Head
  Government Documents and Maps Dept.
  Robert E. Kennedy Library
  California Polytechnic State University
  San Luis Obispo, Cal. 93407
  Diablo Canyon nuclear plant

## COLORADO

- Ms. Sue Safarik
  Weld Library District, Lincoln Park Branch
  919 7th Street
  Greeley, Colo. 80631
  Fort St. Vrain nuclear plant

## CONNECTICUT

- Ms. Marcella Kenney, Reference Librarian
  Russell Library
  123 Broad Street
  Middletown, Conn. 06457
  Haddam Neck nuclear plant

- Dr. Paul S. Price
  Director of Learning Resources
  Three Rivers Community Technical College
  Thames Valley Campus
  574 New London Turnpike
  Norwich, Conn. 06360
  Millstone nuclear plant

## FLORIDA

- Ms. Joyce Shiver
  Coastal Region Library
  8619 W. Crystal Street
  Crystal River, Fla. 32629
  Crystal River nuclear plant

- Ms. Peggy Peterson, Librarian
  Charles S. Miley Learning Resources Ctr.
  Indian River Community College
  3209 South Virginia Avenue
  Ft. Pierce, Fla. 34981
  St. Lucie nuclear plant

- Ms. Sherry Mosley, Librarian
  Library Documents Department
  Florida International University
  University Park Miami, Fla. 33199
  Turkey Point nuclear plant

## GEORGIA

- Ms. Aloice Coleman
  Appling County Public Library
  301 City Hall Drive
  Baxley, Ga. 31513
  Edwin I. Hatch nuclear plant

- Mrs. Gwen Jackson, Librarian
  Burke County Library
  412 4th Street
  Waynesboro, Ga. 30830
  Alvin W. Vogtle nuclear plant

## ILLINOIS

- Mrs. Yvonne Jaycox,
  Assistant Librarian
  Byron Public Library District
  109 N. Franklin Street
  Byron, Ill. 61010
  Byron nuclear plant

- Mrs. Malinda Evans
  Vespasian Warner Public Library
  310 N. Quincy Street
  Clinton, Ill. 61727
  Clinton nuclear plant

- Mrs. Nancy Gillfillian
  Library Director
  Dixon Public Library
  221 Hennepin Avenue
  Dixon, Ill. 61021
  Quad Cities nuclear plant
  Sheffield low-level waste
  burial site

- Ms. Deborah Steffes
  Reference Assistant
  Morris Area Public Library District
  604 Liberty Street
  Morris, Ill. 60450
  Dresden nuclear plant
  Morris spent fuel storage facility

- Ms. Evelyn Moyle,
  Documents Librarian
  Jacobs Memorial Library
  Illinois Valley Community College
  Rural Route 1
  Oglesby, Ill. 61348
  LaSalle nuclear plant

- Ms. Mary Jane Anderson,
  Library Director
  Government Documents Collection
  Wilmington Public Library
  201 South Kankakee Street
  Wilmington, Ill. 60481
  Braidwood nuclear plant

- Ms. Tiffany Severns
  Reference Librarian
  Waukegan Public Library
  128 N. County Street
  Waukegan, Ill. 60085
  Zion nuclear plant

- Ms. Ann Bergstrom,
  Library Assistant
  West Chicago Public Library
  118 W. Washington Street
  West Chicago, Ill. 60185
  Kerr-McGee West Chicago

## IOWA

- Mr. Roger Rayborn
  Cedar Rapids Public Library
  500 1st Street, S.E.
  Cedar Rapids, Ia. 52401
  Duane Arnold nuclear plant

## KANSAS

- Ms. Nannette Martin,
  Documents Librarian
  Government Documents Dept.
  William Allen White Library
  Emporia State University
  1200 Commercial Street
  Emporia, Kans. 66801
  Wolf Creek Generating Station

- Mr. Paul Arrigo
  NRC-LPDR Documents Collection
  Washburn University School of Law
  Topeka, Kans. 66621
  Wolf Creek Generating Station

## LOUISIANA

- Mrs. Smittie Bolner, Head
  Government Documents
  Department
  Troy H. Middleton Library
  Louisiana State University
  Baton Rouge, La. 70803
  River Bend nuclear plant

- Mr. Kenneth E. Owen, Head
  Louisiana Collection
  Earl K. Long Library
  University of New Orleans
  Lakefront Drive
  New Orleans, La. 70148
  Waterford nuclear plant

- Ms. Pam Suggs, Director
  Claiborne Parish Library
  901 Edgewood Drive
  Homer, La. 71040
  Louisiana Energy
  Services, Inc., facility

## MAINE

- Ms. Sue Cereste, Assistant Librarian
  Wiscasset Public Library
  High Street
  P.O. Box 367
  Wiscasset, Me. 04578
  Maine Yankee nuclear plant

## MARYLAND

- Ms. Mildred Ward, Library Assistant
  Calvert County Public Library
  30 Duke Street
  P.O. Box 405
  Prince Frederick, Md. 20678
  Calvert Cliffs nuclear plant

## MASSACHUSETTS

- Mrs. Carol Letson
  Library/Learning Resource Center
  Greenfield Community College
  One College Drive
  Greenfield, Mass. 01301
  Yankee Rowe nuclear plant

- Ms. Grace E. Karbott,
  Reference Librarian
  Plymouth Public Library
  132 South Street
  Plymouth, Mass. 02360
  Pilgrim nuclear plant

## MICHIGAN

- Mr. David O'Brien,
  Reference Librarian
  Van Wylen Library
  Hope College
  137 E. 12th Street
  Holland, Mich. 49423
  Palisades nuclear plant

- Mr. Eric Grandstaff,
  Library Director
  North Central Michigan College
  1515 Howard Street
  Petoskey, Mich. 49770
  Big Rock Point nuclear plant

- Mr. Carl Katafiasz
  Government Documents Librarian
  Monroe County Library System
  3700 S. Custer Rd.
  Monroe, Mich. 48161
  Enrico Fermi nuclear plant

- Ms. Anne Vandermolen,
  Library Assistant
  Maud Preston Palenske
  Memorial Library
  500 Market Street
  St. Joseph, Mich. 49085
  Donald C. Cook nuclear plant

## MINNESOTA

- Mr. William L. Johnston, Librarian
  Technology and Science Department
  Minneapolis Public Library
  300 Nicollet Mall
  Minneapolis, Minn. 55401
  Monticello nuclear plant
  Prarie Island nuclear plant

## MISSISSIPPI

- Ms. Donna Janky, Director
  Judge George W. Armstrong Library
  220 South Commerce
  Natchez, Miss. 39120
  Grand Gulf nuclear plant

## MISSOURI

- Mrs. Evelyn Hillard
  Public Services Librarian
  Callaway County Public Library
  710 Court Street
  Fulton, Mo. 65251
  Callaway nuclear plant

- Ms. Loretta Ponzar
  Jefferson College Library
  1000 Viking Drive
  Hillsboro, Mo. 63050
  Combusion Engineering, Inc.
  Hematite Uranium Fuel facility

## NEBRASKA

- Mrs. Donna Ellis
  Auburn Public Library
  1118 15th Street
  P.O. Box 324
  Auburn, Neb. 68305
  Cooper nuclear plant

- Ms. Margaret Blackstone,
  Librarian
  Business, Science and
  Technology Dept.
  W. Dale Clark Library
  215 S. 15th Street
  Omaha, Neb. 68102
  Fort Calhoun nuclear plant

## NEVADA

- Ms. Susan Jarvis
  James R. Dickinson Library
  University of Nevada-Las Vegas
  4505 Maryland Parkway
  Las Vegas, Nev. 89154
  Yucca Mountain high-level
  waste geologic repository site

- Ms Janita Jobe
  Government Publications Dept.
  University Library
  University of Nevada-Reno
  Reno, Nev. 89557
  Yucca Mountain high-level
  waste geologic repository site

## NEW HAMPSHIRE

- Ms. Pamela Gjettum
  Exeter Public Library
  Founders Park
  Exeter, N.H. 03833
  Seabrook nuclear plant

## NEW JERSEY

- Ms. Ida Mangifesta
  Pennsville Public Library
  190 S. Broadway
  Pennsville, N.J. 08070
  Hope Creek nuclear plant

- Ms. Elizabeth C. Fogg, Director
  Salem Free Public Library
  112 West Broadway
  Salem, N.J. 08079
  Salem nuclear plant

- Ms. Ro Kamsar
  Reference Librarian
  Reference Department
  Ocean County Library
  101 Washington Street
  Toms River, N.J. 08753
  Oyster Creek nuclear plant

## NEW YORK

- Mr. Alexander Beattie
  Reference and Documents
  Department
  Penfield Library
  State University of New York
  Oswego, N.Y. 13126
  James A. Fitzpatrick
  nuclear plant
  Nine Mile Point nuclear plant

- Ms. Carolyn Johnson, Head
  Business and Social Science Division
  Rochester Public Library
  115 South Avenue
  Rochester, N.Y. 14610
  Robert Emmet Ginna
  nuclear plant

- Mr. Erich Mayer, Assistant
  Librarian
  Buffalo and Erie County
  Public Library
  Lafayette Square
  Buffalo, N.Y. 14203
  West Valley Demonstration
  Project

- Ms. Laurie Strick
  Shoreham-Wading River
  Public Library
  Route 25 A
  Shoreham, N.Y. 11786
  Shoreham nuclear plant

- Mr. Oliver F. Swift
  Municipal Reference Librarian
  White Plains Public Library
  100 Martine Avenue
  White Plains, N.Y. 10601
  Indian Point nuclear plant

## NORTH CAROLINA

- Ms. Dawn Hubbs, Documents
  Librarian
  J. Murrey Atkins Library
  University of North Carolina
  at Charlotte—UNCC Station
  Charlotte, N.C. 28223
  William B. McGuire nuclear plant

- Ms. Marsha Proctor, Head
  Adult Services
  Cameron Village Regional Library
  1930 Clark Avenue
  Raleigh, N.C. 27605
  Shearon Harris nuclear plant

- Mrs. Eileen Brown
  Reference/Documents Librarian
  William Madison Randall Library
  University of North Carolina
  at Wilmington
  601 S. College Road
  Wilmington, N.C. 28403-3297
  Brunswick steam electric plant

## OHIO

- Ms. Donnie Potelicki, Director
  Garfield Heights Branch Library
  5409 Turney Road
  Garfield Heights, Ohio 44125
  Chemetron Corporation

- Ms. Ann Freed, Reference Librarian
  Perry Public Library
  3753 Main Street
  Perry, Ohio 44081
  Perry nuclear plant

- Mrs. Julia Baldwin, Documents
  Librarian
  Government Documents Collection
  William Carlson Library
  University of Toledo
  2801 West Bancroft Avenue
  Toledo, Ohio 43606
  Davis-Besse nuclear plant

## OKLAHOMA

- Ms. O.J. Grosclaude
  Stanley Tubbs Memorial Library
  101 E. Cherokee St.
  Sallisaw, Okla. 74955
  Kerr-McGee Sequoyah

## OREGON

- Mr. Joseph J. Kohut
  Science Librarian
  Branford P. Millar Library
  Portland State University
  P.O. Box 1151
  10th and Harrison
  Portland, Ore. 97207
  Trojan nuclear plant

## PENNSYLVANIA

- Ms. Mary Ann Paulin,
  Reference Librarian
  B.F. Jones Memorial Library
  663 Franklin Avenue
  Aliquippa, Pa. 15001
  Beaver Valley nuclear plant

- Ms. Judy Weinrauch
  Government Publications Section
  State Library of Pennsylvania
  Walnut Street and
  Commonwealth Avenue
  Box 1601
  Harrisburg, Pa. 17105
  Three Mile Island nuclear plant
  Peach Bottom nuclear plant

- Ms. Vicki Held
  Apollo Memorial Library
  219 N. Pennsylvania Avenue
  Apollo, Pa. 15613
  Babcock & Wilcox Parks
  Township and B&W Apollo

- Mr. Scott Elmer
  Pottstown Public Library
  500 High Street
  Pottstown, Pa. 19464
  Limerick nuclear plant

- Mr. Ernest Fuller
  NRC Materials Aide
  Saxton Community Library
  911 Church Street
  Saxton, Pa. 16678
  Saxton nuclear experimental
  facility

- Ms. Sandra Schimmel
  Reference Librarian
  Reference Department
  Osterhout Free Library
  71 South Franklin Street
  Wilkes-Barre, Pa. 18701
  Susquehanna steam
  electric station
  Susquehanna low-level
  waste storage

## RHODE ISLAND

- Ms. Ann Crawford, Director
  Cross Mill Public Library
  4417 Old Post Road
  Charlestown, R.I. 02813
  Wood River Junction

## SOUTH CAROLINA

- Mrs. Margaret Cannon, Director
  Barnwell County Public Library
  Hagood Avenue
  Barnwell, S.C. 29812
  Barnwell reprocessing plant
  Barnwell low-level waste
  burial site

- Ms. Liz Watford, Librarian
  Nuclear Information Depository
  Hartsville Memorial Library
  220 N. Fifth Street
  Hartsville, S.C. 29550
  H.B. Robinson nuclear plant
  Robinson independent spent
  fuel storage

- Mrs. Mary Mallaney
  Assistant Reference Librarian
  York County Library
  138 East Black Street
  P.O. Box 10032
  Rock Hill, S.C. 29730
  Catawba nuclear plant

- Ms. Joyce Lusk, Librarian
  Oconee County Library
  501 W. South Broad Street
  Walhalla, S.C. 29691
  Oconee nuclear plant

- Ms. Sarah D. McMaster, Director
  Fairfield County Library
  300 Washington Street
  Winnsboro, S.C. 29180
  Virgil C. Summer nuclear plant

## TENNESSEE

- Ms. Patricia Maroney, Head
  Business, Science and
  Technology Dept.
  Chattanooga-Hamilton County
  Library
  1001 Broad Street
  Chattanooga, Tenn. 37402
  Sequoyah nuclear plant
  Watts Bar nuclear plant
  TVA Sequoyah low-level
  waste storage

## TEXAS

- Mrs. Terry Wang
  Library--Documents
  University of Texas
  at Arlington
  702 College
  P.O. Box 19497
  Arlington, Tex. 76019
  Comanche Peak steam
  electric station

- Ms. Patsy G. Norton, Director
  Wharton County Junior College
  J.M. Hodges Learning Center
  911 Boling Highway
  Wharton, Tex. 77488
  South Texas Project

## VERMONT

- Mr. Jerry Carbone
  Brooks Memorial Library
  224 Main Street
  Brattleboro, Vt. 05301
  Vermont Yankee nuclear plant

## VIRGINIA

- Mr. Gregory A. Johnson
  Senior Public Services Assistant
  Manuscripts Dept.
  Alderman Library
  University of Virginia
  Charlottesville, Va. 22903
  North Anna nuclear plant

- Mr. Alan Zoellner
  Documents Librarian
  Swem Library
  College of William and Mary
  Williamsburg, Va. 23187
  Surry nuclear plant
  Surry independent spent
  fuel storage

## WASHINGTON

- Mrs. Lois McCleary
  Library Assistant
  W.H. Abel Memorial Library
  125 Main Street, South
  Montesano, Wash. 98563
  WPPSS Nuclear Projects 3 & 5

- Ms. Judy McMakin
  Richland Public Library
  955 Northgate Street
  Richland, Wash. 99352
  WPPSS Nuclear Projects 1, 2,
  & 4
  Richland low-level waste
  burial site

## WISCONSIN

- Ms. Ann Kasuboski
  Government Documents Section
  Cofrin Library
  University of Wisconsin
  2420 Nicolet Drive
  Green Bay, Wis. 54311
  Kewaunee nuclear plant

- Ms. Nancy Steinhoff
  Reference Librarian
  LaCrosse Public Library
  800 Main Street
  LaCrosse, Wis. 54601
  LaCrosse nuclear plant

- Ms. Connie Kocian
  Adult Services Assistant
  Joseph Mann Library
  1516 16th Street
  Two Rivers, Wis. 54241
  Point Beach nuclear plant

# Appendix 4

# Regulations and Amendments—Fiscal Year 1993

## REGULATIONS AND AMENDMENTS PUT INTO EFFECT

### Departures From Manufacturer's Instructions; Elimination of Recordkeeping Requirements—Parts 30 and 35

On October 2, 1992 (57 FR 45566), the NRC published an amendment to its regulations that eliminates certain recordkeeping requirements related to the preparation and use of radiopharmaceuticals. The final rule, effective immediately, eliminates recordkeeping requirements related to the justification for and a precise description of the departure and the number of departures from the manufacturer's instructions approved by the Food and Drug Administration.

### Receipt of Byproduct and Special Nuclear Material—Part 50

On October 21, 1992 (57 FR 47978), the NRC published an amendment to its regulations, effective November 20, 1992, governing the conditions of licenses for production and utilization facilities to allow a reactor licensee to receive reactor-generated byproduct and special nuclear material being returned after off-site processing, such as compaction or incineration.

### Clarification of Statutory Authority for Purposes of Criminal Enforcement—Parts 11, 19, 20, 21, 25, 26, 30, 31, 32, 33, 34, 35, 39, 40, 50, 52, 53, 54, 55, 60, 61, 70, 71, 72, 73, 74, 75, 95, 110, 140, and 150

On November 24, 1992 (57 FR 55062), the NRC published an amendment to its regulations, effective December 24, 1992, that clarifies the applicability of the existing criminal penalty provisions of the Atomic Energy Act of 1954, as amended, to willful violations of certain Commission regulations.

### Fitness-for-Duty Programs: NRC Partial Withdrawal of NRC Information Collection Requirements—Part 26

On November 25, 1992 (57 FR 55443), the NRC published an amendment to its regulations, effective December 28, 1992, on the status of information collection requirements contained in 10 CFR Part 26. The Commission obtained approval from the Office of Management and Budget (OMB) for the information collection requirements contained in 21.24(d)(iv) and partial approval of the information collection requirements contained in 26.71(d) of the final rule entitled "Fitness-for-Duty Programs" (August 26, 1991; 56 FR 41922). The Commission is withdrawing the portion of 26.71(d) that contains the information collection requirements not approved by OMB because no compelling need exists for the additional data at this time.

### Disposal of Waste Oil by Incineration—Part 20

On December 7, 1992 (57 FR 57649), the NRC published an amendment to its regulations, effective January 6, 1993, that permits the on-site incineration of contaminated waste oils generated at licensed nuclear power plants without amending existing operating licenses.

### Revised Standards for Protection Against Radiation; Minor Amendments—Part 20

On December 8, 1992 (57 FR 57877), the NRC published an amendment to its regulations, effective immediately, that corrects errors in the text of the revised standards for protection against radiation, conforms portions of the regulatory text to the Commission's decision to defer mandatory implementation of the revised standards until 1994, and reflects the recent OMB approval of the use of NRC Forms 4 and 5.

### Combined Construction Permits and Operating Licenses; Conforming Amendments—Part 52

On December 23, 1992 (57 FR 60975), the NRC published an amendment to its regulations governing the issuance of combined construction permits and operating licenses for nuclear power plants. The amendments, effective January 22, 1993, serve to conform the regulations to the provisions of Title XXVIII of Public Law 102-486, the "Energy Policy Act of 1992," signed into law on October 24, 1992.

### Acquisition Regulation (NRCAR)—48 CFR Chapter 20

On December 23, 1992 (57 FR 61152), the NRC published an amendment revising its Nuclear Regulatory Commission Acquisition Regulation (NRCAR) that establishes requirements for the procurement of goods and services within the NRC to satisfy the particular needs of the agency. This amendment, effective January 22, 1993, expands the existing NRCAR to implement and supplement the government-wide Federal Acquisition Regulation.

### Exclusion of Attorneys From Interviews Under Subpoena—Part 19

On December 29, 1992 (57 FR 61780), the NRC published an amendment to its regulations, effective March 1, 1993, that provides for the exclusion of counsel from a subpoenaed interview when that counsel represents multiple interests in the investigation and concrete evidence exists that the counsel's presence at the interview would obstruct and impede the investigation.

**Material Approved for Incorporation by Reference; Maintenance and Availability—Parts 34, 35, 50, 73, and 110**

On December 29, 1992 (57 FR 61785), the NRC published an amendment to its regulations that clarifies previously published requirements governing the availability of material approved for incorporation by reference. This amendment, effective immediately, indicates that copies of material that has been incorporated by reference are maintained and available for review at the NRC Library.

**Conduct of Employees; Conforming Amendments—Part 0**

On January 12, 1993 (58 FR 3825), the NRC published an amendment to its regulations, effective February 3, 1993, that removes provisions that have been superseded by recently issued Office of Government Ethics regulations, which take effect on February 3, 1993.

**Licenses and Radiation Safety Requirements for Irradiators—Parts 19, 20, 30, 36, 40, 51, 70, and 170**

On February 9, 1993 (58 FR 7715), the NRC published an amendment to its regulations, effective July 1, 1993, that establishes a new 10 CFR Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators.

**Export and Import of Nuclear Equipment and Material; Clarifying Amendments—Part 110**

On March 9, 1993 (58 FR 12999), the NRC published an amendment to its regulations that clarifies the Commission's licensing requirements governing the export and import of nuclear equipment and material. This amendment, effective immediately, makes NRC's regulations consistent with the physical security guidelines contained in IAEA INFCIRC/225, and conforms NRC's regulations for export and import to the Solar, Wind, Water, and Geothermal Power Production Incentives Act of 1990 and to U.S. Government foreign relations commitments and changing circumstances.

**Clarification of Physical Protection Requirements at Fixed Sites—Part 73**

On March 15, 1993 (58 FR 13699), the NRC published an amendment to its general physical protection requirements for fixed sites. This amendment, effective April 14, 1993, makes it clear that the Commission's regulations do not require protection against both radiological sabotage and theft of special nuclear material at all facilities. The amendment also requires that nonpower reactor licensees who operate at or above two megawatts thermal protect against radiological sabotage where deemed necessary.

**List of Approved Spent Fuel Storage Casks: Additions—Part 72**

On April 7, 1993 (58 FR 17948), the NRC published an amendment to its list of approved spent fuel storage casks that adds one spent fuel storage cask to the list of approved casks. This amendment, effective May 7, 1993, will allow holders of power reactor operating licenses to store spent fuel in this approved cask under a general license.

**Training and Qualification of Nuclear Power Plant Personnel—Parts 50 and 52**

On April 26, 1993 (58 FR 21904), the NRC published an amendment to its regulations that requires each applicant for and each holder of a license to operate a nuclear power plant to establish, implement, and maintain a training program for nuclear power plant personnel based on a systems approach to training. This amendment, effective May 26, 1993, meets the directives of Section 306 of the Nuclear Waste Policy Act of 1982.

**Nuclear Regulatory Commission Acquisition Regulation; Minor Amendments—48 CFR Parts 2012, 2015, 2030, and 2052**

On May 3, 1993 (58 FR 26253), the NRC published an amendment to its regulations, effective immediately, that corrects errors in the text of the NRC's acquisition regulation and to conform portions of regulatory text to recodified regulations of the Cost Accounting Standards Board.

**Licensees' Announcements of Safeguards Inspections—Parts 73 and 74**

On May 21, 1993 (58 FR 29521), the NRC published an amendment to its regulations, effective June 21, 1993, concerning safeguards inspections at facilities that possess a formula quantity of strategic special nuclear material in unirradiated form. The purpose of this amendment is to ensure that the presence of NRC safeguards inspectors at affected facilities is not announced or widely communicated to licensee and contractor personnel without an express request to do so by the safeguards inspector.

**Repeal of NRC Standards of Conduct Regulations—Part 0**

On May 25, 1993 (58 FR 29951), the NRC published an amendment to its regulations, effective June 24, 1993, that repeals the delegations of authority and other miscellaneous regulations in 10 CFR Part 0 that are now contained in NRC internal management directives and handbooks or are no longer necessary.

**Fitness-for-Duty Requirements for Licensees Authorized to Possess, Use, or Transport Formula Quantities of Strategic Special Nuclear Material—Parts 26, 70, and 73**

On June 3, 1993 (58 FR 31467), the NRC published an amendment to its regulations that requires licensees who are authorized to possess, use, or transport formula quantities of strategic special nuclear material (SSNM) to institute fitnessfor-duty programs. This amendment, effective November 30, 1993, is necessary to provide greater assurance that individuals who have a drug or alcohol problem do not have access to or control over SSNM.

**Licensing Requirements for Land Disposal of Radioactive Wastes—Part 61**

On June 22, 1993 (58 FR 33886), the NRC published an amendment to its regulations containing licensing requirements for low-level radioactive waste (LLRW) disposal facilities. This amendment, effective July 22, 1993, is necessary to clarify that these regulations also apply to the licensing of above-ground disposal facilities; replace the phrase "quality control program" in these regulations with the phrase "quality assurance program," tailored to LLRW disposal; update the Paperwork Reduction Act Statement in the regulations; and identify the correct NRC recipient of copies of the licensee's annual reports.

**Monitoring the Effectiveness of Maintenance at Nuclear Power Plants—Part 50**

On June 23, 1993 (58 FR 33993), the NRC published an amendment to its regulations for monitoring the effectiveness of maintenance programs at commercial nuclear power plants. This amendment, effective July 10, 1996, changes the time interval for conducting evaluations from a mandatory once every year to at least once every refueling cycle, but not to exceed 24 months.

**Duplication Fees—Part 9**

On July 20, 1993 (58 FR 38665), the NRC published an amendment to its regulations that revises the charges for copying records publicly available at the NRC Public Document Room in Washington, DC. This amendment, effective immediately, is necessary to reflect the change in copying charges resulting from the Commission's award of a new contract for the copying of records.

**FY 1991 and 1992 Final Rule Implementing the U.S. Court of Appeals Decision and Revision of Fee Schedules; 100 percent Fee Recovery, Fiscal Year 1993—Parts 170 and 171**

On July 20, 1993 (58 FR 38666), the NRC published an amendment to its regulations that revises the licensing, inspection, and annual fees charged to its applicants and licensees. This amendment, effective August 19, 1993, is necessary to implement Public Law 101-508, enacted November 5, 1990, which mandates that the NRC recover approximately 100 percent of its budget authority in fiscal year 1993, less amounts appropriated from the Nuclear Waste fund.

**Prepare Radiopharmaceutical Reagent Kits and Elute Radiopharmaceutical Generators; Use of Radiopharmaceuticals for Therapy; Extension of Expiration Date—Parts 30 and 35**

On July 22, 1993 (58 FR 39130), the NRC published an amendment to its regulations that extends the expiration date of the interim final rule related to the preparation and therapeutic use of radiopharmaceuticals from August 23, 1993, to December 31, 1994. This amendment, effective August 23, 1993, allows licensees to continue to use byproduct material under the provisions of the interim final rule until the NRC completes a related rulemaking to address broader issues for the medical use of byproduct material, including those issues addressed by the interim final rule.

**Decommissioning Recordkeeping and License Termination: Documentation Additions—Parts 30, 40, 70, and 72**

On July 26, 1993 (58 FR 39628), the NRC published an amendment to its regulations, effective October 25, 1993, that requires holders of a specific license for possession of certain byproduct material, source material, special nuclear material, or for independent storage of spent nuclear fuel and high-level radioactive waste to prepare and maintain additional documentation that identifies all restricted areas in which licensed materials and equipment were stored or used, all areas outside of restricted areas for which documentation is required under current decommissioning regulations for unusual occurrences or spills, all areas outside of restricted areas in which waste has been buried, and all areas containing material, outside of restricted areas, in which the licensee would be required if the license were terminated, to decontaminate the area or seek special approval for disposal.

**Adjustment of the Maximum Standard Deferred Premium—Part 140**

On August 12, 1993 (58 FR 42851), the NRC published an amendment to its regulations, effective August 20, 1993, that increases the maximum standard deferred premium, presently established at $63 million-per-reactor-per-accident (but not to exceed $10 million in any one year), in accordance with the aggregate percentage change of 19.9 percent in the Consumer Price Index from August 1988 through March 1993.

**Access Authorization Fee Schedule for Licensee Personnel—Parts**

**11 and 25**

On August 23, 1993 (58 FR 44435), the NRC published an amendment to its regulations that revises the fee schedule for background investigations of licensee personnel who require access to National Security Information and/or Restricted Data and access to or control over Special Nuclear Material. These amendments, effective September 22, 1993, comply with current regulations that provide that NRC will publish fee adjustments concurrent with notifications of any changes in the rate charged the NRC by the Office of Personnel Management for conducting investigations.

**FSAR Update Submittals—Parts 50 and 54**

On August 27, 1993 (58 FR 45243), the NRC published an amendment to its regulations that consistently applies the requirement that nuclear power plant licensees submit final safety analysis report updates annually or six months after each refueling outage. These amendments, effective September 27, 1993, will eliminate the confusion caused by the conflicting requirements in different sections of the regulations.

**Day Firing Qualification Courses for Tactical Response Team Members, Armed Response Personnel, and Guards at Category I Licensees—Part 73**

On August 31, 1993 (58 FR 45781), the NRC published an amendment to its regulations, effective February 28, 1994, that requires armed security force personnel at fuel cycle facilities possessing formula quantities of strategic special nuclear material (Category I licensees), qualify and annually requalify for use of their assigned weapons using new day firing qualification courses.

**Nuclear Regulatory Commission Acquisition Regulation; Minor Amendments—48 CFR Parts 2017 and 2052**

On September 8, 1993 (58 FR 47220), the NRC published an amendment to its regulations, effective immediately, that makes minor corrective and conforming amendments to the NRC's acquisition regulation.

## REGULATIONS AND AMENDMENTS PROPOSED

**Reactor Site Criteria; Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants and Proposed Denial of Petition for Rulemaking From Free Environment, Inc. et al.—Parts 50, 52, and 100**

On October 20, 1992 (57 FR 47802), the NRC published an amendment to its regulations that would update the criteria used in decisions regarding power reactor siting, including geologic, seismic, and earthquake engineering considerations for future nuclear power plants. The proposed rule would allow NRC to benefit from experience gained in the application of the procedures and methods set forth in the current regulation and to incorporate rapid advancements in the earth sciences and earthquake engineering.

**Licensees' Announcements of Safeguards Inspections—Parts 73 and 74**

On November 3, 1992 (57 FR 49656), the NRC published an amendment to its regulations concerning fuel cycle facilities. The proposed rule would ensure that the presence of NRC safeguards inspectors at certain fuel cycle facilities is not announced or widely communicated to licensee and contractor personnel without an express request to do so by the inspector.

**Requirements Concerning the Accessible Air Gap for Generally Licensed Devices—Parts 31 and 32**

On November 27, 1992 (57 FR 56287), the NRC published an amendment to its regulations governing the safe use of radioactive byproduct material in certain measuring, gauging, and controlling devices. The proposed rule would provide for additional regulatory control over devices with both an accessible air gap and radiation levels that exceed specified values.

**Availability of Official Records—Part 2**

On December 23, 1992 (57 FR 61013), the NRC published an amendment to its regulations pertaining to the availability of official records to conform the regulations to existing case law and agency practice. The proposed rule would inform the public of three additional exceptions to a submitter's right to withdraw submitted information; provide more specific guidance for marking proprietary information; and inform the public of agency practice regarding reproduction and distribution of submitted copyrighted material.

**Self-Guarantee as an Additional Financial Assurance Mechanism—Parts 30, 40, 50, 70, and 72**

On January 11, 1993 (58 FR 3515), the NRC published an amendment to its regulations that would allow certain nonelectric utility licensees to use self-guarantee as a means of financial assurance. This proposed rule is intended to reduce the cost burden of financial assurance while providing the NRC with sufficient assurance that decommissioning costs will be funded. This proposed rule also responds to a petition for rulemaking (PRM-30-59) from General Electric Company and Westinghouse Electric Corporation.

**Timeliness in Decommissioning of Materials Facilities—Parts 30, 40, 70, and 72**

On January 13, 1993 (58 FR 4099), the NRC published an amendment to its regulations that would establish specific time periods for decommissioning unused portions of operating nuclear materials facilities and for decommissioning the entire site upon termination of operations. The proposed rule would also require that licensees provide a description of the conditions of the site as part of the information to be submitted in support of decommissioning plans.

**Licensee Submittal of Data in Computer-Readable Form—Parts 40, 72, 74, 75, and 150**

On January 26, 1993 (58 FR 6098), the NRC published an amendment to its regulations that would require certain licensees to submit data to the NRC in computer-readable form. The proposed rule is intended to streamline the collection of nuclear material transaction data and increase the accuracy of the reported information.

**Procedures and Criteria for On-site Storage of Low-Level Radioactive Waste—Parts 30, 40, 50, 70, and 72**

On February 2, 1993 (58 FR 6730), the NRC published an amendment to its regulations that would establish a regulatory framework containing the procedures and criteria that would apply to on-site storage of low-level radioactive waste beyond January 1, 1996. The proposed rule is intended to support the goals that have been established by the Low-Level Radioactive Waste Policy Amendments Act of 1985 and is consistent with the June 19, 1992, United States Supreme Court decision in New York v. United States.

**Specific Licensing of Exports of Certain Alpha-Emitting Radionuclides and Byproduct Material—Part 110**

On March 17, 1993 (58 FR 14344), the NRC published an amendment to its regulations that would conform the export controls of the United States to international export control guidelines and treaty obligations. The NRC is also proposing that Appendix A to 10 CFR Part 110 be restructured for clarification and to emphasize the distinction between nuclear reactor equipment controlled by the NRC and the Department of Commerce.

**Monitoring the Effectiveness of Maintenance at Nuclear Power Plants—Part 50**

On March 22, 1993 (58 FR 15303), the NRC published an amendment to its regulations that would monitor the effectiveness of maintenance programs at commercial nuclear power plants. The proposed amendment would change the time interval for conducting evaluations from once every year to at least once every refueling cycle, but not to exceed 24 months.

**Modifications to Fitness-for-Duty Program Requirements—Part 26**

On March 24, 1993 (58 FR 15810), the NRC published an amendment to its regulations that would modify current Fitness-for-Duty Program requirements. The proposed rule would permit licensees to reduce the random testing rate for licensee employees but maintain the 100 percent random testing rate for contractor and vendor employees.

**NRC Fee Policy; Request for Public Comment—Parts 170 and 171**

On April 19, 1993 (58 FR 21116), the NRC published a notice soliciting public comment on the need for changes to its fee policy and associated legislation. This action responds to legislation that requires the NRC to review its policy for assessment of annual fees, solicit public comment on the need for changes to this policy, and recommend to the Congress the changes in existing law the NRC finds are needed to prevent the placement of an unfair burden on NRC licensees. This notice also announces the receipt of and requests comment on a petition for rulemaking submitted by the American Mining Congress (PRM-170-4) that requests that the NRC conduct a rulemaking to evaluate its fee policy.

**FY 1991 and 1992 Proposed Rule Implementing the U.S. Court of Appeals Decision and Revision of Fee Schedules; 100 Percent Fee Recovery, Fiscal Year 1993—Parts 170 and 171**

On April 23, 1993 (58 FR 21662), the NRC published an amendment to its regulations that would amend the licensing, inspection, and annual fees charged to its applicants and licensees. The proposed amendments would implement Public Law 101-508, enacted November 5, 1990, which mandates that the NRC recover approximately 100 percent of its budget authority in fiscal year 1993, less amounts appropriated from the Nuclear Waste Fund. The amount to be recovered for fiscal year 1993 is approximately $518.9 million.

**Authorization to Prepare Radiopharmaceutical Reagent Kits and Elute Radiopharmaceutical Generators; Use of Radiopharmaceuticals for Therapy; Extension of Expiration Date—Parts 30 and 35**

On May 6, 1993 (58 FR 26938), the NRC published an amendment to its regulations that would extend the expiration date of the interim final rule (August 23, 1990; 55 FR 34513) related to the preparation and therapeutic use of radiopharmaceuticals from August 23, 1993, to December 31, 1994. The proposed extension would allow licensees to continue to use byproduct material under the provisions of the interim final rule until the NRC completes a related rulemaking to address broader issues for the medical use of byproduct material (including those issues address by the interim final rule).

**FSAR Update Submittals—Parts 50 and 54**

On May 14, 1993 (58 FR 28523), the NRC published an amendment to its regulations that would amend the power reactor safety regulations in order to consistently apply the requirement that nuclear power plant licensees submit final safety analysis report updates annually or six months after each refueling outage.

**Operator's Licenses—Part 55**

On May 20, 1993 (58 FR 29366), the NRC published an amendment to its regulations that would delete the requirement that each licensed operator at power, test, and research reactors pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's 6-year license as a prerequisite for license renewal.

**Emergency Planning Licensing Requirements for Independent Spent Fuel Storage Facilities (ISFSI) and Monitored Retrievable Storage Facilities (MRS)—Part 72**

On May 24, 1993 (58 FR 29795), the NRC published an amendment to its regulations that would provide, as directed by the Nuclear Waste Policy Act of 1982, for the emergency planning licensing requirements for independent storage facilities and monitored retrievable storage facilities. The proposed rule would ensure that local authorities would be notified in the event of an accident so that they may take appropriate action.

**Interim Storage of Spent Fuel in an Independent Spent Fuel Storage Installation; Site-Specific License to a Qualified Applicant—Parts 2 and 72**

On June 3, 1993 (58 FR 31478), the NRC published an amendment to its regulations in which the Director of Nuclear Material Safety and Safeguards would be able to issue a sitespecific license to a qualified applicant for the interim storage of spent fuel in an independent spent fuel storage installation following satisfactory completion of NRC safety and environmental reviews and after any public hearing on the application.

**Whistleblower Protection for Nuclear Power Plant Employees—Parts 19, 30, 40, 50, 60, 61, 70, 72, and 150**

On June 15, 1993 (58 FR 33042), the NRC published an amendment to its regulations regarding the protection of employees who provide information to the NRC or to their employers concerning safety issues. The proposed rule would conform current regulations to reflect the new nuclear whistleblower protection provisions of the Energy Policy Act of 1992, which was enacted on October 24, 1992.

**Preparation, Transfer for Commercial Distribution, and Use of Byproduct Material for Medical Use—Parts 30, 32, and 35**

On June 17, 1993 (58 FR 33396), the NRC published an amendment to its regulations regarding the medical use of byproduct material. The proposed rule would provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians greater discretion to prepare radioactive drugs containing byproduct material for medical use. The proposed rule would also allow research involving human subjects and byproduct material and the medical use of radiolabeled biologics.

**Production and Utilization Facilities; Emergency Planning and Preparedness—Exercise Requirements—Part 50**

On June 28, 1993 (58 FR 34539), the NRC published an amendment to its regulations that would revise NRC's emergency planning regulations. The proposed rule would update and clarify ambiguities that have arisen in the implementation of the Commission's emergency planning exercise requirements.

**Notification of Spent Fuel Management and Funding Plans by Licensees of Prematurely Shut Down Power Reactors—Part 50**

On June 30, 1993 (58 FR 34947), the NRC published an amendment to its regulations that would clarify the timing of notification to the NRC of spent fuel management and funding plans by licensees of nuclear power reactors that have been shut down before the expected end of their operating lives. The proposed rule would require that a licensee submit notification either within two years after permanently ceasing operation of its licensed power reactor or no later than five years before the reactor operating license expires, whichever event occurs first.

**Disposal of High-Level Radioactive Wastes in Geologic Repositories; Investigation and Evaluation of Potentially Adverse Conditions—Part 60**

On July 9, 1993 (58 FR 36902), the NRC published an amendment that would clarify its regulations with respect to the consideration of certain defined geologic and other conditions that, if present, are potentially adverse to the ability of a geologic repository to meet the prescribed performance objectives with respect to isolation of high-level radioactive waste.

**Equal Access to Justice Act: Implementation—Part 12**

On August 2, 1993 (58 FR 41061), the NRC published an amendment to its regulations that would add new provisions to implement the Equal Access to Justice Act.

**Notification of Events at Independent Spent Fuel Storage Installations and the Monitored Retrievable Storage Installation—Part 72**

On September 14, 1993 (58 FR 48004), the NRC published an amendment to its regulations that would revise licensee reporting requirements regarding the notification of events related to radiation safety at independent spent fuel storage installations and a monitored retrievable storage installation.

**Informal Hearing Procedures for Materials Licensing Adjudications—Part 2**

On September 29, 1993 (58 FR 50858), the NRC published an amendment to its regulations that would provide that requests for a hearing in certain material license proceedings be filed within 30 days of actual notice of the amendment application.

**Restoration of the Generic Exemption from Annual Fees for Nonprofit Educational Institutions—Part 171**

On September 29, 1993 (58 FR 50859), the NRC published an amendment to its regulations that would address the question on whether nonprofit educational institutions should receive a generic exemption from annual fees.

## ADVANCE NOTICES OF PROPOSED RULEMAKING

**Medical Use of Byproduct Material; Training and Experience Criteria—Part 35**

On October 9, 1992 (57 FR 46522), a document was published that withdrew an advance notice of proposed rulemaking on the training and experience criteria for all individuals who use byproduct material for clinical procedures in the practice of medicine (May 25, 1988; 53 FR 18845).

**Licensing of Source Material—Part 40**

On October 28, 1992 (57 FR 48749), an advance notice of proposed rulemaking was published in which the NRC announced that it is considering amending its regulations governing the licensing of source material and mill tailings. The contemplated rulemaking would consider revisions to improve control of source material through more specific regulation and to update the applicable requirements to conform with the revised standards for protection against radiation.

**Radioactive Waste Below Regulatory Concern; Generic Rulemaking, Withdrawal—Parts 2 and 20**

On August 24, 1993 (57 FR 44620), a document was published that withdrew an advance notice of proposed rulemaking concerning the submittal of petitions for disposal of radioactive waste streams below regulatory concern that was set out in the Commission's regulations (December 2, 1986; 51 FR 43367).

# Appendix 5

# Regulatory Guides—Fiscal Year 1993

NRC regulatory guides describe methods acceptable to the NRC staff of implementing specific parts of the NRC's regulations and also, in some cases, describe techniques used by the staff in evaluating specific problems or postulated accidents. Guides also may advise applicants regarding information the NRC staff needs in reviewing applications for permits and licenses.

Comments on the guides are encouraged, and the guides are revised whenever appropriate to reflect new information or experience. The NRC issues the guides for public comment in draft form before they have received complete staff review and an official staff position has been established.

Once issued, regulatory guides may be withdrawn when superseded by Commission regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes make them obsolete.

When guides are issued, reviewed, or withdrawn, notices are placed in the *Federal Register.*

To reduce the burden on the taxpayer, the NRC has made arrangements for the sale of active regulatory guides by both the U.S. Government Printing Office (on an individual guide basis) and the National Technical Information Service (on a standing order basis). Draft guides issued for public comment receive free distribution. NRC licensees receive, at no cost, pertinent draft and active regulatory guides as they are issued.

The following guides were issued, revised, or withdrawn during the period from October 1, 1992, to September 30, 1993.

**Division 1—Power Reactor Guides**

1.9 Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants (Revision 3)

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 29)

1.85 Materials Code Case Acceptability—ASME Section III, Division 1 (Revision 29)

1.108 Withdrawn—Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Revision 1)

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 10)

1.160 Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

**Division 2—Research and Test Reactor Guides**

None

**Division 3—Fuels and Materials Facilities Guides**

None

**Division 4—Environmental and Siting Guides**

None

**Division 5—Materials and Plant Protection Guides**

None

**Division 6—Product Guides**

None

**Division 7—Transportation Guides**

None

**Division 8—Occupational Health Guides**

8.9 Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Revision 1)

8.37 ALARA Levels for Effluents from Materials Facilities

8.38 Control of Access to High and Very High Radiation Areas in Nuclear Power Plants

**Division 9—Antitrust and Financial Review Guides**

None

**Division 10—General Guides**

None

## DRAFT REGULATORY GUIDES

**Division 1**

DG-1010 Proposed Revision 4 to Regulatory Guide 1.28, Quality Assurance Program Requirements

DG-1015 Identification and Characterization of Seismic Sources, Deterministic Source Earthquakes, and Ground Motion

DG-1016 Second Proposed Revision 2 to Regulatory Guide 1.12, Nuclear Power Plant Instrumentation for Earthquakes

DG-1017 Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions

DG-1018 Restart of a Nuclear Power Plant Shut Down by a Seismic Event

DG-1020 Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

DG-1023 Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 ft-lb

DG-1025 Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

**Division 3**

DG-3006 Standard Format and Content for Fire Protection Sections of License Applications for Fuel Cycle Facilities

DG-3008 Nuclear Criticality Safety Training

DG-3009 Topical Guidelines for the Licensing Support System

**Division 4**

DG-4003 (Proposed Revision 2 to Regulatory Guide 4.7) General Site Suitability Criteria for Nuclear Power Stations

**Division 8**

DG-8005 (Withdrawn) Assessing External Radiation Doses from Airborne Radioactive Materials

DG-8013 ALARA Levels for Effluents from Materials Facilities (October 1992)

# Appendix 6

# Civil Penalties and Orders—Fiscal Year 1993

**CIVIL PENALTIES PROPOSED, IMPOSED AND/OR PAID IN FISCAL YEAR 1993**
(Listed according to Enforcement Action (EA) numbers)

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Lafayette Clinic<br>Detroit, MI<br>(EA 91-017) | \$11,500 proposed in FY92, \$7,500 imposed in FY92 and paid in FY93 | Discrimination for participating in protected activities. |
| Power Authority of the State of New York<br>Fitzpatrick<br>(EA 92-033) | \$500,000 proposed in FY92, \$300,000 imposed and paid in FY93 | Violations of Appendix R, fire protection program, inadequate corrective actions, unqualified reactor protection system relays and submittal of inaccurate information. |
| Carolina Power & Light<br>Brunswick<br>(EA 92-075) | \$225,000 proposed and paid in FY93 | Inadequate corrective action and seismic concerns regarding missing bolts in concrete walls. |
| Midwest Industrial X-Ray<br>Fargo, ND<br>(EA 92-091) | \$8,000 proposed in FY92, paid in FY93 | Deliberate failure to use alarm ratemeters. |
| Sequoyah Fuels Corporation<br>Gore, OK<br>(EA 92-100) | \$12,500 proposed in FY92, paid in FY93 | In-plant release of $UF_6$; inadequate response to plant alarms. |
| American Testing & Inspection<br>Joliet, IL<br>(EA 92-102) | \$15,000 proposed in FY93, settled and civil penalty withdrawn, Order issued | Noncompliance with previous order concerning failure to file Form 241. |
| Metals Evaluation & Testing<br>Oakland, CA<br>(EA 92-105) | \$7,500 proposed in FY92, imposed and paid in FY93 | Failure to survey, recharge dosimeters, meet equipment safety standards. |
| Cleveland Clinic Foundation<br>Cleveland, OH<br>(EA 92-110) | \$1,250 proposed and paid in FY93 | Unauthorized repair of teletherapy unit by unqualified technician. |
| Baystate Medical Center<br>Springfield, MA<br>(EA 92-114) | \$2,000 proposed in FY92, imposed and paid in FY93 | Therapeutic misadministration. |
| Missouri Department of Highways<br>Jefferson City, MO<br>(EA 92-126) | \$1,250 proposed in FY92, paid in FY93 | Damaged moisture density gauge. |
| CTI Incorporated<br>Martinez, CA<br>(EA 92-127) | \$12,500 proposed in FY92, imposed and paid in FY93 | Failure to use alarm rate meters, conduct surveys, and post high radiation areas. |
| City of Columbus<br>Columbus, OH<br>(EA 92-132) | \$2,000 proposed and imposed in FY93, pending | Unauthorized cleaning and maintenance of moisture density gauges by former Radiation Safety Officer and other employees. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| MQS Inspection, Inc.<br>Elk Grove, IL<br>(EA 92-133) | $5,000 proposed and paid in FY93 | Overexposure to hand while locking camera, not having alarm rate meter operational. |
| Power Authority of the State of New York<br>Indian Point 3<br>(EA 92-134) | $100,000 proposed in FY92, paid in FY93 | Non-code repair of emergency service water system, degraded service water system. |
| Power Authority of the State of New York<br>Indian Point 3<br>(EA 92-135) | $100,000 proposed and paid in FY93 | Inaccurate information given at Enforcement Conference. |
| Eastern Testing & Inspection, Inc.<br>Thorofare, NJ<br>(EA 92-136) | $7,500 proposed in FY92, imposed in FY93 and being paid over time | Transportation and program violations due to careless disregard for requirements. |
| Grinnell Corporation<br>Cranston, RI<br>(EA 92-141) | $25,000 proposed in FY92, paid in FY93 | Numerous radiography violations, failure to file Form 241. |
| Howard, Needles, Tammen, and Bergendoff<br>Indianapolis, IN<br>(EA 92-144) | $875 proposed in FY92, paid in FY93 | Loss of control of material and management breakdown. |
| South Dakota Dept. of Transportation<br>Pierre, SD<br>(EA 92-150) | $3,400 proposed and paid in FY93 | Lost gauge, willful failure to report, programmatic breakdown. |
| Consolidated Engineering Laboratory<br>Pleasanton, CA<br>(EA 92-154) | $5,000 proposed in FY92, paid in FY93 | Rate meter, surveillance, survey, posting and other violations. |
| Tennessee Valley Authority<br>Sequoyah<br>(EA 92-155) | $62,500 proposed in FY92, paid in FY93 | Inoperable safety injection pump. |
| Power Authority of the State of New York<br>Indian Point 3<br>(EA 92-159) | $37,500 proposed and paid in FY93 | Failure to drug test operator prior to returning to duties; failure to set up followup testing program for two individuals. |
| Department of the Army<br>Rock Island, IL<br>(EA 92-162) | $15,000 proposed, imposed, and paid in FY93 | Breakdown in control of licensed activities. |
| Siemens Power Corp.<br>Richland, WA<br>(EA 92-163) | $18,750 proposed and paid in FY93 | Six violations of license conditions involving maintenance of dual criticality contingency controls. |
| Philadelphia Electric Company<br>Limerick<br>(EA 92-164) | $25,000 proposed in FY93, pending | Discrimination for engaging in protected activities. |

| Licensee, Facility and EA Number | Civil Penalties Proposed, Imposed and/or Paid in FY 93 | Summary |
|---|---|---|
| Northeast Nuclear Energy Co. Millstone (EA 92-166) | $62,500 proposed and paid in FY93 | Inoperable auxiliary filter system and inoperable hydrogen recombiner due to open plenum door and variable vane fans. |
| Carolina Power & Light Company Robinson (EA 92-167) | $50,000 proposed and paid in FY93 | Inadequate foreign material exclusion control resulted in inoperable safety injection system. |
| University of Missouri Columbia, MO (EA 92-170) | $625 proposed and paid in FY93 | Transportation violations involving mislabeling of radioactive shipments. |
| Houston Lighting & Power Company South Texas (EA 92-175) | $75,000 proposed and paid in FY93 | Procedural violations involving generic shutdown requirements (technical specification 3.03). |
| Philadelphia Electric Company Limerick (EA 92-179) | $62,500 proposed and paid in FY93 | Substantial potential for overexposure. |
| University of Michigan Ann Arbor, MI (EA 92-185) | $3,750 proposed and paid in FY93 | P-32 contamination event in public domain; failure to survey. |
| Triad Engineering Consultants, Inc. St. Albans, WV (EA 92-186) | $375 proposed and paid in FY93 | Loss of moisture density gauge. |
| Wolf Creek Nuclear Operating Corporation Wolf Creek (EA 92-191) | $50,000 proposed and paid in FY93 | Degraded service water flow. |
| St. Clares Riverside Medical Center Denville, NJ (EA 92-196) | $10,000 proposed and paid in FY93 | Misadministration and unplanned exposure due to the failure to instruct medical personnel. |
| Soil Consultants, Inc. St. Peters, MO (EA 92-201) | $125 proposed and paid in FY93 | Failure to control material; gauge left unattended and vehicle ran over it. |
| Kaiser Aluminum Oakland, CA (EA 92-202) | $625 proposed and paid in FY93 | Improper transfer of licensed material. |
| Capital Materials Testing, Inc. Ballston Spa, NY (EA 92-203) | $7,500 proposed and imposed in FY93, settlement reached at $5,000 and being paid over time. | Failure to survey, post, and restrict high radiation area. |
| Department of Veterans Affairs Birmingham, AL (EA 92-204) | $10,000 proposed in FY93, pending | Failure to report, failure to maintain records. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Wisconsin Electric Power Company Point Beach (EA 92-205) | $75,000 proposed and paid in FY93 | Foreign material in containment spray. |
| Washington Public Power Supply System Washington Nuclear (EA 92-206) | $75,000 proposed and paid in FY93 | Violations resulting in reactor power oscillation. |
| Gulf States Utilities Company River Bend (EA 92-207) | $100,000 proposed and paid in FY93 | Breakdown in control of radiation protection program. |
| Duke Power Company Oconee (EA 92-211) | $100,000 proposed, imposed, and paid in FY93 | Degraded service water flow due to inoperable butterfly valve; inadequate corrective actions. |
| Northeast Nuclear Energy Company Millstone (EA 92-212) | $100,000 proposed and paid in FY93 | Discrimination for engaging in protected activities. |
| Western Technologies Albuquerque, NM (EA 92-216) | $8,000 proposed, imposed, and paid in FY93 | Failure to wear alarm rate meters. |
| Carolina Power & Light Company Brunswick (EA 92-217) | $50,000 proposed and paid in FY93 | Failure to perform adequate surveys and evaluations before cutting a neutron source. |
| Overhoff Technology Corporation Milford, OH (EA 92-219) | $1,200 proposed and paid in FY93 | Program breakdown; careless disregard for requirements. |
| Individual Midland Park, NJ (EA 92-230) | $3,800 proposed, imposed, and paid in FY93 | Willful administration of doses to patients when dose calibrator reading exceeded 10% error. |
| Nuclear Fuel Services Erwin, TN (EA 92-231) | $37,500 proposed and paid in FY93 | Chemical fire event due to process control deficiencies; transfer of material to unfavorable geometry vessel. |
| Department of Agriculture Greenbelt, MD (EA 92-232) | $10,000 proposed, imposed, and paid in FY93 | Violations involving repetitive problems that collectively indicate programmatic problems. |
| R. S. Scott Associates Alpena, MI (EA 92-236) | $250 proposed and paid in FY93 | Unattended gauge. |
| Geo Cim, Inc. Hato Rey, PR (EA 92-238) | $375 proposed and paid in FY93 | Improper storage of licensed material. |
| Caribbean Soil Testing Company, Inc. (EA 92-239) | $375 proposed and paid in FY93 | Theft of licensed material after recovery. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Ponce I&M Engineering<br>Ponce, PR<br>(EA 92-240) | $2,000 proposed and paid in FY93 | Use of licensed material with expired license. |
| Yale-New Haven Hospital<br>New Haven, CT<br>(EA 92-241) | $10,000 proposed in FY93, pending | Failure to control/secure brachytherapy source, therapeutic misadministrations, quality management violations. |
| American Cyanamid Company<br>Princeton, NJ<br>(EA 92-243) | $1,000 proposed and paid in FY93 | Radiation safety program breakdown. |
| Pike Community Hospital<br>Waverly, OH<br>(EA 92-247) | $3,750 proposed, imposed, and paid in FY93 | Breakdown in control of radiation safety program. |
| Tennessee Valley Authority<br>Sequoyah<br>(EA 92-251) | $50,000 proposed and paid in FY93 | Twelve incidents of failure to properly secure safeguards information. |
| Indiana & Michigan Electric<br>D. C. Cook<br>(EA 92-252) | $37,500 proposed and paid in FY93 | Inoperable diesel generator. |
| Washington Public Power Supply System<br>Washington Nuclear<br>(EA 92-254) | $5,000 proposed and paid in FY93 | Transportation violations. |
| G.R. Osterland Co.<br>Cleveland, OH<br>(EA 92-255) | $125 proposed and paid in FY93 | Loss of control of material (moisture density gauge). |
| Tennessee Valley Authority<br>Sequoyah<br>(EA 92-257) | $125,000 proposed and paid in FY93 | Misposition of essential Raw Cooling Water throttle valve. |
| Wahiawa General Hospital<br>Oahu, HI<br>(EA 92-259) | $1,250 proposed, $750 imposed and paid in FY93 | Breakdown in control involving improper packaging for transport prior to disposal. |
| Pacific Radiopharmacy<br>Oahu, HI<br>(EA 92-260) | $2,500 proposed and paid in FY93 | Breakdown in control involving improper radiopharmaceutical disposal. |
| Southwest X-Ray Corp.<br>Little Rock, AK<br>(EA 93-001) | $2,500 proposed and paid in FY93 | Failure to use alarming rate meters. |
| Cameo Diagnostic Center, Inc.<br>Springfield, MA<br>(EA 93-005) | $1,750 proposed in FY93, pending | Willful use of licensed material at unauthorized location. |
| Sequoyah Fuels Corp.<br>Gore, OK<br>(EA 93-010) | $18,000 proposed and paid in FY93 | Procedural violations resulting in offsite release of non-radioactive toxic material. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Memorial Hospital of Laramie City<br>Cheyenne, WY<br>(EA 93-011) | $1,250 proposed and paid in FY93 | Therapeutic misadministration. |
| Babcock & Wilcox Co.<br>Lynchburg, VA<br>(EA 93-012) | $37,500 proposed, imposed, and paid in FY93 | Management breakdown in the areas of nuclear criticality limits and controls, and the audit program. |
| Edwards Pipeline Testing, Inc.<br>Tulsa, OK<br>(EA 93-015) | $12,000 proposed in FY93, pending | Willful failure to conduct quarterly audits of radiographers. |
| Commonwealth Edison<br>Dresden<br>(EA 93-019) | $75,000 proposed and paid in FY93 | Safety review program (10 CFR 50.59) problems identified in review of containment cooling service water modification. |
| Professional Services Industries, Inc.<br>Pittsburgh, PA<br>(EA 93-021) | $7,500 proposed and paid in FY93 | Failure to properly post and maintain surveillance of high radiation area resulting in unauthorized entry; failure to notify radiation safety officer of entry; shipping violation. |
| Community Hospital South<br>Indianapolis, IN<br>(EA 93-022) | $6,875 proposed, $5,625 imposed and paid in FY93 | Management breakdown. |
| Houston Lighting & Power Company<br>South Texas Project<br>(EA 93-023) | $25,000 proposed and paid in FY93 | Numerous procedural errors involving wrong train/unit; inadequate independent verification. |
| Papastavros Associates Medical Imaging<br>Wilmington, DE<br>(EA 93-027) | $250 proposed and paid in FY93 | Violation of medical quality management rule. |
| Nebraska Public Power District<br>Cooper<br>(EA 93-030) | $200,000 proposed, imposed, and paid in FY93 | Inaccurate information and inadequate corrective actions related to start-up strainers. |
| N.V. Enterprises<br>Evanston, WY<br>(EA 93-033) | $4,000 proposed in FY93, pending | Willful violation of ratemeter requirements. |
| Tennessee Valley Authority<br>Sequoyah<br>(EA 93-034) | $100,000 proposed and paid in FY93 | Failure to follow procedures resulting in loss of reactor coolant pump seal injection. |
| Power Authority of the State of New York<br>Indian Point 3<br>(EA 93-036) | $300,000 proposed and paid in FY93 | Significant weakness in design, testing, procedure adherence, corrective actions. |
| Castle Medical Center<br>Kailua, HI<br>(EA 93-040) | $7,500 proposed, imposed, and paid in FY93 | Breakdown in control of licensed activities; violations of medical quality management rule. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Westinghouse Electric Corporation (EA 93-044) | $18,750 proposed and paid in FY93 | Failure to perform nuclear criticality safety analysis, use of nonfavorable geometry containers. |
| Houston Lighting & Power South Texas (EA 93-047) | $75,000 proposed and paid in FY93 | Inadequate corrective actions related to motor operated valve; failure of management personnel to follow procedure. |
| Standard Nuclear Consultants, Ltd. Elburn, IL (EA 93-048) | $500 proposed and paid in FY93 | Failure to file Form 241. |
| Jersey Technology Laboratories, Inc. Newark, NJ (EA 93-049) | $250 proposed and paid in FY93 | Failure to secure licensed material (gauge). |
| Mercy Catholic Medical Center Philadelphia, PA (EA 93-052) | $12,500 proposed and paid in FY93 | Breakdown in control of licensed activities; violations of medical quality management rule. |
| Duke Power Company Catawba (EA 93-054) | $75,000 proposed and paid in FY93 | Service water system degraded because certain valves would not open against design pressures. |
| GPU Nuclear Corp. Oyster Creek (EA 93-055) | $50,000 proposed and paid in FY93 | Degraded shutdown cooling. |
| Houston Lighting & Power Company South Texas (EA 93-057) | $325,000 proposed and paid in FY93 | Technical specification violation involving emergency diesel generators and auxiliary feedwater pumps. |
| Indiana & Michigan Electric Company D.C. Cook (EA 93-059) | $25,000 proposed in FY93, pending | Discrimination against an employee by a contractor. |
| Gulf States Utilities River Bend (EA 93-060) | $50,000 proposed and paid in FY93 | Violation of 10 CFR 50.59 involving containment air lock door electrical interlocks. |
| Northern Virginia Endocrinologists Annandale, VA (EA 93-061) | $500 proposed and paid in FY93 | Quality management program failure to provide written directive prior to administration and to instruct staff. |
| Michigan State University East Lansing, MI (EA 93-062) | $3,750 proposed andpaid in FY93 | A visitor handled an unmarked carbon-14 target and spread contamination off-campus. |
| Commonwealth Edison Co. Braidwood (EA 93-063) | $50,000 proposed and paid in FY93 | Inoperable reactor vessel head vent valve; valve was locked closed when it should be open. |
| Commonwealth Edison Co. Zion (EA 93-064) | $50,000 proposed in FY93, pending | Auxiliary building door open and .25-inch negative pressure could not be maintained, as described in Final Safety Analysis Report. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| University of Michigan<br>Ann Arbor, MI<br>(EA 93-069) | $3,750 proposed and paid in FY93 | Licensee exceeded maximum power level. |
| Gray Wireline Services<br>Levelland, TX<br>(EA 93-073) | $1,500 proposed and paid in FY93 | Agreement State licensee deliberately used radioactive material in NRC jurisdiction without paying fee and provided false statements to NRC. |
| Metropolitan Hospital<br>Richmond, VA<br>(EA 93-076) | $5,000 proposed and paid in FY93 | Licensed material improperly stored in an unrestricted area. |
| Mayo Clinic<br>Rochester, MN<br>(EA 93-079) | $6,000 proposed, imposed, and paid in FY93 | Willful failure to survey, resulting in P-32 contamination off-site. |
| Twin Falls Clinic & Hospital<br>Twin Falls, ID<br>(EA 93-082) | $5,000 proposed and imposed in FY93, pending | Licensee did not develop and submit a Quality Management Program. |
| Siemens Power Corp.<br>Richland, WA<br>(EA 93-085) | $12,500 proposed and paid in FY93 | Violation of criticality control requirements. |
| Chestnut Hill Hospital<br>Philadelphia, PA<br>(EA 93-086) | $6,250 proposed and paid in FY93 | Breakdown in control of licensed activities. |
| ATEC Associates of Va.<br>Chantilly, VA<br>(EA 93-089) | $375 proposed and paid in FY93 | Failure to maintain control of a moisture density gauge. |
| Winchester Medical Center<br>Winchester, VA<br>(EA 93-090) | $1,250 proposed and paid in FY93 | Violation of medical quality management rule. |
| Wayne General Hospital<br>Wayne, NJ<br>(EA 93-093) | $6,250 proposed and paid in FY93 | Failure to implement quality management program; management breakdown. |
| Ingham Medical Center<br>Lansing, MI<br>(EA 93-109) | $12,250 proposed in FY93, pending | Misadministration; violation of medical quality management rule. |
| Vermont Yankee Nuclear<br>Vermont Yankee<br>(EA 93-112) | $50,000 proposed and paid in FY93 | Scram response time violation. |
| Steel Warehouse<br>South Bend, IN<br>(EA 93-115) | $250 proposed and paid in FY93 | General licensee shipped a gauge to the vendor without proper Dept of Transportation labelling or shipping containers. |
| Scientific Inspection Tech., Inc.<br>Hixson, TN<br>(EA 93-116) | $4,000 proposed and paid in FY93 | Extremity overexposure |

| Licensee, Facility and EA Number | Civil Penalties Proposed, Imposed and/or Paid in FY 93 | Summary |
|---|---|---|
| Cassia Memorial Hospital<br>Burley, ID<br>(EA 93-121) | $2,500 proposed and paid in FY93 | Breakdown in control of licensed activities; violation of medical quality management rule. |
| Commonwealth Edison<br>Quad Cities<br>(EA 93-127) | $100,000 proposed and paid in FY93 | Diesel generator technical specification violations. |
| Wolf Creek Nuclear Operating Corporation<br>Wolf Creek<br>(EA 93-129) | $50,000 proposed and paid in FY93 | Entry into Mode 3 with switches for motor-driven auxiliary feedwater pumps in pull-to-lock position. |
| Northeast Nuclear Energy Company<br>Millstone<br>(EA 93-130) | $50,000 proposed in FY93, pending | Inadequate requalification training on all units. |
| GPU Nuclear Corporation<br>Oyster Creek<br>(EA 93-136) | $75,000 proposed in FY93, pending | Failure to follow plant technical specification and procedures in initiating radiation work permit. |
| Mallinckrodt Medical, Inc.<br>St. Louis, MO<br>(EA 93-140) | $1,000 proposed in FY93, pending | Failure to survey, or inadequate survey, of package at pharmacy. |
| Hazelton Wisconsin, Inc.<br>Madison, WI<br>(EA 93-141) | $500 proposed and paid in FY93 | Improper disposal of Ni-63 sources in gas chromatographs. |
| Mobile Cardiovascular Testing<br>Milwaukee, WI<br>(EA 93-150) | $2,500 proposed and paid in FY93 | Program breakdown involving twelve violations. |
| University of Virginia<br>Charlottesville, VA<br>(EA 93-153) | $2,000 proposed and paid in FY93 | Operating reactor without automatic trip factors. |
| St. Joseph Radiology Associates, Inc.<br>St. Joseph, MO<br>(EA 93-155) | $25,000 proposed in FY93, pending | Abandonment of licensed material. |
| Commonwealth Edison Co.<br>Quad Cities<br>(EA 93-162) | $50,000 proposed and paid in FY93 | Program breakdown in fire protection. |
| Columbus Hospital<br>Great Falls, MT<br>(EA 93-164) | $2,500 proposed and paid in FY93 | Failure to comply with quality management plan. |
| St. Elizabeth Medical Center<br>Dayton, OH<br>(EA 93-165) | $1,250 proposed and paid in FY93 | Transferral of licensed material to nonlicensee for incineration. |
| Gulf States Utilities<br>River Bend<br>(EA 93-167) | $100,000 proposed and paid in FY93 | Main steam isolation valve not able to close. |

| *Licensee, Facility and EA Number* | *Civil Penalties Proposed, Imposed and/or Paid in FY 93* | *Summary* |
|---|---|---|
| Tulsa Gamma Ray, Inc.<br>Tulsa, OK<br>(EA 93-172) | $5,000 proposed in FY93, pending | Failure to maintain control of licensed material; failure to secure radiography exposure device during transport. |
| Cleveland Electric<br>Perry<br>(EA 93-176) | $200,000 proposed and paid in FY93 | Strainers for residual heat removal pumps from suppression pool became clogged, reducing the flow area of the strainers to less than 50% during normal operation. |
| Consumers Power Company<br>Palisades<br>(EA 93-178) | $50,000 proposed and paid in FY93 | Failure to follow procedures, ineffective procedures, and inadequate control of work practices resulted in the failure to uncouple one control rod prior to the removal of the reactor vessel head. |
| Mercy Memorial Medical Center, Inc.<br>St. Joseph, MI<br>(EA 93-179) | $6,250 proposed and paid in FY93 | Brachytherapy misadministration; failure to train nurse to recognize brachytherapy source. |
| Environmental Protection Agency<br>Port Orchard, WA<br>(EA 93-181) | $1,000 proposed and paid in FY93 | Loss of Ni-63 gas chromatograph sources. |
| Berkshire Health Systems, Inc.<br>Pittsfield, MA<br>(EA 93-186) | $7,500 proposed and paid in FY93 | Violations of medical quality management rule. |
| Individual<br>Toledo, OH<br>(EA 93-204) | $2,000 proposed in FY93, pending | Breakdown in control of licensed activities; violation of medical quality management rule. |
| Nondestructive Inspec tion Service<br>Hurricane, WV<br>(EA 93-205) | $5,000 proposed in FY93, pending | Failure to perform an adequate survey following a radiographic exposure. |
| Commonwealth Edison<br>Quad Cities<br>(EA 93-210) | $125,000 proposed in FY93, pending | High pressure coolant injection turbine exhaust rupture disc failed. |
| Princeton Community Hospital<br>Princeton, WV<br>(EA 93-212) | $5,000 proposed and paid in FY93 | Violation of medical quality management rule; materials left unattended. |
| Department of Veterans Affairs<br>Dallas, TX<br>(EA 93-217) | $3,750 proposed in FY93, pending | Violations of medical quality management rule. |
| Schnabel Engineering Associates, Inc.<br>Richmond, VA<br>(EA 93-219) | $375 proposed in FY93, pending | Failure to control material; moisture density gauge run over by a bulldozer. |
| Wayne City Office of Public Service<br>Detroit, MI<br>(EA 93-220) | $12,500 proposed in FY93, pending | Gauge fell from truck, picked up by member of public, not reported to the NRC, another gauge run over by a bulldozer. |

**ORDERS ISSUED IN FISCAL YEAR 1993 (Listed according to Enforcement Action (EA) numbers.)**

| *Licensee, Facility and EA Number* | *Order Issued in FY 1993* | *Facts* |
|---|---|---|
| Individual<br>St. Joseph, MO<br>(EA 92-172) | Order Modifying License issued October 16, 1992 | Unauthorized material possession. |
| Amoco Oil Co.<br>Whiting, IN<br>(EA 92-198) | Order Modifying License issued December 1, 1992 | Falsified audit records. |
| Harrisburg Cancer Center<br>State College, PA<br>(EA 93-006) | Order Suspending License issued January 20, 1993 | Therapeutic misadministration. |
| Yale-New Haven Hospital<br>New Haven, CT<br>(EA 93-016) | Confirmatory Order issued April 26, 1993 | Failure to control/secure brachytherapy source, therapeutic misadministration, quality management violations. |
| Department of Agriculture<br>Greenbelt, MD<br>(EA 93-028) | Confirmatory Order issued March 26, 1993 | Broadscope licensee with repetitive violations involving improper transfer of licensed material and failure to perform audits. |
| Radiation Oncology Center at Marlton<br>Marlton, NJ<br>(EA 93-041) | Order Modifying License issued March 9, 1993 | Breakdown in control of licensed activities. |
| Individual<br>(EA 93-042) | Order Limiting Licensed Activities issued May 4, 1993 | Individual submitted false employment history and transcript reflecting bachelors degree which had not been obtained. |
| Innovative Weaponry, Inc.<br>Albuquerque, NM<br>(EA 93-067) | Order Modifying License issued June 18, 1993 | Distribution of unauthorized and improperly labelled gun sights. |
| Department of Veterans Affairs<br>Birmingham, AL<br>(EA 93-174) | Order Modifying License issued September 13, 1993 | Failure to report, failure to maintain records. |

# Appendix 7

# Nuclear Electric Generating Units in Operation or Under Construction

(As of December 31, 1993)

The following is a listing of the 116 nuclear power reactor electrical generating units which were in operation or under construction in the United States as of December 31, 1993, representing a total capacity of 107,591 MWe (megawatts-electric; one megawatt is 1,000 kilowatts), of which 8,513 MWe was not yet licensed for operation. There are two reactor types represented, abbreviated PWR—pressurized water reactor, and BWR--boiling water reactor. Of the 116 reactor units listed, 78 are PWRs and 38 are BWRs. Plant status is indicated as follows: OL—has operating license (not necessarily for full-power operation), CP—has construction permit. The dates for operation are either actual (in the case of operating licenses) or as scheduled by the utilities, for plants not yet licensed for operation, as of December 31, 1993. At that time, there were 109 commercial nuclear reactors in the United States with operating licenses and operating; these units had been operating for a cumulative 1,550 reactor-years (an additional 155 reactor-years had been accumulated by reactors now permanently shut down). At the end of 1993, there were seven units for which construction permits were in effect (although construction of some of these has been postponed indefinitely). See the last page of this appendix for an alphabetic listing of all nuclear plants in the United States, with information on power ratings and dates of licensing.

| Site | Plant | Capacity (Net MWe) | Type | Status | Utility | Commercial Operation |
|---|---|---|---|---|---|---|
| **ALABAMA** | | | | | | |
| Decatur | Browns Ferry Unit 1 nuclear power plant | 1,065 | BWR | OL 1973 | Tennessee Valley Authority | 1974 |
| Decatur | Browns Ferry Unit 2 nuclear power plant | 1,065 | BWR | OL 1974 | Tennessee Valley Authority | 1975 |
| Decatur | Browns Ferry Unit 3 nuclear power plant | 1,065 | BWR | OL 1976 | Tennessee Valley Authority | 1977 |
| Dothan | Joseph M. Farley Unit 1 nuclear power plant | 804 | PWR | OL 1977 | Alabama Power Co. | 1977 |
| Dothan | Joseph M. Farley Unit 2 nuclear power plant | 814 | PWR | OL 1981 | Alabama Power Co. | 1981 |
| Scottsboro | Bellefonte Unit 1 nuclear power plant | 1,235 | PWR | CP 1974 | Tennessee Valley Authority | 1993 |
| Scottsboro | Bellefonte Unit 2 nuclear power plant | 1,235 | PWR | CP 1974 | Tennessee Valley Authority | 1995 |
| **ARIZONA** | | | | | | |
| Wintersburg | Palo Verde Unit 1 nuclear power plant | 1,304 | PWR | OL 1984 | Arizona Public Service Co. | 1986 |

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|---|---|---|---|---|---|---|
| Wintersburg | Palo Verde Unit 2 nuclear power plant | 1,304 | PWR | OL 1985 | Arizona Public Service Co. | 1986 |
| Wintersburg | Palo Verde Unit 3 nuclear power plant | 1,304 | PWR | OL 1987 | Arizona Public Service Co. | 1988 |
| **ARKANSAS** | | | | | | |
| Russelville | Arkansas Nuclear One Unit 1 nuclear power plant | 836 | PWR | OL 1974 | Arkansas Power & Light Co. | 1974 |
| Russelville | Arkansas Nuclear One Unit 2 nuclear power plant | 858 | PWR | OL 1978 | Arkansas Power & Light Co. | 1980 |
| **CALIFORNIA** | | | | | | |
| San Clemente | San Onofre Unit 2 nuclear power plant | 1,100 | PWR | OL 1982 | So. Calif. Ed. & San Diego Gas & Electric Co. | 1983 |
| San Clemente | San Onofre Unit 3 nuclear power plant | 1,100 | PWR | OL 1983 | So. Calif. Ed. & San Diego Gas & Electric Co. | 1984 |
| Diablo Canyon | Diablo Canyon Unit 1 nuclear power plant | 1,084 | PWR | OL 1984 | Pacific Gas & Electric Co. | 1985 |
| Diablo Canyon | Diablo Canyon Unit 2 nuclear power plant | 1,106 | PWR | OL 1985 | Pacific Gas & Electric Co. | 1986 |
| **CONNECTICUT** | | | | | | |
| Haddam Neck | Haddam Neck nuclear power plant | 555 | PWR | OL 1967 | Conn. Yankee Atomic Power Co. | 1968 |
| Waterford | Millstone Unit 1 nuclear power plant | 654 | BWR | OL 1970 | Northeast Nuclear Energy Co. | 1971 |
| Waterford | Millstone Unit 2 nuclear power plant | 864 | PWR | OL 1975 | Northeast Nuclear Energy Co. | 1975 |
| Waterford | Millstone Unit 3 nuclear power plant | 1,156 | PWR | OL 1985 | Northeast Nuclear Energy Co. | 1986 |
| **FLORIDA** | | | | | | |
| Florida City | Turkey Point Unit 3 nuclear power plant | 646 | PWR | OL 1972 | Florida Power & Light Co. | 1972 |
| Florida City | Turkey Point Unit 4 nuclear power plant | 646 | PWR | OL 1973 | Florida Power & Light Co. | 1973 |
| Red Level | Crystal River Unit 3 nuclear power plant | 806 | PWR | OL 1977 | Florida Power Corp. | 1977 |
| Ft. Pierce | St. Lucie Unit 1 nuclear power plant | 817 | PWR | OL 1976 | Florida Power & Light Co. | 1976 |
| Ft. Pierce | St. Lucie Unit 2 nuclear power plant | 842 | PWR | OL 1983 | Florida Power & Light Co. | 1983 |

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|---|---|---|---|---|---|---|
| **GEORGIA** | | | | | | |
| Baxley | Hatch Unit 1 nuclear power plant | 757 | BWR | OL 1974 | Georgia Power Co. | 1975 |
| Baxley | Hatch Unit 2 nuclear power plant | 771 | BWR | OL 1978 | Georgia Power Co. | 1979 |
| Waynesboro | Vogtle Unit 1 nuclear power plant | 1,100 | PWR | OL 1987 | Georgia Power Co. | 1987 |
| Waynesboro | Vogtle Unit 2 nuclear power plant | 1,100 | PWR | OL 1989 | Georgia Power Co. | 1989 |
| **ILLINOIS** | | | | | | |
| Morris | Dresden Unit 2 nuclear power plant | 772 | BWR | OL 1969 | Commonwealth Edison Co. | 1970 |
| Morris | Dresden Unit 3 nuclear power plant | 773 | BWR | OL 1971 | Commonwealth Edison Co. | 1971 |
| Zion | Zion Unit 1 nuclear power plant | 1,040 | PWR | OL 1973 | Commonwealth Edison Co. | 1973 |
| Zion | Zion Unit 2 nuclear power plant | 1,040 | PWR | OL 1973 | Commonwealth Edison Co. | 1974 |
| Cordova | Quad-Cities Unit 1 nuclear power plant | 769 | BWR | OL 1972 | Comm. Ed. Co. -Iowa-Ill. Gas & Elec. Co. | 1973 |
| Cordova | Quad-Cities Unit 2 nuclear power plant | 769 | BWR | OL 1972 | Comm. Ed. Co. -Iowa-Ill. Gas & Elec. Co. | 1973 |
| Seneca | LaSalle Unit 1 nuclear power plant | 1,078 | BWR | OL 1982 | Commonwealth Edison Co. | 1984 |
| Seneca | LaSalle Unit 2 nuclear power plant | 1,078 | BWR | OL 1983 | Commonwealth Edison Co. | 1984 |
| Bryon | Byron Unit 1 nuclear power plant | 1,120 | PWR | OL 1984 | Commonwealth Edison Co. | 1985 |
| Byron | Byron Unit 2 nuclear power plant | 1,120 | PWR | OL 1986 | Commonwealth Edison Co. | 1987 |
| Braidwood | Braidwood Unit 1 nuclear power plant | 1,120 | PWR | OL 1986 | Commonwealth Edison Co. | 1988 |
| Braidwood | Braidwood Unit 2 nuclear power plant | 1,120 | PWR | OL 1987 | Commonwealth Edison Co. | 1988 |
| Clinton | Clinton Unit 1 nuclear power plant | 950 | BWR | OL 1986 | Illinois Power Co. | 1987 |
| **IOWA** | | | | | | |
| Pala | Arnold Unit 1 nuclear power plant | 515 | BWR | OL 1974 | Iowa Elec. Power & Light Co. | 1975 |

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|---|---|---|---|---|---|---|
| **KANSAS** | | | | | | |
| Burlington | Wolf Creek nuclear power plant | 1,150 | PWR | OL 1985 | Kansas Gas & Electric Co. | 1985 |
| **LOUISIANA** | | | | | | |
| Taft | Waterford nuclear power plant | 1,151 | PWR | OL 1984 | Louisiana Power & Light Co. | 1985 |
| St. Francisville | River Bend Unit 1 nuclear power plant | 934 | BWR | OL 1985 | Gulf States Utilities Co. | 1986 |
| **MAINE** | | | | | | |
| Wiscasset | Maine Yankee Atomic Power Atomic Power Co. | 810 | PWR | OL 1972 | Maine Yankee | 1972 |
| **MARYLAND** | | | | | | |
| Lusby | Calvert Cliffs Unit 1 nuclear power plant | 825 | PWR | OL 1974 | Baltimore Gas & Electric Co. | 1975 |
| Lusby | Calvert Cliffs Unit 2 nuclear power plant | 825 | PWR | OL 1976 | Baltimore Gas & Electric Co. | 1977 |
| **MASSACHUSETTS** | | | | | | |
| Plymouth | Pilgrim Unit 1 nuclear power plant | 670 | BWR | OL 1972 | Boston Edison Co. | 1972 |
| **MICHIGAN** | | | | | | |
| Big Rock | Big Rock Point nuclear power plant | 69 | BWR | OL 1964 | Consumers Power Co. | 1963 |
| South Haven | Palisades nuclear power plant | 635 | PWR | OL 1971 | Consumers Power Co. | 1971 |
| Laguna Beach | Fermi Unit 2 nuclear power plant | 1,093 | BWR | OL 1985 | Detroit Edison Co. | 1988 |
| Bridgman | Cook Unit 1 nuclear power plant | 1,044 | PWR | OL 1974 | Indiana & Michigan Electric Co. | 1975 |
| Bridgman | Cook Unit 2 nuclear power plant | 1,082 | PWR | OL 1977 | Indiana & Michigan Electric Co. | 1978 |
| **MINNESOTA** | | | | | | |
| Monticello | Monticello nuclear power plant | 525 | BWR | OL 1970 | Northern States Power Co. | 1971 |
| Red Wing | Prairie Island Unit 1 nuclear power plant | 503 | PWR | OL 1973 | Northern States Power Co. | 1973 |
| Red Wing | Prairie Island Unit 2 nuclear power plant | 500 | PWR | OL 1974 | Northern States Power Co. | 1974 |
| **MISSISSIPPI** | | | | | | |
| Port Gibson | Grand Gulf Unit 1 nuclear power plant | 1,250 | BWR | OL 1982 | Mississippi Power & Light Co. | 1985 |

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|---|---|---|---|---|---|---|
| **MISSOURI** | | | | | | |
| Fulton | Callaway Unit 1 nuclear power plant | 1,188 | PWR | OL 1984 | Union Electric Co. | 1985 |
| **NEBRASKA** | | | | | | |
| Fort Calhoun | Fort Calhoun Unit 1 nuclear power plant | 478 | PWR | OL 1973 | Omaha Public Power District | 1973 |
| Brownville | Cooper nuclear power plant | 764 | BWR | OL 1974 | Nebraska Public Power District | 1974 |
| **NEW HAMPSHIRE** | | | | | | |
| Seabrook | Seabrook Unit 1 nuclear power plant | 1,198 | PWR | OL 1989 | Public Service of New Hampshire | 1990 |
| **NEW JERSEY** | | | | | | |
| Toms River | Oyster Creek Unit 1 nuclear power plant | 620 | BWR | OL 1969 | GPU Nuclear Corp. | 1969 |
| Salem | Salem Unit 1 nuclear power plant | 1,079 | PWR | OL 1976 | Public Service Electric & Gas Co. | 1977 |
| Salem | Salem Unit 2 nuclear power plant | 1,106 | PWR | OL 1980 | Public Service Electric & Gas Co. | 1981 |
| Salem | Hope Creek Unit 1 nuclear power plant | 1,067 | BWR | OL 1986 | Public Service Electric & Gas Co. | 1986 |
| **NEW YORK** | | | | | | |
| Indian Point | Indian Point Unit 2 nuclear power plant | 864 | PWR | OL 1973 | Consolidated Edison Co. | 1974 |
| Indian Point | Indian Point Unit 3 nuclear power plant | 891 | PWR | OL 1975 | Power Authority of the State of New York | 1976 |
| Scriba | Nine Mile Point Unit 1 nuclear power plant | 610 | BWR | OL 1969 | Niagara Mohawk Power Co. | 1969 |
| Scriba | Nine Mile Point Unit 2 nuclear power plant | 1,080 | BWR | OL 1986 | Niagara Mohawk Power Co. | 1988 |
| Ontario | Ginna Unit 1 nuclear power plant | 470 | PWR | OL 1969 | Rochester Gas & Electric Co. | 1970 |
| Scriba | FitzPatrick nuclear power plant | 810 | BWR | OL 1974 | Power Authority of the State of New York | 1975 |
| **NORTH CAROLINA** | | | | | | |
| Southport | Brunswick Unit 2 nuclear power plant | 790 | BWR | OL 1974 | Carolina Power & Light Co. | 1975 |
| Southport | Brunswick Unit 1 nuclear power plant | 790 | BWR | OL 1976 | Carolina Power & Light Co. | 1977 |

| | | | | | | |
|---|---|---|---|---|---|---|
| Cowans Ford Dam | McGuire Unit 1 nuclear power plant | 1,180 | PWR | OL 1981 | Duke Power Co. | 1981 |
| Cowans Ford Dam | McGuire Unit 2 nuclear power plant | 1,180 | PWR | OL 1983 | Duke Power Co. | 1984 |
| Bonsal | Harris Unit 1 nuclear power plant | 915 | PWR | OL 1986 | Carolina Power & Light Co. | 1987 |
| **OHIO** | | | | | | |
| Oak Harbor | Davis-Besse Unit 1 nuclear power plant | 874 | PWR | OL 1977 | Toledo Edison–Cleveland Electric Illuminating Co. | 1977 |
| Perry | Perry Unit 1 nuclear power plant | 1,205 | BWR | OL 1986 | Toledo Edison–Cleveland Electric Illuminating Co. | 1987 |
| Perry | Perry Unit 2 nuclear power plant | 1,205 | BWR | CP 1977 | Toledo Edison–Cleveland Electric Illuminating Co. | Indef. |
| **PENNSYLVANIA** | | | | | | |
| Peach Bottom | Peach Bottom Unit 2 nuclear power plant | 1,051 | BWR | OL 1973 | Philadelphia Electric Co. | 1974 |
| Peach Bottom | Peach Bottom Unit 3 nuclear power plant | 1,035 | BWR | OL 1974 | Philadelphia Electric Co. | 1974 |
| Pottstown | Limerick Unit 1 nuclear power plant | 1,065 | BWR | OL 1984 | Philadelphia Electric Co. | 1986 |
| Pottstown | Limerick Unit 2 nuclear power plant | 1,065 | BWR | OL 1989 | Philadelphia Electric Co. | 1990 |
| Shippingport | Beaver Valley Unit 1 nuclear power plant | 810 | PWR | OL 1976 | Duquesne Light Co. Ohio Edison Co. | 1976 |
| Shippingport | Beaver Valley Unit 2 nuclear power plant | 852 | PWR | OL 1987 | Duquesne Light Co. Ohio Edison Co. | 1987 |
| Goldsboro | Three Mile Island Unit 1 nuclear power plant | 776 | PWR | OL 1974 | GPU Nuclear Corp. | 1974 |
| Berwick | Susquehanna Unit 1 nuclear power plant | 1,052 | BWR | OL 1982 | Pennsylvania Power & Light Co. | 1983 |
| Berwick | Susquehanna Unit 2 nuclear power plant | 1,052 | BWR | OL 1984 | Pennsylvania Power & Light Co. | 1985 |
| **SOUTH CAROLINA** | | | | | | |
| Hartsville | Robinson Unit 2 nuclear power plant | 665 | PWR | OL 1970 | Carolina Power & Light Co. | 1971 |
| Seneca | Oconee Unit 1 nuclear power plant | 860 | PWR | OL 1973 | Duke Power Co. | 1973 |
| Seneca | Oconee Unit 2 nuclear power plant | 860 | PWR | OL 1973 | Duke Power Co. | 1974 |

| | | | | | | |
|---|---|---|---|---|---|---|
| Seneca | Oconee Unit 3 nuclear power plant | 860 | PWR | OL 1974 | Duke Power Co. | 1974 |
| Broad River | Summer Unit 1 nuclear power plant | 900 | PWR | OL 1982 | So. Carolina Electric & Gas Co. | 1984 |
| Lake Wylie | Catawba Unit 1 nuclear power plant | 1,145 | PWR | OL 1984 | Duke Power Co. | 1985 |
| Lake Wylie | Catawba Unit 2 nuclear power plant | 1,145 | PWR | OL 1986 | Duke Power Co. | 1986 |
| **TENNESSEE** | | | | | | |
| Daisy | Sequoyah Unit 1 nuclear power plant | 1,128 | PWR | OL 1980 | Tennessee Valley Authority | 1981 |
| Daisy | Sequoyah Unit 2 nuclear power plant | 1,148 | PWR | OL 1981 | Tennessee Valley Authority | 1982 |
| Spring City | Watts Bar Unit 1 nuclear power plant | 1,165 | PWR | CP 1973 | Tennessee Valley | 1988 |
| Spring City | Watts Bar Unit 2 nuclear power plant | 1,165 | PWR | CP 1973 | Tennessee Valley Authority | 1989 |
| **TEXAS** | | | | | | |
| Glen Rose | Comanche Peak Unit 1 nuclear power plant | 1,150 | PWR | OL 1990 | Texas Utilities | 1990 |
| Glen Rose | Comanche Peak Unit 2 nuclear power plant | 1,150 | PWR | OL 1994 | Texas Utilities | 1994 |
| Bay City | South Texas Unit 1 nuclear power plant | 1,250 | PWR | OL 1987 | Houston Lighting & Power Co. | 1988 |
| Bay City | South Texas Unit 2 nuclear power plant | 1,250 | PWR | OL 1989 | Houston Lighting & Power Co. | 1989 |
| **VERMONT** | | | | | | |
| Vernon | Vermont Yankee nuclear power plant | 504 | BWR | OL 1972 | Vermont Yankee Nuclear Power Corp. | 1972 |
| **VIRGINIA** | | | | | | |
| Gravel Neck | Surry Unit 1 nuclear power plant | 775 | PWR | OL 1972 | Virginia Electric & Power Co. | 1972 |
| Gravel Neck | Surry Unit 2 nuclear power plant | 775 | PWR | OL 1973 | Virginia Electric & Power Co. | 1973 |
| Mineral nuclear power plant | North Anna Unit 1 | 865 | PWR | OL 1976 | Virginia Electric & Power Co. | 1978 |
| Mineral | North Anna Unit 2 nuclear power plant | 890 | PWR | OL 1980 | Virginia Electric & Power Co. | 1980 |
| **WASHINGTON** | | | | | | |
| Richland | WPPSS No. 1 (Hanford) nuclear power plant | 1,266 | PWR | CP 1975 | Wash. Public Power Supply System | Indef. |

| | | | | | | |
|---|---|---|---|---|---|---|
| Richland | WPPSS No. 2 (Hanford) nuclear power plant | 1,103 | BWR | OL 1983 | Wash. Public Power Supply System | 1984 |
| Satsop | WPPSS No. 3 | 1,242 | PWR | CP 1978 | Wash. Public Power Supply System | Indef. |
| **WISCONSIN** | | | | | | |
| Two Creeks | Point Beach Unit 1 nuclear power plant | 495 | PWR | OL 1970 | Wisconsin Electric Power Co. | 1970 |
| Two Creeks | Point Beach Unit 2 nuclear power plant | 495 | PWR | OL 1971 | Wisconsin Electric Power Co. | 1972 |
| Kewaunee | Kewaunee nuclear power plant | 515 | PWR | OL 1973 | Wisconsin Public Service Corp. | 1974 |

## U.S. Nuclear Power Plants with Operating Licenses

(Plant - type - MWe - cp - ol)*

**Arkansas 1 = pwr, 836, 12/68, 5/74.**
**Arkansas 2 = pwr, 858, 12/72, 12/78.**
**Beaver Valley 1 (Pa.) = pwr, 810, 6/70, 7/76.**
**Beaver Valley 2 = pwr, 833, 5/74, 8/87.**
**Big Rock Point (Mich.) = bwr, 69, 5/60, 5/64.**
**Braidwood 1 (Ill.) = pwr, 1120, 12/75, 7/87.**
**Braidwood 2 = pwr, 1120, 12/75, 5/88.**
**Browns Ferry 1 (Ala.) = bwr, 1065, 5/67, 12/73.**
**Browns Ferry 2 = bwr, 1065, 5/67, 8/74.**
**Browns Ferry 3 = bwr, 1065, 5/67, 8/76.**
**Brunswick 1 (N.C.) = bwr, 790, 2/70, 11/76.**
**Brunswick 2 = bwr, 790, 2/70, 12/74.**
**Byron 1 (Ill.) = pwr, 1105, 12/75, 2/85.**
**Byron 2 = pwr, 1105, 12/75, 1/87.**
**Callaway (Mo.) = pwr, 1145, 4/76, 10/84.**
**Calvert Cliffs 1 (Md.) = pwr, 825, 7/69, 7/74.**
**Calvert Cliffs 2 = pwr, 825, 7/69, 11/76.**
**Catawba 1 (S.C.) = pwr, 1129, 8/75, 1/85.**
**Catawba 2 = pwr, 1129, 8/75, 5/86.**
**Clinton (Ill.) = bwr, 930, 2/76, 4/86.**
**Comanche Peak 1 (Tex.) = pwr, 1150, 12/74, 4/90.**
**Comanche Peak 2 (Tex.) = pwr, 1150, 12/74.**
**Cook 1 (Mich.) = pwr, 1020, 3/69, 10/74.**
**Cook 2 = pwr, 1060, 3/69, 12/77.**
**Cooper (Neb.) = bwr, 764, 6/68, 1/74.**
**Crystal River 3 (Fla.) = pwr, 821, 9/68, 1/77.**
**Davis-Besse ((Ohio) = pwr, 860, 3/71, 4/77.**
**Diablo Canyon 1 (Cal.) = pwr, 1073, 4/68, 11/84.**
**Diablo Canyon 2 = pwr, 1087, 12/70, 8/85.**
**Dresden 2 (Ill.) = bwr, 772, 1/66, 12/69**
**Dresden 3 = bwr, 773, 10/66, 3/71.**
**Duane Arnold (Iowa) = bwr, 515, 6/70, 2/74.**
**Farley 1 (Ala.) = pwr, 813, 8/72, 6/77.**
**Farley 2 = pwr, 823, 8/72, 3/81.**
**Fermi 2 (Mich.) = bwr, 1093, 9/72, 7/85.**
**Fitzpatrick (N.Y.) = bwr 778, 5/70, 10/74.**
**Fort Calhoun 1 (Neb.) = pwr, 478, 6/68, 8/73.**
**Ginna (N.Y.) = pwr, 470, 4/66, 12/84.**
**Grand Gulf 1 (Miss.) = bwr, 1142, 9/74, 11/84.**
**Haddam Neck (Conn.) = pwr, 569, 5/64, 12/74.**
**Harris 1 (N.C.) = pwr, 860, 1/78, 1/87.**
**Hatch 1 (Ga.) = bwr, 860, 9/69,10/74.**
**Hatch 2 = bwr, 768, 12/72, 6/78.**
**Hope Creek 1 (N.J.) = bwr, 1067, 11/74, 7/86.**
**Indian Point 2 (N.Y.) = pwr, 849, 10/66, 9/73.**
**Indian Point 3 = pwr, 965, 8/69, 4/76.**
**Kewaunee (Wis.) = pwr, 503, 8/68, 12/73.**
**LaSalle 1 (Ill.) = bwr, 1036, 9/73, 8/82.**
**LaSalle 2 = bwr, 1036, 9/73, 3/84.**
**Limerick 1 (Pa.) = bwr, 1055, 6/74, 8/85.**
**Limerick 2 = bwr, 1065, 6/74, 7/89.**
**Maine Yankee = pwr, 810, 10/68, 6/73.**
**McGuire 1 (N.C.) = pwr, 1129, 2/73, 7/81.**
**McGuire 2 = pwr, 1129, 2/73, 5/83.**
**Millstone 1 (Conn.) = bwr, 654, 5/66, 10/86.**
**Millstone 2 = pwr, 863, 12/70, 9/75.**
**Millstone 3 = pwr, 1142, 8/74, 1/86.**
**Monticello (Minn.) = bwr, 536, 6/67, 1/81.**
**Nine Mile Point 1 (N.Y.) = bwr, 610, 4/65, 12/74.**
**Nine Mile Point 2 = bwr, 1080, 6/74, 7/87.**
**North Anna 1 (Va.) = pwr, 915, 2/71, 4/78.**
**North Anna 2 = pwr, 915, 2/71, 8/80.**
**Oconee 1 (S.C.) = pwr, 846, 11/67, 2/73.**
**Oconee 2 = pwr, 846, 11/67, 10/73.**
**Oconee 3 = pwr, 846, 11/67, 6/74.**
**Oyster Creek (N.J.) = bwr, 620, 12/64, 8/69.**
**Palisades (Mich.) = pwr, 730, 3/67, 10/72.**
**Palo Verde 1 (Ariz.) = pwr, 1221, 5/76, 6/85.**
**Palo Verde 2 = pwr, 1221, 5/76, 4/86.**
**Palo Verde 3 = pwr, 1221, 5/76, 11/87.**
**Peach Bottom 2 (Pa.) = bwr, 1051, 1/68, 12/73.**
**Peach Bottom 3 = bwr, 1035, 1/68, 7/74.**
**Perry 1 (Ohio) = bwr, 1205, 5/77, 11/86.**
**Pilgrim 1 (Mass.) = bwr, 670, 8/68, 9/72.**
**Point Beach 1 (Wis.) = pwr, 485, 7/67, 10/70.**
**Point Beach 2 = pwr, 485, 7/68, 3/73.**
**Prairie Island 1 (Minn.) = pwr, 503, 6/68, 4/74.**
**Prairie Island 2 = pwr, 503, 6/68, 10/74.**
**Quad Cities 1 (Ill.) = bwr, 769, 2/67, 12/72.**
**Quad Cities 2 = bwr, 769, 2/67, 12/72.**
**River Bend 1 (La.) = bwr, 936, 3/77, 11/85.**
**Robinson 2 (S.C.) = pwr, 665, 4/67, 9/70.**
**Salem 1 (N.J.) = pwr, 1106, 9/68, 12/76.**
**Salem 2 = pwr, 1106, 9/68, 5/81.**
**San Onofre 2 = pwr, 1070, 10/73, 9/82.**
**San Onofre 3 = pwr, 1080, 10/73, 9/83.**
**Seabrook 1 (N.H.) = pwr, 1198, 7/76, 5/89.**
**Sequoyah 1 (Tenn.) = pwr, 1148, 5/70, 9/80.**
**Sequoyah 2 = pwr, 1148, 5/70, 9/81.**
**South Texas 1 = pwr, 1250, 12/75, 3/88.**
**South Texas 2 = pwr 1250, 12/75, 12/88.**
**St. Lucie 1 (Fla.) = pwr, 839, 7/70, 3/76.**
**St. Lucie 2 = pwr, 839, 5/77, 6/83.**
**Summer (S.C.) = pwr, 885, 3/73, 11/82.**
**Surry 1 (Va.) = pwr, 781, 6/68, 5/72.**
**Surry 2 = pwr, 781, 6/68, 1/73.**
**Susquehanna 1 (Pa.) = bwr, 1032, 11/73, 11/82.**
**Susquehanna 2 = bwr, 1032, 11/73, 6/84.**
**Three Mile Island 1 (Pa.) = pwr, 776, 5/68, 4/74.**
**Turkey Point 3 (Fla.) = pwr, 666, 4/67, 7/72.**
**Turkey Point 4 = pwr, 666, 4/67, 4/73.**
**Vermont Yankee = bwr, 504, 12/67, 2/73.**
**Vogtle 1 (Ga.) = pwr, 1079, 6/74, 3/87.**
**Vogtle 2 = pwr, 1165, 6/74, 2/89.**
**Washington Nuclear 2 = bwr, 1095, 3/73, 4/84.**
**Waterford 3 (La.) = pwr, 1075, 11/74, 3/85.**
**Wolf Creek 1 (Kans.) = pwr, 1128, 5/77, 6/85.**
**Zion 1 (Ill.) = pwr, 1040, 12/68, 10/73.**
**Zion 2 = pwr, 1040, 12/68, 11/73.**
**Total as of 12/31/93 = 109.**

Reactor projects for which construction permits were in effect** as of 12/31/93 (cp date shown):

**Bellefonte 1 (Ala.) = pwr, 1235, 12/74.**
**Bellefonte 2 = pwr, 1235, 12/74.**
**Perry 2 (Ohio) = bwr, 1205, 5/77.**
**Washington Nuclear 1 = pwr, 1266, 12/75.**
**Washington Nuclear 3 = pwr, 1242, 4/78.**
**Watts Bar 1 (Tenn.) = pwr, 1165, 1/73.**
**Watts Bar 2 = pwr, 1165, 1/73.**
**Total as of 12/31/93 = 7.**

*Name of plant; type of plant: pressurized water reactor = pwr, boiling water reactor = bwr; electric power output in megawatts (MWe); date of construction permit (cp) issuance; date of operating license (ol) issuance.

**Construction has been halted on a number of these projects.

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