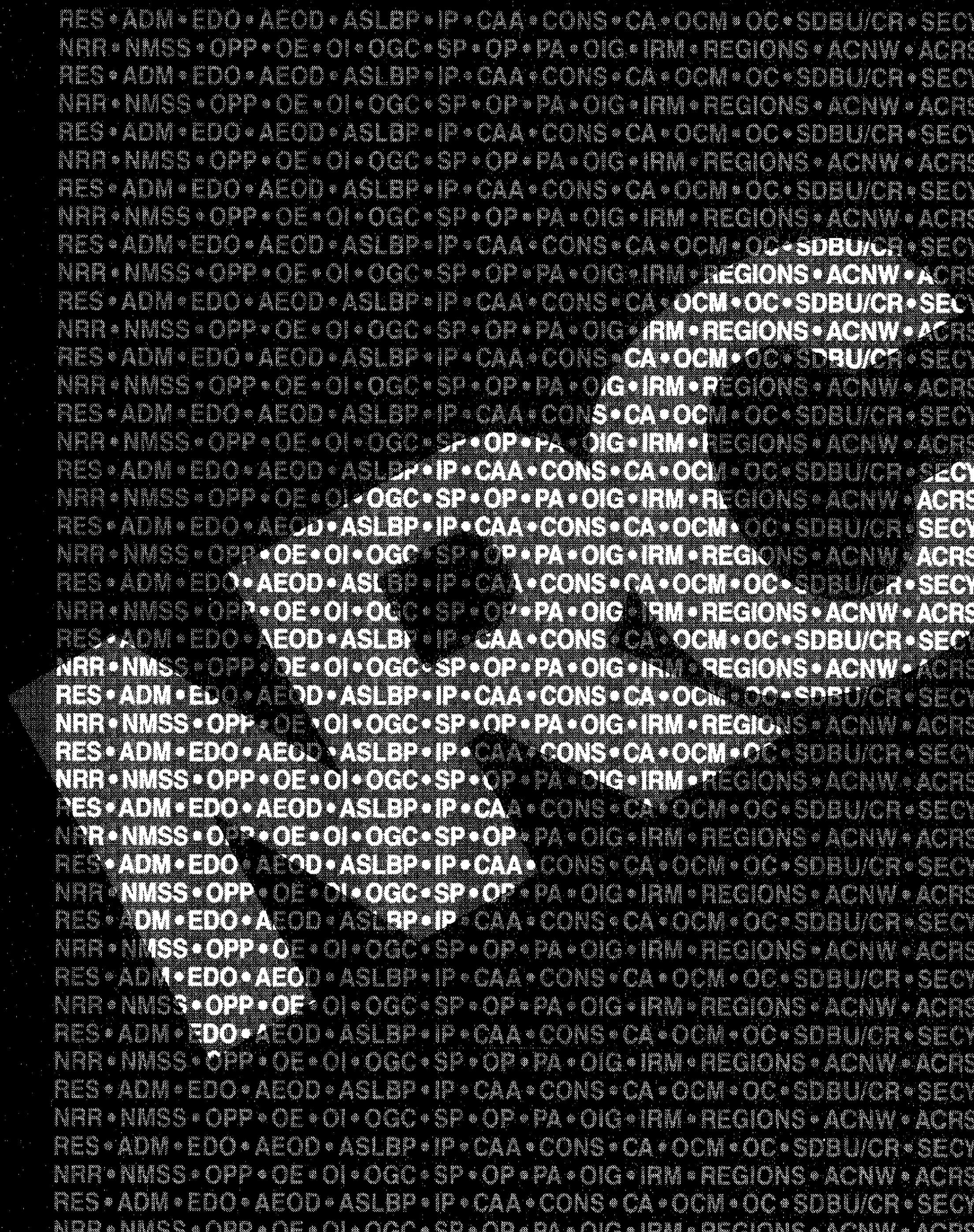
ANNUAL REPORT 1992

United States Nuclear Regulatory Commission



July 31, 1993

The President
The White House
Washington, DC 20500

Dear Mr. President:

This Annual Report for 1992 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 1992, with additional treatment of events after that period where circumstances warranted.

Respectfully,

Ivan Selin

Ivan Selin
Chairman

ANNUAL REPORT 1992

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

NUREG–1145, Vol. 8, *1991 NRC Annual Report*, published July 1992

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Table of Contents

Chapter 1: 1992 Highlights/Licensing and Inspection Summary
(1-6)

Chapter 2: Nuclear Reactor Regulation
(7-53)

Chapter 3: Operational Information/Investigations and Enforcement Actions
(55-96)

Chapter 4: NUCLEAR MATERIALS REGULATION
(97–106)

Chapter 5: SAFEGUARDS AND TRANSPORTATION
(107–113)

Chapter 6: Waste Management
(115–126)

Chapter 7: Communicating With the Public, the Government, and Other Nations
(127–148)

Chapter 8: NUCLEAR REGULATORY RESEARCH
(149–203)

Chapter 9: PROCEEDINGS AND LITIGATION
(205–215)

Chapter 10: MANAGEMENT AND ADMINISTRATIVE SERVICES
(217-230)

APPENDICES:

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, 8 and 10.)

"...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 8; for materials facilities, devices, and transportation packaging, see Chapters 4 and d5; for waste disposal facilities, see Chapters 6 and 8.)

"(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, 7 and 8.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing consistency plans for dealing with such incidents...." (See Chapters 5 and 8.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities...." (See Chapters 6 and 8.)

"(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 8.)

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 8.)

NUCLEAR NONPROLIFERATION 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 7.)

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 10.)

1992 Highlights/Licensing and Inspection Summary

Chapter 1

This is the 18th annual report of the U.S. Nuclear Regulatory Commission (NRC), covering events and activities occurring in fiscal year 1992 (the year ending September 30, 1992), with some treatment of events from the last quarter of calendar year 1992.

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.

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 proper regard and provision for the 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 from theft and 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 1992 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 changes during the report period in the makeup of the Commission and of the creation of a new office in the NRC, and it provides a summary of the licensing and inspection activity treated in detail throughout the report, as well as reporting the status of agency consolidation.

Changes in the Commission

Early in the fiscal year, on December 16, 1991, Dr. E. Gail de Planque was sworn in as a member of the Nuclear Regulatory Commission, bringing the Commission back to a full complement of five members. (Commissioner Thomas Roberts had completed his second term on the Commission in June of 1991.) Commissioner de Planque had formerly served as Director of the Environmental Measurements Laboratory of the Department of Energy, in New York City. (The appointment of Dr. de Planque was covered in the *1991 NRC Annual Report*, pp. 1, 207-208.)

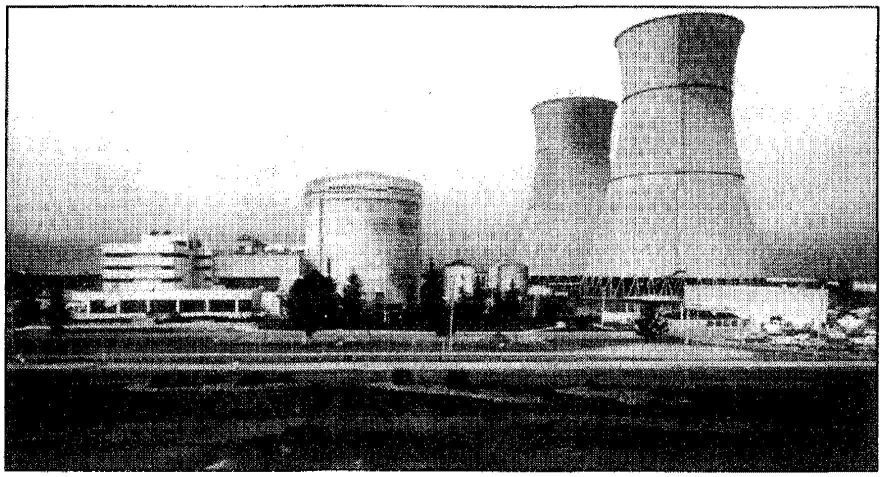
Commissioner Kenneth C. Rogers was reappointed by the President—and confirmed by the U.S. Senate, on May 21, 1992—to a second five-year term on the Nuclear Regulatory Commission, beginning July 1. Dr. Rogers, a physicist, had served for 15 years as President of the Stevens Institute of Technology, in Hoboken, N.J., before his initial appointment to the Commission in 1987.

New NRC Component—Office of Policy Planning

In fiscal year 1992, the NRC Office of Policy Planning (OPP) was created for the purpose of evaluating relevant long-range policy issues from a broad perspective, including consideration of the viewpoints of industry and of public interest groups. OPP serves as the principal advisor to the Commission and to the Executive Director for Operations for policy planning; the Director of OPP is Chairman of the NRC's Steering Committee for Strategic Planning. Richard H. Vollmer was appointed Director of OPP in May 1992, and office operations commenced in July. Mr. Vollmer, who returned to the agency from industry, had formerly held a number of positions within the NRC, since beginning his service (with the former Atomic Energy Commission) in 1968. (See Chapter 10 for background.)

Power Reactor Regulation

Power Reactor Licensing Summary. During fiscal year 1992, the NRC issued no new operating licenses. Two plants that have been permanently shut down—Rancho Seco (Cal.), and Yankee Rowe (Mass.)—were issued "possession-only" licenses. Those actions bring the number of reactors licensed to operate at full power in



The Rancho Seco (Cal.) nuclear power plant, shown at left, was one of two plants permanently shut down during fiscal year 1992 (the other being Yankee Rowe (Mass.); both were issued "possession only" licenses during the period. The Rancho Seco facility, first licensed for operation in 1974, was initially taken our of service in June 1989, as the result of a negative vote by ratepayers/owners on a referendum to allow continued operation by the licensee, the Sacramento Municipal Utility District.

the United States to 110, as of September 30, 1992. There are a total of eight plants, as of the same date, for which Construction Permits have been issued. Most of these are projects which have been halted and/or deferred. There were no new applications for Operating Licenses or Construction Permits during the period, and no Construction Permits were issued.

Licensing Actions for Operating Power Reactors. Either routine activity or unexpected events at a nuclear facility can result in a need for "licensing actions" on the part of the NRC. 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 1992, the Office of Nuclear Reactor Regulation (NRR) completed about 1,620 licensing actions. About 75 percent of these actions were plant-specific and predominantly licensee-oriented. The balance were multi-plant actions deriving from the imposition of NRC requirements. The total licensing action inventory has decreased from about 1,400 to 1,100 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 licensing actions related to major safety issues. The annual report includes the status, as of September 30, 1992, of implementation and verification of all safety-issue actions affecting multiple facilities: TMI Action Plan Requirements, Unresolved Safety Issues (USI), Generic Safety Issues (GSI), and, for the first time, all other multi-plant actions. As reported in the annual report, published in December 1992, more than 99 percent of the TMI Action Plan items have been implemented at the 110 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 the other multi-plant action items have been implemented. (See Chapter 2.)

Renewal of Operating Licenses. The first operating license of a current active plant will expire in the year 2000, 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 an application 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 also on establishing the regulatory framework needed to process such applications. A final rule was published 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 potential age-related degradation and to determine what actions, if any, are needed to ensure continued plant safety during a period of extended operation.

Improving the Licensing Process. The Commission has strongly encouraged the nuclear industry to standardize power reactor designs and to identify sites for nuclear power plants early in the licensing process, and it has issued a rule (10 CFR Part 52) addressing these matters. The focus of the rule is design certification, a regulatory instrument that will permit the early resolution of many licensing issues, with provisions for a combined license and early site permit. Aspects of the effort currently under development include the content of a design certification and the inspections, tests, analyses, and acceptance criteria needed to ensure that the facility is built and can be operated in accordance with the certification. The NRC is reviewing safety analysis reports and a number of other documents pertaining to standardized designs. The guidelines for NRC performance of safety analysis reports is contained in the "Standard Review Plan (SRP) for the Review of Safety Analysis Reports."

Power Plant Maintenance. On July 10, 1991, the Commission published, in the *Federal Register* (56 FR 31306), a new maintenance rule, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (10 CFR 50.65). The rule, expected to be fully implemented by 1996, requires 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.

During fiscal year 1992, the NRC and the industry developed parallel implementing guidance documents, with the prospect that the NRC would endorse the NUMARC guidelines in a Regulatory Guide, at such time as it found those guidelines to be acceptable. After a number of public meetings to discuss the subject with NUMARC, and after further revision of the NUMARC guidelines, the NRC determined that the industry guidelines were sufficiently acceptable to proceed with a trial verification-and-validation program for the guidelines.

Special Reactor Plant Inspections. During fiscal year 1992, the NRC headquarters and regional staffs performed 48 special team inspections, each involving 4-to-10 inspectors and requiring 2-to-4 weeks to complete. The objective of these inspections was to determine whether, when called upon to do so in an emergency, nuclear plant systems and personnel would perform their safety functions in the way set forth in the facility's Safety Analysis Report.

The program of Electrical Distribution System Functional Inspections, developed in 1990, was continued in 1992. As of the end of the fiscal year, these specialized inspections had been performed at the plants on 55 sites. NRC plans to complete the program at the remaining 14 sites by mid-1993.

Development was continued in 1992 of two new types of team inspections, Service Water Systems Operational Performance (SWSOP) and Shutdown Risk and Outage Management (SROM). Four SWSOP and two SROM pilot inspections were completed, testing and developing the methodology. The NRC plans to proceed with the SWSOP inspections at sites with perceived service water problems, problem plants, and older facilities. Additional SROM pilot inspections are also planned.

Grand Gulf OSART Inspection. From August 3-21, 1992, an Operational Safety Assessment Review Team (OSART) from the International Atomic Energy Agency visited the Grand Gulf nuclear power plant—a 1,250 megawatt, single unit, boiling water reactor facility, located near Port Gibson, Miss. The visit came about at the request of the United States, for the purpose of reviewing facility operating practices and exchanging technical knowledge and experience regarding ways and means to achieve excellence in operational safety. The OSART comprised 12 international experts and three observers, whose collective nuclear experience totaled over 260 years.

The OSART inspection report stated that the team was greatly impressed with the commitment of management and staff to the achievement of high levels of safety in the operation and maintenance of the plant. The OSART found that the utility (Entergy Operations, Inc.) was well managed and actively supported the nuclear power operation by providing clear policy direction and adequate resources. The OSART also found the plant management and supervisory staff to be dedicated to their tasks, the operating and maintenance personnel to be well trained and highly motivated, and good technical support to be available at both the corporate and plant levels.

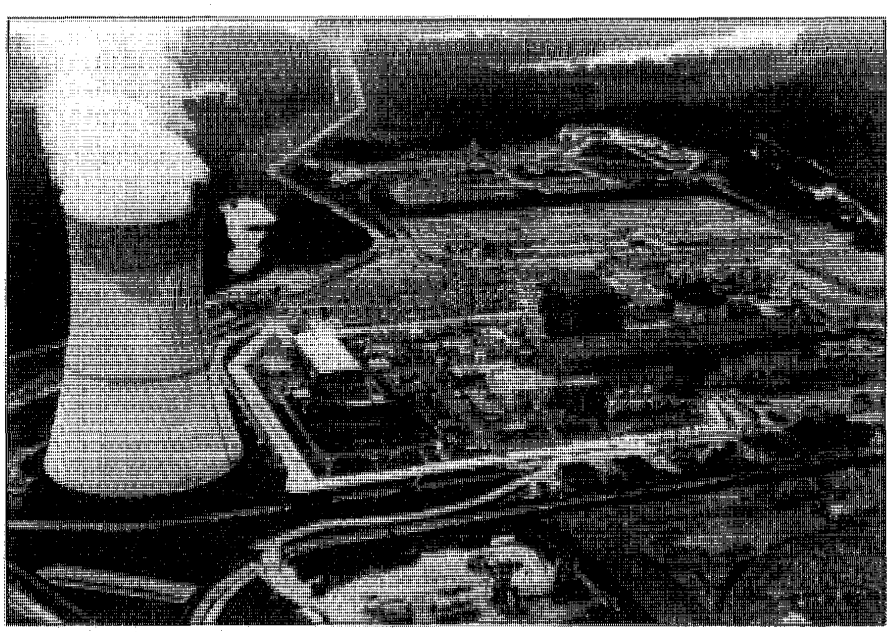
The OSART made a number of recommendations for the management of the licensee and of the Grand Gulf plant to consider. The utility will prepare a detailed response to the final OSART report. The NRC will take cognizance of the status of the licensee's response to the OSART recommendations. Conclusions reached by the OSART are in substantial agreement with the NRC's assessment of the performance of the Grand Gulf nuclear power plant and of the licensee over the past several years.

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 in a number of ways, one of them involving installation of 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, currently, Thermo-Lag fire barriers are 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 resistive ratings and ampacity derating factors (lowering the current-carrying capacity of cables to account for the insulating effects of the fire barrier) for Thermo-Lag were indeterminate, and that some evaluations of test results and some procedures employed in 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 resistive protection needed to satisfy the NRC's requirements. The staff has incorporated these issues into an action plan to ensure that the issues are tracked, evaluated and resolved. The staff has also issued five information notices to the industry, a generic letter, a bulletin and a bulletin supplement; developed a proposed staff position for fire endurance test criteria; reviewed various industry full-scale test programs; and conducted toxicity, combustibility, and small- scale fire tests. For the short-term, licensees have addressed the fire endurance problem by implementing compensatory measures, such as fire watches, where Thermo-Lag has been installed. Long term action to correct the problem ranges from barrier upgrades and repairs to complete replacement of some barriers or relocation of the affected cables. Additional plant-specific analyses may also be required to address cable derating. Meanwhile, the staff is evaluating other fire barrier materials and systems used by licensees. 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.)

Technical Specifications Improvements. The NRC has undertaken an extensive effort to improve power reactor technical specifications, since issuing a proposed policy statement on the subject in 1987. Under NRC regulations, technical specifications are incorporated into the operating license for a power reactor facility to specify safety limits, limiting conditions for operation, surveillance requirements, design features and administrative controls that are necessary to ensure the safe operation of the facility. Improved Standard Technical Specifications (STS) were completed in June 1992 for each of the major nuclear steam supply system vendor designs. These improved STS are currently being implemented at "lead plants" on a voluntary basis. Improvements to the STS include applying human factors principles in the formatting of technical specifications; focusing more on safety in defining the scope of the technical specifications; expanding the bases of the technical specifications, to make them more explanatory; and achieving greater consistency among the STS of the different vendor owners groups. These efforts are intended to facilitate plant operator use and understanding of the technical specifications, and thereby to lead to improved safety. Interim improvements to current technical specifications, and the use of risk insights for sharpening technical specifications, are being encouraged.

Vendor Inspections. In fiscal year 1992, the NRC vendor inspection staff conducted 34 vendor and licensee inspections. Several other vendor inspections were carried out by the vendor inspection staff in providing technical support to the NRC Office of Investigations. Five inspections of licensees were conducted to review vendor procedures and their implementation for the procurement of commercial grade parts, components and materials for use in safety-related applications. The vendor inspection staff also assisted the NRC Office of Investigations and various U.S. Attorneys in ensuing criminal cases. (See Chapter 2.)



An Operational Safety Assessment Review Team (OSART) from the International Atomic Energy Agency visited the Grand Gulf (Miss.) nuclear power plant for a 19-day review of operating practices at the U.S. facility and for exchanges of technical information. The boiling water reactor plant is located on the Mississippi River and has been licensed to operate since 1984. The OSART was impressed with the high level of safety achieved in operations and maintenance at the plant.

Table 1. License and Annual Fee Collections FY 1992

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR Part 170	$93.1 million	$13.4 million	$106.5 million
10 CFR Part 171	$341.4 million	$41.4 million	$382.8 million
TOTAL FEES	$434.5 million	$54.8 million	$489.3 million

Nuclear Materials Regulation

Nuclear materials regulation during fiscal year 1992 included 70 licensing actions involving fuel cycle plants, facilities, and spent fuel issues; about 2,700 fuel facility and materials licensee inspections; and about 6,100 licensing actions on applications for new byproduct materials licenses, amendments and renewals of existing licenses, and reviews of sealed sources and devices.

Materials Licensing and Inspection. The NRC currently administers approximately 7,200 licenses for the possession and use of nuclear materials in medical and industrial applications. This total represents a reduction of about 600 licenses in the past year, attributable in part to the State Agreement reached with Maine (shifting some licensing activity to the State), and also to the full-cost recovery license fee rule (causing some licensees to decline renewal). NRC regional staff completed approximately 2,700 inspections of materials facilities in fiscal year 1992.

The NRC completed over 6,100 licensing actions during the fiscal year. Of this total, over 400 were new license issuances, 4,400 were license amendments, 900 were license renewals, and 400 were sealed source and device design reviews. (See Chapter 4.)

Safeguards Activity—Protecting Against Theft/Sabotage

Reactor Safeguards Inspection and Licensing. Within the five NRC Regional Offices, a total of 170 safeguards inspections were conducted at licensed nuclear power reactors under NRC safeguards requirements. Approximately 212 revisions to licensee security, contingency, and guard training plans were reviewed and found acceptable by both regional and headquarters staff.

Non-power Reactors (NPRs). NRC conducted 33 safeguards inspections of non-power reactors (NPRs) during fiscal year 1992. Efforts are continuing toward converting 25 NPRs from the use of HEU to low-enriched uranium (LEU) fuel.

Fuel Cycle Facility Inspections. Comprehensive physical security and material- control-and-accounting (MC&A) inspections were conducted at the major U.S. fuel fabrication facilities. Newly implemented physical security improvements were thoroughly inspected at the two facilities possessing significant quantities of HEU. Performance-based inspection procedures were followed for both MC&A and physical security inspections.

Transport Inspection and Enforcement. The NRC continued to conduct safeguards inspections of selected shipments of spent fuel. No significant problems were identified from inspections carried out during the report period. The NRC also continued its transportation-related safety inspection program. The total effort involved approximately 1,400 individual inspections covering byproduct, source, and SNM licensees, and including fuel cycle facilities and shippers of spent reactor fuel.

An inspection program to ensure that transportation containers certified by the NRC are fabricated in accordance with the NRC-approved design and quality assurance programs of the container suppliers continued in fiscal year 1992. Inspections were conducted at eight facilities, representing a broad spectrum of the industry. (See Chapter 5.)

NRC License and Annual Fees

The Omnibus Budget Reconciliation Act of 1990 (Public Law 101–508) requires that, in fiscal year 1992, 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 1992, a total of $512.5 million was appropriated to the NRC (Public Law 102–104), of which $19,962,000 was derived from the Nuclear Waste Fund. Of the remaining $492,538,000, approximately 99 percent, or $489,265,320, was collected through license fees and annual charges, resulting in a net appropriation to the NRC of $3,272,680, for fiscal year 1992. Table 1 shows the amounts collected through license and annual fees in fiscal year 1992.

Consolidation of NRC Headquarters

At the close of fiscal year 1991, the first stages of site clearing and excavation for Two White Flint North (TWFN) had begun. By the end of fiscal year 1992, the base-building construction of the 10-story, 364,000 square foot building was nearing completion. Installation of the exterior concrete pre-cast panels and windows had commenced.

Occupancy for more than 1,300 people is scheduled for early calendar year 1994. During fiscal year 1992, preliminary space and furniture plans were developed for the 12 offices that will occupy TWFN. In addition, design layouts were developed for the state-of-the-art Emergency Operations Center, central computer facility, multi-purpose auditorium, day-care facility, physical fitness center, an expanded staff training facility, and other resources for the use of the 2,450 people working in the two-building complex.

Nuclear Reactor Regulation

Chapter 2

The Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC) is responsible for the development and issuance of regulations for the safe operation of the nation's operating nuclear power and research reactors and for the assessment of applications to construct and operate new reactors and the issuance of 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. (Reactors operated by the Department of Energy (DOE) for the purpose of furnishing fissionable materials for use in nuclear weapons are not regulated by the NRC.) 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 provides direction to, and oversight of, the NRC Regional Offices in the areas of reactor licensing and inspection activity.

The licensing activity of NRR begins with the extensive review given to 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. (See "Licensing the Nuclear Power Plant," on the next page.)

In recent years, the steady increase in the number of licensed operating nuclear plants and decrease in the number of plants still under construction have brought about a substantial shift in NRC activity. NRC staff energies are now directed mainly to the safety regulation of the 110 nuclear power plants licensed for operation in the United States, as of the close of fiscal year 1992. At the same time, increased attention is being given to the development of criteria and procedures for conducting safety reviews of the advanced reactor designs proposed for nuclear plants of the future.

Regulatory activity related to nuclear power plants during fiscal year 1992 is treated in this chapter under the following headings:

- Status of Licensing
- Plant License Renewal
- Improving the Licensing Process
- Inspection Programs
- Performance Evaluation
- Quality Assurance
- Operator Licensing
- Emergency Preparedness
- Safety Reviews
- Antitrust Activities
- Property Insurance
- Advisory Committee on Reactor Safeguards.

Where appropriate, the NRC may decide to grant a Limited Work Authorization to an applicant in advance of a final decision on the construction permit, in order to allow certain work to begin at the site; such a step can save months in overall construction time. This authorization will not be given until the NRC staff has completed its environmental impact and site suitability reviews, and the ASLB for the project 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.

LICENSING THE NUCLEAR POWER PLANT

The first step in the nuclear power plant licensing process is the filing with the NRC of an application by a utility for a construction permit. 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 next phase encompasses various safety, environmental, safeguards (from theft or sabotage), and antitrust reviews undertaken by the NRC staff. Thereafter, as required by law, the independent Advisory Committee on Reactor Safeguards, or ACRS, carries out an assessment of the proposed project and of 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, or 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 initial application of a utility is accepted (or "docketed") by the NRC staff, a notice of the fact is published in the *Federal Register*, and copies of the application are furnished to the appropriate State and local authorities and to a local public document room, or LPDR, established by the NRC in the vicinity of the proposed plant site, and also to the NRC public document room in Washington, D.C. At the same time, a notice of a public hearing is published 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 the guidance of the Standard Format (Regulatory Guide 1.70), the applicant for a construction permit lays out the proposed nuclear plant design in a Preliminary Safety Analysis Report, or PSAR. If and when this report has been made sufficiently complete to warrant review, the application is docketed and the NRC staff evaluations, mentioned above, begin. The staff's safety, environmental, safeguards, and antitrust review proceed in parallel. Even before submission of a safety report, NRC staff will conduct a substantive review and inspection of the applicant's quality assurance program with respect to design and procurement activities. 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 employed and accurately carried out; whether the applicant has conducted its analysis and evaluation in sufficient depth and breadth to support a staff conclusion that adequate levels of safety are assured. When the NRC staff is satisfied that the acceptance criteria of the Standard Review Plan have been met by the applicant's preliminary report, Safety Evaluation Report is prepared by the staff summarizing results of its review with regard to 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, in the form of a 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. At this stage, 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 be held, generally in a community near the proposed facility site, on the safety aspects of the licensing decision.

Where appropriate, the NRC may decide to grant a Limited Work Authorization to an applicant in advance of a final decision on the construction permit, in order to allow certain work to begin at the site; such a step can save months in overall construction time. This authorization will not be given until the NRC staff has completed its environmental impact and site suitability reviews, and the ASLB for the project 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 complete to warrant review, it is docketed, and an analysis of the consequences to the environment from the construction and operation of the proposed facility is undertaken. 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 is conducting analyses and preparing a report on the site suitability concerns of the proposed licensing action. Upon completion of 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. (In the alternative, where indicated, 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 Attorney General in advance of, or concurrent with, 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 1992

Low-Power Operating License issued	0
Full-Power Operating Licenses issued	0
Operating License applications under review	8

STATUS OF LICENSING

Reorganization of the Office of Nuclear Reactor Regulation

On October 4, 1992, the Office of Nuclear Reactor Regulation (NRR) was reorganized along structural lines that reflect the increasing commitment of NRR resources to "evolutionary" and advanced reactor activities. The organizational realignments were needed to more effectively provide for NRR's dual focus—giving regulatory oversight to operating reactors while carrying out a thorough review and evaluation of advanced plant and license renewal activities.

The new organization places management of technical activities under the Associate Director for Inspection and Technical Assessment (ADT), transfers some functions to the Associate Director for Projects (ADP), and transfers the responsibility for conducting environmental reviews from ADP to the Associate Director for Advanced Reactors and License Renewal.

The number of technical divisions within NRR remained at six. Responsibility for generic activities, event follow-up and standard technical specifications was realigned from ADT to ADP. The organizational change includes realigning six branches, creating a new branch, and consolidating environmental functions into a single branch level organization.

Reactor Engineer Intern Program

The Reactor Engineer Intern Program was established in 1988 to help 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. Approximately two-thirds of the 49 interns currently in the program are based in Headquarters Offices.

By means of a series of individually tailored assignments at Headquarters, Regional Offices, and plant sites—coupled with extensive formal training in nuclear reactor technology—Reactor Engineer Interns are given wide exposure to the NRC's activity, so that they may acquire a broad grasp of the various concerns, roles and tasks of the agency. Upon completion of a rigorous two-year program, interns are given permanent technical professional assignments, based on their educational background, personal and career preferences, and on the needs of the agency. Fourteen interns completed the program during the past year and were recognized at a graduation ceremony in May 1992; they have assumed permanent positions in Headquarters and the Regions. A total of 18 interns have graduated from the program since its inception.

License Applications, Issuances and Decommissioning

During fiscal year 1992, the NRC issued no new operating licenses. Two plants were changed from "indefinitely" to "permanently" shut down. These plants were the Fort St. Vrain (Colo.) and Rancho Seco (Cal.) nuclear power plants. This brings the number of reactors licensed to operate at full power in the United States to 110 as of September 30, 1992. (See Appendix 7 for a complete listing of plants in operation or under construction, with location, reactor type and other data.) There were no new applications for operating licenses or construction during the report period and no construction permits were issued.

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. An underlying assumption embodied in the rule is that a permanent cessation of operations at a given facility would not occur before completion of the full 40-year term of a nuclear power plant operating license.

More recently, several licensees have announced their decisions to permanently cease power operations at nuclear power generating facilities before expiration of their operating licenses. The reasons for these decisions are related to various political, technical, or economic problems, but the result is that the facilities have, in terms of the decommissioning requirements rule, prematurely entered into the decommissioning process. During fiscal year 1992, the NRC staff completed most of its review efforts necessary to support premature decommissioning of the Shoreham (N.Y.), Fort St. Vrain (Colo.), and Rancho Seco (Cal.) facilities. The NRC staff issued an order to the current Shoreham licensee, Long Island Power Authority, approving the Shoreham decommissioning plan. The Long Island Power Authority is now in the process of dismantling that facility. The NRC staff expects to complete the reviews of both the Fort St. Vrain and Rancho Seco decommissioning plans early in fiscal year 1993. Also during fiscal year 1992, the Yankee Atomic Electric Company, the Southern California Edison Company, and the Portland General Electric Company announced their decisions to prematurely shut down and decommission the Yankee-Rowe (Mass.), San Onofre Unit 1 (Cal.), and Trojan (Ore.) facilities, respectively. In February of 1992, Yankee Atomic Electric declared that Yankee-Rowe had permanently ceased operation. Southern California Edison Company planned to cease operations at San Onofre Unit 1 in November of 1992, after the close of the report period. The Trojan facility will not be permanently shut down until 1996. The NRC staff has begun actions to ensure the orderly decommissioning of these facilities.

Licensing Actions for Operating Power Reactors

Either routine activity or unexpected events at a nuclear power facility can result in a need for "licensing actions" on the part of NRC. Routine post-licensing activities affecting reactor operations include such matters as license amendment requests, possibly involving public hearings; requests for exemptions from regulations; backfit modification requests based on new NRC regulations, or orders for modifications of a license. During Fiscal year 1992, NRR completed about 1,620 licensing actions. About 75 percent of these were plant-specific and predominantly licensee-initiated. The balance were multiplant actions resulting from NRC-imposed requirements. The inventory of licensing actions has decreased from about 1,400 under review at the end of fiscal year 1991 to about 1,100 under review at the end of fiscal year 1992.

Special Cases

FitzPatrick. The FitzPatrick (N.Y.) nuclear power plant is a boiling water reactor plant owned and operated by the New York Power Authority, the licensee. In light of declining performance at the facility, in June 1991, NRC senior management concluded that an NRC diagnostic evaluation was warranted to provide an independent assessment of licensee performance. This evaluation was conducted in September and October 1991, by a Diagnostic Evaluation Team which found deficiencies in the following areas: the operator requalification program, the support facilities, plant material condition, the procedure change process, the use of operating experience for plant improvements, QA effectiveness, the fire protection program, the preventive maintenance program, the root cause analysis program for equipment failures, engineering evaluations, the configuration control program, headquarters support and oversight, site management oversight and control, and the corrective action program.

The team determined that there were six root causes for the apparent poor performance at the plant. These were:

- Failure of corporate management to adequately and effectively plan for the operational support of FitzPatrick.
- Inadequate management oversight and direction.
- Ineffective resource allocation and utilization.
- Ineffective use of industry experience.
- Insufficient standards for performance.
- Ineffective communications and teamwork between the plant and corporate headquarters.

The plant was shut down on November 27, 1991, in order to resolve a design deficiency in the core spray system containment isolation logic. On December 6, 1991, the licensee decided not to restart the plant before the scheduled January 11, 1992 refueling outage, as initially planned. The licensee concluded that restart of the plant would not be prudent without a thorough assessment of fire protection and certain other deficiencies (under 10 CFR Part 50, Appendix R), identified by the NRC and their own staff. This judgment was precipitated by an unexpectedly high failure rate uncovered during penetration seal inspections. The plant remained in an extended refueling outage throughout the remainder of this period.

In December 1991, the licensee implemented a comprehensive, long term Results Improvement Program designed to improve overall performance at the plant and the corporate office. Furthermore, the licensee implemented significant organizational and management changes. Concurrently, the NRC established a FitzPatrick Assessment Panel to assess the adequacy of the Results Improvement Program and to coordinate agency resources for monitoring and assessing licensee performance. 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.

In the NRC staff's Systematic Assessment of Licensee Performance (SALP) report for the period of February 1, 1991, through April 18, 1992, decline in performance was noted in the functional areas of Operations, Maintenance/Surveillance, and Engineering/Technical Support. The level of performance in four functional areas was determined to be only adequate, a conclusion calling for increased management attention to ensure a continued acceptable level of performance. The four areas included those noted above, as well as Safety Assessment/Quality Verification. However, "superior" performance was demonstrated in the areas of Security and Emergency Preparedness, and "good" performance was noted in the area of Radiological Controls.

Besides the weaknesses identified by the Diagnostic Evaluation Team, deficiencies in a number of programs were found by the licensee and by the NRC staff. The NRC conducted several inspections, including team inspections, of the fire protection program and emergency service water system. The inspections identified several violations, including ones associated with:

(1) Inadequate control of a design modification for the analog transmitter trip unit system.

(2) The failure to identify and correct certain conditions adverse to quality.

(3) Inadequate implementation of the fire protection program.

(4) The failure to meet certain 10 CFR Part 50, Appendix R, requirements.

(5) The submittal of incomplete and inaccurate information to the NRC.

The NRC held conferences on March 18 and June 24, 1992, to discuss the violations and their causes, as well as proposed corrective actions. Subsequently, the NRC undertook escalated enforcement against the licensee, on September 15, 1992. Five Severity Level III violations with a cumulative Civil Penalty of $500,000 were assessed. The licensee requested that the NRC reconsider the Civil Penalty in light of the extensive corrective actions and improvement efforts implemented. This request was under consideration by the NRC at the close of the report period.

The plant remained in an extended refueling outage throughout this entire period. The outage was significantly extended, in order 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 August and September 1992, the licensee conducted its own assessment of the plant's readiness for start-up. The licensee's Restart Readiness Report was submitted on September 28, 1992, indicating that, pending completion of certain specified items, the unit could be restarted. In October 1992, the NRC conducted a Restart Readiness Team Inspection, in order to have an independent, in-depth evaluation of the readiness of plant management, programs, equipment and staff to carry out the safe restart and operation of the FitzPatrick plant. The team noted significant improvement in performance in each of the areas reviewed, while identifying several issues requiring resolution prior to start-up. However, the team also concluded that, following resolution of the start-up issues identified by the licensee and the NRC, the management controls, programs, plant equipment and personnel were adequate to conduct safe restart and operation of the plant. Upon resolution of these issues, the licensee will seek authorization for restart from the NRC, after the close of this report period.

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), comprises two General Electric 849 megawatt (electric) boiling water reactors. In July 1992, the plant 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.

In April 1992, the licensee shut down both units when it was determined that a number of interior walls in the diesel generator building had not been installed according to requirements regarding the "design basis" seismic qualification. It was found that the failure of these walls during an earthquake could affect operation of the diesel generators in the building. Subsequent to the shutdown, the licensee proceeded to perform a thorough review of the structural integrity of all the safety walls and initiated a program to review the structural steel in the reactor buildings and the reactor containments (drywells).

The list of physical deficiencies at the Brunswick plant is not limited, however, to seismic qualification issues. A number of pieces of safety equipment and their supports were found to have sustained damage from corrosion, and the diesel generators to be in need of repair, because of damage discovered during surveillance and performance tests. An extensive list of backlog maintenance and equipment "trouble-ticket" items also remained to be dealt with. The NRC is closely monitoring the licensee's correction of the physical deficiencies and is performing inspections and audits to oversee the proper implementation of corrective action programs. The licensee has agreed to a complete implementation of programs to correct physical inadequacies before returning either of the units to service, expected to occur sometime in 1993.

Over and above concerns about the physical condition of the plant, the NRC has informed the licensee of a need to correct the root cause of the decline in performance by improving management effectiveness in controlling the processes and activities at the plant. The licensee has taken the following actions in response to the concerns of NRC management:

(1) The management of nuclear generation has been restructured both at the corporate level and at the Brunswick site.

(2) Increased resources—both in capital improvements and in the strengthening of management and technical professionals—have been committed to the nuclear facilities.

(3) Improvement initiatives were begun covering three major elements: BSEP start-up actions, BSEP three-year business plan covering specific performance improvement initiatives, and a corporate improvement initiative.

The NRC is continuing its oversight during the outage and the various maintenance and repair activities taking place in that period. By November 30, 1992, the licensee was to provide the staff with a statement of its corporate improvement initiatives, a description of the scope of work to be performed prior to start-up of Units 1 and 2, and an integrated schedule for Unit 2 start-up. By December 15, 1992, the licensee was to provide the staff with the utility's three-year business plan and the integrated schedule for unit start-up. Periodic meetings of the NRC staff and licensee management will continue.

Turkey Point. The Turkey Point (Fla.) nuclear power plant is situated on the shores of Biscayne Bay, about 25 miles south of Miami, and is the site of four electric generation units, owned and operated by the Florida Power and Light Company (FPL). Turkey Point Units 1 and 2 are oil- and gas-fired. Units 3 and 4 are pressurized light water nuclear units, each designed to produce 760 megawatts of electrical power. The area within 10 miles of the site encompasses an approximate population of 100,000. This area undergoes tropical storms about once every two years and hurricane winds once every seven years.

On August 24, 1992, Class 4 Hurricane Andrew hit south Florida. In its preliminary report, the National Hurricane Center estimated that the storm, during landfall over south Florida, generated sustained surface wind speeds (the one-minute average at 10 meters elevation) of 145 miles-per-hour. Several unofficial reports estimated stronger gusts. The eye of the storm passed over the Turkey Point site and caused extensive on-site and off-site damage throughout the 10-mile emergency planning zone around the plant. South Florida was declared a disaster area by the President and a Federal Response Plan was activated, bringing assistance from various

When Hurricane Andrew struck south Florida, in August 1992, with surface winds of 145 m.p.h. and above, the storm passed directly over the Turkey Point nuclear power plant, causing very extensive damage at the site and throughout the area. The plant, shown here, is the site of two pressurized water reactors, as well as non-nuclear generating units. Operations were resumed after the close of the report period, following repair and restoration of the units.

Federal agencies, including the Federal Emergency Management Agency (FEMA) and the NRC. The plant safety systems functioned as designed and the plant remained in a stable condition throughout the storm and thereafter. An NRC/Industry Task Force has been formed to collect and organize information to be gained from the event. The Task Force was expected to complete its review and issue its report within the next few months.

Two hours prior to the estimated arrival of the storm—in accordance with its emergency plan procedures for severe weather conditions—the licensee brought Units 3 and 4 to a "hot shutdown" condition. The storm caused extensive damage to the site, including complete loss of off-site power; loss of communications; loss of access by road; and damage to the fire protection and security systems, to the material warehousing facility, and to the fossil-fuel units' smoke stacks. But there was no damage to safety-related systems at Units 3 and 4, and there was no radioactive release to the environment. FPL declared an "Unusual Event," upon issuance of the hurricane warning for south Florida. The warning was subsequently upgraded to an "Alert," because of a degradation of the fire protection system after Hurricane Andrew hit the site. The utility promptly activated the Turkey Point Operation Support Center and the Technical Support Center. Similarly, the NRC activated its monitoring mode on August 24, 1992, and remained in that mode until August 31, 1992, when off-site power to both Units 3 and 4 was restored from a single off-site power source. NRC representatives were sent to the State Emergency Operations facility in Tallahassee and to FPL's Emergency Operations Facility in Miami. NRC resident inspectors provided 24-hour coverage at the site during significant plant operations.

After the storm, the licensee began implementing its recovery plan, which comprised three distinct priorities—stabilization of plant, damage assessment, and restart. Following its and NRC staff's damage assessments and inspections, the licensee determined which items to repair, restore, retest, or address as a pre-requisite to returning the units to service. NRC Headquarters and Region II staffs closely monitored plant recovery and restart activities. Restoration activities concentrated on Unit 4 systems to facilitate restart. Storm damage repairs on Unit 3, which was scheduled to commence during a regular refueling outage on August 24, 1992, were put on hold pending completion of Unit 4 restart.

As a result of the storm, an elevated service water storage tank collapsed and caused damage to a water supply system. During the recovery operation, this tank was eliminated. The storm also caused damage to the fossil-fuel Units 1 and 2 stacks. The Unit 1 stack, which sustained significant damage, was dismantled for reasons of personnel safety, using controlled demolition techniques. The Unit 2 stack sustained only minor cracks. The utility demonstrated by analyses that the unit 2 stack, in its then current condition, had adequate strength margins to withstand its original "design wind loading" without adverse interaction with the nuclear units. The licensee plans to modify the Unit 2 stack with new structural reinforcements prior to the next hurricane season.

To minimize future potential risk of loss of communication, the utility introduced improved designs into its existing site communication systems. Aerial wire communications systems to the site were replaced by a buried fiber-optic cable system containing dedicated circuits for State, Federal and local agency notifications. Backup microwave links between the site and FPL's corporate office and high frequency automatic long-range communication links between the facility and the NRC Offices, as well as with State and local governments, were also established. Antennas associated with these systems are designed to withstand a Class 5 hurricane.

Upon completion of the storm damage repairs to Unit 4, the staff performed an evaluation of on-site issues relevant to the restart, and Unit 4 was restarted on September 28, 1992. On October 1, 1992, the licensee executed a voluntary shutdown when it learned that FEMA had not completed its post-hurricane re-verification of the adequacy of off-site emergency planning facilities and equipment located within the 10-mile EPZ around the site. FPL suspended operation of the unit until the FEMA re-verification was completed. Upon reaffirmation by FEMA of the adequacy of off-site emergency planning, on October 25, 1992, power operations at Unit 4 were resumed.

Commonwealth Edison Company. The Commonwealth Edison Company 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 22 years, for Dresden, to five years, for Braidwood. Each site houses two operating reactors, giving the utility a total nuclear generating capacity of 11,500 megawatts-electric.

In the course of monitoring and evaluating operations at these plants under the Systematic Assessment of Licensee Performance (SALP) program, the NRC found that activities at the Byron plant exhibited generally excellent performance, and the Braidwood and LaSalle operations demonstrated good performance. Performance of the three older plants—Dresden, Quad Cities, and Zion—was deemed acceptable but in need of added NRC attention. In 1991, regulatory concerns with declining performance at both Dresden and Zion prompted the NRC to add these plants to its list of operating plants that warrant increased NRC attention.

The NRC has been closely monitoring the corrective action programs and efforts to improve plant performance at Dresden and Zion. Close surveillance is maintained through increased inspections by the resident and region-based inspectors, and by a Dresden Oversight Team (DOT) and a Zion Review Team (ZRT). The DOT and ZRT consist of personnel from Headquarters and Region III, management and staff, who periodically visit the sites to evaluate licensee performance and appraise the status of the improvement programs. More recent surveys showed improvements in many areas at both facilities. Still, the NRC has determined that continued close monitoring is warranted until the licensee has demonstrated sustained improved performance at Dresden and Zion.

During the first half of 1992, the NRC performed an extensive review and assessment of the utility's overall performance and concluded the following:

(1) Recent performance weaknesses at Dresden were similar to, but not as severe as, those identified in 1987, when Dresden was first placed on the list of plants warranting increased NRC attention. Although there have been improvements in many areas—such as scram rates, equipment and material condition, communications, modifications, and design support—some weaknesses were not fully corrected, with a resultant decline in performance in 1991, compared with the performance appraisal of 1988, when the plant was removed from the watch list.

(2) Deficiencies at Dresden in 1991, although similar to those at Zion, were not as extensive. The same kinds of problems were found at Quad Cities, but they were neither as severe nor extensive as at the other two, and the licensee appears to be effectively addressing them. Although there have also been some performance problems at Byron, Braidwood and LaSalle, overall performance at Byron has been excellent, and at Braidwood and LaSalle, performance has been good.

(3) The probable root causes for the utility's difficulties were determined by the NRC staff to be the following: (a) insufficient management attention and resources were committed to operating sites, while the construction of new facilities was causing hardware and programmatic problems that are still in evidence; (b) the limited effectiveness of corporate level oversight of nuclear operations brought about disparities in the quality of operations at the various sites; (c) the licensee was slow to recognize situations requiring increased management attention and to ensure permanent correction of problems, as they came to light; (d) weak engineering support to the operating reactor plants brought on equipment operability concerns that were not being addressed promptly, as well as modifications that were not being properly implemented; and (e) the utility had not substantially benefitted from experiences of other utilities or from experience at its own sites.

The licensee has developed an Integrated Management Action Plan to assist it in making fundamental changes in its nuclear organization, culture, communications, and management processes; the plan is intended to bring about:

(1) Improved operational and managerial effectiveness and efficiency, through focused accountability, clarity of organizational roles and responsibilities, enhanced executive oversight, and preventive management planning and action.

(2) A management philosophy centered on improved plant performance through "customer oriented" support; re-emphasis on managing, rather than engineering; solving problems promptly; and providing rational resource allocations, as well as appropriate motivations, to achieve goals.

(3) Clear and mutually supportive interaction between the nuclear department and other company elements, through improved liaison, clarity and simplicity of oversight and support functions, and improved vertical and horizontal communications.

(4) Improvements in key management systems and processes.

NRC staff has expanded its monitoring effort regarding the exercise of the licensee's corporate oversight of the six nuclear sites, and of corporate appraisal of priorities among issues affecting each plant.

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), pointing up significant continuing weaknesses in TVA performance and indicating 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 surfaced at the TVA's Watts Bar (Tenn.) project.

Sequoyah. Sequoyah Units 1 and 2 were restarted in November and May 1988, respectively, following NRC staff inspection, approval of TVA's corrective actions, and NRC authorization for restart. In June 1989, NRC senior management decided to remove the Sequoyah site from the category of plants requiring special attention. (New concerns arising after the close of the report period will be covered in the next NRC annual report.)

Browns Ferry. Unit 2 was shut down in September of 1984 for a planned refueling outage. Units 1 and 3 were shut down 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. From the beginning, TVA had decided to focus its efforts exclusively on Unit 2 in developing and implementing necessary corrective actions; restoration of Unit 3, and then Unit 1, would follow. In August of 1991, Unit 2 returned to normal full power commercial operation, after successful completion of a Power Ascension Test Program. By letter dated June 30, 1992, the NRC notified TVA that Unit 2 had demonstrated excellent plant performance over the past year, 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 operate.

Browns Ferry Unit 3 restart is currently scheduled for early 1994. In general, TVA is applying the same corrective action plans and criteria as were employed to effect the Unit 2 restart. Any changes proposed by TVA will be reviewed by the NRC staff. In September 1992, all Unit 1 and 3 recovery activities were consolidated under the site vice president.

The current goal for returning Unit 1 to service is July 1996. TVA does not yet have a formal schedule for implementing its corrective action plans at Unit 1. Although some preliminary "work-scope" review and system "walkdown" activities are in progress at Unit 1, serious efforts to achieve restart of Unit 1 will not begin until after restart of Unit 3.

Watts Bar. TVA had announced that its priorities for startup of its facilities would be in this order—Sequoyah, Browns Ferry, and 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 of January 1994. Unit 2 has a projected fuel loading date of 1999.

Although Unit 1 was virtually complete in 1985, significant corrective programs were required to resolve deficiencies identified through allegations, employee concerns, inspections and audits. The staff has reviewed and approved all but four of 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). All corrective action programs are required to be implemented by TVA before issuance of an operating license.

Virtually all licensing-related activity ceased in 1985, as a result of the problems cited above. The staff re-initiated licensing activity related to Unit 1 in 1990. The activity mainly consists of reviewing amendments to the Final Safety Analysis Report and other support documents, in the context of the Standard Review Plan, applicable bulletins, and certain generic letters. The staff's review findings are documented in periodic supplements to NUREG–0847.

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.). The action was taken on the basis of several factors—a 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 a number of 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 proceed with the completion of the two Bellefonte units as nuclear units.

In a letter dated December 4, 1990, TVA provided the NRC staff with its plans to submit position papers in selected technical areas for NRC review. The technical areas were those in which differences between expectations of TVA and those of the NRC—in technical approach or criteria—could affect significantly the schedule and the scope of work necessary to complete and to license the two Bellefonte units. TVA requested that the NRC staff review TVA's position papers and provide docketed agreements or comments on each of TVA's positions.

TVA indicated that, if it decides to reactivate the Bellefonte nuclear licensing process, it would document the agreed upon positions in future amendments to the final safety analysis report (FSAR) and adhere to the agreements in the Bellefonte licensing process.

On January 17, 1991, TVA met with the NRC staff to describe several position papers that would summarize TVA positions, with which it would seek NRC agreement, regarding the technical approach and criteria that would be proposed in updating its operating license application. Subsequent to that meeting, 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 agreements and providing commentary where agreement could not yet be reached. TVA has incorporated the agreements reached with the NRC staff into the FSAR (Amendment 30 dated December 20, 1991). NRC staff is also defining the inspection activity that will be needed if TVA resumes construction of Bellefonte Nuclear Plant.

During fiscal year 1993, TVA plans to resolve the outstanding Bellefonte issues previously raised by the NRC, provide the TVA Board with information and a recommendation on the "restart decision," organize independent teams for reviewing systems and components at Bellefonte, and conduct pilot studies using independent teams. TVA expects to submit its request for NRC approval of its proposal to reactivate the licensing of Bellefonte during the spring of 1993. (These measures were taken and the request submitted after the close of the report period.)

GE Generic BWR Power Uprate Program

In late 1990, the General Electric Company (GE) submitted a topical report describing a standardized approach for increasing the rated power level of GE boiling water reactors (a procedure called "uprating") by approximately 5 percent, without requiring significant modifications to existing plant systems. Approximately 5 percent of added generating capacity can be obtained by increasing the licensed power level of a reactor from the warranty rating, considered during initial plant licensing, to the actual design rating of the reactor, considered in the design and specification of the reactor vessel, reactor internals, and connected components and systems. The difference between the warranty and design ratings is commonly called "stretch power."

The GE proposal reflects a measure which could increase generating capacity throughout the country. The staff has worked closely with both GE and industry representatives to develop a generic program under which to implement these kinds of power increases in a safe, efficient manner. In September of 1991, the staff formally endorsed the generic program described in the GE topical report.

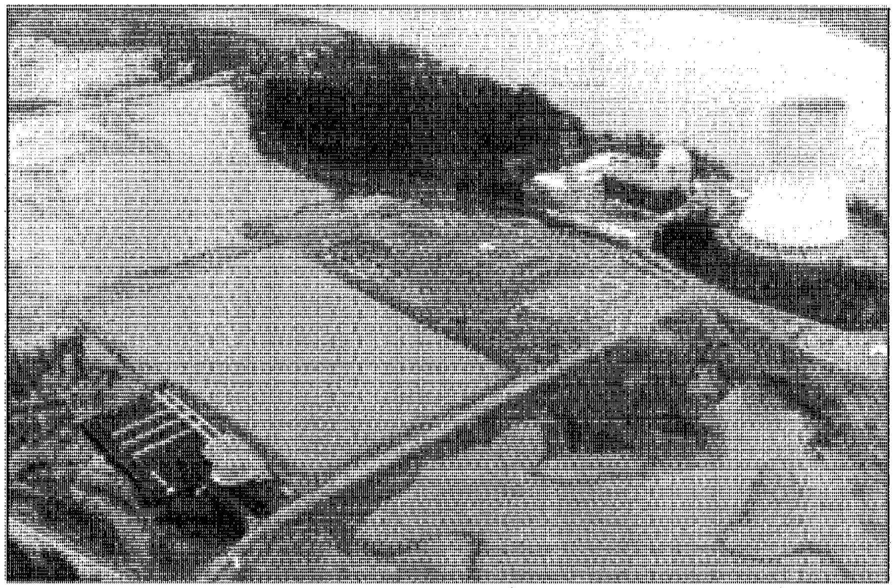
In July of 1992, the NRC staff issued its evaluation of generic analyses performed and submitted by GE regarding selected plant systems and components common to the various boiling water reactor (BWR) product lines. Staff review of these generic analyses leads to a reduction in the size and complexity of subsequent licensee submittals, as well as in the scope of staff review needed for each plant-specific submittal.

On September 9, 1992, the NRC's Director of Nuclear Reactor Regulation, Thomas Murley, signed the first License Amendment sought under the BWR power uprate program, to the license of Fermi Unit 2 (Mich.). Two additional submittals, for the Susquehanna (Pa.) and FitzPatrick (N.Y.) facilities, have been received and were under staff review at the close of the report period. Licensees for another 20 BWR's have expressed interest in requesting license amendments under the generic uprate program.

Trojan Phase-out Planned

The Trojan (Ore.) nuclear power plant is a Westinghouse, four-loop, pressurized water reactor, with a net capacity of 1095 megawatts-electric, located in Columbia County, Oregon. The operator and primary owner of the facility is the Portland General Electric (PGE) Company. Trojan is licensed to operate until the year 2011.

Because of continuing steam generator degradation at the plant, long term operation of Trojan would require that they be replaced. PGE evaluated 14 discrete scenarios for the future, with combinations of various energy sources, and involving (1) continued Trojan operation with steam generator replacement, (2) phaseout operation of Trojan until 1996, and (3) immediate shutdown of the facility. The utility concluded that, while continued operation of Trojan with steam generator replacement did not compare favorably to the alternatives in any scenario, an immediate shutdown would not be a prudent decision until replacement generating capacity was obtained. Thus, PGE decided on phaseout operation of Trojan to 1996.



The Portland General Electric Company has decided to phase out operation of its Trojan nuclear power plant, with shutdown planned for 1996. The 1,095-megawatt facility has experienced continuing steam generator degradation in recent years, and the licensee has concluded that, for reasons described in the text, a phase-out is the best course of action. The plant has been in operation since 1975.

A number of factors were involved in this decision. PGE determined that the return on the investment would not be significant enough if the steam generators were replaced; nor could the company assume that increased plant performance would follow from steam generator replacement. PGE believed that the operation and maintenance costs associated with Trojan operation were too high, and might not be readily brought under control. Also, company management concluded that operating a nuclear power plant carried significant economic, political and regulatory uncertainties which could not be adequately anticipated or controlled. In addition, the Trojan plant had to compete with numerous hydroelectric plants on the Columbia River, which are able to produce relatively inexpensive electricity. PGE has secured long term gas contracts from British Columbia, Canada, and has determined that it can build gas-fired turbine plants to replace the electrical generation from Trojan at lower overall cost. Lastly, the utility has factored conservation efforts into its energy strategy to help restrain future residential and commercial demand for electrical generation.

PLANT LICENSE RENEWAL

By the end of 1991, about 20 percent of the nation's electricity was being generated by nuclear energy (about 110,000 megawatts). The Department of Energy has projected an increase in demand for electricity of another 100,000 megawatts in the next decade. In light of this 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 given plant is not renewed, the licensee will need some 10-to-12 years prior to expiration of that license to plan for and decide on such matters as 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 currently permits the NRC staff to renew operating licenses but does not set forth a process to be followed; thus, the immediate focus of NRC effort has been to define the process for review of licensee renewal applications.

In order to help meet the electrical energy needs of the nation into the early 21st century, some utilities are now carefully examining what steps will be needed to extend the useful life of their nuclear power plants beyond 40 years. The NRC is actively engaged in a number of parallel activities, including rulemaking proceedings, regulatory guidance development, industry technical report reviews, and lead plant reviews (see below).

As of the end of fiscal year 1992, there were 110 licensed nuclear power plants in the United States providing about a fifth of the nation's electrical power. The first of the 40-year operating licenses issued for these plants will expire in the year 2000. The timely renewal of operating licenses, where appropriate and acceptable, would extend overall operating life of a nuclear unit to 60 years; this option represents an important potential contribution to ensuring an adequate energy supply for the nation during the first half of the 21st century.

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—with the exception of age-related degradation unique to the period of extended operation—the regulatory process assures 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.

A keystone requirement in the rule, with respect to management of age-related degradation, is that each renewal applicant must evaluate the extent of such degradation through an integrated plant assessment. The requirements for the plant-specific assessment furnish a guide through a process which demonstrates that age-related degradation of the facility's systems, structures, and components (SSCs) has been identified, evaluated, and accounted for, so as to ensure that the facility's licensing basis will be maintained throughout the term of the renewed license. The required assessment consists of: a screening process to select SSCs, important to license renewal; an evaluation and demonstration of the effectiveness of ongoing licensee actions and plant-specific programs that address aging concerns; and the implementation, as needed, of supplemental actions to prevent or mitigate age-related degradation during the period of extended operation.

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. The proposed amendments and a draft GEIS were published for public comment in September 1991, and a public workshop on them was held in November 1991. The NRC received numerous public comments on the amendments and the GEIS and was revising the documents and preparing responses to the comments at the close of the report period.

Regulatory Guidance Development

To facilitate implementation of 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); these efforts have been proceeding in parallel with (1) the renewal rulemaking, (2) reviews of industry technical reports, and (3) lead plant 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 in early 1993. The staff anticipates completion of the final regulatory guide and SRP-LR their trial employment on a lead plant application is completed.

The NRC also 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 projects that the final regulatory guide and ESRP-LR will be completed about six months after the issuance of the final Part 51 rule and the GEIS.

Industry Technical Report Reviews

The Nuclear Management and Resources Council (NUMARC) has 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 NRC has completed the first round review of all 11 industry reports. The staff provided NUMARC with a number of 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 has revised and resubmitted eight of the reports. At the close of the report period, the NRC staff was reviewing three of the revised reports to determine whether staff questions have been resolved and was preparing draft safety evaluation reports (SERs) documenting the staff's conclusions; the draft SERs will be published for public comment. When the appraisal of the three reports under review is completed, the staff will review the remaining reports and will also prepare safety evaluation reports documenting the staff's conclusions.

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. During the report period, 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 and Wilcox (B&W) Owner's Group announced that it was initiating 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 first of many technical reports were scheduled to be submitted to the NRC by spring of 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 1997.

IMPROVING THE LICENSING PROCESS

Standardization

The Commission strongly endorses regulatory policies that encourage the industry to pursue standardization of power reactor designs. Standard designs are expected to benefit pubic health and safety in a number of ways: concentrating industry resources on common approaches to design problems that have wide application; stimulating adoption of sound construction practices and quality assurance; fostering constantly improving maintenance and operating procedures; and permitting a more efficient and effective licensing and inspection process. In this regard, on April 18, 1989, the Commission issued 10 CFR Part 52 which codified the "Statement of Policy on Nuclear Power Plant Standardization" into a rule. The Commission will revise 10 CFR Part 52 to conform with the language of the Energy Policy Act of 1992. The current focus of the rule is Subpart B, which provides a regulatory framework for certification through rulemaking of standard plant designs, a regulatory process that will bring about early resolution of licensing issues.

The NRC staff continues to work with the Nuclear Management and Resources Council (NUMARC), Electric Power Research Institute (EPRI), and the individual vendors to develop the procedures and practices for implementation of 10 CFR Part 52. A key issue in the effort is the development of inspections, test, analyses, and acceptance criteria (ITAAC) to verify that the facility was built and will operate in accordance with the license and the NRC's rules and regulations. The staff was evaluating applications for design certification and technical requirements for future advanced light water reactors (ALWR), discussed below, at the close of the report period.

Future Reactor Designs

EPRI Advanced Light Water Reactor Program. The Electric Power Research Institute (EPRI) has 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 any 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 requirements document, 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) whose design would comprise various passive safety features and systems. The URD also proposes resolution of certain unresolved safety issues and generic safety issues and delineates ways of complying with 10 CFR Part 52.

The NRC staff issued the Final Safety Evaluation Report (FSER) on Volumes I and II (NUREG-1242) on the EPRI ALWR URD in August 1992. That event marked a major milestone of the staff's review effort. The Draft Safety Evaluation Report (DSER) on Volume III was issued in April 1992 and FSER on that volume is scheduled to be published in 1993.

GE Advanced BWR. General Electric Nuclear Energy, in cooperation with its international technical associates, is developing an advanced boiling water reactor (ABWR). The ABWR will incorporate such innovative features as digital controls, internal recirculation pumps, and control rod drives which feature manifold means of controlling rod motion, as well as special procedures to prevent and mitigate severe accidents. The ABWR is expected to be the first standard design to conform to the EPRI URD for Evolutionary Light Water Reactors. The NRC staff issued a draft final safety evaluation report (SER) in October 1992 and expects to issue the final SER in 1993.

CESSAR-DC, System 80+. During the period 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's draft safety evaluation report (DSER) was issued in September 1992. The staff and CE are working to resolve the issues identified in the DSER before issuance of a final safety evaluation report (FSER). Design certification rulemaking will follow that issuance and the associated final design approval.

Westinghouse AP600. Westinghouse Electric Corporation submitted an application for final design approval and design certification of its "AP600" design, on June 26, 1992. The AP600 is a 600 megawatts-electric pressurized water reactor plant incorporating passive safety systems and features. The NRC staff performed an acceptance review of the AP600 application and reported that, although the application contained a voluminous amount of information, it did not include all the information required by 10 CFR Part 52. While the staff continues to review the AP600 application, it will establish a formal review schedule only after Westinghouse completes the application.

Simplified BWR. GE Nuclear Energy submitted an application for final design approval and design certification of its simplified boiling water reactor (SBWR) design, on August 27, 1992. The SBWR is a 600 megawatts-electric reactor that employs passive features, such as gravity flow and natural convection, to perform essential safety functions. The staff was conducting an acceptance review of the SBWR application material at the close of the report period.

Westinghouse RESAR SP/90. The NRC completed its review of the Westinghouse Electric Corporation's application for preliminary design approval (PDA) of its reference safety analysis report SP/90, a design developed independently of the EPRI's Utility Requirements Document. NRC staff completed its review in April 1991 and issued a PDA, based upon a Safety Evaluation Report (NUREG–1413) setting forth the staff's evaluation of the design. Westinghouse has indicated that it does not intend to pursue certification for this design.

Non-Light-Water Reactors

The staff is conducting pre-application reviews of four non-light-water reactor designs (MHTGR, PRISM, CANDU 3, and PIUS), pursuant to the Commission's Advanced Reactor Policy Statement. The following discussion describes the status of each of these reviews. At the close of the period, the staff was in the process of reappraising pre-application completion schedules, following public meetings conducted with each of the pre-applicants; for that reason, completion schedules are not included in the pre-application reviews.

MHTGR. Modular High Temperature Gas-Cooled Reactor (MHTGR) design information was submitted to the NRC by the Department of Energy (DOE)—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 licensing of advanced designs. The MHTGR concept features a helium-cooled, graphite moderated 350-megawatt (thermal) standard reactor module. One objective of the design is to meet the accident dose limits at the exclusion area boundary, laid down in the Protective Action Guidelines of the Environmental Protection Agency, with a minimal reliance on active systems and without reliance on operator actions. A high reliance is placed on the containment strength and reliability of the individual fuel particles, which are 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 passive reactor shutdown characteristics and a passive decay heat removal system. (The MHTGR design may not require the conventional low-leakage containment building.) A draft pre-application safety evaluation report was issued in March 1989.

PRISM. The Power Reactor Innovative Small Module (PRISM) design concept was also submitted by DOE 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 425 megawatts (thermal) each, with three steam turbine generator sets, producing a total plant electrical output of 1,245 megawatts (electric). Plant design and performance is characterized as 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 are kept small and shutdown and decay heat removal are assured with high reliability.

The NRC issued a draft pre-application safety evaluation report (PSER) in November 1989. In 1990, DOE submitted two new amendments to their preliminary safety information document, in response to open issues identified in the draft PSER. The staff is reviewing the two new amendments.

CANDU 3. 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, in a letter dated May 25, 1989.

The CANDU 3 design is a single-loop pressurized-water reactor, rated at 450 megawatts (electric), with two steam generators and two heat transport pumps connected in series. The design employs natural uranium fuel, heavy-water moderator and reactor coolant, computer-controlled operation, and refueling without shutdown. Major technical issues to be resolved include those involving reactivity feedback and control, reactor shutdown reliability, and on-line refueling.

NRR staff has had a number of meetings with AECL Technologies and with the Atomic Energy Control Board, the Canadian regulatory body, to explore various aspects of the CANDU 3 design with them. The staff has also planned and initiated review activity. AECL Technologies expects to make a design certification decision sometime in 1996.

PIUS. In October 1989, ABB Atom (Sweden) asked that the NRC perform a review of the Preliminary Safety Information Document (PSID) related to the Process Inherent Ultimate Safety (PIUS) reactor design, under provisions of the Advanced Reactor Policy Statement, for the purpose of determining whether the design could be licensed. ABB/Combustion Engineering, the U.S. representative for ABB Atom, is presenting the PIUS design to the NRC for pre-application review.

PIUS is an advanced pressurized-water reactor (PWR) design which exploits certain 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 effecting reactor shutdown. The reactor module is open at the bottom and again at the high point of the "hot leg." At these two openings, density locks are provided to prevent mixing of the coolant and pool water, under normal operating conditions. There is no physical flow barrier in the density locks, but the difference in density between the reactor water and the cooler borated pool water provides a relatively stationery 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.

The review of the PIUS PSID began in June 1991. In August 1991, NRR issued a contract to a Department of Energy (DOE) National Laboratory to analyze the core physics of PIUS. The NRC's Office of Nuclear Regulatory Research is participating in the computer code development and modeling effort.

Early Site Permits

Another element of 10 CFR Part 52, issued by the Commission on April 18, 1989, provides the regulatory framework for obtaining early resolution of site-related issues. The NRC has conducted a series of public meetings with the Industry Siting Group (ISG) on the environmental protection, emergency planning, and site-safety elements of early site permits. These meetings provided a forum to raise technical and regulatory issues identified by the ISG, as part of the early site permit demonstration program. The NRC participated in the Siting Conference for prospective applicants sponsored by the Department of Energy under the demonstration program. The NRC continues its program improvement activities to enhance its capability to accept and review an application for an early site permit.

Standard Review Plan Update And Development Program

In fiscal year 1991, the NRC established the Standard Review Plan Update and Development Program (SRP-UDP) to update the Standard Review Plan (SRP) for use in reviewing future applications for power reactor design certification. In 1975, the staff first published a "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," to give guidance in the conduct of safety reviews of applications to construct or operate nuclear power plants. The SRP continues to be the most definitive means available for specifying NRC's interpretation of an "acceptable level of safety" for light water reactor facilities.

In 1981, the NRC staff revised the entire SRP and published it as NUREG–0800. Since 1981, the NRC has introduced extensive changes in the regulation of the nuclear power reactor industry. But it has not fully revised the SRP to reflect these changes and thus has undertaken the SRP-UDP. When complete, the revised SRP 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 completed procedures for updating the SRP and published them as NUREG–1447. The staff completed the management and information data base for the program and installed the data base on the NRR AUTOS LAN, for the use of NRC's technical divisions. The staff purchased a full text search and retrieval computer system of generic regulatory documents and installed it on the NRR AUTOS LAN, in support of this program. The staff compiled a table of all codes and standards referenced in regulatory documents—including their endorsement status, referenced and latest version. The staff can use this table when it reviews references to new versions of codes and standards in future design applications. The staff compared the code and standard version referenced in regulatory guidance to the current version for the American Concrete Institute standards, and similar collations are under way for the Institute of Electrical and Electronic Engineers codes and standards. The task of determining which issues in regulatory documents could change specific SRP sections is about 80 percent complete. The program's computer data base contains these data for the use of the technical divisions.

The information described above allows staff reviewers to find those changes to the current SRP that will be analyzed and incorporated in the revised SRP. In fiscal year 1993, the staff will continue to consolidate and analyze these data, evaluate codes and standards for other code groups, find information from design certification reviews that also applies to the SRP, and maintain the program to ensure that previously collected data remains current. The reviewer can use these data with the SRP text retrieval system of generic regulatory documents in conducting certification design reviews.

Technical Specifications Improvements

In 1987, the NRC issued a proposed policy statement on technical specification improvements for nuclear power reactors. Under NRC regulations, technical specifications are made part of a utility's actual operating license for a nuclear reactor facility; they specify the safety limits, limiting conditions for operation, surveillance requirements, design features, and administrative controls that are deemed necessary to ensure the safe operation of the facility. The Technical Specifications Improvement Program, established by the policy statement, entailed the improvement of Standard Technical Specifications (STS) and a parallel program of short-term improvements to existing Technical Specifications, the latter for interim application pending completion of the improved STS. The development of improved STS was undertaken to raise the level of safety in nuclear power plants by making technical specifications clearer and easier to use, and more sharply focused on safety concerns.

Improved STS were completed in September 1992 for each of the major vendors of nuclear steam supply systems, namely, 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 new STS encompass 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 licensee-controlled documents; (4) the bases for technical specifications more explicitly posit the relationship between operational requirements and safety analyses; and (5) there is greater consistency in the technical specifications among designs of the NSSS vendor owners groups. The plants within the various owners groups that have volunteered for "lead plant" conversion to the improved STS include Crystal River Unit 3 (Fla.; Babcock & Wilcox owners group), Zion Units 1 and 2 (Ill.; Westinghouse owners group), San Onofre Units 2 and 3 (Cal.; Combustion Engineering owners group), and Hatch Unit 2 (Ga.; General Electric (BWR/4) owners group). All plants of the General Electric (BWR/6) owners group have chosen to convert to the improved STS at the same time.

Adoption of the improved STS will lead to the assignment of certain requirements to licensee-controlled 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 any such relocated requirements. In this regard, the Nuclear Management and Resources Council (NUMARC) coordinated an industry-wide effort to develop guidance for carrying out internal evaluations of plant design changes or changes to procedures, in accordance with 10 CFR 50.59. The guidance was published in June 1989 as NSAC–125, "Guidelines for 10 CFR 50.59 Safety Evaluations," and is followed by many licensees.

The NRC is also continuing work in the parallel program noted above for improving existing Technical Specifications. In this regard, certain STS improvements may be selectively adopted by licensees as "line-item" improvements to existing Technical Specifications, by submitting license amendment requests. Examples of line-item improvements are extending surveillance intervals and outage times (as for the instrumentation used in reactor protection systems and engineered safety features actuation systems) or transferring the contents of technical specifications that address administrative matters—such as component lists, organization charts, or the reactor vessel material specimen withdrawal schedule—to more appropriate, licensee-controlled documents.

The NRC is continuing to evaluate means to improve technical specifications through the application of risk insights. Such insights are currently being applied in the development of technical specification requirements for low-power and shutdown operations. The NRC staff is also continuing to encourage, and to monitor, industry initiatives to apply risk insights to regulatory requirements and develop risk-based technical specifications.

IMPROVING NRC ANALYTICAL CAPABILITY

The nuclear industry has used computer codes for analyzing the performance of engineered structures and systems for many years. These codes allow the structures and systems to be modeled, and their design capabilities to be determined, without subjecting a facility to the actual conditions in question. In the case of reactor accidents, actual tests may be impractical or physically impossible. Models are the only practical means available for examining the response of a facility to accident conditions.

Evolving reactor designs and the limited operational experience of commercial reactors in the early 1970's provided a strong impetus to develop advanced thermal-hydraulic computer codes. Such codes are used in sensitivity studies and independent audit calculations to verify vendor models. As a result of past experience, the NRC staff has developed considerable expertise using the codes, and has also improved its technical understanding of thermal-hydraulic phenomena. Similar technical capabilities are available for structural and mechanical analyses associated with the large number of plants licensed.

Although computer codes are important tools and have contributed significantly to evaluating the safety of nuclear facilities, they are only one aspect of the regulatory process. Computer codes may be used to check the calculations of an applicant or licensee, but their most important function is to help staff reviewers understand the safety significance and performance of the structures, systems and components important to safeguarding the health and safety of the public.

Recently, with the start of licensing reviews of advanced reactor designs, and also the emergence of very powerful computer work-stations, the NRC has determined that significant benefits can accrue from a strong analytical capability within the staff. Several new projects were launched, therefore, in 1992, to re-establish a high level of technical expertise in several offices, and these have become the prototypes of efforts to improve the analytical capabilities of the entire technical staff. With the introduction of a high-performance computer environment throughout the agency, every technical reviewer eventually should have access to a wide range of sophisticated and powerful computer codes, along with the data bases needed to use them.

The prototypic efforts cited included the following:

(1) A small reactor analysis group of about four staff members was formed in NRR's Division of Systems Technology. In fiscal year 1992, this group of analysts began to provide computational support for the review of the advanced light-water reactor designs in the areas of thermal-hydraulics, containment behavior, and associated disciplines. They also provided analytical services on an as-needed basis for operating reactor events and issues. During their first seven months of operation, the new group prepared preliminary models of the AP600 containment, and performed initial calculations of the performance of that reactor during design basis accident conditions. It also used severe accident models prepared by the Office of Research (RES) to analyze the sensitivity of the ABWR containment during severe accidents. Finally, in response to a water-hammer event at the Harris (N.C.) plant, it began calculations of the effectiveness of the emergency core cooling system under degraded operating conditions.

(2) To improve staff understanding of the computer codes that they may be developing, assessing, and maintaining, and to better manage its code development contractors, RES has initiated an in-house analyses capability using the RELAP5/SCDAP and MELCOR codes (see Chapter 8.). Particular attention is being devoted to code assessment issues for the passive advanced light-water reactors. Supporting analyses of operating and advanced passive plants are also being performed for ongoing accident management and probabilistic risk assessment studies.

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

(4) A pilot program in the Division of High-Level Waste Management in NMSS is using high-performance computer work-stations integrated with staff personal computers and special peripheral equipment to support 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 ALWR Designs

The requirements for design certification for reactors utilizing "simplified, inherent, passive, or other innovative means" to accomplish their safety functions include (under 10 CFR 52.47(b)(2)(A)) the ability to demonstrate 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 Westinghouse

Electric Corporation's AP600 and the General Electric Company's Simplified Boiling Water Reactor (SBWR) rely on passive systems for reactor safety. Accordingly, both vendors have developed testing programs to provide data to satisfy the requirements of 10 CFR 52.47. A process for NRC monitoring of the vendors' test programs has been developed (described in SECY-91-273). NRR has implemented the process and is in the process of reviewing these test programs to determine if they are adequate to provide the necessary data. The staff will also examine the experimental data, when they are available, to ensure that the testing has been carried out properly.

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. The one-quarter linear scale loop will be used 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 SIET laboratories in Piacenza, Italy. This loop, called SPES-2, constitutes a 1:395 scale representation of the AP600, and will examine the behavior of the passive safety systems during the high-pressure phase of accidents. Testing in both integral facilities is due to begin in early 1993.

The NRC staff performed a preliminary evaluation of the SE and low-pressure IST programs (reported in SECY-91-273). The deficiencies in the test facilities and the planned experimentation were discussed, resulting in considerable alteration to the programs by Westinghouse. The staff also presented the need for high-pressure, full-height integral testing in SECY-92-030, discussion of which led to development of Westinghouse's SPES-2 test program. The staff will continue to evaluate these programs as more detailed design information and test plans become available.

General Electric has also designed a broad-based testing program to support its SBWR design. Much of the testing planned for SBWR has already been completed (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 passive containment cooling system (PCCS)). IST programs have also been carried out at the Toshiba facility in Japan and at GE's San Jose site, in order to study the behavior of the PCCS facility and the gravity-driven cooling system (GDCS), respectively. Additional testing is planned at Toshiba in a modified test facility that will include a scaled isolation condenser (IC) decay heat removal heat exchanger. Further SE tests are planned at SIET in a new facility called PANTHERS, using full-scale modules of the PCCS and IC heat exchangers, and a new 1:25-scale, full-height integral test loop, PANDA, is under construction at the Paul Sherrer Institute (PSI) in Wuerenlingen, Switzerland. The PANDA tests will look at the long term cooling behavior of the SBWR, concentrating on the performance of the integral behavior of the PCCS. The PANTHERS tests will begin in 1993, and the PANDA loop is expected to be ready in 1994. The staff has maintained a continuing dialogue with GE regarding the SBWR testing program. (GE's testing program is reviewed in SECY 92-339, in which some deficiencies and planned tests are noted. GE is addressing these deficiencies.)

The NRC will also conduct confirmatory research for both the AP600 and SBWR designs. The research will provide valuable confirmatory data to aid in the validation of NRC's analytical codes used to audit the vendors' calculations, and will also provide valuable experimental experience 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.) NRR is involved in 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 a U.S. site to be selected.

Design Bases Reconstitution

The Commission published a policy statement in the *Federal Register* on August 10, 1992 that set forth the Commission's expectations on the availability and adequacy of design bases information. The policy statement emphasized that nuclear facilities should not be modified without a clear understanding of the applicable design bases. Without that knowledge, the possibility exists that a plant modification could adversely affect the safety functions of a system or structure.

The policy statement described four actions the NRC will take:

- The staff will issue a generic letter requesting all licensees to describe the programs that are in place to ensure that design information is correct, accessible and current. Those licensees who are not implementing a design reconstitution program will be requested to provide their rationale for not doing so. If a reconstitution program is under way, the schedule for program implementation and completion will be requested.

- The staff will set priorities for NRC inspections of licensees' management of design and configuration by means of the kinds of techniques employed in safety system functional inspections (SSFIs), based upon responses to the generic letter and other plant specific information known to the NRC. Additional staff guidance will be developed, where needed, for the design bases aspects of these inspections.

- The NRC systematic assessment of licensee performance (SALP) process will be modified so as to explicitly address assessment of licensee programs for the control of design bases information, to reflect NRC inspection activity in this area, and to assure consistent evaluations.

- The staff will continue to encourage self-identification of design bases issues through application of related provisions of the Commission's enforcement policy. The staff will, however, pursue enforcement actions for engineering deficiencies whose root cause lies in the inadequacy or unavailability of design bases information and which are identified during NRC inspections.

INSPECTION PROGRAMS

The Office of Nuclear Reactor Regulation (NRR) is responsible for administering the agency's reactor inspection program, which encompasses all applicant and licensee activity carried out in connection with the constructing and operating of nuclear facilities. Most of the inspection effort is dedicated to operations at the 110 plants for which operating licenses are in effect (as of September 30, 1992), with additional coverage of the eight facilities with construction permits. Responsibility for developing, maintaining, and assessing the effectiveness of the reactor inspection program is shared among NRR staff.

Improvements continued to be made to the operating reactor program throughout fiscal year 1992 on the basis of field experience in implementing the current program. The objectives of the inspection program are (1) to ensure that a minimum level of inspection is conducted at every plant, (2) to integrate headquarters and regional programs, (3) to provide more flexibility for Regional Administrators to allocate resources on the basis of plant performance, and (4) to explicitly allocate resources to respond to safety issues and regulatory concerns. Pursuant to these objectives, 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, as well as to gather information related to Systematic Assessment of Licensee Performance (SALP) Program evaluations (see "Performance Evaluation," below). In the "initiatives" phase of the inspection program, Regional Offices redirected certain of their inspection resources away 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 determine the state of reactor safety, in order to confirm that the operations are in compliance with the provisions of the license, and to ascertain whether other conditions exist which have safety implications serious enough to warrant corrective action. The inspection programs of the NRC are mainly conducted through the five NRC Regional Offices. As described later in the report, a limited number of inspection programs are conducted directly by NRC Headquarters. 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 Offices are under the supervision of the NRC Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Resources.

Besides the routine or planned program of inspections for reactor, fuel cycle facility, and materials licensees, the NRC is committed to dealing aggressively with unsafe or potentially unsafe events or conditions occurring at individual plant sites or other facilities involving licensed operations (by means of "reactive" inspections). In conducting reactive inspections, the NRC seeks to determine the root cause of the event or condition; evaluate the licensee management's response to it, including action to prevent recurrence; and decide whether the problem is one that could occur at other facilities.

Reactor Inspection Program

The operating reactor inspection program is conducted by headquarters and region-based inspectors. Headquarters inspectors conduct, or support the Regional Office in the conduct of, inspections under the Team Inspection Program, discussed below. The Regional Offices conduct most of the required program inspections, and regional inspections are conducted by both region-based and resident inspectors. In general, region-based inspectors are

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 rules, 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 responsibility for safe operation of a nuclear plant lies, as noted, with the licensee. The NRC inspection program is designed, by means of selective examinations, to ensure that the licensee is meeting his responsibility. The NRC inspection program is audit-oriented, that is, it is not an attempt to examine every activity or item involved, but rather 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. What to sample, the sizes of the samples, and the frequencies of the inspection efforts are all judgments 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 and latent vulnerabilities. The inspection process monitors the licensee's activity and gives feedback to licensee's management, so as to allow it to take appropriate corrective actions. However, implementation of the NRC inspection program does not supplant the licensee's programs or attenuate its responsibilities. What the inspection program seeks to provide is a feedback mechanism and an independent verification of 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 NRC 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 the Regional Offices.

Inspections are a vital part of the NRC's review of applications for licenses, and also of the process leading to issuance of construction permits and operating licenses. Inspections continue throughout the operating life of a nuclear facility.

Prior to 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 materials, 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 of its staff. Inspections during this phase seek to determine whether the licensee has developed adequate test plans—both to assure 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 management's overall test procedure, (2) examining selected test procedures for technical adequacy, and (3) witnessing and assessing selected tests to certify results 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 carried out.

About six months before the operating license is issued, a startup phase begins, preparatory to fuel loading and "power ascension." After issuance of the operating license, fuel is loaded into the reactor and an actual startup test program begins. As with 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.

When startup testing is satisfactorily completed, routine operations may begin. Thereafter, the NRC continues its inspection program for the rest of the operating life of the plant.

As previously affirmed, the responsibility for safe operation of the nuclear power plant lies with the licensee, and the NRC's role is to make sure that the licensee is meeting that responsibility. The NRC does this 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 resident inspector is also the key contact when the Regional Office is determining when additional inspection efforts are indicated for a given plant. The activity of the resident inspector is supplemented by the work of engineers and specialists from the Regional Office staff 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 inspection program for operating reactor plants is defined in the NRC Inspection Manual, in terms of its frequency, scope and depth. 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 of emphasis" inspections—special inspections which focus on a specific issue; and discretionary inspections— those which are required to resolve safety issues brought to light by other inspections or as a result of plant operational experience. The program is structured to ensure that, among other considerations, the resources available for inspection are used efficiently and effectively, with particular attention accorded those plants where, based on past performance, improvements in the level of protection and safety-consciousness may be in order.

The inspection program is an essential element in the NRC's regulatory operations. Its results are factored into the NRC's overall evaluation of licensee performance under the Systematic Assessment of Licensee Performance (SALP) program, designed to ensure that nuclear power plants are constructed and operated safely and in compliance with regulatory requirements. 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.

specialists, and resident inspectors are generalists. The resident inspectors provide the major on-site NRC presence for direct observation and verification of licensee activity. Their work comprises 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 generally assess radiation control, security, equipment condition, housekeeping, and the like. The resident inspector also acts as the primary on-site evaluator in the NRC inspection effort regarding licensee event reports, actual events and incidents, and other general inspection of licensee activity. Resident inspectors also serve as the NRC contact with local officials, the press and the public. Region-based inspectors, on the other hand, perform technically detailed inspections in such areas as system modifications, inservice inspection, fire protection, physics testing, radiation protection, security/safeguards, and licensee management systems.

The inspection program allows headquarters and regional inspections to focus on those plant operations which contribute most to ensuring reactor safety and on the identification of safety problems. Program improvements continued to be made in fiscal year 1992, based on knowledge gained from implementation of the current program.

The inspection program comprises the following three elements:

(1) **Core Inspections.** These inspections are 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:

(2a) Generic Area Team Inspections are team inspection efforts addressing a subject area selected by identification of an emerging safety concern, or of an area calling for increased attention because of a history of long-standing or recurring problems. Inspections of this kind are scheduled to be conducted at all sites. The area of emphasis for generic area team inspections for fiscal year 1992 was directed toward electrical distribution systems, and this effort will continue for fiscal year 1993, until scheduled inspections are completed. At this time a new area of safety concern will be selected for these team inspections.

(2b) Safety Issues Inspections are one-time inspection efforts to address a specific safety issue. The inspection effort is instituted 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 effort. During fiscal year 1992, three TIs were issued, affecting such issues as reliable decay heat removal during outage, inservice testing programs and verification of plant records.

(3) **Initiative Inspections.** These are inspections that go beyond those performed under the core and area-of-emphasis inspections. The Regional Administrator identifies those plants where these inspections are required to be performed 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 reactive inspections which generally are unplanned inspections conducted at the discretion of the Regional Administrator in response to various plant events or issues.

Special Team Inspections

During fiscal year 1992, the NRC headquarters and regional staffs performed 48 special team inspections. A special team inspection usually involves a team of 4-10, or more, inspectors with several engineering disciplines and requires 2-to-4 weeks to complete the 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 a plant in a safe condition after shutdown. Depending on the nature of the inspection, the team examines, as appropriate, design, installation, testing, maintenance and operation of the selected systems. The overall objective of these inspections is to determine whether, when called on to do so in an emergency, the plant systems and personnel would perform their safety functions as described in the Safety Analysis Report.

Headquarters staff develop the concept for each new type of team inspection, test it in a limited number of pilot inspections, and when ready, incorporate the inspection methodology into the NRC Inspection Manual. The responsibility to perform most special team inspections is assigned to the Regional Offices. In special circumstances Headquarters may lead a team inspection, such as the Independent Design Inspection performed at Watts Bar in late 1992.

Some types of team inspections are performed on an "as needed" basis at particular plants, while others become an "area of special emphasis" inspection and are performed at all plants. Established types of special team inspections may cover emergency operations, maintenance, ability of systems to perform safety functions as designed, testing of motor operated valves, modification of safety systems during reactor outages, operational safety, readiness to begin initial plant operation or resume plant operation after an extended outage, and independent review of selected plant designs.

The NRC conducted a special team inspection at the Harris (N.C.) nuclear power plant to investigate the facts and circumstances regarding a reported unavailability of the "high head" safety injection system. The event was reported to the NRC in 1991 by the licensee and was flagged as safety significant by the NRC's Accident Sequence Precursor (ASP) program (see Chapter 3). The inspection team determined that the licensee's actions with respect to this event were incomplete. Consequently, the licensee performed further system testing which revealed that system modifications were warranted.

Electrical Distribution System Functional Inspections. A recent type of special team inspection, called and 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 initial inspections, NRC decided in early 1991 to conduct an EDSFI at essentially all plants in the country (with the possibility of an exception where an in-depth inspection had been performed recently in the same program area). This program has been continued during 1992 with the result that, as of the end of fiscal year 1992, the inspection had been completed for the plants at 55 sites. Currently, NRC plans to complete the program by mid-1993 at the other 14 sites included in the program plan.

The NRC has developed a computerized data base of EDSFI findings that allows them to be tracked and trended by plant, component, and technical issue. The data base and associated software is being made available to the industry.

EDSFI inspection results indicated the 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. Because of NRC's attention to electrical distribution systems, licensee's are conducting their own electrical inspections, have devoted more effort to evaluating the design basis for their electrical distribution systems, and are improving the functional capability of these systems.

New Initiatives. Development was begun in 1991 on two new types of team inspections in areas of concern to the NRC. The two new types were Service Water Systems Operational Performance (SWSOP) and Shutdown Risk and Outage Management (SROM). Development continued in 1992 with the performance of four SWSOP pilot inspections and two SROM inspections to test the methodology and scope of each type of inspection. The NRC plans to proceed with the SWSOP inspections at sites with perceived service water problems, at problem plants, and at older facilities. Other SROM pilot inspections are also planned.

Inspection of Emergency Operating Procedures

During fiscal year 1992, the staff continued to perform routine inspections of emergency operating procedures (EOPs). The objectives of the inspections were to assess the usefulness of the EOPs by evaluating their technical accuracy and by taking human factors into account. Results of continuing EOP inspections indicate improvement in the implementation of EOP programs. Inspections show that, in general, EOPs effectively allow for operators to bring the reactor to a safe shutdown condition following an "off-normal" event. Plant operators have also satisfactorily demonstrated their proficiency in dealing with EOPs, during the periodic operator licensing requalification examinations. However, some deficiencies previously identified in "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures" (NUREG–1358, April 1989) continue to be in evidence. Findings to that effect from recent EOP inspections will be addressed in a supplement to NUREG–1358, to be published early in fiscal year 1993. These findings include: 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.

Vendor Inspection Program

The Vendor Inspection Program is centered in NRC Headquarters and is principally a reactive program (see above) structured to respond to vendor and licensee reports of deviations and defects in vendor-supplied parts, components, materials, and services provided to nuclear power plants. The program devises tasks and sets priorities by which to identify and deal with issues, according to their safety significance and generic applicability.

Inspections during fiscal year 1992 addressed reports from industrial organizations and allegations from members of the public concerning defective and sometimes misrepresented parts, components and materials. Licensees and vendors are required to report problems and defects in safety-related equipment, materials and services to the NRC, by provisions of 10 CFR 21, 10 CFR 50.55e, and 10 CFR 50.72 and 50.73, as appropriate. The NRC determined the validity, extent and safety significance of each reported and alleged deficiency and assured that licensees were apprised of potential problems so that appropriate action could be taken to prevent the use of defective components in nuclear plant safety systems.

In fiscal year 1992, the NRC vendor inspection staff conducted 34 vendor and licensee inspections. Several other vendor inspections were carried out by the vendor inspection staff in providing technical support to the NRC Office of Investigations. These inspections covered vendors and distributors who manufacture/supply relays, circuit breakers, batteries, electrical inverters, power supplies, pressure transmitters and switches, printed circuit assemblies, valves, motor operated valve actuators, pipe supports, snubbers, actuators, fuel assemblies and parts, diesel generators, fire barrier material, concrete and grout products, radiation protection technician services, and commercial grade dedication and equipment qualification services. Five inspections of licensees were conducted to review vendor procedures and their implementation for the procurement of commercial grade parts, components and materials for use in safety-related applications. The vendor inspection staff also assisted the NRC Office of Investigations and various U.S. Attorneys in ensuing criminal cases.

The Vendor Inspection Program also included inspection of foreign vendors who supply components for use in U.S. nuclear power plants. In this phase of the program, the NRC inspected actuators and spare parts being manufactured in England by Rotork Controls and pipe supports and hydraulic snubbers being fabricated in Germany by LISEGA GmbH for domestic nuclear plants.

As a result of inspection findings and other information in the vendor program area, the NRC issued 23 information notices and supplements to previously issued notices informing the nuclear industry of problems. The information notices dealt with substandard and refurbished relays represented as new; relay failures; misapplication of relays; distributor modification of relays; an incorrect relay used in emergency diesel generator circuitry; thermally induced accelerated aging and failure of relays; remote trip failures in molded case circuit breakers; misapplication and inadequate testing of molded case circuit breakers; contact adjustment problems with circuit breakers; improper installation of instrumentation modules; electrical connection problems with controllers; defective armature carriers on contactors; non-conservative errors in over-temperature delta-temperature setpoints; failed batteries; cracked transformer insulators; counterfeit valves; motor operated valve data; hydrogen embrittlement of couplings; deficiencies in actuator design modifications; potential substandard flanges; criminal prosecution of a commercial grade valve supplier; and access denied to NRC inspectors.

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 and inaugurate in 1988 and 1989. An interagency data base for the interchange of information on counterfeit/misrepresented parts is in development.

During 1992 the NRC conducted five inspections of licensees' programs for procurement of commercial grade parts and components for use in safety-related applications, including the potential for inadvertent procurement of fraudulent and misrepresented vendor products. The NRC is continuing to work with utility representatives and the Nuclear Management and Resources Council (NUMARC) to address the nature, extent and safety significance of licensee procurement problems, as well as to resolve differences in the interpretation of the requirements of Appendix B for these procurement activities.

PERFORMANCE EVALUATION

The performance evaluation process is focused on licensee performance at nuclear power plants and is intended to improve the NRC's ability to continuously evaluate and track levels of performance in those facilities. The effort involves the integration of information from a variety of the NRC's ongoing activities, such as the Systematic Assessment of Licensee Performance (SALP) program (see below), enforcement actions, performance indicator tracking, trend analysis, event evaluation, operator examinations, and inspection findings. The process germinates in semiannual meetings of NRC senior management for discussion and appraisal of operating plant performance. On these occasions, plants of greatest concern to the agency are identified, and a coordinated course of action for dealing with each is drawn up, including recommendations for special inspections and intensified management attention. The results of each meeting are reported to the Commission, and each licensee involved is informed of NRC senior management's characterization of its overall performance.

Systematic Assessment of Licensee Performance

A principal and regular source of data by which licensee performance is judged is the Systematic Assessment of Licensee Performance, or SALP, program. Under this program, the performance of each licensee with a nuclear power facility under construction or in operation in the

United States is evaluated through the periodic, comprehensive examination of available data—including inspection reports, special reviews, 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 nuclear 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.

One phase of the SALP assessment involves a review of the past year's licensee event reports, inspection reports, 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 essentially comprises performance evaluations in specific functional areas, such as plant operations, maintenance and surveillance, and engineering and technical support.

The SALP program supplements normal regulatory processes and is intended to be sufficiently diagnostic to give meaningful guidance to utility management regarding NRC concerns about quality and safety in plant construction or plant operation.

Human-Systems Interface

During fiscal year 1992, the staff continued its efforts to review 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 (SBWR), the ABB-Combustion Engineering (CE) System 80+, and the Westinghouse AP–600.

Human factors is 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 work-stations with computerized display and control functions, as well as some conventional hardwired controls. Staff guidance for reviewing these new designs has been and continues to be under development. The staff has prepared a model and acceptance criteria for use in reviewing the advanced control room design implementation processes being proposed by the advanced reactor applicants. And the staff continues its work to develop new guidelines on human-system interface design that will incorporate information gathered from other industries that use advanced control and display technology. In the summer of 1992, an international workshop, sponsored by the Office of Research, was conducted to evaluate draft guidance for reviewing advanced control room designs. The guidance is expected to be published in early 1993.

During fiscal year 1992, the staff prepared draft and final safety evaluations for the human-systems interface portion of the GE ABWR advanced reactor design, a draft safety evaluation for the human-system interface portion of the CE System 80+ design, a draft safety evaluation for the human factors portion of the EPRI Passive ALWR Requirements Document; the staff also developed preliminary questions on the Westinghouse AP–600 and GE SBWR human-systems interfaces, and has continued exchanges with foreign utilities, research, and regulatory organizations to examine their efforts to design and evaluate advanced control room designs.

The staff has also continued to conduct follow-up investigations of selected human performance-related events. These investigations focus on identifying and evaluating the contributions to plant safety made by human performance and on analyzing the conditions which cause human error. The investigations consider a various kinds of human-system interfaces—including the design of control and monitoring stations, procedures, communications and training. In this regard, the staff has been engaged in collecting data on operator performance during operator licensing requalification examinations; the data collected will be analyzed, beginning in fiscal year 1993.

Training

During fiscal year 1992, 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 staffs. As part of the evaluation, NRC staff personnel attended some INPO accreditation team visits as observers, when utilities' training programs are under review by INPO. NRC management personnel are also present as observers during utility presentations to the National Nuclear Accrediting Board. The staff also continues to conduct training inspections when conditions at particular licensee sites warrant staff evaluation. During the report period, the staff conducted training inspections at eight sites.

The staff has concluded that the industry continues to make progress in bringing about improvements in training and in implementing the Commission's training policy statement. Although training improvements have been observed, training deficiencies continue to be found and corrective action required. The Commission continues to endorse the industry accreditation program as an effective means of ensuring proper nuclear power plant personnel training.

During fiscal year 1992, the staff developed 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. On January 7, 1992, the proposed training rule was published for public comment in the *Federal Register*. Based on comments from the Commission and the public, the staff is revising the proposed rule and supporting documents. The Commission expects to issue the final training rule in early 1993. It is expected that the rule will have no more than minimal impact on current industry training initiatives.

QUALITY ASSURANCE

The NRC continued to emphasize performance-based quality assurance (QA) verification during fiscal year 1992, carrying out QA reviews of plant safety and reliability throughout the industry. In this regard, the area of Software Quality Assurance (SQA) has emerged as one of substantial NRC interest and involvement. As digital systems replace analog systems in operating nuclear power plants—and assume an important part in the design of advanced plants—the acceptability of digital systems, and of SQA practices with respect to the software associated with these systems, must be assessed in terms of their effect on public health and safety.

To augment NRC experience with SQA practices, the NRC staff met with members of the Nuclear Management and Resources Council (NUMARC) and Nuclear Utility Software Management Group (NUSMG) to discuss the results of a licensee survey conducted by NUSMG on SQA practices in the nuclear industry. The NRC is actively involved with industry in efforts to document a methodology for reviewing, approving and inspecting future applications which involve software and its 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 1992, the NRC staff reviewed the Quality Assurance portions of Standard Safety Analysis Reports (SSAR) for the GE-ABWR, CE System 80+, and Westinghouse AP600 advanced reactor applications. In addition, review of topical reports for quality assurance programs for future applicants, vendors and foreign companies were also conducted.

In response to several requests from the Regional Offices, the staff has drafted generic operating plant procedure review schedules. This guidance is intended to assist the Regional Offices in their review of future revisions to NRC approved Quality Assurance Programs.

Maintenance

Proper maintenance is essential to nuclear power plant safety, and the results of plant maintenance activities must be monitored and evaluated to assure that they remain effective, particularly as plants continue to age.

On July 10, 1991, the Commission published, in the *Federal Register* (56 FR 31306), a new maintenance rule, 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The rule requires 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 or other events caused by the lack of effective maintenance. Concomitant with issuance of the Maintenance Rule, the Commission directed that regulatory implementing guidance be developed and issued within two years of the date of the rule, and that, for the period prior to the effective implementation date of the rule, NRC maintenance inspection procedures be revised to be consistent with the results-based philosophy of the rule.

During fiscal year 1992, NRC's maintenance efforts were primarily dedicated to supporting the above two initiatives. The NRC regulatory guidance effort was headed by a steering group of senior managers from the Offices of the Executive Director for Operations (EDO), NRR, and RES. The NRC steering group also agreed to consider endorsement of the guidance for maintenance rule implementation to be developed by the industry through the Nuclear Management and Resources Council (NUMARC). The expectation of the NRC was to be able to endorse NUMARC guidelines in a regulatory guide without exceptions. After initial agreement with NUMARC to develop guidance documents in parallel, the NRC steering group conducted numerous public meetings with NUMARC to discuss the guidance. In the June-July 1992 time period, the NRC steering group determined that the NUMARC guidelines would be acceptable, provided additional changes were made to resolve certain significant issues; the steering group directed that NRC staff from NRR, RES, and Regions III and IV, would serve as a working group to meet with a NUMARC working group to resolve remaining significant issues. Subsequently, the NRC determined that the industry guidelines were sufficiently acceptable to proceed with a verification and validation (V&V) program. The NUMARC V&V program involved trial implementation of the industry guidelines at nine reactor plants. It was scheduled to be completed by December 1992, and the NRC regulatory guide was scheduled to be completed by June 1993.

Concurrent with the development of the maintenance regulatory guide, the staff began work on proposed revisions to inspection procedures for the inspection of licensee maintenance activities in the period prior to the effective date of the maintenance rule (July 1996). The proposed revisions to inspection procedures adopt a "results-oriented" inspection approach, while remaining within the scope of existing regulations. They also emphasize inspection of those areas generally noted as weaknesses during the conduct of maintenance team inspections (1988–1991), including the lack of trending, root-cause analysis, and adequate engineering support.

Because of the relationship of these interim maintenance procedures to a fundamental transition from what was a process-oriented focus to a results-based maintenance rule, the Commission directed the staff to obtain public comment on this inspection guidance. In August 1992, the staff held a public workshop to solicit comments on the proposed revised inspection procedure. NRC staff from Headquarters and the Regions conducted the workshop, which was attended by about 120 representatives of nuclear utilities, vendors and contractors. Comments from the workshop, as well as written comments, were considered for incorporation into the procedure. Subsequently, the interim maintenance inspection procedure was submitted to the Commission for their review.

OPERATOR LICENSING

The NRC is continuing 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 maintains the responsibility for managing the program and administering the examinations at non-power facilities.

During fiscal year 1992, the NRC issued initial licenses for 328 ROs and 340 SROs, and administered Generic Fundamentals Examinations (GFEs) to 403 prospective ROs and SROs at power reactor facilities. The GFE tests prospective licensed operators on their understanding of theoretical knowledge required for operating a nuclear power plant; the GFE must be passed before the applicant can take the site-specific written examination. The NRC also issued initial licenses for 29 ROs and 28 SROs at non-power reactor facilities.

The NRC administered 805 requalification examinations to ROs and SROs at 53 power reactor facilities and 23 examinations at nine non-power reactor facilities. The NRC requalification examination process ensures the continued competency of individual licensed operators and also evaluates the quality of the facility licensees' requalification program.

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, 1992, the NRC received 19 reports of licensed individuals exceeding their facility licensee's cutoff levels for drugs or alcohol. The NRC has issued Notices of Violation to 11 of the individuals and letters of reprimand to two others. The NRC was evaluating the appropriate actions for the six remaining cases, at the close of the report period.

The NRC received 10 plant-referenced simulator certifications during fiscal year 1992. Eight of these were submitted by facilities in accordance with approved schedule exemptions, and two others were submitted by a two-unit facility that had previously certified another simulator. The NRC approved one facility's request to use a simulation facility other than a plant-referenced simulator and extended one facility's schedule exemption until December 31, 1992. As of September 1992, only that one facility licensee had not certified a plant-referenced simulator or obtained NRC approval of a simulation facility.

The NRC is continuing 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 effected during fiscal year 1992 or are in progress:

(1) The NRC completed an evaluation of a revised requalification examination dynamic simulator test grading method that focuses on crew, rather than individual, performance. The NRC is now developing an amendment to 10 CFR Part 55 that will delete the

requirement for each licensed operator to pass a comprehensive written examination and an operating test conducted by the NRC during the term of the operator's six-year license. The new dynamic simulator test grading procedure and the proposed rule change are described below.

(2) The NRC is now administering the Generic Fundamentals Examinations at the individual facilities, rather than in the NRC's Regional Offices. The change in venue has significantly reduced the regulatory impact of that program on the participating facilities.

Operator Licensing Requalification Changes

During fiscal year 1991, the staff proposed to the Commission that the NRC evaluate the feasibility of substituting crew grading criteria for the individual grading criteria used in the NRC's dynamic simulator tests. The staff believed that such a change would encourage better teamwork, communications, command and control among the control room operators and thereby provide a more accurate measure of the operators' abilities. With the Commission's approval, the staff developed and implemented a pilot simulator test procedure to evaluate the process.

The staff tested the pilot procedure with more than 100 licensed operators at six facilities. The staff completed its evaluation of the revised dynamic simulator testing procedure early in fiscal year 1992 and concluded that the revised method was a significant improvement over the method contained in the current revision of the Operator Licensing Examiner Standards (NUREG–1021). The revised simulator testing method improved operator teamwork, overall performance, and operational safety, because it provided an incentive for the operators to independently verify critical activities and take compensatory measures, when necessary. The pilot examination method also gave the facility licensees greater responsibility for ensuring the competence of each individual operator and helped reduce undue examination stress.

The staff briefed the Commission on the results of the pilot examinations in June 1992 and recommended that it approve the implementation of the dynamic simulator crew grading method at each facility. The Commission approved the staff's recommendation and authorized the staff to begin conducting the dynamic simulator requalification examinations using the crew grading method, as soon as the proposed procedure was published for industry comment. The staff will formally promulgate the crew-based dynamic simulator grading method in the next revision of NUREG–1021.

The NRC is also developing a proposed amendment to 10 CFR Part 55 that will delete the requirement for each licensed operator to 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 amendment will also require facility licensees to submit to the NRC copies of the annual operating test and the comprehensive written examination it uses for licensed operator requalification. Finally, the proposed amendment will revise the scope of the regulation to include facility licensees.

The NRC believes that the proposed amendment will decrease the cost and the regulatory impact of the NRC's requalification oversight program, while increasing operational safety. Eliminating the requirement for the NRC to examine each operator every six years will enable NRR to apportion resources for this regulatory task to actively oversee the requalification program at a given facility based upon the program performance there, rather than on the number of individuals licensed to operate the facility. The proposed amendment will allow the NRC to conduct requalification examinations at the facility in accordance with existing NRC procedures or to inspect the facility's requalification program in accordance with a newly proposed procedure.

Under the proposed requalification oversight program, the NRC will periodically conduct selected portions of the requalification examinations at each facility and conduct a requalification program inspection during each year in which it does not conduct examinations. The NRC will retain the authority to conduct examinations "for cause" at any facility.

EMERGENCY PREPAREDNESS

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

An event occurring in fiscal year 1992 with significant emergency planning implications was Hurricane Andrew, which caused the shutdown of the two nuclear reactors at the Turkey Point (Fla.) power facility, Units 3 and 4, on August 24, 1992. Following substantial effort on the part of the licensee, Florida Power & Light Company, to repair storm-related damage, Unit 4 was ready for restart by late September. On October 1, 1992, the NRC requested that the utility suspend restart activities pending further consideration by FEMA of the status of off-site emergency planning in an area of about a 10-mile radius, the Emergency Planning Zone, (EPZ) surrounding the plant. On October 23, 1992, FEMA reaffirmed that, on the basis of its assessment of restorative measures taken, there was reasonable assurance that the public health and safety would be protected in the event of a radiological emergency at the Turkey Point plant. The FEMA reassessment was the product of extensive coordination among FEMA, the NRC, the State of Florida, Dade and Monroe County emergency management officials, and the licensee. Analyses of the event disclosed a number of lessons of potential benefit to the NRC staff in any future disaster-related shutdown of an operating nuclear power plant.

The staff continued to address issues related to the Seabrook (N.H.) nuclear power plant, which received a full-power operating license in fiscal year 1990, following a lengthy proceeding that involved numerous emergency preparedness issues. In fiscal year 1991, the Commonwealth of Massachusetts re-entered into emergency preparedness planning for Seabrook, occasioning a transition from what was a utility-based off-site response organization to a State off-site response. The NRC staff, with the assistance of FEMA, is reviewing the licensee's transition plans, including the proposal to convert the Vehicular Alert and Notification System in the Massachusetts portion of the 10-mile Emergency Planning Zone to a fixed pole mounted system. The Massachusetts Radiological Emergency Response Plan (MARERP) has been submitted to FEMA for review and the implementation of the MARERP was demonstrated as part of the full participation exercise conducted at Seabrook in June 1992.

The NRC staff also reviewed a methodology proposed by the Nuclear Utilities Management and Resources Council (NUMARC) for categorizing events based on plant conditions (emergency action levels) at power reactors. NRC, NUMARC and industry personnel participated in discussions and workshops which included "walk-throughs" of scenarios designed to test the methodology and a pilot test at a licensed BWR. The NRC endorsed the NUMARC alternative emergency action level (EAL) methodology, in Revision 2 to Reg Guide 1.101, as an acceptable means of meeting NRC requirements. The staff participated in a NUMARC-sponsored national workshop, in St. Louis, on implementation of the alternative EAL, in fiscal year 1992. Many licensees are planning to revise their EAL schemes in fiscal year 1993 to bring them into conformance with the new method for classifying emergencies developed by NUMARC and the NRC staff. During fiscal year 1992, the staff identified a need to develop guidance for the development of shutdown EALs, and documented its proposed plan of action in NUREG–1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the U.S."

Additional emergency preparedness licensing actions in fiscal year 1992 included review of advanced reactor submittals, and of emergency preparedness issues related to the decommissioning of the Yankee-Rowe (Mass.) nuclear power plant. The staff also addressed the emergency planning aspects of actual events occurring at operating plants during the year, including an analysis of the loss-of-annunciator events which led to the declaration of Alerts at the Quad Cities (Ill.), Nine Mile Point (N.Y.), and Palo Verde (Ariz.) facilities.

The staff worked closely with FEMA in fiscal year 1992 in addressing the following tasks: (1) the development of emergency planning guidance for early site permit applicants, under 10 CFR 52; (2) the development of a standard for portal monitors used at reception centers; (3) the identification of areas where there is a need to issue future guidance; and (4) the review and response to petitions concerning off-site emergency preparedness.

SAFETY REVIEWS

Applications of Probabilistic Risk Assessment

In fiscal year 1992, 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. 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 plant performance.

With respect to advanced reactor design assessments, the NRC staff is in the final stage of reviewing the PRA for General Electric's ABWR design, and the EPRI Key Assumptions and Ground Rules for Advanced Designs PRAs. The staff is also maintaining significant progress in the review of the Combustion Engineering System 80+ Advanced Design PRA, and has started preliminary PRA reviews for the Westinghouse AP600 and the General

Electric SBWR advanced passive design. Reported estimates of the core damage frequency for these last two designs are significantly lower than those of conventional operating PWRs and BWRs.

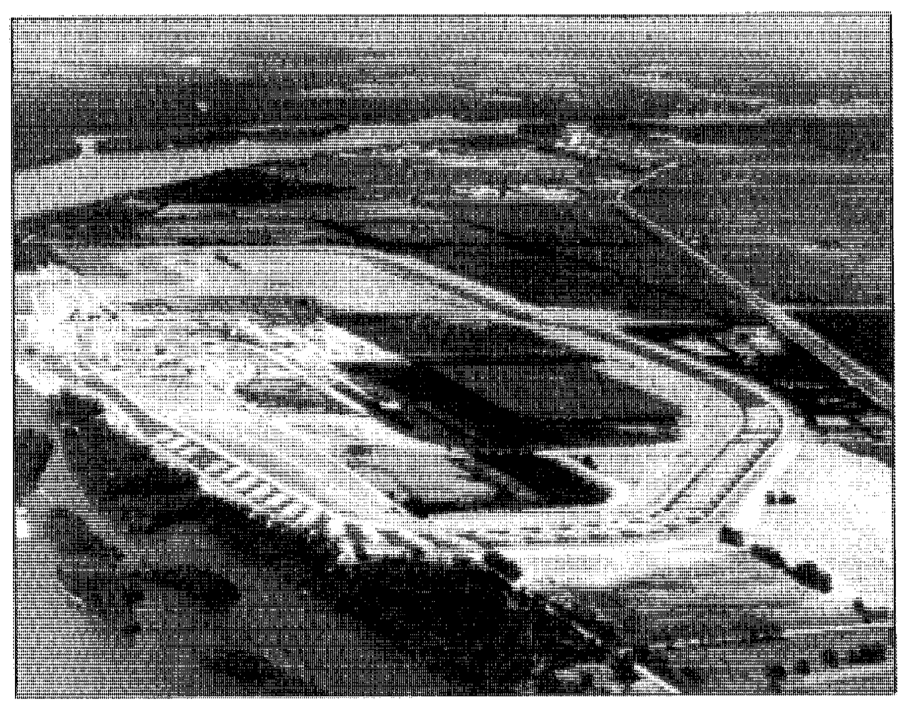
Individual Plant Examinations (IPEs) continued during the report period, with approximately one-half of the IPEs, required for all plants in operation or under construction, now completed. (See discussion of the IPE program later in the section.) The staff completed reviews of five submittals and started review of six more. These 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 submittals are expected within the next three years.

The application of PRA results and insights to inspection and licensing activities continues to prove its worth. PRA-based information contributed to the planning and the conduct of two setpoint inspections, one service water system inspection, one Integrated Performance Appraisal Team Inspection, one ATWS team inspection, one Risk-based Operational Safety and Performance Assessment Team Inspection, and 22 Electrical Distribution System Functional Inspections (EDSFIs). Risk insights into "4-KV bus momentary interrupting current" was provided for Regions on the basis of EDSFI findings. Eleven risk-based inspection guides (RIGs) were also completed for plants and issued as NUREGs. These documents address auxiliary feedwater systems and high-pressure coolant injection systems. Before publishing the RIGs, draft copies were released to utilities and "as-built" conditions were verified during a plant walkdown. This walkdown verification and the utilities' comments were incorporated into the final NUREGs. Risk-based inspection procedure IP 93804 was also revised on the basis of lessons learned from various RIGs.

Interfacing Systems Loss-of-Coolant Accident Program

In 1989, NRC initiated a study of the potential for off-normal events associated with low-pressure systems that interface with high-pressure systems. The specific concern was that conditions could exist for an "unisolable interfacing systems loss-of-coolant accident" (ISLOCA) which bypasses the containment. Inadvertent exposure of a low-pressure system to pressures beyond its design capacity could lead to breaks outside the containment. That possibility implies, in the event of a severe accident, a potentially significant release of radioactivity directly to the environment.

The new study investigated the factors that contribute to the likelihood and severity of an ISLOCA. It involved detailed evaluations of operating events, team inspections of selected assessments, thermo-hydraulics, human factors, and environmental qualification of equipment. The inspections disclosed a number of areas affecting ISLOCA risks and appropriate measures for reducing them. The former include various inadequacies observed regarding maintenance, surveillance, and testing, as well as human factors involving man- machine interfaces, procedures, and training.



Analyses of emergency planning implications related to actual events was undertaken during the fiscal year, including such events as the declaration of Alerts at several nuclear power plants, including the Quad Cities plant, shown here. The plant on the Mississippi River serves the area it is named for, which comprises the cities of Davenport and Bettendorf in Iowa, and Moline and Rock Island in Illinois.

In 1992, findings of the program were summarized and issued as Information Notice 92-36. Specifically, the information notice describes several staff observations regarding ISLOCA risk at nuclear power plants. One was that the estimated core damage frequency for ISLOCAs could be greater than previous PRA had estimated forsome plants. Other findings include the observations that plants generally do not have contingency backup water-supplies for long term cooling after an ISLOCA, that a relatively high importance should be assigned to human errors, and that certain ISLOCA precursors can have a detrimental effect on plant operations. The ISLOCA research program is continuing under Generic Issue 105, "Intersystem Loss of Coolant Accidents in Light Water Reactors." Upon completion of this research, the staff may issue further generic correspondence to the licensees.

Performance of Motor-Operated Valves

The NRC staff is actively sustaining efforts to improve the performance of motor-operated valves (MOVs) in nuclear power plants. Despite these ongoing efforts, MOV problems continue to occur or to be identified. These 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.

In "Action Plans for Motor-Operated Valves and Check Valves" (NUREG-1352, June 1990), the NRC staff describes its plans to improve MOV performance and also industry activities related to MOVs. A significant aspect of the MOV action plan is the staff's review of the implementation of Generic Letter 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and Surveillance," and its supplements, by nuclear power plant licensees. In Generic Letter 89-10, the staff requested that licensees help ensure 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 tracking 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.

The staff issued Supplement 1 to Generic Letter 89-10 on June 13, 1990, providing detailed information on the results of public workshops held to discuss the generic letter. On August 3, 1990, the staff issued Supplement 2 to Generic Letter 89-10, allowing licensees additional 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 to Generic Letter 89-10 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 (1) containment isolation in the steam supply line of the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems, and in the supply line of the Reactor Water Cleanup system; and (2) in other systems directly connected to the reactor vessel. In responding to Supplement 3 to Generic Letter 89-10, BWR licensees have reported that approximately half of the MOVs within the scope of Supplement 3 have been, or will be, modified or adjusted to provide additional assurance of their capability to close under design basis conditions. On February 12, 1992, the staff issued Supplement 4 to Generic Letter 89-10 removing 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.

As an integral part of their Generic Letter 89-10 programs, most licensees are using MOV diagnostic equipment to obtain information on the thrust required to open or close the valve and on the thrust delivered by the motor actuator. On March 2, 1992, the NRC staff held a public meeting with representatives of ITI-MOVATS to discuss the accuracy of the Thrust Measuring Device (TMD) manufactured by ITI-MOVATS for estimating the stem thrust based on spring pack displacement. At that meeting, ITI-MOVATS informed the staff of the increased inaccuracy of the TMD, this resulting from a calibration of the equipment in the valve *opening* direction, while relying on the diagnostic equipment to predict the thrust in the valve *closing* direction. On July 8, 1992, the staff published for public comment proposed Supplement 5 to Generic Letter 89-10, which would request that licensees address the increased inaccuracy of MOV diagnostic equipment resulting from valve stem directional effects. The staff was reviewing the public comments received on the proposed supplement to Generic Letter 89-10, at the close of the report period.

Temporary Instruction 2515/109 (January 14, 1991) was issued to provide guidance for the conduct of regional inspections of programs being developed by nuclear power plant licensees in response to Generic Letter 89-10. The inspections under the temporary instruction are being carried out in two phases: an initial review of the Generic Letter 89-10 program, and a subsequent evaluation of program implementation at each nuclear power plant. The staff has already conducted inspections to review the Generic Letter 89-10 programs at most nuclear power plants. A summary of the results of those inspections was provided in NRC Information Notice 92-17 (February 26, 1992), "NRC Inspections of Programs Being Developed at Nuclear Power Plants in Response to Generic Letter

89–10." The temporary instruction is being revised to reflect the lessons learned from the initial inspections of the Generic Letter 89–10 programs. The staff will begin the inspections of program implementation in 1993.

The staff is monitoring the industry's efforts toward resolving concerns about the performance of MOVs at nuclear power plants. In 1992, it became clear from nuclear plant operating events, from Generic Letter 89–10 program development and implementation, from industry research, and from NRC inspections that nuclear power plant licensees will need to continue to apply substantial resources to improving MOV performance. While the industry is resolving concerns about MOV performance, the NRC staff will remain ready to institute regulatory action wherever necessary to assure that public health and safety are protected.

Evaluation of Shutdown and Low-Power Risk Issues

As discussed in the *1991 NRC Annual Report,* the 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.

On February 25, 1992, the staff issued SECY–92–067 regarding the shutdown risk program and 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 from the evaluation of shutdown and low-power operations. The most significant technical findings are the following:

(1) Outage planning is crucial to safety during shutdown conditions since it establishes (a) if and when a licensee will enter circumstances likely to challenge safety functions, and (b) the kinds of mitigation equipment available.

(2) Some current NRC requirements in the area of fire protection (e.g., 10 CFR 50, Appendix R) do not apply to shutdown conditions, but significant maintenance activities do occur during shutdown that can increase the potential for fire.

(3) Well-trained and well-equipped plant operators can play a very significant role in accident mitigation for shutdown events.

(4) All probabilistic risk assessments for shutdown conditions in pressurized water reactors (PWRs) have found that accident sequences involving loss of decay heat removal (DHR) during operation with a reduced inventory (e.g., midloop operation) are dominant contributors to the core-damage frequency.

(5) Extended loss of DHR capability in PWRs can lead to a loss-of-coolant accident (LOCA) caused by failure of temporary pressure boundaries in the reactor coolant system (RCS) or rupture of DHR system piping. In either case, the containment may be open and emergency core cooling system (ECCS) recirculation capability may not be available.

(6) Passive methods of decay heat removal can be very effective in delaying or preventing a severe accident in a PWR; however, procedures and training for such methods have been lacking.

(7) All PWR containments and boiling water reactor (BWR) Mark III primary containments are capable of offering significant protection under severe core damage conditions if the containment is closed or can be closed quickly. However, analyses show that the steam and radiation environment in the containment, which can result from an extended loss of DHR or LOCA, would make it difficult to close the containment in many cases. BWR Mark I and II secondary containments offer little protection against a severe accident, but this is offset by a significantly lower likelihood of core damage in BWRs than in PWRs.

The comment period on NUREG–1449 ended on April 30, 1992, and a large number of comments were received from utilities and industry organizations. Subsequent to the end of the comment period, the staff held five public meetings to discuss the comments. These meetings included representatives of each of the nuclear steam supply system (NSSS) owners groups, representatives of individual utilities, representatives of the Nuclear Management and Resources Council (NUMARC), and members of the public. The staff has considered the public comments and is currently modifying NUREG–1449, as appropriate, prior to issuing it as a final report.

The fundamental conclusion of the evaluation of reactor shutdown issues is that public health and safety has been adequately protected while plants were in shutdown conditions, but that numerous and significant events have occurred which indicate that substantial safety improvements are possible and appear warranted.

The NRC staff is currently considering improvements in the following areas to enhance shutdown and low power operation:

(1) Improvements in outage planning and control.

(2) Improvements in fire protection.

(3) Improvements in operations, training, procedures, and other contingency plans.

(4) Improvements in technical specifications.

(5) Improvements in instrumentation.

Pressurized Thermal Shock and Reactor Vessel Materials

Reactor pressure vessel integrity is essential in assuring reactor safety. During operation, a reactor vessel is bombarded by 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. (Charpy energy refers to a material's ability to resist breaking under impact, at various termperatures.) 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 demonstrate, by performing an "equivalent margins" analysis, that lower values maintain margins against failure equivalent to those required by Appendix G of the ASME Code.

The issue of pressurized thermal shock (PTS) arises because, in pressurized water reactors (PWRs), transients and postulated accidents can result in overcooling (thermal shock) of the reactor vessel, concurrent with or followed by significant pressure. In such conditions, rapid cooling of the reactor vessel internal surface results in thermal stress, with a maximum tensile stress at the inside of the vessel. The effects of this thermal stress are compounded by pressure stresses if the vessel is pressurized. These types of stresses could cause propagation of postulated pre-existing defects, if they are located in areas where the material fracture resistance has decreased because of irradiation.

In July 1985, the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events." This rule established screening criteria to evaluate whether a reactor vessel has adequate fracture toughness to withstand 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 RTPTS value, calculated using the methodology in the rule. The RTPTS value is a measure of the fracture resistance of the material. As the RTPTS value increases, the fracture resistance decreases.

Analyses performed by the NRC staff indicate that the risk from PTS events for reactor vessels with RTPTS 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 operation 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 RTPTS in excess of the screening criteria.

On May 15, 1991, the PTS rule was amended to revise the method of calculating the RTPTS value and to require licensees to consider the effect of surveillance test results and vessel operating temperature on the RTPTS value. The methodology for calculating the RTPTS value was revised based on additional data that improve the characterization of the empirical relationships between copper, nickel and neutron fluence, which were derived from statistical analysis of an updated surveillance data base.

Yankee-Rowe Nuclear Power Plant

On February 27, 1992, Yankee Atomic Electric Company (licensee, YAEC) notified the Commission that it would permanently cease power operation of its Yankee-Rowe (Mass.) facility and would begin developing plans to decommission the facility, in accordance with 10 CFR Part 50. The plant had been voluntarily shut down by YAEC on October 1, 1991, after the NRC staff issued a memorandum to the Commission recommending that Yankee-Rowe be shut down until actual data from the test and surveillance program show the reactor pressure vessel has adequate margin against vessel failure from a pressurized thermal shock (PTS) event. The staff's recommendation was based on a revised PTS analyses performed by the licensee. The revised analysis used more realistic models and showed less conservative results. Ultimately, the licensee's decision to decommission the plant was based on a combination of factors, most importantly the economic outlook and the degree of uncertainty associated with resolution of the reactor vessel issues. The licensee has subsequently applied for a Possession Only License.

Station Blackout Rule

The term "station blackout" means the loss of off-site alternating current (a.c.) power to safety-related and non-safety-related electrical buses concurrent with turbine trip and the unavailability of the emergency diesel generators. The "Reactor Safety Study" (WASH-1400) showed that, for some plants, a station blackout event could be an important contributor to the total risk from nuclear power plant accidents. To deal with the issue, the NRC amended its regulations by adding a new requirement (10 CFR 50.63) that all nuclear power plants be capable of coping with station blackout for a specified duration of time, as determined by the design characteristics and site-specific considerations of each plant. (The bases for and the development of the "station blackout" rule (10 CFR 50.63) are set forth in the *1988 NRC Annual Report*, p. 30.)

The NRC staff performed initial reviews of licensee responses, and conducted a limited number of site audits of the documentation supporting the responses. Based on these audits, the NRC staff, in conjunction with the Nuclear Management and Resources Council (NUMARC), developed augmented guidance that was provided to the industry in January 1990.

As of September 30, 1992, the staff had conducted safety reviews of 69 plant sites (108 units) and was continuing reviews for one remaining operating plant and two plants nearing readiness to apply for operating licenses. At many of the sites, those for which only procedural changes or minor modifications are required, these changes or modifications have already been implemented. At sites which require major modifications, such as the addition of diesel generators, licensees are nearing completion of the modifications or are moving toward completion. (The types of modifications that are being made to comply with the station blackout rule are set forth in the *1991 NRC Annual Report*, p. 35.)

To close the issue, the staff is developing a plan to conduct inspections to verify licensees' implementation of the rule.

Steam Generator Replacement at Millstone

The Millstone Unit 2 (Conn.) nuclear power plant was shut down on May 30, 1992, for a refueling and the replacement of the steam generators. The replacement project was projected to take approximately seven months, with startup to occur at the end of December 1992. The old steam generators were cut at the mid-section of the cone area and removed through the containment access opening. The upper steam drum sections were completely refurbished in the containment.

The old steam generators were prepared for shipment to a burial site in South Carolina. The preparation included removing all liquids from within the steam generators and the tubes, sealing all openings, filling the steam generators with a light-weight concrete, and barge shipping the steam generators on a special multi-wheel transporter to the South Carolina burial site. A certificate of compliance for transport of radioactive material was issued.

The NRC staff oversight and inspection of the Steam Generator Replacement Project has covered all aspects of the program. Numerous meetings have been held at the NRC Headquarters and the Region to assess the licensee's progress in planning and evaluating each phase of the replacement project. Inspections have been held both at the offices of the utility's contractor, and at the plant site. Inspections continued through the completion of the replacement.

A notable aspect of the project was the decision to employ a somewhat unusual welding technique called narrow-gap welding. The technique involves the use of a conventional welding process, in this case automatic gas-tungsten arc welding, along with a special joint configuration. This method was selected for making the butt welds in the reactor coolant system pipes. The advantages of this technique are significant reductions in weld shrinkage, weld end prep machining, deposited filler metal quantities, and welding time.

As of September 30, 1992 the replacement of the steam generators was on schedule and going as planned. The only unexpected occurrence was some greater than expected movement of the piping when it was cut. That matter was under review at the close of the report period.

Radiation Protection at Nuclear Reactors

Daily monitoring of licensee and Regional Office reports to the NRC Operations Center alerts the NRC staff to potential problems developing in radiation safety, ranging from major repair problems involving highly radioactive components to contamination from small leaks of liquid and gaseous materials. These initial insights are followed by telephone discussions with regional representatives and eventual follow-through on any health physics problems in regional inspections. Deeper involvement of headquarters staff in regional and licensee problems is effected by the staff participation in routine environmental and radiological inspections, as well as in special team inspections investigating significant licensee problems, or the issuance of generic communications.

During fiscal year 1992, the NRC staff provided radiation protection oversight by licensing activity at most of the operating nuclear power reactors, as well as through reviews of design criteria and conceptual designs for advanced reactors. This work included acceptance reviews for Westinghouse AP-600 and the General Electric SBWR, a task continuing for the Advanced Boiling Water Reactor (ABWR), CESSAR SYSTEM 80+ and for the evolutionary and passive designs of the Electric Power Research Institute (EPRI). The activity included detailed evaluations of occupational radiation protection design features, systems, and equipment. In conducting the review, the staff resolved the shielding deficiency problem in the ABWR drywell. Design evaluation continued for the off-site consequences of design basis accidents for the ABWR CESSAR System 80+, and EPRI projects. Also included were reviews of control room habitability problems for such plants as Catawba (S.C.) and Surry (Va.). Licensing action during the period also included a generic re-examination of the radiological aspects of the steam generator tube analysis (SGTR) issue. Plant specific SGTR reviews were completed on Byron (Ill.), Braidwood (Ill.), Farley (Ala.), and Seabrook (N.H.). An important NRC staff function has included radiation-protection evaluations on the decommissioning activity at the Fort St. Vrain (Colo.) power reactor, as well as the U.S. Army Materials Technology Laboratory research reactor (Mass.). In addition, the staff has evaluated a proposal from the Kewaunee (Wis.) licensee for the disposal of wastes contaminated with very low levels of radioactivity. Another important staff function is in the area of generic communications on radiation-protection matter. During the report period, an Information Notice was prepared and issued on occupational worker hazards from high airborne radiation material fields created during maintenance activities at the Limerick (Pa.) and the Fitzpatrick (N.Y.) plants.

Inspection support was provided during the year for radiation-protection inspection at the Calvert Cliffs (Md.) plant, and a special team inspection covering the ALARA ("as low as reasonably achievable") radiation exposure reduction program at the Perry (Ohio) plant. Additional inspection support was provided in areas of procedure development for radiochemical analysis of waste oil associated with the San Onofre (Cal.) licensee and reviews of the adequacy of radiation monitoring systems at the Palo Verde (Ariz.) and Trojan (Ore.) licensees.

In response to the major revision of 10 CFR Part 20, the NRC staff conducted technical training sessions for the radiation protection inspectors at each of the five Regional Offices. The staff also developed question and answer (Q&A) packages (over 150 Q&As) in response to licensee and NRC staff queries on the new Part 20 implementation. The NRC staff held numerous regional conferences with the licensees, gaining their feedback and providing them implementation guidance on the revised Part 20.

The Office of Nuclear Reactor Regulation (NRR) staff continued to provide support to the NRC Office of Nuclear Regulatory Research (RES) through participation in a public workshop on the 10 CFR Part 51 rule change, which would limit the scope of environmental issues that a licensee needs to address when applying for a renewal of an operating license, under the provisions of 10 CFR Part 54. Additional significant technical support continued to RES that focused on the development, preparation and issuance of 10 regulatory guides associated with the revised 10 CFR Part 20.

Environmental Radioactivity Near Nuclear Power Plants

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 from the plant. This environmental monitoring program verifies 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.

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 have the overall level of effluent management and control required by the 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 range from lake, river and well water for water-borne contaminants; to radio-iodine and particulate dusts for airborne contaminants; to 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 quarter.

Results of all licensee measurements in their radiological environmental monitoring program are recorded in an annual radiological environmental report, which is 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 measurements for each power reactor site from this "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 them to carry out environmental monitoring. The purpose of the State contracts is to establish policies and procedures under which the States independently monitor the environs of the NRC licensed facilities. The States provide assistance by collecting samples or making radioactivity measurements in the environs of licensed facilities. These measurements duplicate, as closely as possible, certain parts of the licensee's environmental monitoring efforts, but they are executed independently of the licensee. The results of State monitoring are used to check the accuracy of licensee monitoring programs and to aid in verifying the ability of the licensee to measure radioactivity in environmental media.

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 mandated plant upgrades imposed on all LWRs shortly after the 1979 accident at Three Mile Island. Since 1983, the annual average doses for both PWRs and BWRs have been steadily declining.

In 1991, the average dose-per-unit for LWRs was 253 person-rems. This is 24 percent less than the 1990 average of 333 person-rems. In 1991, the average dose-per-unit for PWRs was 223 person-rem, down 22 percent from the average dose-per-unit of 285 person-rem in 1990. The activities which most frequently involved exposure to radiation in the nuclear plants in 1991 were steam generator-related work, valve maintenance and repair, installation and removal of scaffolding and insulation, and inservice inspection work.

In 1991, the average dose-per-unit for BWRs was 314 person-rem. This is 26 percent lower than the average dose-per-unit for BWRs of 426 person-rem in 1990. Major contributors to BWR doses in 1991 included valve maintenance and repair, inservice inspection work, control rod drive replacement and repair, installation and removal of scaffolding and insulation, and refueling activities.

The 1991 dose compilation includes data from 74 PWRs and 37 BWRs. The total reflects the addition of two new PWRs, Comanche Peak (Tex.) and Seabrook (N.H.). New plants which have not been in commercial operation for a full year are not included in this compilation. One PWR, Rancho Seco (Cal.), has been dropped from this year's annual listing, since this plant has been permanently defueled. Other plants no longer 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.), and Three Mile Island Unit 2 (Pa.).

The NRC has ongoing contracts with Brookhaven National Laboratory (BNL) related to occupational dose reduction at LWRs. The purpose of one of the NRC-sponsored studies is to monitor U.S. and foreign nuclear power plant efforts to reduce occupational dose. Other BNL studies involve the compilation of an ongoing annotated bibliography of selected readings in radiation protection and reduction, a study of the impact of reduced dose limits, and hot particle production, mitigation and chemistry. The NRC also has an ongoing contract to evaluate the affects of hydrogen water chemistry on shutdown radiation levels in BWRs.

Implementation Status of Safety Issues

The NRC publishes a document annually giving the status of the implementation of actions dealing with major safety issues. Volume 1 of this document—setting forth the status of implementation and verification of actions addressing the Three Mile Island (TMI) Action Plan Requirements—was published in March 1991. Volume 2—describing the status of implementation and verification of unresolved safety issues (USIs)—was published in May 1991. Volume 3—which addresses the status of implementation and verification of generic safety issues (GSIs)—was published in June 1991. These reports constituted the basis for a combined, updated annual report to the Commission, which was published in December 1991 as Supplement 1. Supplement 2, published in

December 1992, includes the status as of September 30, 1992, of implementation and verification of all such multi-plant actions. As reported in Supplement 2, approximately 99 percent of the TMI Action Plan items have been implemented at the 111 licensed plants. Of the 13,408 applicable items, 13,322 have been completed or closed out, and only 86 remain open. About 47 percent of the remaining 86 open items are projected to be implemented by the end of calendar year 1993.

Fill-Oil Loss in Rosemount Pressure Transmitters

On April 21, 1989, the NRC issued Information Notice (IN) 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 led 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, in which it requested that licensees promptly identify and take appropriate corrective actions regarding Model 1153 Series B, Model 1153 Series D, and Model 1154 transmitters manufactured by Rosemount that may have the potential for leaking fill-oil. These actions included removing certain transmitters from reactor protection (RPS) and engineered safety feature (ESF) actuation systems.

Rosemount has made improvements to the manufacturing process and the post-production screening for transmitters produced after July 11, 1989. These improvements included making process changes to reduce stresses on the sensor modules and pressure testing the sensors to identify any incipient failures caused by leaking fill-oil.

During the summer and fall of 1990, the Nuclear Management and Resources Council (NUMARC) surveyed the industry to gather data on all installed Rosemount Model 1153 and 1154 transmitters and safety-related Model 1151 and 1152 transmitters at commercial nuclear facilities. NUMARC also requested data on all suspected or confirmed failures of Rosemount transmitters attributed to a loss of fill-oil from these same facilities.

The staff continued to review the Rosemount transmitter loss of fill-oil issue by analyzing data gathered from (1) licensee event reports, (2) the licensees' 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," and (6) numerous meetings with representatives from the industry, NUMARC, and Rosemount. The Brookhaven National Laboratory (BNL) assisted the NRC staff in evaluating the data by assessing the failure rates for various types of transmitters according to operating pressure, time-in-service, and suspect or non-suspect lot classifications. This additional information was collected and analyzed. 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. In addition, new techniques were developed to identify failing transmitters. With this information, the staff wrote an updated supplement to the original Bulletin. On April 7, 1992, the proposed Supplement 1 to Bulletin 90–01, was published in the *Federal Register* for public comment. The staff received 12 replies to this notice. The comments received primarily concerned the scope of coverage for the transmitters to be addressed, and clarification of the exact nature of requested actions. On July 23, 1992, the staff held a public meeting to discuss the comments received and their disposition; there were more comments recorded at this meeting. The staff has reviewed all comments received and has modified the supplement where warranted. On September 8, 1992, the Committee to Review Generic Communications (CRGR) reviewed the proposed supplement and supplied comments, which the staff incorporated into the supplement.

The supplement to the Bulletin was nearing final form and readiness for presentation to the Commissioners for their approval, at the close of the report period. The Supplement requests utilities to perform enhanced surveillance testing on the Rosemount transmitters, commensurate with their importance to safety and demonstrated failure rate. In general, transmitters used in high-pressure applications, above 1,500 pounds-per-square-inch (psi), or in safety systems, will require testing more frequently than transmitters in medium pressure applications, between 1,500 and 500 psi, and transmitters in low-pressure applications will require testing only with normal surveillance techniques. Transmitters that have reached the appropriate time-in-service may be excluded from the enhanced surveillance. On a case-by-case basis, licensees may request to monitor transmitters on a more extended cycle, if sufficient justification is provided based upon transmitter performance in service and its specific safety function. In all instances, a high degree of confidence must be maintained for detecting failure of these transmitters caused by a loss of fill-oil, and a high degree of reliability has to be maintained for the function consistent with its safety significance. The improved transmitters manufactured after July 11, 1989, are not subject to these requirements. The enhanced surveillance monitoring program should provide early identification of problem Rosemount pressure transmitters.

Salem Unit 2 Nuclear Power Plant

On November 9, 1991, a catastrophic failure of a turbine-generator occurred at the Salem Unit 2 (N.J.) nuclear power plant. Unit 2 was operating at 100 percent power when the plant operators initiated a routine test procedure to verify the operability of steam turbine protection features. During the test, a momentary oil pressure perturbation occurred in the control oil system. Though of short duration, the oil pressure decrease was sufficient to open the interface valve. This valve functions to relieve the emergency trip oil pressure from the turbine steam admission valves. As a result of the pressure decrease, the steam admission valves closed and isolated the steam flow to the high and low-pressure turbines.

The control oil pressure perturbation also caused the reactor trip breakers (RTBs) to open, bringing about an immediate reactor "trip," or shutdown. Because of the test in progress, the primary turbine trip system was isolated and incapable of providing turbine trip assurance. That fact shifted reliance to the backup emergency turbine trip system.

The opening of the RTBs caused the emergency trip solenoid valve to be electrically energized. The reactor trip also initiated a 30-second delay, for opening the output breakers from the main generator. Although it was energized, the emergency trip solenoid valve failed to open. When the control oil pressure returned to normal, the interface valve closed. Because the emergency trip solenoid valve did not open, the emergency trip oil pressure increased, which started the reopening of the turbine steam admission valves. Steam was admitted to the turbine at about the same time that the output breakers from the main generator opened. The disconnection of the main generator from the grid effectively removed all load resistance from the turbine-generator system. Consequently, as high energy steam was readmitted to the turbine, the turbine-generator began to overspeed. At 103 percent of the normal rated turbine speed of 1,800 rpm, two additional solenoid valves were electrically energized. However, these valves failed to open to relieve the emergency trip oil pressure, and the turbine-generator unit continued to overspeed.

When the turbine speed reached approximately 2,900 rpm, several blades in the No. 22 low-pressure turbine section separated from the rotor disc, penetrated the 1.25-inch thick steel turbine casing, and became projectiles. Because the Salem turbine generators are outside on the turbine building roof, the projectiles landed on the roof and the ground around the turbine building. In addition, some of the blade pieces were propelled into the main condenser where they severed or damaged the condenser tubes. No nuclear safety systems were affected by the turbine projectiles.

The resulting eccentric motion of the rotor shaft caused severe vibration at the main generator. Consequently, the generator's hydrogen seals failed and the seal oil lines ruptured. Hydrogen gas (used to cool the generator) and seal oil (used to pressurize the generator hydrogen seals) were released and ignited. A fire erupted in the immediate area of the generator.

When the operators performing the turbine test recognized the situation, they restored the control oil system to normal. An operator manually tripped the turbine, to assure that the control oil system functioned to open the interface valve and thus relieve the emergency trip oil pressure that was holding the steam admission valves open. These actions isolated the turbine from further steam admission. The event duration was about 74 seconds.

In accordance with its emergency plan, the licensee initially declared an Unusual Event, with a brief upgrade to an Alert, until the licensee determined that the turbine projectiles had not affected any safety-related systems. All reactor plant systems operated normally and the reactor was brought to a safe shutdown condition. No radiological releases occurred. The fire was extinguished within 20 minutes, by a combination of automatically actuated fire suppression systems and rapid response from the on-site fire brigade. The Unusual Event was terminated in about three hours.

In response to this event, Region I dispatched an Augmented Inspection Team (AIT) to the site to review and evaluate the circumstances and significance of this occurrence. The team arrived on-site on November 10, 1991. Between November 10, 1991, and December 3, 1991, the team conducted an independent inspection, review and evaluation of circumstances and events associated with this occurrence. The AIT concluded that the proximate cause of the event was the failure of the turbine control solenoid valves to function as designed to prevent turbine overspeed, and to effect and maintain closure of steam admission valves. The solenoid valves failed to function because of mechanical binding, caused by a combination of foreign material, sludge build-up, and general corrosion, that prevented the functioning of the solenoids' internal components. The solenoid valve malfunction was not detected or corrected by the licensee because of ineffective surveillance test methods and a lack of any preventive maintenance.

In addition, a number of precursor events were identified that were pertinent to the solenoid valve failures experienced at Salem Unit 2. The most significant was the failure of the same solenoid valves at Salem Unit 1, on September 10, 1990. As a result of those failures, the licensee had agreed to replacing the valves at Salem Unit 2 at the first outage of sufficient duration. Though such an outage occurred in May 1991, the valves were not replaced at that time, and action was deferred until the scheduled January 1992 refueling outage; this delay was caused by a deficiency in the commitment tracking system.

On October 20, 1991, an opportunity to repair the solenoid valves was missed, when the Salem Unit 2 turbine was being placed in service. A part of that procedure required the operators to perform a test of the control oil system by verifying that certain steam admission valves close when the test switch was placed in TEST. The test was attempted at two different times, and the valves did not close either time. Although five individuals were involved, to varying degrees, none clearly understood the implications of the test results. Subsequently, the turbine was placed in service without resolving the indicated problem in the control oil system.

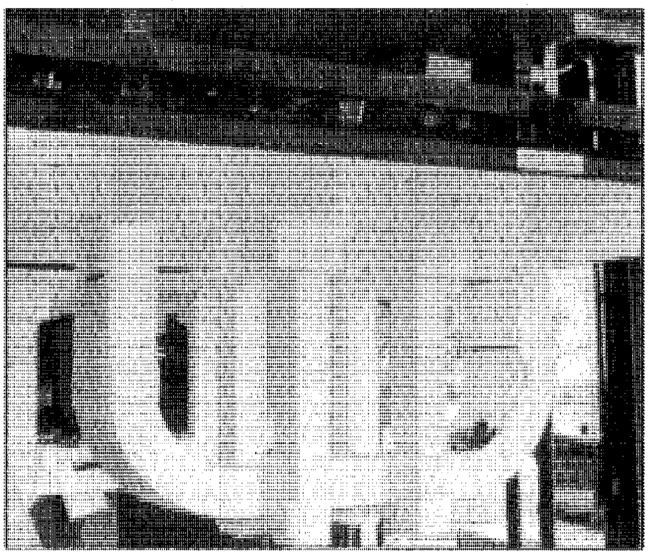
Salem Unit 2 remained shut down until the end of May 1992. During that time, the unit was refueled, the three low-pressure turbine rotors were replaced, the generator was replaced and the exciter was rebuilt. Also, extensive repairs to the main condenser were required as a result of broken pieces of turbine blades severing tubes in the main condenser.

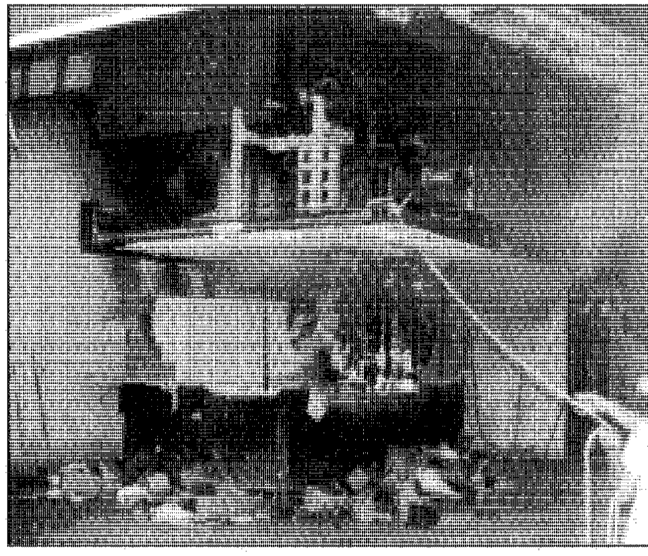
Thermo-Lag Fire Barrier Systems

Following a fire at the Browns Ferry (Ala.) nuclear power plant 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, the fire protection rule for operating plants. The rule applied to all nuclear power plants licensed to operate before January 1979; however, three sections in Appendix R of the rule were later made applicable to all plants. These provisions deal with 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 the systems necessary to achieve and maintain hot shutdown conditions, from either the control room or emergency control stations, be free from fire damage. Licensees can satisfy this requirement by separating redundant safe shutdown trains located within the same fire area outside primary containment, achieving that separation by providing one of the following conditions: (1) a horizontal distance of at least 20 feet with no intervening combustibles and installed fire detectors and 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 acceptance of a substance called Thermo-Lag 330-1, manufactured by Thermal Science, Inc., in St. Louis, Mo., to satisfy the NRC's new fire protection requirements. During the period 1982-to-1991, a number of concerns were brought to the NRC's attention about the acceptability of Thermo-Lag. If these concerns had been fully pursued by the NRC, the generic issues that were later identified could have been addressed earlier than 1991. Another concern has been the failure of the NRC to perform an adequate initial review of Thermo-Lag. Currently, Thermo-Lag fire barriers are installed in a majority of operating plants to meet the requirements of 10 CFR 50.48 for safe shutdown capability. Unlike a barrier, such as a masonry wall, which relies on endurance to provide protection, Thermo-Lag acts as a sacrificial material. In a fire, it sublimates, expands and chars, and is partially consumed. It is mainly used to separate redundant cable trays and conduits by surrounding the cable trays or conduits within a Thermo-Lag enclosure. Some licensees have also used Thermo-Lag to construct walls, ceilings and vaults.

By 1991, the NRC had received information about problems at the River Bend (La.) nuclear power plant which raised questions as to the adequacy of Thermo-Lag as an effective fire barrier. In response, the NRC established a Special Review Team to review the issues identified and make recommendations for their resolution. The Special Review Team completed its activities in February 1992 and issued its final report in April 1992. The Special Review Team concluded that: (1) the fire resistive ratings and ampacity derating factors (lowering the current-carrying capacity of cables to account for the insulating effects of the fire barrier) for Thermo-Lag were indeterminate, (2) some licensees had performed an inadequate review and evaluation of fire endurance test results and ampacity derating factors to determine the validity of the tests and 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 fire tests of cable tray and conduit barriers conducted by the nuclear industry, and small-scale panel tests performed for the NRC staff at the National Institute of Standards and Technology, demonstrated that certain Thermo-Lag fire barrier configurations may not provide the level of fire resistive protection needed to satisfy the NRC's



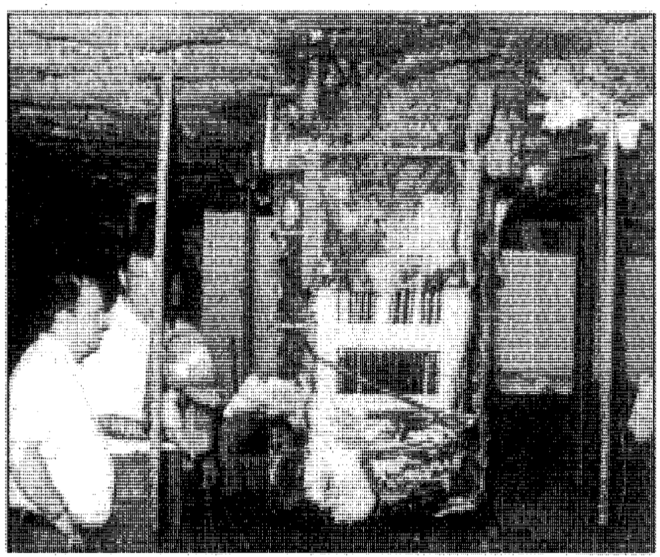


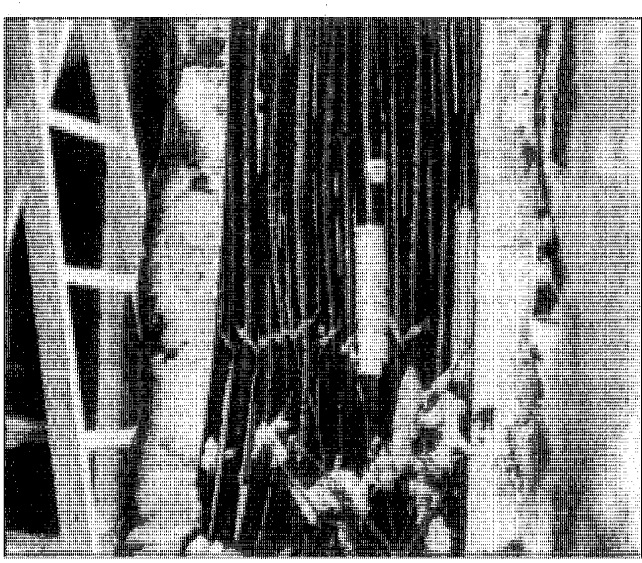
A set of "one-hour" fire endurance tests using the Thermo-Lag material were conducted by Omega Point Laboratories, in San Antonio, Tex., for the Texas Utilities Generating Company, in connection with its application for licensing of the Comanche Peak Unit 2 plant. The material, approved by the NRC in 1981 for use as a "fire barrier" and installed in numerous plants around the country, has come increasingly into question with respect to its ability to retard the spread of fire for a period of time sufficient to satisfy regulations. (See text for discussion of resultant investigations and corrective action.)

The tests, involving 17 separate configurations, were carried out between June and December of 1992. Results showed that, when installed and configured properly, the material will perform the "one-hour" fire barrier function, which, in combination with other fire protection provisions, may meet regulatory requirements (see text).

The photo at top left shows a 12-inch wide cable tray test specimen protected by the Therm-Lag material. In the photos at the right, top to bottom, the specimen is removed from a test furnace, having been exposed to heat of about 1,700 degrees Fahrenheit for one hour, and examined by technicians. Below is a close-up of the thermal degradation and exposure of the cables in the cable tray.

Further regulatory investigation and coordination with industry will be pursued until the technical and programmatic issues set out in the NRC staff's action plan have been resolved.





requirements. Further, some Thermo-Lag barrier applications used by some licensees, such as applications in walls and ceilings, have not been qualified as fire barriers by test.

The staff incorporated these issues into an action plan to ensure that the issues are tracked, evaluated and resolved. In addition to the special review team report, the NRC staff issued five information notices to the industry, a generic letter, a bulletin and a bulletin supplement; developed a proposed staff position for fire endurance test criteria; reviewed various industry full-scale test programs; and conducted toxicity, combustibility, and small-scale fire tests.

As part of the corrective action taken on issues related to Thermo-Lag, most licensees are participating in an industry program coordinated by the Nuclear Utilities Management and Resources Council (NUMARC). Efforts to date have concentrated on development of acceptance criteria for fire barrier testing. The staff is working closely with NUMARC and individual licensees (e.g., Texas Utilities Electric and Tennessee Valley Authority) to review and monitor industry fire tests, derating tests, and other industry activities. For the short-term, the licensees have addressed the fire endurance problem by implementing compensatory measures, such as fire watches, where Thermo-Lag has been installed. Long term action to correct the Thermo-Lag fire barrier problems range from barrier upgrades and repairs to complete replacement of some barriers or relocation of the affected cables. Additional plant-specific analyses may also be required to address cable derating. The staff is also evaluating other fire barrier materials and systems used by the licensees. Regulatory action and coordination with the industry will continue until the technical and programmatic issues in the staff's action plan have been resolved.

Besides the technical issues involved in the use of Thermo-Lag, there has been a high level of Congressional and intervenOr interest in this 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 fire barriers can be effectively demonstrated. After the close of the report period, the staff completed a reassessment of the NRC reactor fire protection program that had been initiated in response to the programmatic issues identified during the staff's acceptance review of Thermo-Lag fire barrier materials. This self-assessment, which began in June 1992, was undertaken to identify strengths and weaknesses and possible areas for improvement. The staff is preparing a task action plan to address the issues identified in the reassessment.

Boiling Water Reactor Stability

Boiling Water Reactors (BWRs) may be subject to thermal-hydraulic and neutronic-driven power oscillations when operating at low flow and relatively high power, as, for example, during 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 initiated by an instability event on March 9, 1988, at the LaSalle Unit 2 (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, (3) the possible effects of large oscillations on anticipated transients without scram (ATWS), and (4) a more recent power oscillation event at Washington Public Power Supply System Nuclear Plant No. 2 (WNP-2).

Previous staff actions have include the issuance of NRC Bulletin 88–07, and Supplement 1, requesting BWR licensees to take specified interim actions to prevent significant oscillations until long term resolutions could be developed. Interim actions have been generally effective in increasing awareness of the potential problem among reactor operators. The staff and its consultants, and the BWROG, have engaged in a coordinated effort to improve understanding of stability phenomena and of the principal fuel, core design and operating parameters contributing to instability. Substantial effort was required to develop computer codes and to validate them.

Based on improved understanding, analyses have been performed by the BWROG and the staff to develop and evaluate long term solutions by which to detect and suppress oscillations and to evaluate ATWS events. Also, in September 1992, the staff issued "Density-Wave Instabilities in Boiling Water Reactors" (NUREG/CR–6003 (ORNL/TM-12130)), to document knowledge gained regarding BWR stability and design and operating sensitivities.

Staff understanding of the potential complications from adverse operating parameters was further enhanced by investigation of the August 15, 1992 instability event at WNP-2. The operators manually shut down the reactor after observing power oscillations of 25 percent, peak-to-peak, at operating conditions well below the stability exclusion region boundary. The instability has been found to arise, in part, from thermal-hydraulic characteristics of the fuel and the core loading pattern, and primarily from the power distribution, which involved large radial and axial peaking. Subsequent operation with more appropriate restrictions on power distribution produced stable operation. This experience has been factored into the staff's long term solution review.

The BWROG has proposed to resolve the stability 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 proposed several types of options to implement this solution. The two primary options involve (a) 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 (b) 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 online, using diverse characteristics of oscillation signals to detect instability and cause rod insertion. Staff review has found these solutions acceptable when augmented, in some cases, by procedures to monitor core power distribution and, in other cases, by an on-line stability monitor. The staff is developing a generic letter setting out requirements for the long term solution and reinforcing interim administrative controls.

The staff review has led to the conclusion that, for some ATWS events, large oscillations are possible and they could lead to melting of a small fraction of the fuel; however, containment integrity would be maintained, and the radiological consequences would remain within 10 CFR Part 100 limits. Revisions to Emergency Operating Procedures have been proposed to limit power oscillations during ATWS events.

Boiling Water Reactor Water Level Instruments

The staff is reviewing the potential for inaccurate reactor vessel water level indication in boiling water reactors (BWRs). It has been postulated that under certain conditions, the reference leg of the water level instruments could become saturated with dissolved non-condensible gases, such as hydrogen and/or oxygen. In the event of a rapid reactor vessel depressurization, the non-condensible gases could come out of solution and thereby displace water from the reference leg. This postulated phenomenon would result in a false high indication of reactor vessel water level.

On July 22, 1992, the staff requested activation of the BWR Owners Group (BWROG) Regulatory Response Group (RRG), in order to address this phenomenon. NRC Information Notice 92–54 was issued on July 24, 1992, to alert licensees to the potential for level instrument inaccuracies following a rapid depressurization event. On July 29, 1992, the staff held a public meeting with the RRG to discuss the issue, and on August 19, 1992, the staff issued Generic Letter (GL) 92–04.

GL 92–04 requested that each BWR licensee: (1) evaluate the impact of potential errors on automatic safety system response during all licensing basis transient and accident events; (2) evaluate the impact of potential level indication errors on operator's short and long term actions, during and after all licensing basis accidents and transients; (3) evaluate the impact of potential level indication errors on operator actions prescribed in emergency operating procedures; (4) notify the staff of any corrective actions taken; and (5) provide plans and schedules for corrective actions. All licensees have responded to GL 92–04. The licensees' responses: (1) confirmed that their level instruments are expected to fulfill their function of initiating safety systems prior to a significant depressurization; (2) committed to perform system modifications, should the BWROG experimental programs confirm the phenomenon and identify optimum corrective modification, if needed; and (3) confirmed that the appropriate plant personnel were sensitized to the potential for such occurrences. The staff found the licensees' responses acceptable.

The BWROG submitted a generic report dealing with short term action items, as identified at the July 29, 1992 meeting. The report concluded that there was not an immediate safety concern associated with the postulated effects of non-condensible gases on reference leg behavior. The NRC staff agreed with this conclusion. However, the staff considers the matter a significant issue that must be resolved. Level instrumentation should be of high functional reliability for long term operation. The staff believes that operators should not be distracted from responding to postulated depressurization events as a result of significant instrument errors. Addressing NRC staff concerns, the BWROG submitted a long term action plan and schedule for resolving the issue. The plan includes full-scale testing, analysis and review of potential modifications. The schedule for completion of the long term plan is July 1993. The BWROG transmitted two letters to all plant operations superintendents giving guidance to operators in the event they encounter this particular problem.

The staff will closely monitor the activities of the BWROG. The oversight will include review of their test plans, observation of some of the tests, and review the proposed generic corrective actions. The staff will prepare a Temporary Instruction to confirm that licensees have sensitized their operators to this phenomenon and ensure that the operators have received adequate guidance to respond properly to such an event.

Individual Plant Examination

The Commission issued Generic Letter No. 88–20, in November 1988, requiring each licensee and construction permit holder to conduct an individual plant examination (IPE) to systematically search for any significant contributors to core damage risk. The Commission encouraged use of probabilistic risk analysis (PRA) methodology in these examinations. 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). Additional 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. Thirty-one IPE reports were submitted to the NRC during fiscal year 1992, covering 40 nuclear power plants, compared to a total of seven IPE reports submitted during the two previous fiscal years. The NRC staff completed its review of three of these reports and issued staff evaluation reports to the licensees for two of these facilities (Seabrook (N.H.) and Millstone Unit 3 (Conn.)) in fiscal year 1992. The staff report for the Turkey Point (Fla.) facility was scheduled for issuance in October 1992. 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.

The staff has included in the IPE program provisions for licensees to resolve certain Unresolved Safety Issues (USIs), most notably USI A–45, "Shutdown Decay Heat Removal Requirements." The purpose behind identification of USI A–45 was to determine whether the decay heat removal function at operating plants is adequate and if cost-beneficial improvements in the function could be effected. The staff had concluded in the late 1980's that a generic requirement to require an additional dedicated decay heat removal system at all plants was not cost effective in terms of added safety benefit. However, the staff decided that, since the IPE program would call for each plant's decay heat removal systems to be examined for vulnerabilities, the most efficient way to resolve this USI would be to subsume it into the IPE program. Licensees have generally complied with staff's expectation that they address potential decay heat removal system vulnerabilities in their IPE reports. One notable plant modification to improve decay heat removal capability, made as a result of the IPE effort, was introduced at Turkey Point Units 3 and 4 (Fla.). The licensee decided that, by adding service water hose connections to provide backup cooling to two of the three reactor coolant system charging pump seals, the contribution of reactor coolant pump seal failures to core damage frequency would be reduced by about a factor of five (3E-4 to 6E- 5/yr). The staff is considering closure of USI A–45 in its plant-specific evaluation reports, following review of each plant's IPE report.

Operational Safety Assessment

The NRC headquarters staff participates with NRC regional staff in the review and follow-up of events at nuclear facilities, for the purpose of determining probable cause(s) and identifying generic issues that need to be communicated to the industry. The reviews involve evaluating events against existing safety analyses, appraising facility and operator performance during events, reviewing licensee analyses, and assessing any need for corrective action.

In fiscal year 1992, the NRC assigned augmented inspection teams, i.e., groups of regional technical experts augmented by personnel from Headquarters or other Regions, to determine the facts regarding the following events at nuclear facilities:

- Inadvertent reactor vessel draindown at Vogtle Unit 1 (Ga.) in October 1991.
- Moisture separator reheater drain line rupture at Millstone Unit 2 (Conn.) in November 1991.
- Turbine failure and electric generator fire at Salem Unit 2 (Del.) in November 1991.
- Partial loss of off-site power at Palo Verde Unit 3 (Ariz.) in November 1991.
- Flooding caused by circulation water line break at Perry Unit 1 (Ohio) December 1991.
- Equipment failures in reactor scram at Quad Cities Unit 1 (Ill.) in February 1992.
- Loss of shutdown cooling at Prairie Island Unit 2 (Minn.) in February 1992.
- Loss of ultimate heat sink at Nine Mile Point Unit 1 (N.Y.) in February 1992.
- Loss of off-site power and loss of control room annunciation at Nine Mile Point Unit 2 (N.Y.) in March 1992.
- Loss of control room panel annunciator windows and audio alarms at Palo Verde Unit 3 (Ariz.) in May 1992.
- Movement of fuel while critical at University of Michigan—Ford Reactor in June 1992.
- Failure of pressurizer safety valve to reseat at Fort Calhoun (Neb.) in July 1992.

- Unusual event and manual scram resulting from power oscillations at Washington Nuclear Unit 2 (Wash.) in August 1992.

- Scram without feedwater trip and other equipment failures at LaSalle Unit 2 (Ill.) in August 1992.

When generic problems are identified in the course of a staff review of reported events and problems, there are a number of actions that can be taken by the NRC. If warranted, Information Notices are issued to notify utilities of events or problems that could affect their plants. Utilities are expected to determine whether the problems described are applicable to their plants and to take appropriate corrective action. Bulletins have a similar function, but they request that specific actions be taken by utilities, and they require written confirmation when actions have been completed. In fiscal year 1992, the staff issued 105 Information Notices, including nine supplements, and four Bulletins, including one supplement. Generic Letters may also be issued to address operational safety matters having broad applicability. In fiscal year 1992, the staff issued 14 Generic Letters, including one revision and four supplements.

Cleanup at Three Mile Island

During fiscal year 1992, preparations continued for placing the damaged Unit 2 reactor at the Three Mile Island (Pa.) nuclear power plant (TMI-2) in post-defueling monitored storage (PDMS). PDMS constitutes a passive, monitored state similar to the SAFSTOR option of decommissioning. The NRC staff estimates that the plant will be physically ready to enter PDMS by August of 1993. The licensee, GPU Nuclear (GPUN), plans to keep TMI-2 in the PDMS state until both TMI-1 and TMI-2 are decommissioned, expected in 2014.

In August of 1988, GPUN submitted a safety analysis report (SAR) to the NRC documenting its proposal to amend the TMI-2 license to allow the facility to enter PDMS. Throughout fiscal year 1992, GPUN submitted 15 amendments to this SAR. The NRC staff and contractor consultants from Battelle Memorial Institute's Pacific Northwest Laboratory (PNL) have evaluated the licensee proposals, and a Safety Evaluation (SE) addressing the license conditions and technical specifications necessary to implement PDMS was issued on February 20, 1992. As part of the evaluation, the staff published a technical evaluation report (TER) which appraised PDMS as an integrated process and assessed licensee commitments that were not in the technical specifications. The staff published a notice of opportunity for a prior public hearing regarding the license change to implement PDMS, on April 25, 1991. One individual petitioned to intervene. The petitioner, the licensee, and the NRC staff reached a settlement on September 25, 1992, and the request to intervene was withdrawn; on October 16, 1992, the Atomic Safety and Licensing Board dismissed the proceeding.

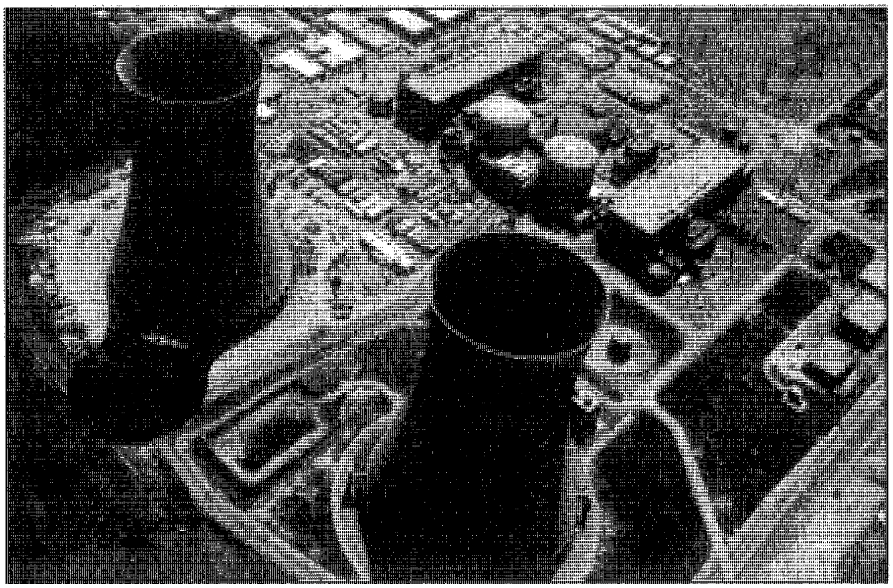
In early fiscal year 1992, final neutron measurements of the residual fuel remaining in the vessel were completed. The reactor vessel fuel measurement program is the final step in the special nuclear materials (SNM) accountability program at TMI-2. The SNM inventory is being taken in accord with agreements between GPUN, the Department of Energy (DOE), and the NRC regarding the core material accountability and fuel transfer to DOE. The NRC staff and consultants from PNL have performed independent evaluations and made independent measurements of GPUN's earlier fuel measurements in the auxiliary and reactor buildings. The staff and PNL will evaluate the final results of GPUN's reactor vessel fuel measurement program.

The evaporator used to decontaminate and dispose of the 2.3 million gallons of accident generated water (AGW) continued processing and vaporizing AGW during fiscal year 1992. During a large portion of fiscal year 1992, the evaporator system was used in a "decoupled" mode, i.e., the evaporators decontaminate incoming water, but no water is sent to the vaporizer. This mode is used to pre-process water for later reprocessing in the "coupled" mode, where it is vaporized. At the end of fiscal year 1992, a total of approximately 1,282,000 gallons of AGW had been decontaminated and vaporized.

The 11-member Advisory Panel for the Decontamination of Three Mile Island Unit 2, 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 membership.) During fiscal year 1992, the panel held two meetings in Harrisburg, Pa. Major topics discussed at these meetings included the NRC staff's SE and TER addressing PDMS, the status and progress of cleanup at the TMI-2 facility, and the decommissioning funding status and plans.

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.



Among the antitrust activities conducted by the NRC during the report period was one involving the Perry (Ohio) nuclear power plant. In the spring of 1991, the staff had denied the request of the Cleveland Electric Illuminating Company and the Toledo Edison Company, joint licensees, to suspend the antitrust license conditions for the Perry facility on the grounds that the plant was a "high cost" facility and that the agency was not authorized to impose antitrust conditions on a plant of that nature. The Perry plant is located on Lake Michigan, east of Cleveland.

In previous years, the Commission's antitrust review responsibility has been primarily confined to reviews of construction permit and operating license applications. Over the past several years, however, the staff's antitrust activities have been concentrated in the areas of license amendment reviews—usually associated with new owners or operators—and compliance proceedings initiated by requests to enforce antitrust license conditions.

During fiscal year 1992, the staff initiated or conducted the following activities associated with the NRC's antitrust review responsibility: (1) completed operating license amendment reviews in Seabrook (N.H.) and Millstone Unit 3 (Conn.), resulting from the merger of Public Service Company of New Hampshire and Northeast Utilities; (2) initiated two other operating license amendment reviews pursuant to requests by Georgia Power Company (GPCO) to change plant operators at the Hatch and Vogtle plants, both in Georgia; (3) performed a re-evaluation review of its "No Significant Antitrust Change Finding," pursuant to the change in ownership of the Seabrook nuclear power plant; (4) completed the operating license significant change review for Comanche Peak Unit 2 (Tex.) nuclear power plant; (5) settled a long-standing Section 2.206 compliance proceeding involving the Diablo Canyon (Cal.) licensee, Pacific Gas and Electric Company, and the Northern California Power Agency; and (6) initiated a hearing before the Atomic Safety and Licensing Board pursuant to certain licensees' requests to suspend antitrust license conditions as part of the Perry and Davis-Besse (both in Ohio) operating licenses.

The staff found that the merger involving Northeast Utilities and Public Service Company of New Hampshire, a Seabrook and Millstone Unit 3 licensee, would not adversely impact the competitive bulk power market in New England. A significant factor in the staff's decision was the fact that extensive license conditions limiting the ability of the surviving company, Northeast Utilities, to abuse its market power in the region were imposed upon Northeast Utilities, as a result of a final decision in a related hearing at the Federal Energy Regulatory Commission. As a result of its own review and the existence of these license conditions, the NRC staff approved the change in owners of Millstone Unit 3 and Seabrook, as well as the change of operator at Seabrook.

In late fiscal year 1992, the Georgia Power Company (GPCO), licensee for both the Hatch Nuclear Plant and Vogtle Nuclear Plant (both in Georgia), submitted license amendment requests to the NRC requesting that staff approve the proposed change in operators of both Hatch and Vogtle from GPCO to Southern Nuclear. (A similar request was submitted by Alabama Power Company in fiscal year 1991.)

Pursuant to a request by the City of Holyoke Gas and Electric Department to re-evaluate its "No Significant Antitrust Change Finding" in the Seabrook case, the staff reviewed the request and determined that the City of Holyoke, in its request, had provided no new information or identified any information that was overlooked by the staff in its original "significant change" review of Seabrook. As a result, the staff reaffirmed its No Significant Antitrust Change Finding.

The staff completed its operating license significant change review of Texas Utilities' Comanche Peak Unit 2 and found that there had been no significant changes in the licensee's competitive activities since a similar review was completed for the licensee in 1989 for Unit 1 of Comanche Peak. (At the close of the report period, there had been no requests for re-evaluation of the staff's No Significant Antitrust Change Finding.)

After several years of negotiations and legal proceedings at Federal and State levels, the Northern California Power Agency (NCPA) and Pacific Gas and Electric Company (PG&E) resolved their contract dispute by assurance of NCPA's ability to purchase partial requirements power from PG&E and to use PG&E's transmission facilities. As a result of the settlement, NCPA withdrew its outstanding Section 2.206 petition that had alleged that PG&E was not in compliance with its Diablo Canyon antitrust license conditions.

In the spring of 1991, the staff denied the request by Cleveland Electric Illuminating Company (CEI) and Toledo Edison Company (TE) to suspend the antitrust license conditions as part of the Perry and Davis-Besse operating licenses. Generally, the licensees argued that their Perry nuclear plant was a high cost facility and that Section 105c of the Atomic Energy Act of 1954, as amended, did not authorize the NRC to impose antitrust license conditions on a high cost nuclear power plant. Staff denied the licensees' requests, and the licensees requested a limited hearing to determine whether the NRC does in fact have the jurisdiction under Section 105c to impose license conditions on a nuclear facility that, compared with available alternative measures, entail a high cost. A hearing was initiated in the summer of 1992 and a decision was expected from the Atomic Safety and Licensing Board early in fiscal year 1993.

INDEMNITY, FINANCIAL PROTECTION, AND PROPERTY INSURANCE

The Price-Anderson System

Under NRC regulations implementing the Price-Anderson Act (which became law on September 2, 1957, and was extended on 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.

Insurance Premium Refunds

The two private nuclear energy liability insurance pools—American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid policyholders a 26th 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 1992 totaled $15,313,036, which is approximately 68.1 percent of all premiums paid on the nuclear liability insurance policies issued in 1982 and covers the period 1991–1992. The refunds represent 74 percent of the premiums placed in reserve in 1982.

Property Insurance

The 10th annual property insurance reports submitted by power reactor licensees indicated that, of the 76 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 six 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 $2.515 billion currently available.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS), established by statute, in a 1957 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 also review any matter related to the safety of nuclear facilities when specifically requested to do so by the Department of Energy. And, 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, 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.

Consistent with the statutory charter of the committee, ACRS reports, except for 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, selected with a view to achieving broad and balanced reviews, is drawn from scientific and engineering disciplines and includes individuals experienced in conducting safety-related appraisals of nuclear plant design, construction and operation.

During fiscal year 1992, 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:

- Testing Requirements of the Westinghouse AP600 and General Electric SBWR Designs.
- General Electric Advanced Boiling Water Reactor Design.
- Key Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements.
- Electric Power Research Institute's Advanced Light Water Reactor Requirements Document.

The Advisory Committee on Reactor Safeguards (ACRS) was established by law to advise the Nuclear Regulatory 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. Current members of the ACRS (early 1993) are shown in the photo (see Appendix 2 for background of the members).

Seated (left-to-right) are Mr. Charles J. Wylie, Dr. J. Ernest Wilkins, Jr., Dr. Paul G. Shewmon (ACRS Chairman), Mr. James C. Carroll (ACRS Vice-Chairman), Dr. Harold W. Lewis.

Standing (left-to-right) are Dr. Robert L. Seale (new member in 1993, Professor, University of Arizona), Mr. Carlyle Michelson, Dr. Thomas S. Kress, Mr. Peter R. Davis, Mr. William J. Lindblad, Dr. Ivan Catton.

- Inspections, Tests, Analyses, and Acceptance Criteria Program for the General Electric Advanced Boiling Water Reactor Design.

- Power Increases for General Electric Reactors.

- Diablo Canyon Nuclear Power Plant Long Term Seismic Programs.

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

  - Trends in Estimated Core Melt Probability for Operating Reactors.
  - Use of Design Acceptance Criteria During 10 CFR Part 52 Design Acceptance Reviews.
  - Priority Rankings for New NRC Generic Issues.
  - NRC Staff Probabilistic Risk Assessment Working Group Program Plan.
  - Implementation of the NRC Safety Goal Policy.
  - Reliability of Emergency AC Power at Nuclear Power Plants.
  - Individual Plant Examination Programs.
  - Accident Management Programs.
  - Elimination of Regulatory Requirements Marginal to Safety.
  - Generic Implications of the Salem ATWS Event.
  - Piping and the Use of Highly Combustible Gases in Vital Reactor Plant Areas.
  - Reliability of ATWS Recirculation Pump Trip in Boiling Water Reactors.
  - Severe Accident Research Program Plan.
  - Digital Instrumentation and Control System Reliability.
  - Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers.
  - Hydrogen Control for PWR Dry Containments.
  - Vender Test Programs for the Support of Design Certification of Passive Light Water Reactors.
  - Steam Generator Tube Repair Limits.
  - Consistent Use of Probabilistic Risk Assessment.
  - Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants.
  - Technical and Severe Accident Issues Associated With Evolutionary Light-Water Reactor Designs.

  -Diesel Generator Reliability.

In addition, the committee also provided advice to the NRC on proposed rules, criteria, and regulatory guides, including:

- Proposed Revision to Appendix A to 10 CFR 100, Seismic and Geologic Siting Criteria for Nuclear Power Plants.

- Proposed 10 CFR Part 50 and Part 100 (Non-seismic) Rule Changes and Proposed Update of Source Term.

- Proposed Revisions to 10 CFR Parts 50 and 100 and Proposed Regulatory Guides Relating to Seismic Siting and Earthquake Engineering Criteria.

- Final Rule to Amend 10 CFR 50.55a—Codes and Standards.

- Advanced Notice of Proposed Rulemaking on Severe Accident Performance Criteria for Future LWRs.

- Proposed Amendments to the Fitness-For-Duty Rule (10 CFR Part 26).

- Proposed Revision 3 to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors."

- Proposed Regulatory Guide and Interim Standard Review Plan for License Renewal and Related Branch Technical Position on Fatigue Evaluation Procedures.

- Regulatory Guides for the Implementation of the Revised 10 CFR Part 20.

- Proposed Rulemaking to Modify 10 CFR Part 50.72 and Part 50.73 Operating Power Reactor Event Reporting Requirements.

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 Action

Chapter 3

This chapter covers activities of three NRC offices concerned with (1) gaining the fullest possible understanding of actual 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 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 system.
- 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 Report 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 non-reactor 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 for Operations (EDO) as to whether or not the requirements should be imposed.

The members of the CRGR, as of the end of fiscal year 1992 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 office.

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, and (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 1992, the CRGR has held 229 meetings and taken up a total of 391 separate issues. In fiscal year 1992, the CRGR held 15 regular meetings and considered 35 issues, including 13 generic "backfits" in the form of two Rules, two Generic Letters, four Bulletins, and five Regulatory Guides. A listing of the 35 issues considered by CRGR at its regular meetings follows.

Generic letter on reactor vessel integrity.

Proposed rule amendment and regulatory guide on diesel generator reliability.

Generic letter on relaxing technical specification surveillance intervals to reduce testing at power.

Supplement to generic letter on seismic adequacy of equipment in operating nuclear power plants.

Supplement to generic letter on motor-operated valves to relax staff position regarding position-changeable valves.

Generic letter on revising plant technical specifications to allow certain types of hydrostatic testing at boiling water reactors.

Revision to a regulatory guide on emergency preparedness to endorse industry-developed guidance on emergency action levels.

Supplement to generic letter on reconstituting fuel assemblies to restrict the definition of approved methods which may be used in justifying reconstitution.

Proposed rule amendment on reactor site criteria. Generic letter on installation of digital-based safety systems to require submittal of safety-related analog-to-digital replacements for NRC review.

Revision to a regulatory guide on recording and reporting of occupational radiation exposure.

Proposed rule amendment to add two cask designs to the list of approved casks for dry storage of spent fuel at power reactor sites.

Revision of a regulatory guide on quality assurance programs for the design, construction, modification and decommissioning of nuclear power reactors.

Supplement to generic letter to relax staff position on life testing and periodic replacement of reactor trip breakers.

Final rule amendment to update references in the rule to the ASME Code, to augment reactor vessel weld inspection requirements, to augment containment leakage test requirements, and to separate in-service inspection and inservice test requirements in the rule.

Generic letter on augmented inspection requirements for boiling water reactors with Mark I and Mark II containment designs.

Proposed rule to modify existing requirements for fitness-for-duty programs at operating nuclear power plants. Regulatory guide on radiation dose to the embryo/fetus.

Regulatory guide on monitoring criteria for methods to calculate occupational doses.

Regulatory guide on planned special exposures.

Revision to a regulatory guide on post-accident monitoring equipment to relax the staff position on Category I neutron flux monitoring systems.

Advance notice of proposed rulemaking on severe accident performance criteria for advanced reactors.

Emergency bulletin on adequacy of Thermo-Lag fire barrier systems.

Proposed rule amendment to revise emergency planning regulations related to emergency response exercises.

Proposed regulatory guide on standard format and content for license renewal applications.

Proposed standard review plan for license renewal applications.

Proposed branch technical position on equipment qualification requirements for review of license renewal applications.

Proposed branch technical position on fatigue for review of license renewal applications.

Expedited bulletin on effects of non-condensible gases on reactor vessel water level instrumentation in boiling water reactors.

Emergency bulletin supplement on adequacy of Thermo-Lag fire barrier systems.

Proposed standard technical specifications for operating nuclear power plants.

Bulletin supplement on loss of fill oil in pressure transmitters manufactured by Rosemount Inc.

Generic letter on the risk associated with leakage of combustible gases in nuclear power plants during normal operation.

Generic letter on availability and adequacy of design basis information for operating nuclear power plants.

During fiscal year 1992, at the specific request of the Commission (responding to an initiative of the President), the CRGR conducted a Special Review of NRC Regulations to identify existing requirements that could be reduced or eliminated without undue reduction of the protection of public health and safety. In conducting the Special Review, the Committee held six meetings, including one public meeting, and examined more than 100 regulations for possible alteration or elimination. Following this phase, the CRGR distinguished eight areas for potentially substantial reduction of requirements, and the Commission approved initiation of rulemaking actions to modify regulations in those categories, as appropriate.

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 1992, AEOD initiated a minor rulemaking to modify operating reactor event reporting requirements. The rulemaking was intended to relax reporting requirements regarding certain types of events, pursuant to 10 CFR 50.72 and 10 CFR 50.73. Specifically, the objective of the rule change was to exempt from reporting certain types of events—primarily those involving invalid actuations of a limited set of, narrowly defined, engineered safety features (ESFs). Such events include the invalid actuation/isolation or realignment of the following ESFs: the reactor water clean-up system, the control room emergency ventilation systems; the reactor building ventilation system; the fuel building ventilation system; and the auxiliary building ventilation system, or their equivalent ventilation systems. 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.

Based on the staff's review of several hundred reactor-years of operational experience, it was determined that these types of events do not contribute significantly to an understanding of reactor operational safety, and thus unnecessarily consume the industry's and the NRC's resources. It is anticipated that this rule change will result in about 150 (5–10 percent) fewer LERs per year and the same reduction in the 10 CFR 50.72 reports. The removal of these reporting requirements will not adversely affect the agency's ability to carry out its mission to protect public health and safety.

On June 26, 1992, the NRC published a Notice for Proposed Rulemaking on regulation reduction in the *Federal Register*. Subsequently, 10 comments were received from the public. The industry supported the agency's initiative to reduce unnecessary reporting, and two respondents opposed the rule change. Following evaluation and resolution of received comments, the final rule was published on September 10, 1992, and became effective on October 13, 1992.

The public comment period for the draft NUREG–1022, Revision 1, "Event Reporting Systems, 10 CFR 50.72 and 50.73, Clarification of NRC Systems and Guidelines for Reporting," closed on January 31, 1992. Thirty-seven letters of comments were received.

Public comments generally indicated that consolidation of reporting requirements was helpful, but that the clarifications constituted new and different guidance in major areas. Some comments also expressed concern that the guidance, contrary to the stated intent, would result in significant increases in reporting, with no apparent safety benefit. The comment letters generally expressed the view that the revised NUREG should be further amended, allowing for continuing interaction with the public to reach consensus on those clarifications that would benefit both the NRC and the industry. Areas of concern to the commenters included apparent increased reporting of ESF actuations, conditions outside the design basis, conditions that alone could prevent fulfillment of safety functions, administrative requirements of technical specifications, and internal and external threats.

A public meeting was held on May 7, 1992, to clarify the draft material and issues, to clarify major industry comments, and to discuss future activities. It was determined that most of the issues raised by commenters were problems in communication that could be resolved by clarification of the NUREG. However, substantial differences remained in the interpretation of reporting requirements for ESF actuations, and, in part, for conditions that alone could prevent fulfillment of a safety function.

The NRC staff is continuing the process of resolving and incorporating public comments to complete the document and plans another public meeting prior to publication of the final document.

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 the average, 150 data related to 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; the office participated in international meetings during the report period, as described under "International Activities," later in this chapter.

Reports of operational events received from the Nuclear Energy Agency (NEA) of the Organization of Economic Cooperation and Development, from the International Atomic Energy Agency (IAEA), 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 performance in the United States.

During the report period, the AEOD staff and contractors reviewed about 115 reports on foreign events submitted to the NEA/Incident Reporting System (NEA/IRS), and nearly 200 event reports received by the NRC as part of bilateral exchanges with various countries throughout the world. The NRC continued to participate in the NEA/IRS to share the U.S. reactor operational experience with the world nuclear community. In fiscal year 1992, about 50 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 entitled "Annual Industry Averages, 1988–1991" has been selected for quarterly review in the NRC Performance Indicator (PI) Program.

Selected industry trends are developed by analysis of operational experience data from 1988 through 1992. 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 five of these indicators may have stabilized (automatic reactor scrams while critical, safety system actuations, significant events, safety system failures, and forced outage rate). The equipment forced outages indicator has continued to trend downward, over the past five years. 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 either startup or operation above low power.

Reactor Scrams. As an essential element of basic reactor safety systems, the reactor shutdown or "scram" can result from initiating events that range from relatively minor incidents to events that are precursors of accidents. The 1992 data reflecting average automatic scrams is the same as in 1991. The last three annual industry averages for this indicator have remained almost constant.

In 1992, equipment failure remained the leading cause of scrams, causing approximately three times more than the next leading cause (personnel error). For scrams occurring at operating plants during 1992, the systems initiating the most scrams, in descending order, were the feedwater, the turbine, electrical, and reactor protection systems.

Safety System Actuations. AEOD monitors a subset of engineered safety features 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 engineered safety features systems.

The annual industry average for SSAs in 1992 was slightly less than in 1991. The averages for the past three years indicate a leveling 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 average number of significant events-per-plant has been decreasing since 1988. In 1991 and 1992, the number of significant events was the same, indicating a possible stabilizing of this indicator.

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. Safety system failures can have implications for a plant's readiness to respond to anticipated events and postulated accidents.

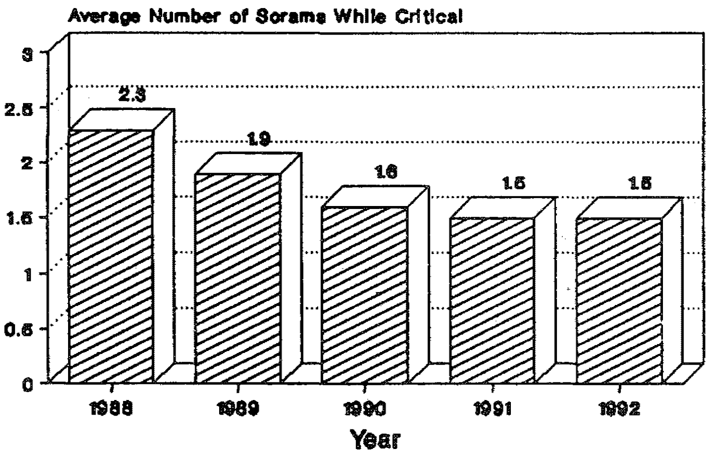
From 1988 through 1991, the trend in the average number of SSFs-per-plant was essentially flat. Although the data for 1992 show a slight increase over 1991, that may be within a normal statistical variance for this indicator.

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 forced outage rate has remained between 7.2 percent and 9.9 percent for the past six 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.

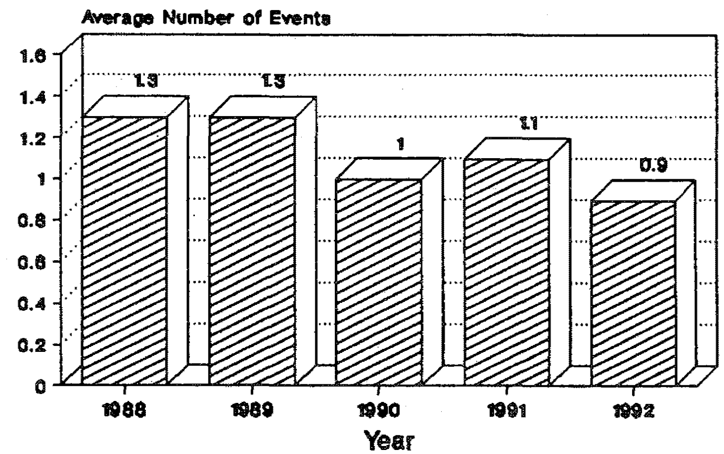
The industry average EFO rate has declined from 1987 through 1991. The slight decrease observed 1990 through 1992 may indicate the rate has stabilized.

Average Number of Reactor Scrams While Critical



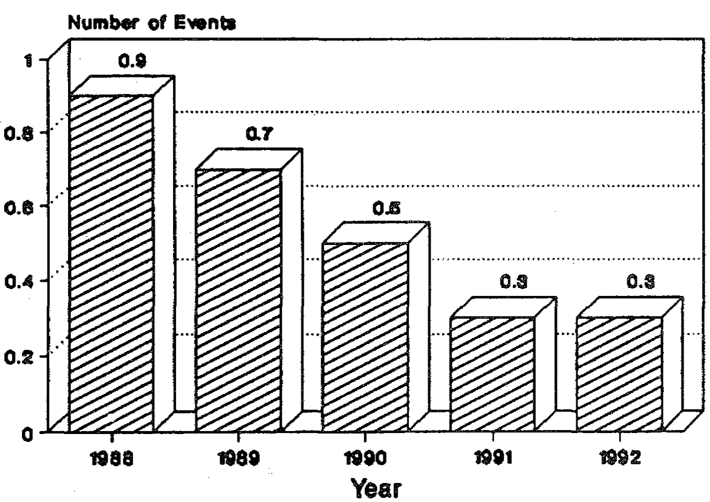
(1992 PROJECTED FROM 9 MONTHS OF DATA)

Average Number of Safety System Actuations



(1992 PROJECTED FROM 9 MONTHS OF DATA)

Average Number of Significant Events



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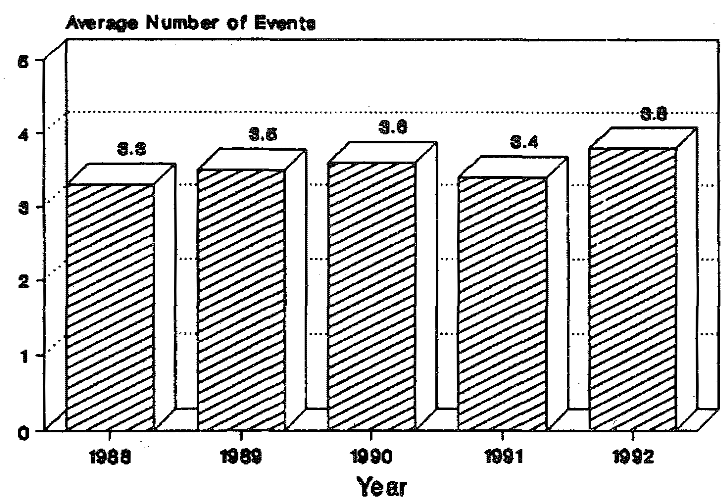
Average Number of Safety System Failures



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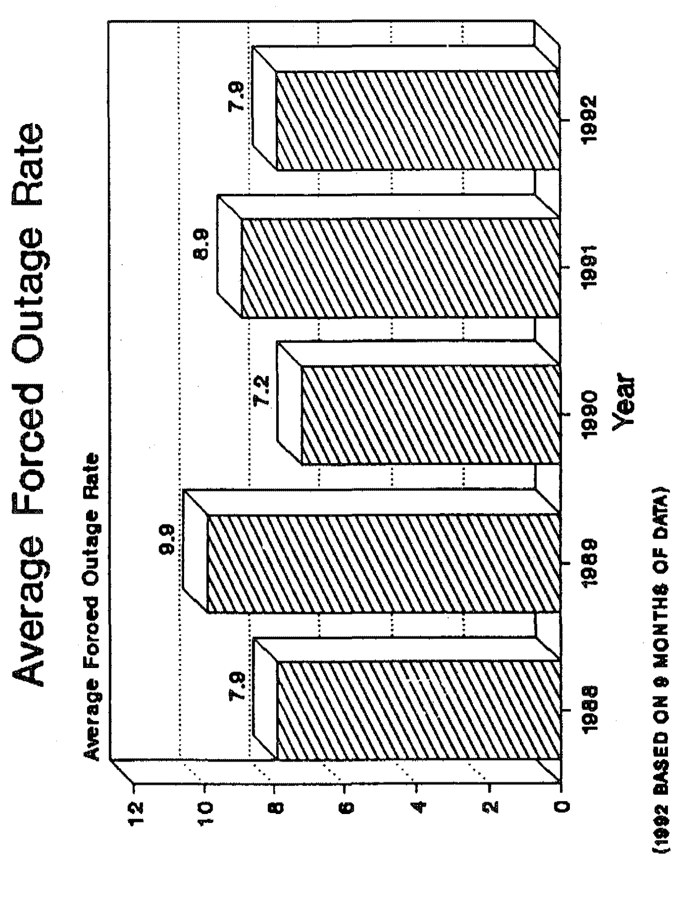
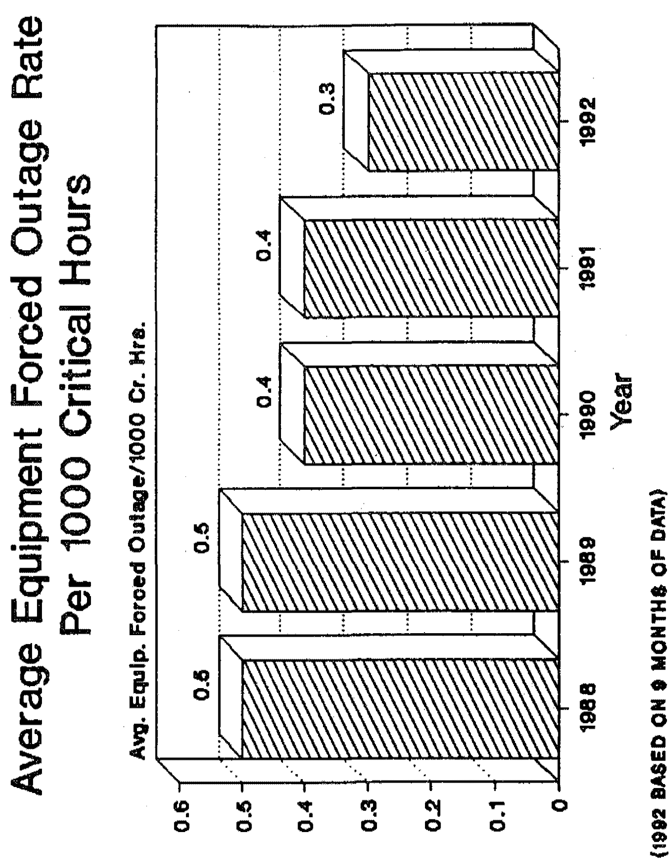
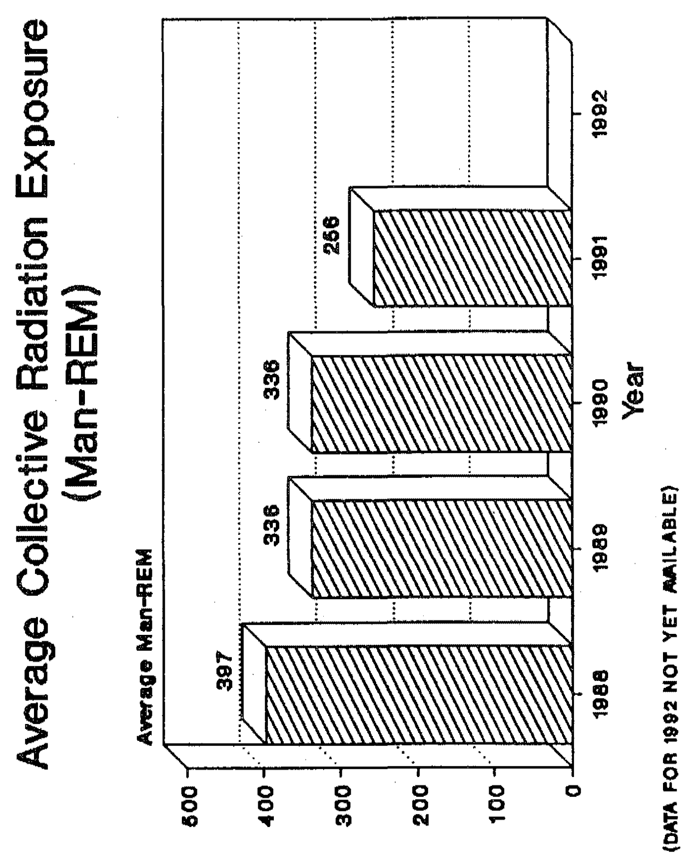






The graphs on these two pages display five-year trends (1988 through 1992) for Performance Indicators (PIs) 1-to-7 (the eighth indicator, "Cause Codes," is not subject to industry-wide calculation), beginning top-left and top-right for PI-1 and PI-2 and proceeding to PI-7, above. The averages shown do 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. The trends indicate that most of the PIs may have stabilized.

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.

The peer groups cited were developed on the bases of plant age and of differences in nuclear steam supply system vendor designs. These categorical refinements will improve the usefulness of the plant trend and deviation comparisons, by accounting for differences in event vulnerabilities that are due to unique differences in plant designs and operational cycle characteristics. Calculational techniques and display methods were also developed for the incorporation of these refinements into the PI report. The computational techniques included development of methods to indicate the statistical significance of the plant performance trends and deviations being observed, including the "cause-codes" indicator. The results of these efforts were outlined in early 1992 in a Commission paper, "Performance Indicator (PI) Program—Progress on Incorporating Peer Groups and Operating Cycle Phases" (SECY-92-083).

During fiscal year 1992, the AEOD staff began a test and evaluation program of the methods developed to incorporate these peer groups and operating cycle effects into the Performance Indicator Program. Three draft PI reports using the new methodology were produced in parallel with the approved quarterly PI reports. These drafts were reviewed by the Interoffice Task Group on Performance Indicators to evaluate the content, calculational methods, and display techniques of the draft reports. The AEOD staff, assisted by the Idaho National Engineering Laboratory and the Oak Ridge National Laboratory, evaluated the effectiveness of the new methods to identify plant performance trends. The results of these evaluations were used to improve subsequent draft reports. The program will be completed and recommendations presented to the Commission in fiscal year 1993.

The Office of Nuclear Regulatory Research (RES) completed work on the development of a risk-based indicator of safety system unavailability, in fiscal year 1992. The indicator is the product of the fractions of time of plant operation during which each train of selected safety systems would not have functioned on demand. AEOD will 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 1991, the IAEA convened a Technical Committee Meeting on "Experience with the Use of Plant Specific Safety Indicators" to review the current status of and exchange experience in the use of PIs around the world, and to identify future developments in the area of PIs. AEOD was represented at the meeting. A working group for regulators provided recommendations to the IAEA to continue sponsoring meetings for the exchange of experience in the development and use of safety indicators, and to convene consultants' meetings to further their acceptance and use worldwide.

Collective Radiation Exposure

Data on the industry's collective occupational radiation exposure for 1992 was not available at the close of the report period. The industry's collective radiation exposure declined from 1988 through 1991. 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 Reactors and Non-reactors

The six main sources of radiation exposure to people are natural radiation (82 percent) and radiation from the following five man-made sources (18 percent): medical uses, occupational activities, nuclear production of electricity, miscellaneous environmental sources, and consumer products. The average person in the United States receives an effective dose equivalent of about 50 millirems-per-year from medical applications. The whole fuel cycle, including the operation of nuclear reactors, contributes less than one millirem-per-year to the average person's exposure. All the other man-made sources of radiation combined add up to approximately six millirems-per-year effective dose equivalent.

Almost all of the radiation dose to human beings from nuclear power plant operations is occupational dose, i.e., the dose to the nuclear power plant employees and their contractors who work at the plant. Because the economics of operating a plant create a strong incentive to lower exposures and achieve ALARA objectives (exposure "As Low As Reasonably Achievable") objectives, utility violations of NRC limits on personnel exposure are rare; the vast majority of nuclear power plant personnel have

Table 1. Annual Exposure Data for Certain Categories of Licensees for 1990

Licensee Category	Number of Licensees Reporting	Number of Monitored Individuals	Number of Workers With Measurable Doses	Collective Dose (personDrems or personDcSv)	Average Individual Dose (rems or cSv)	Average Measurable Dose-per-Worker (rems or cSv)
Reactors	109	203,434	109,702	36,947	0.18	0.34
Industrial Radiography	258	6,523	4,458	2,120	0.33	0.48
Manufacturing & Distribution	55	4,195	2,272	693	0.17	0.31
Fuel Fabrication	10	13,756	3,233	287	0.02	0.09
Low Level Waste	2	784	115	26	0.03	0.23
Independent Spent Fuel Storage	2	56	22	6	0.11	0.27

annual exposures far below NRC regulatory limits, specified in 10 CFR Part 20.

Table 1 lists the exposure data for licensee categories for 1990. Of the six classes of licensees required to report collective exposures for monitored individuals, reactor licensees had the highest collective exposure (36,947 rems to 203,434 people) in 1990, by virtue of the large number of employees; they were followed by radiographers (2,120 rems to 6,523 people), manufacturers and distributors (693 rems to 4,195 people), and fuel fabrication licensees (287 rems to 13,756 people). Low-level waste disposal (26 rems to 784 people) and independent spent-fuel storage (six rems to 56 people) licensees have relatively low collective doses.

People who do not work at a nuclear power plant receive some radiation doses from plant operation. These doses are trivial compared with doses from nature and medical applications. The non-occupational collective doses are usually less than 0.2 percent of occupational exposures.

Although worker occupational exposures have been maintained at a low level, over- exposures continue to occur. Between 1986 and 1990, licensees reported 22 events at nuclear power plants involving 28 individuals who received exposures that exceeded the quarterly limits specified in 10 CFR Part 20. The rate of overexposures of radiographers is greater by more than a factor of 10 than that of personnel working at a reactor site. A summary of the data on the number of reports and the number of individuals overexposed in NRC-licensed facilities for reactors and non-reactors for the years 1986 through 1990 is given in Table 2. Data for Agreement State licensees are not included in this table, because they are not readily available. Every year the number of events and the number of individuals overexposed in non-reactor applications exceeded those exposed at reactor sites.

Accident Sequence Precursor Program

The Accident Sequence Precursor (ASP) Program was established at the Nuclear Operations Analysis Center at Oak Ridge National Laboratory in the summer of 1979 to provide a structured and systematic means of evaluating the safety significance of nuclear plant operating experience. The ASP program evaluates selected licensee

Table 2. Overexposure Events at Reactor and NRC Non-reactor Licensees–D1986–1990

Type of Licensee	1986		1987		1988		1989		1990	
	No. of Reports	No. of People	No. of Reports	No. of People	No. of Reports	No. of People	No. of Reports	No. of People	No. of Reports	No. of People
Reactors	4	4	4	4	10	14	2	4	2	2
Medical & Academic	2	2	4	4	6	6	10	17	7	8
Radiography	7	9	2	2	3	3	11	14	9	12
Commercial & Industrial	3	3	2	2	3	3	0	0	4	4
Fuel Cycle	1	1	1	2	1	1	0	0	1	3
Other	3	3	2	2	3	4	4	4	0	0

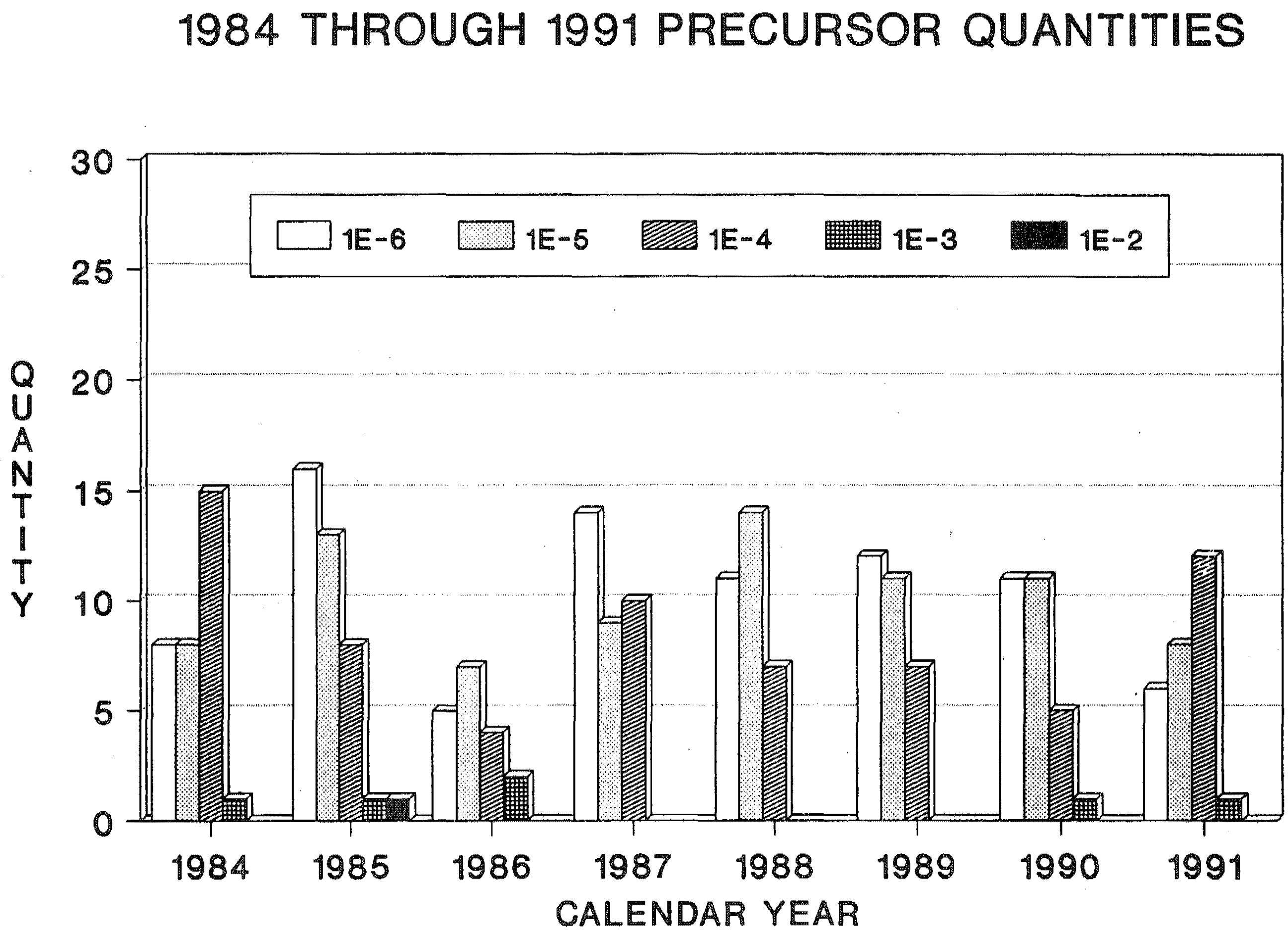
event reports of plant problems, equipment failures, or other operational incidents that the operators of the nuclear plants are required to report. An Accident Sequence Precursor is an operational event, or events, or a plant condition that is an important part of a postulated nuclear plant core damaging accident sequence.

The ASP program identifies and evaluates operational occurrences that involve portions of postulated core damage sequences. It evaluates plant equipment and human responses that could affect the progression of an accident, evaluating the actual failures that have occurred along with the probabilities for other, postulated, failures that could occur. It uses "event tree" models and probabilistic risk assessment techniques to provide a quantitative estimate of the significance of the reported data and, hence, provides perspective for evaluation. The event trees model plant responses to challenges such as transients, loss-of-coolant accidents (LOCAs), or loss-of-off-site power (LOOP).

Accident sequences considered in the ASP program are those associated with inadequate core cooling. ASPs are important elements in such sequences. Such precursors can be infrequent initiating events or equipment failures, that when coupled with one or more postulated events could result in a plant condition involving inadequate core cooling. The precursor method couples and evaluates seemingly disparate elements of operational experience with random failures assumed for other branches of the "event tree" models being evaluated. All actual or potentially concurrent failures, degradations, or outages of safety systems or related plant systems are accounted for in these evaluations.

The precursor events from the ASP program form a unique data base of historical system failures, multiple losses of redundancy, and infrequent core damage initiators. Several of the precursor events involved failure of equipment caused by factors or conditions or phenomenology that affected the ability of safety equipment to perform its function. These mechanistic failures are different from "random" failure or unavailability of equipment. The precursor results can help show whether plant designs and capabilities can cope with actual operational events.

Commercial nuclear power reactors in the United States now have over 1,600 reactor years of operating



The graph above shows quantities of Accident Sequence Precursor incidents identified from U.S. nuclear power plant operational data. The legend indicates ranges of the conditional core damage probability (CCDP) associated with these precursors. The ranges are rounded off to decades, e.g., the legend marker of 1E–4 denotes precursors having a CCDP in the range of 1E–4 to just under 1E–3. (The Vogtle (Ga.) plant precursor of 3–20–90 has been rounded upward from 9.7E–4 and is shown as a 1E–3 event in 1990.)

Events whose CCDP is in the 1E–3 decade are considered highly significant. As the chart indicates, the number of precursor events has remained relatively constant or has decreased in recent years, with the exception of the increase in 1E–4 events in 1991. The trends will continue to be closely monitored by the NRC.

experience. The precursor program utilizes information from this valuable nuclear plant experience data to provide an ongoing assessment of operating experience. This assessment helps indicate how well plant designs and capabilities cope with actual operational events.

Table 3 lists precursor events with estimated conditional core damage probabilities (CCDPs) greater than 10–6 that occurred in 1991, that is, given the event or condition, the estimated probability that the event or condition could have led to core damage was greater than one in one million.

In 1991, six events involved unavailability or potential unavailability of high-or low- pressure safety injection at pressurized water reactors. In two of these cases, hydrogen gas could have potentially made the high pressure injection pumps inoperable. It should be noted that in 1990, several ASP events involving gas entrainment occurred.

The event with the highest CCDP in 1991 was the unrecognized unavailability pressure safety injection for about a year at the Harris Unit 1 facility. In this event, both relief valves in the alternate minimum flow path for the safety injection pumps and some associated piping were found failed. These failures in the alternate minimum flow lines would have diverted sufficient flow such that flow requirements stated in the plant's FSAR would not have been met.

For 1991, six of the 27 ASP events were losses of off-site power. Losses of off-site power were also important ASP events in previous years. ASP analyses indicated that the loss of off-site power with the highest CCDP occurred at Yankee Rowe on June 15, 1991. In this event, caused by lightning, off-site power was lost for 24 minutes. All of the emergency diesel generators (EDGs) operated as designed. In another loss of off-site power event at Vermont Yankee, off-site power was lost for 13 hours. The EDGs worked properly and a tie line from off-site was available through operator actions, to power one-half the emergency equipment at the site.

In addition, three events involved unavailability of the EDGs which would be required to respond to loss of off-site power events. In two of these events (at the Perry (Ohio) plant, on March 14, 1991, and at Millstone (Conn.) Unit 2 on August 21, 1991), two diesel generators were potentially inoperable for some period of time.

In 1991, unavailability of equipment needed to mitigate the consequences of anticipated events (e.g., small break loss-of-coolant accident, losses of off-site power) made a significant contribution to the ASP precursors. The contribution of this type event compared to previous years (since 1984) appears to be on an upward trend. This trend will be watched closely to evaluate whether agency action is required if the trend continues upward.

Additional information and detailed analysis may be obtained in a set of publications, NUREG/CR-4674, Vols. 1 through 16, "Precursors to Potential Core Damage Accidents."

Results of AEOD Studies

In 1992, the AEOD staff continued to analyze and evaluate operating experience, publishing a major study on safety valve performance, and several technical reports describing equipment problems. Emergency diesel generator performance continued to be studied. Considerable effort was expended on the quantitative analysis of risk associated with operational events and conditions and in 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 that the NRC issued, 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 4 provides a list of 1992 reports.

"Safety and Safety/Relief Valve Reliability" (Special Report S92-02). This report, issued in April 1992, derived from a project in which the AEOD staff analyzed 1,100 events that had occurred from January 1981 to December 1989 in which a safety valve or a safety/relief valve (SRV) malfunctioned, during operation or during surveillance testing. The correct safety valve and SRV function is to open the valve within setpoint pressure tolerance limits to relieve overpressure conditions, and to reclose it to maintain system boundary integrity. The systems of interest were the reactor coolant system and the main steam system. The valves involved were Crosby and Dresser pressurizer safety valves (PSVs), Crosby and Dresser main steam safety valves (MSSVs), and Target Rock two-stage SRVs.

PSVs, SRVs, and MSSVs are usually allowed 1 percent tolerance on either side of the setpoint, i.e., setpoint q1 percent pounds-per-square-inch (psi). Approximately 70 percent of all the reported safety valve malfunctions were attributed to a condition called "setpoint drift," when the valves do not meet the q1 percent psi tolerance. The safety significance of safety valve or SRV malfunction was shown to be a degradation of overpressure protection for higher-than-required setpoints or a loss of coolant event for lower-than-required setpoints. Significant complications can occur during a post-scram (reactor shutdown) transient when a safety valve or SRV lifts unexpectedly or fails to reseat.

The study suggests that safety valve performance could be improved if the owners established a program similar to that used previously by another SRV owners' group to identify and correct SRV malfunctions. Other suggestions include development of standard practices for the maintenance and testing of safety valves and SRVs, in order to eliminate testing-induced errors and to establish effective corrective actions.

Findings of the study were forwarded to the Office of Nuclear Reactor Regulation (NRR) for action, and NRR has incorporated them into a proposed Generic Safety Issue, "Spring-Actuated Safety and Relief Valve Reliability," and has requested that the Office of Nuclear Reactor Research (RES) establish a priority for the issue. RES has scheduled work on that task to begin in January 1993.

Table 3. Precursor Events Occurring in FY 1991

Plant	LER Number	Date	CCDP	Description
Harris 1	400/91–008	04/03/91	6.3E–3	HPI unavailability for one refueling cycle due to inoperable alternate miniflow lines
Millstone 3	423/91–011	04/10/91	8.6E–4	Both trains of HPI inoperable due to relief valve failure
Yankee Rowe	029/91–002	06/15/91	6.1E–4	Loop due to lightning strike
Perry 1	440/91–009	03/14/91	5.3E–4	Two EDGs inoperable
Arkansas 2	368/91–012	05/15/91	4.8E–4	Both normal service water trains fouled by debris
Nine Mile Point 2	410/91–017	08/13/91	3.8E–4	Loss of five non-safety UPSs
Peach Bottom 3	278/91–017	09/24/91	3.3E–4	Control wiring for ADS/relief valves found damaged
Vermont Yankee	271/91–009	04/23/91	2.9E–4	Extended LOOP
McGuire 1	369/91–001	02/11/91	2.6E–4	Switchyard breaker test results in LOOP
Zion 2	304/91–002	03/21/91	2.1E–4	LOOP with one EDG out of service
Millstone 2	336/91–009	08/21/91	2.1E–4	Both EDGs unavailable and unit shutdown
Oconee 1	269/91–010	09/19/91	1.2E–4	Potential for hydrogen entrainment in HPI pumps
Pilgrim 1	293/91–024	10/30/91(FY92)	1.2E–4	LOOP and RCIC trip
FitzPatrick	333/91–014	08/05/91	9.5E–5	Hydraulic pressure locking of two LPI valves
Comanche Peak 1	445/91–012	03/26/91	6.2E–5	Potential charging pump unavailability due to hydrogen voids
Brunswick 1	325/91–018	07/18/91	6.0E–5	LOFW with degraded HPCI system
Seabrook	443/91–008	06/27/91	4.4E–5	LOOP
FitzPatrick	333/91–006	05/07/91	2.0E–5	Trip with both LPCI trains inoperable
Oconee 3	287/91–007	07/03/91	1.8E–5	Reactor trip due to LOFW plus degraded EFW

Table 3. Precursor Events Occurring in FY 1991
(continued)

Hatch 1	321/91–001	01/18/91	1.1E–5	LOFW with HPCI degraded and RCIC failed
Zion 2	304/91–004	06/11/91	1.0E–5	Main feedwater pump trip with one AFW pump failed
Harris 1	400/91–010	06/03/91	6.6E–6	Reactor trip breaker fails to open on trip
Salem 1	272/91–030	09/20/91	4.4E–6	Both PORVs failed due to leaking actuators
Surry 2	280/91–017	07/15/91	2.9E–6	Both EDGs for Unit 2 inoperable for 13 hours
San Onofre 1	206/91–014	08/07/91	2.1E–6	Inoperable VCT level transmitters
Diablo Canyon 2	323/91–003	09/01/91	2.1E–6	Containment sump isolation valves and containment spray pumps de-energized during hot shutdown
Indian Point 2	247/91–001	01/07/91	2.0E–6	Reactor trip and AFW pump failure

Performance of Emergency Diesel Generators in Restoring Power to Their Associated Safety Buses DA Review of Events Occurring at Power (Special Study Report AEOD/S91–01). The AEOD staff reviewed actual operating experience to gauge the capability of Emergency Diesel Generators (EDGs) to restore power to a safety bus that had lost power or experienced a no-voltage or sustained undervoltage condition. The staff considered only those events that occurred while the plant was producing power. The data base for the study included approximately 160 LERs from January 5, 1985, to June 25, 1990, a period of 5+ years.

During this period, there were 128 EDG train level demands initiated by an actual loss of power to a safety bus with the plant producing power. None of the events resulted in a station blackout (i.e., loss of all a.c. power). In 114 of the 128 train-level challenges involving a dead safety bus, the EDG successfully started and re-energized the bus and EDG support equipment, and 14 train-level failures occurred. Five of the failures to load occurred when the EDG was out of service for maintenance, and was, therefore, unavailable. In the remaining nine failures, the EDG started but failed to provide emergency power to its associated emergency bus. Of these nine failures, five were caused by various personnel errors. Of the remaining four events, two were attributable to mechanical equipment failures, and two were due to the failure of the EDG's service water pump to start automatically in the load sequence scheme. Of these nine failures, it was decided that most of the situations could be successfully rectified in a reasonable time by operator intervention.

The AEOD staff concluded that the capability of the EDGs to automatically start, load their respective safety buses, and provide power to the engineered safety features is within the range of the reliability goal of 95 percent suggested in Regulatory Guide 1.155, "Station Blackout," to cope with station blackout. It was thought that the level of unavailability occasioned by the EDG's being out of service for maintenance may be higher than was anticipated in the regulatory guide.

Analyses of Human Performance in Operating Events

AEOD continued a program to expand the staff's understanding of human performance during reactor events. Under the program, teams of NRC staff and contractor specialists perform studies of selected events at plant sites shortly after the events occur. During fiscal 1992, studies were completed for the following four events:

(1) Crystal River Unit 3 (Fla.)—Pressurizer Spray Valve Failure (12/8/91).

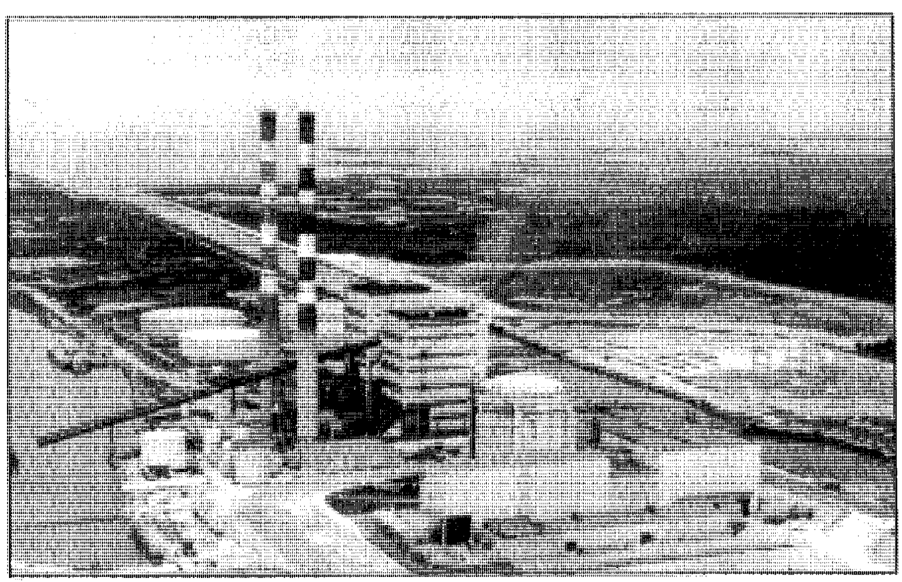
(2) Prairie Island Unit 2 (Minn.)—Loss of Shutdown Cooling (02/2/92).

(3) LaSalle County Unit 2 (Ill.)—RWCU Isolation Bypass (4/20/92).

(4) Fort Calhoun (Neb.)—Stuck-Open Relief Valve (7/3/92).

The first AEOD human factors team study during the period was of a December 8, 1991 event at Crystal River Unit 3 involving the bypassing of the high-pressure injection system (HPI) while reactor coolant system pressure (RCS pressure) was decreasing for unknown reasons. While the plant was starting up, a slow loss of RCS pressure occurred after the pressurizer spray control valve actuator failed, which left the valve partially open while the indicator showed closed. An operator incorrectly withdrew control rods in an effort to control pressure, but the reactor tripped (shut down) on low pressure. The operator bypassed automatic actuation of HPI for six minutes, without procedural or supervisory direction, or any understanding of the cause of the pressure decrease. Operators "unbypassed" the HPI 12 seconds after its automatic demand was alarmed and the HPI activated. The operators manually controlled HPI to maintain RCS pressure. A second bypass of this engineered safety feature (ESF) was in accordance with procedures, but was not conservative for maintaining an adequate subcooling margin. The operators closed the pressurizer spray isolation valve about an hour after the event started, which terminated the pressure decrease. The AEOD team found the operators had failed to follow procedures; that they had not used the annunciator response procedure; that they had not executed all the steps in an abnormal procedure that directed the closure of the pressurizer spray isolation valve, because ESF termination criteria had been met; and that the shift supervisor had made a late declaration of an unusual event.

The second human performance study involved a February 29, 1992 event at Prairie Island Unit 2 involving a loss of shutdown cooling during an RCS drain down to mid-loop. This study was performed as part of an NRC Region III team inspection. During refueling, operators at the plant were draining the reactor coolant system to mid-loop, but they did not understand how nitrogen overpressure would affect the draining process or who had responsibility to stop the drain down. New electronic level instruments remained off-scale on the high side for about two hours during draining, but that reading was not investigated in a timely manner. The operators had difficulty correcting a tygon tube level for nitrogen overpressure effects, while manually calculating reactor vessel water level. Neither procedures nor training provided sufficient direction in nitrogen overpressure control, and the operations did not recognize the significance of round-off errors in the calculation. When a residual heat removal (RHR) suction line vent and the reactor head vessel vent were opened, the electronic level suddenly went from off-scale to about 10 inches below mid-loop, and its low level alarmed. While the RCS temperature was at about 133xF, the operators stopped the RHR pump (after RHR pump low suction pressure, low motor-amps, and low flow had alarmed). The operators started a charging-pump to raise the reactor vessel water level, monitored RCS temperature, and implemented an emergency procedure when it reached 190xF. They aligned the other RHR pump to the refueling water storage tank, injected water to regain reactor vessel level, and realigned the other RHR pump for shutdown cooling. The reactor coolant system temperature reached 221xF before shutdown cooling was re-established.



NRC analyses of "human performance" in nuclear power plant operations during fiscal year 1992 included study of an event at the Crystal River facility, on the Gulf coast of Florida, north of Tampa. When reactor coolant system pressure began to drop, an operator incorrectly withdrew control rods and, when the reactor shut down, bypassed the high-pressure injection system; the actions were taken without supervisory direction or understanding of the cause of the pressure decrease. These kinds of events, in which human performance constitutes either an aggravating or a mitigating factor, are important fields of thorough study for in-depth understanding of their safety implications.

The third study concerned an April 20, 1992 event at the LaSalle Unit 2 nuclear power plant involving the bypassing of a valid automatic RWCU isolation signal, while an RWCU relief valve was open. Both RWCU containment isolation valve motors had failed several weeks earlier, during a spurious RWCU high-differential flow isolation, because of limit switch setpoint drift. Licensee management had criticized the operators for allowing the spurious isolation. After motor replacement, at 8:47 a.m. on April 20, an operator shut down the RWCU, to verify the limit switch settings, while the plant was at 20 percent power. Contrary to procedure, he closed the discharge valve before he shut down the pump, which increased system pressure to the shutoff head of the pump. The RWCU regenerative heat exchanger heated up the closed system and increased its pressure. About a minute later, RWCU high-differential flow alarmed, and the 45-second timer, preceding the automatic RWCU isolation, started. The crew bypassed the isolation logic within 30 seconds, but the 95 gallons-per-minute flow out of the RWCU continued. About three minutes later, the operators identified a flow through an RWCU regenerative heat exchanger relief valve to the reactor building equipment drain tank. The AEOD team study found that there was no direct RWCU relief valve discharge flow indication in the control room, that other instruments used for verification were located on different panels, and that the annunciator response procedure gave no guidance as to how to diagnose this event. The operators "unbypassed" the RWCU isolation logic and allowed the RWCU to automatically isolate and terminate the loss of inventory.

The fourth study was of a July 3, 1992 event at the Fort Calhoun plant involving a reactor trip with a stuck-open pressurizer code safety valve. When a non-safety-related inverter was returned to service, following repairs, its output voltage oscillated and caused a supply breaker to the electrical panel powering main turbine control circuitry to trip and the main turbine control valves to close. With the turbine control valves closed, the heat sink for the reactor coolant system (RCS) was temporarily lost, which resulted in an RCS pressure increase. The reactor shut down automatically at approximately 2,400 psi and the pressurizer relief valves, main steam safety valves, and a pressurizer code safety valve opened to reduce RCS pressure. As RCS pressure decreased, the PORVs closed at 2,350 psi and the pressurizer code safety valve closed at about 1,750 psi. RCS pressure increased to about 1,925 psi, then began to drop rapidly after a pressurizer code safety valve stuck open. The operator shut the pressurizer block valves after the pressurizer quench tank level rose. As the RCS pressure drop continued, safety injection, containment isolation, and ventilation actuations occurred. The licensee declared an alert, implemented emergency procedures and secured the four reactor coolant pumps. The stuck-open pressurizer code safety valve partially closed at approximately 1,000 psi. The plant was subsequently cooled down, using natural circulation and shutdown cooling to "cold shutdown" conditions. Although the AEOD team found several areas in the emergency operating procedures which could be improved, a number of factors contributed to the successful operator response: simulator training included loss of coolant from the RCS; emergency planning actions were practiced in simulator training sessions; and control room organization and staffing provided sufficient personnel and appropriate delegation of responsibilities.

Studies to date have identified such human performance issues as control room organization, teamwork, shift technical advisor role, task awareness, man-machine interface, administrative controls, procedure adequacy and use, training, and the feedback of operating experience information. AEOD plans to continue the program to study events in which human performance constitutes either an aggravating or a mitigating factor. Future issues found will be addressed individually as appropriate.

Analysis of Non-Reactor Operational Experience

Among AEOD responsibilities is the review and evaluation of operating experience of non-reactor programs involving the use of materials licensed by the NRC and the Agreement States, such as source material, natural and enriched uranium, and byproduct materials.

In 1992, AEOD began work on a videotape dealing with desirable practices in the use of cobalt-60 teletherapy. The videotape is to be based on data from reported medical misadministrations and will seek to identify those procedures that lead to the most common kinds of errors and misadministrations. The videotape will also illustrate practices designed to avoid errors in performing teletherapy procedures. The NRC staff is developing the videotape with support from Argonne National Laboratories. The information presented recognizes the commitment of the medical professional community and of the NRC to sound medical practices using byproduct materials. Copies of the videotape will be distributed to all NRC medical licensees and regulatory agencies for the Agreement States.

During 1992, the AEOD issued three surveys that included a review of 1991 non-reactor events and medical misadministrations reported by NRC licensees and Agreement States. These reports were published in the 1991 AEOD Annual Report (NUREG–1272, Vol 6, No. 2).

Report on 1991 Non-reactor Events. The dominant health concern associated with the use of licensed materials is the possible damage that can occur from overexposure to radiation. In 1991, 21 non-reactor events were

Table 4. AEOD Reports Issued During FY 1992

CASE AND SPECIAL STUDIES

Designation	*Subject*	*Issued*
S92–01	Not Issued	
S92–02	Safety and Safety/Relief Valve Reliability	4/92
S92–03	Review of Operational Experience With Molded Case Case Circuit Breakers in U.S. Commercial Nuclear Power Plants	6/92

ENGINEERING EVALUATIONS

Designation	*Subject*	*Issued*
E92–01	Inadequate Management Control of Snubber Surveillance	5/92
E92–02	Insights From Common-Mode Failure Events	6/92

TECHNICAL REVIEWS

Designation	*Subject*	*Issued*
T92–01	Enhanced Setpoint Testing Procedures for Pressurizer Safety Valves at Oconee and Catawba	1/92
T92–02	BWR 5 and 6 Events Applicable to Laguna Verde	1/92
T92–03	Solenoid-operated Valves and Related Equipment D A Status Report	6/92
T92–04	Recent Solenoid-Operated Valve Experiences Involving Maintenance and Testing Deficiencies	6/92
T92–05	Errors in Effective Reactor Trip Settings or Monitoring Associated with Excore Instrumentation	6/92
T92–06	Water Intrusion into Sensitive Control Room Equipment	9/92
T92–07	Inoperability of the Standby Liquid Control System During Surveillance Testing at Nine Mile Point Unit 2	9/92

reported to the NRC, in which 26 individuals received exposures that were greater than those permitted by NRC regulations. All of the individuals were associated with NRC licensees.

Most of the overexposures involved doses exceeding the regulatory limits by a small amount, although one radiographer received an extremity exposure of an estimated 200–714 rems. Also, in 1992, 16 of the 28 Agreement States that provided 1991 data to the NRC, reported the overexposure of 67 individuals. Most of these were whole body overexposures of persons in the course of their employment with Agreement State licensees.

Other types of incidents reported concerned lost, stolen or abandoned materials or leaking sources. None of the events reported to the NRC in 1991 had a significant impact on public health and safety.

Medical Misadministration Reports. The 463 misadministration reports reported during 1991 by NRC licensees involved 520 patients. Of these reports, 444 concerned diagnostic misadministrations and 19 concerned therapy misadministrations. Besides the 19 therapy misadministrations there were two diagnostic misadministrations of iodine-131, in which patients received thyroid doses of more than 1,000 rads, a dose far in excess of that diagnostically intended. Agreement States in 1991 submitted misadministration reports from 103 Agreement State licensees of six therapy and 112 diagnostic misadministrations, involving 148 patients. Besides the six therapy misadministrations, Agreement States submitted three diagnostic misadministrations of iodine-131, in which patients received thyroid doses of more than 1,000 rads, a dose far in excess of that intended and appropriate.

The number of therapy misadministrations reported during 1991 by NRC licensees was about two times the average number reported in the preceding 10 years; the number of diagnostic reports was about the same as 1990 and exceeded the prior nine-year average by about 10 percent. Despite increases in the numbers of reportable events, the rate for all types of misadministrations remained very low.

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)).

There were four abnormal occurrence (AO) reports issued in fiscal year 1992; NUREG-0090, Vol. 14, No. 3 (July–September 1991); Vol. 14, No. 4 (October–December 1991); Vol. 15, No. 1 (January–March 1992); and Vol. 15, No. 2 (April–June 1992). The four reports describe one AO at a nuclear power plant, 14 AOs at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and three AOs reported by the Agreement States. There were no AOs reported at fuel cycle facilities. These four reports also update the status of certain AOs previously reported.

The AOs covered in the reports listed above are listed in Table 4, and each is described below. Five of the events (AOs 91–8, 91–12, 92–5, 92–6, and 92–7) 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

Loss of High-Head Safety Injection Capability. A major degradation of essential safety-related equipment can be considered an abnormal occurrence, as can major deficiencies in management controls.

With the reactor shutdown for refueling, the licensee for the Harris (N.C.) nuclear power plant observed a degradation of high-head safety injection (HHSI) system piping and relief valves, initially reported, on April 3, 1991, as a condition adversely affecting the capacity of the system to perform, during the previous operating cycle, should it have been required to operate and deliver water to the reactor coolant system. The flow rate of water required to neutralize the effects of "design basis" accidents assumed in the licensing analysis would not have been attained, because a significant amount of safety injection flow would have been diverted by the piping failure and by early relief valve opening. Subsequent to initial recognition of the condition, it was analyzed under the NRC's Accident Sequence Precursor Program, and the conditional core damage probability was estimated to be 6×10^{-3} for a particular set of conditions. That estimate is based upon the unavailability of the HHSI for a year prior to discovery. The probability indicates of an event with high safety significance.

Table 5. Abnormal Occurrences Reported During FY 1992

OCCURRENCES AT NUCLEAR POWER PLANTS

AO Number	*Subject*	*NUREG–0090 Issue*
92–4	Loss of High Head Safety Injection Capability at Shearon Harris Nuclear Power Plant	Vol. 15, No. 2 September 1992

OCCURRENCES AT FUEL CYCLE FACILITIES

AO Number	*Subject*	*NUREG–0090 Issue*
		None reported in FY 1992 FY 1992

OCCURRENCES AT OTHER NRC LICENSEES
(Industrial Radiographers, Medical Institutions, Industrial Users, etc.)

AO Number	*Subject*	*NUREG–0090 Issue*
91–8	Radiation Exposures of Members of the Public from a Lost Radioactive Source near Huntsville, Texas	Vol. 14, No. 3 December 1991
91–9	Medical Diagnostic Misadministration at St. John's Mercy Medical Center in St. Louis, Missouri	Vol. 14, No. 3 December 1991
91–10	Medical Diagnostic Misadministration at I. Gonzales Martinez Oncologic Hospital in Hato Rey, Puerto Rico	Vol. 14, No. 4 March 1992
91–11	Medical Therapy Misadministration at William Beaumont Army Medical Center in El Paso, Texas	Vol. 14, No. 4 March 1992
91–12	Medical Therapy Misadministration at St. Joseph's Hospital and Medical Center in Paterson, New Jersey	Vol. 14, No. 4 March 1992
91–13	91–13Medical Therapy Misadministration at University of Pittsburgh Presbyterian-University Hospital in Pittsburgh, Pennsylvania	Vol. 14, No. 4 March 1992
91–14	Medical Therapy Misadministration at University of Wisconsin Hospital in Madison, Wisconsin	Vol. 14, No. 4 March 1992

Table 5. Abnormal Occurrences Reported During FY 1992 (continued)

AO Number	*Subject*	*NUREG–0090 Issue*
92–1	Medical Therapy Misadministration at St. John Medical Center in Tulsa, Oklahoma	Vol. 15, No. 1 July 1992
92–2	Medical Therapy Misadminstration at Harper Hospital in Detroit, Michigan	Vol. 15, No. 1 July 1992
92–3	Multiple Medical Therapy Misadministrations at G. Anthony Doener, M.D., Facility in Freehold, New Jersey	Vol. 15, No. 1 July 1992
92–5	Medical Therapy Misadministration at Beth Israel Hospital in Passaic, New Jersey	Vol. 15, No. 2 September 1992
92–6	Medical Therapy Misadministration at Hospital Metropolitano in Rio Piedras, Puerto Rico	Vol. 15, No. 2 September 1992
92–7	Medical Diagnostic Misadministration at Baystate Medical Center, Inc., in Springfield, Massachusetts	Vol. 15, No. 2 September 1992
92–8	Medical Therapy Misadministration at The Christ Hospital in Cincinnati, Ohio	Vol. 15, No. 2 September 1992

OCCURRENCES AT AGREEMENT STATE LICENSEES

AO Number	*Subject*	*NUREG–0090 Issue*
AS 91–5	Exposure of a Non–Radiation Worker at San Gabriel Valley Medical Center in San Gabriel, California	Vol. 14, No. 4 March 1992
AS 91–6	Exposures of Non–Radiation Workers at Federal Express Los Angeles Airport Hub Sort Facility in Los Angeles, California	Vol. 14, No. 4 March 1992
AS 91–7	Medical Therapy Misadministration at Northridge Hospital Medical Center in Northridge, California	Vol. 14, No. 4 March 1992

The degraded piping and relief valves are part of a subsystem that provides protection against the possibility of having to operate the charging/safety injection pumps against a reactor system pressurized above the pump discharge pressure, by providing a flow path via the relief valves to the refueling water storage tank. The subsystem, which is referred to as the alternate minimum flow (AMF) system, is designed to pass flow only when the pump discharge pressure is above the lift setpoint, which is 2,300 (±69) pounds-per-square-inch for relief valves 1CS–744 and 1CS–755. Extended operation of the pumps with no net flow can cause damage and thereby preclude availability of the safety function flow rate at a later time. The AMF system was installed as part of the original facility, prior to issuance of an operating license. The design proved deficient, in that the physical layout of the AMF piping permits air to be trapped upstream and downstream of the relief valves, when the valves are removed and reinstalled in the system. The upstream isolation valves 1CS–746 and 1CS–752 remain closed until a safety injection signal is received, which prevents water from refilling this piping. Also, piping upstream of the relief valves does not have high point vents for the removal of the trapped air. Water hammer events most likely have occurred, as a result of the presence of trapped air.

The licensee identified the damage to the AMF, as follows: (1) relief valve 1CS–744 had a broken bellows and a cracked spring, and was found to have a reduced relief setpoint of 1,100 psi; (2) relief valve 1CS–755 had a broken bellows (valve seat leakage prevented determining its setpoint with the available equipment); and (3) the piping connection upstream of 1CS–754 failed as a result of a water hammer during engineered safety feature (ESF) testing. (A small leak had previously existed in this weld, and had been scheduled for repair during the 1991 refueling outage.) The degradation of the AMF relief valves and piping was the result of a design change whose implications were not well understood at the time of installation. Subsequently inadequate root cause determinations of recurrent water hammer events are believed to have permitted the equipment to be degraded/damaged to the extent that the HHSI function would not be fulfilled.

The licensee's corrective action included revision of plant procedures to require the piping upstream of the relief valves to be refilled prior to installation of the relief valves and vented through the relief valves by hydraulic pressure to eliminate the air. The licensee has repaired the damage identified above and was evaluating the potential for water hammer downstream of the relief valves, at the close of the report period.

A special NRC inspection team was sent to the Harris site to review the event. The team determined that several water hammer events could have occurred in the AMF system over the past six years. They concluded that water hammer likely did occur during ESF testing and during ESF system actuations, when the pumps pressurized the AMF system piping which contained air. AMF system design lapses at the Harris facility included the facts that: (1) the potential for water hammer events upstream and downstream of the relief valves had not been analyzed, (2) the AMF system piping had not been analyzed for transient or water hammer loads, and (3) the potential for relief valve chatter and setpoint drift had not been analyzed. Similar damage to AMF system components has been identified at other facilities. Information Notice 92–61, "Loss of High Head Safety Injection," was issued to all licensees August 20, 1992. This event was still under NRC staff review at the close of the report period.

Abnormal Occurrences Involving Other NRC Licensees

Radioactive Source Falls Off Truck in Transit. On September 5, 1991, a licensee, Western Atlas International, reported that a two-curie cesium-137 sealed well-logging source had been lost from the licensee's vehicle en route from the its Yukon, Okla., facility to its Houston, Tex., facility. The licensee initiated a search for the source, using radiation detectors and retracing the route of the vehicle. Meanwhile, on the same day, a citizen saw the shipping container lying on the gravel shoulder of the road, about 30 feet from the southeast corner of the intersection of the Interstate 45 Exit 118 road and an underpass road near Huntsville, Tex., and notified the Huntsville Police Department. A police officer was dispatched to the scene.

The radioactive source was found approximately seven feet from its shipping container. The police officer picked up the source and is believed to have held it for about five seconds before dropping it approximately 6-to-12 inches from the container. The area was closed to the public until a member of the city's emergency management services could retrieve the source. The source was replaced in the shipping container, which was missing its shield plug. Licensee personnel subsequently placed the source in a complete shipping container.

The root causes of occurrence were: (1) a pin required to secure the shipping container shield plug was found to be missing; and (2) the bed of the truck from which the shipping container fell was a flat steel deck, with no barriers at the rear of the truck to prevent the source from falling out, other than a canvas cover held in place with four elastic straps. During transportation, several shipping containers were attached to the truck bed by locked chains which had sufficient slack to permit the container to move rapidly when the truck turned corners, thereby breaking a lock and allowing the container to fall off the back of the truck.

The licensee issued a memorandum to all its North American facilities, setting out corrective measures to be effective immediately, and subsequently, took other corrective measures to prevent these kinds of incidents.

Child Given Standard Adult Bone Scan, Dose Ten Times Intended. A bone scan diagnostic study was scheduled by the licensee, St. John's Mercy Medical Center, St. Louis, Mo., for September 9, 1991, to be administered to a 15-month-old male child with possible osteomyelitis (bone inflammation) of the ankle. The child was given an adult dose of technetium-99m MDP (Tc-99m MDP), the radioactive pharmaceutical used for a bone scan, more than 10 times the intended dosage to the child.

The licensee uses a computer system to determine the appropriate amounts of the radiopharmaceutical for a bone scan. Pediatric patients are identified on the licensee's treatment list by an asterisk, accompanied by a handwritten notation of the patient's body weight. The radiopharmacist who prepared the Tc-99m MDP for the bone scan failed to take note of the asterisk and handwritten body weight on the computer printout of scheduled diagnostic studies and prepared the standard adult dosage. The nuclear medicine technician checked the patient's name on the dose ticket accompanying the syringe but did not verify the radio-pharmaceutical and dosage, as required by hospital policy. After the administration, the technician noted the volume of the Tc-99m MDP was greater than expected, rechecked the dose ticket, and discovered the error.

The error did not negate the results of the diagnostic study and the bone scan was completed. The licensee determined that the increased risk of biological effects to the patient was not significant. The calculated radiation dose for the study was about 4.4 rads to the bone and 1.3 rads to the total body. Had the correct dosage been administered, the child would have received about 0.38 rads to the bone and 0.11 rads to the whole body. The cause of the event was human error on the part of the radiopharmacist and of the nuclear medicine technician. The employees involved were counseled, and hospital managers met with the nuclear medicine staff to stress the importance of checking one's own work, as well as the work of others, and the need to adhere to policies and procedures.

Technologist Misreads Calibrator, Administers Millicuries for Microcuries. On June 17, 1991, a patient at the I. Gonzalez Martinez Oncologic Hospital in Hato Rey, Puerto Rico, who was scheduled to receive a diagnostic dose of iodine-131 was mistakenly administered a dose of iodine-131 in the therapeutic range. The misadministration occurred when a nuclear medicine technologist misread the dose calibrator and administered 6.2 millicuries, rather than 6.2 microcuries of radiation. The technologist realized the error nine minutes after the dose was administered, when the printed dose label from the dose calibrator was checked. The physician in charge promptly administered potassium iodide solution to the patient to reduce the uptake of the radioactive iodine. The licensee estimated, based on 24-hour uptake measurements, that the dose to the thyroid was 1,612 rems. The licensee continues to monitor the patient's condition and has advised the NRC that the patient has not experienced any adverse effects because of the misadministration.

The cause was human error by the nuclear medicine technologist. The technologist did not verify the dose by reviewing the printed dose label before administering the dose. The licensee's corrective actions included disciplinary action against the technologist and imposition of a requirement that the nuclear medicine supervisor check each dose before it is administered to a patient. NRC Region II conducted an inspection of the circumstances of the event and found no violations of NRC requirements.

Radiopharmacist Wrongly Assumes Routine Dosage Prescribed. On August 30, 1991, a patient referred to the William Beaumont Army Medical Center, in El Paso, Tex., for therapeutic radioiodine treatment of Graves' disease was mistakenly given a 28.6 millicurie oral dosage of iodine-131 instead of the prescribed oral dosage of 15 millicuries of I-131. The patient's thyroid received about 31,900 rads, instead of the 16,700 rads intended.

Before administering the dosage, the radiopharmacist was informed that a radioiodine treatment for Graves' disease had been requested. He assumed that it was a 29-millicurie treatment rather than a 15-millicurie Medical Center, a 15-millicurie dose is routinely used for Graves' disease while a 29-millicurie dosage is used for thyroid disorders, such as multi-nodular toxic goiters.) He then requested a 29-millicurie dose from a commercial radiopharmacy. The actual dose received was 28.6 millicurie and was labeled as such. When the radiopharmacist logged the dosage into the computer, after it had been measured by the dose calibrator, he failed to note the intended therapy dose in the referring physician's prescription; nor did the counseling nuclear medicine physician verify the dosage to be administered with the intended dosage. The 28.6-millicurie incorrect dosage was then administered to the patient.

The licensee stated that no adverse effects on the patient were noted. The patient's condition will be appropriately monitored in the licensee's Endocrine Clinic. The event was attributed to human error, resulting from the radiopharmacist's and consulting nuclear medicine physician's inattentiveness and brief experience at the facility. The radiopharmacist and consulting nuclear medicine physician were reinstructed about the proper dose verification techniques and safeguards. Consulting physicians will be required to be familiar with a patient's case history before administering a therapeutic dose. Nuclear medicine personnel shall review the NRC videotape, "Good Practices in Preparing and Administering Radiopharmaceuticals," prepared by AEOD. NRC Region IV conducted an inspection to review the event and uncovered no violations of NRC requirements.

Patient Sent to Wrong Room and Given Radiation Therapy. On November 13, 1991, the acting radiation safety officer at St. Joseph's Hospital and Medical Center in Paterson, N.J., notified NRC Region I by letter, dated October 30, 1991, that a therapeutic misadministration involving a strontium-90 (Sr-90) beta applicator—with a nominal activity of 95.5 millicuries—had occurred on October 25, 1991. The therapeutic treatment had been administered to the wrong patient, a 52-year-old male scheduled for a simulation for external beam therapy from a linear accelerator to the head and neck. The misadministration occurred when the radiation oncology department secretary directed the patient to wait in the wrong treatment room, where he went without his chart. The patient spoke minimal English, and the radiation oncologist did not speak the patient's language. The physician asked the patient more than once which area of his body was to be treated. The patient pointed toward his head. On the basis of this exchange of information, and without benefit of a review of the patient's chart, the oncology physician then administered a Sr-90 dose to the patient's eye. The licensee estimates that about 1,000 rads were delivered in 11 seconds to the surface of the right eye. The licensee judges that no harmful effects will ensue to the patient.

An NRC medical consultant was retained to review the event. The consultant agreed with the licensee's estimate of dose to the patient's eye and concluded that the possibility of cataracts is low. The cause of the event was failure to follow the hospital protocol, which requires reviewing the patient's chart before administering treatment.

The licensee's planned corrective actions included the following: (1) patients will only be directed to the treatment area by an aide, who will hand the treatment charts directly to the physician; (2) each patient's chart will include a polaroid photograph of the patient; (3) access to the Sr-90 beta applicator storage area will be limited to the Physics Department and the Chief Technologist; (4) physics staff will accompany the physicians during all Sr-90 beta applicator treatments, and assist in determining the treatment times; (5) staff training and reinforcement of appropriate patient processing procedures and NRC requirements will be conducted.

Patient Receives Dose to Upper Back Rather than Lower Neck. On November 22, 1991, the radiation safety officer at the University of Pittsburgh Presbyterian-University Hospital, in Pittsburgh, Pa., notified NRC Region I that a therapeutic misadministration involving a Co-60 teletherapy unit had occurred at their Presbyterian University Hospital facility, on November 21, 1991. The therapeutic treatment had been administered to the wrong part of a patient's body. The technologist looked at the patient's chart but set up the wrong treatment field. The patient received 287 rads to the thoracic vertebrae (upper back), instead of the prescribed 300 rads to the cervical vertebrae (lower neck). Because the patient had previously undergone thoracic vertebrae treatment, the technologist erroneously assumed that the thoracic treatment was continuing and administered the treatment without adequately reviewing the patient's chart, which indicated the correct treatment area.

The licensee has determined that the treatment will not have any adverse effects on the patient. The patient, who is suffering from metastatic cancer of the breast, was receiving palliative radiation treatments to the spine. The misadministration was attributed to failure to follow the written prescription in the patient's chart. Corrective actions included stressing to technologists the need to carefully read patients' charts and to recognize notations of changes in the fields to be treated. When a field is completed on a patient, the administered dose is to be recorded in the patient's chart, using a different color ink.

Operator Picks Up Another Patient's Chart and Calls For Wrong Dose. A patient was undergoing a series of five treatments for a cancer of the nasal septum—using a high-dose-rate iridium-192 (Ir-192) afterloading unit—at the University of Wisconsin Hospital in Madison, Wis. The initial four treatments were completed without incident. However, before the fifth treatment on November 27, 1991, the operating physicist picked up the wrong patient's chart, located next to the device's control panel, and entered the program information into the computerized device. While the treatment was underway, a student technologist inquired about the length of time to complete the treatment. The prescribing physician and the operating physicist indicated different lengths of time. The physician, realizing there was an error, directed that the treatment be stopped immediately. Subsequently, it was discovered that the physicist had used the chart for the wrong patient and, therefore, entered incorrect treatment program information into the computer. The correct treatment information was then entered into the computer and the treatment series completed.

The erroneous treatment information positioned the Ir-192 source so that the patient's lips received radiation for about one minute. The dose calculation by the licensee indicated the patient received approximately 73 rads to the lips. According to the licensee, the radiation exposure received by the lips during a correctly administered treatment to the nasal septum, would be about 25 rads. The licensee does not expect any adverse consequences to result from the additional exposure to the patient's lips. The causes of the event were the physicist's failure to verify the identity of the patient and the physicist's assuming incorrectly that the chart at the control panel was for the patient undergoing treatment. The licensee has directed that the operating physicist check the identity of each patient before treatment, using patient photographs or other means of verification. Patient charts for treatment series will be placed in a particular location. No exceptions will be made to the training required of a user, which, in the future, will include a general section on high dose rate afterloading devices.

NRC Region III conducted an inspection on December 17, 1991, to review the event and found no violations of NRC requirements.

Ceiling Laser Beam Aimed at Wrong Angle, Missing Area Intended. On January 21, 1992, the licensee, the St. John Medical Center in Tulsa, Okla., notified NRC Region IV that on January 20, 1992, a medical misadministration was discovered that involved two therapeutic radiation doses to a part of a patient's body that was not intended to be treated. The treatments were administered on January 13 and 14, 1992, by a cobalt-60 teletherapy unit. The patient was scheduled to receive ten treatments of 300 rads each to the right scapula. After the second treatment was performed by the therapists, the oncologist reviewed the film and noticed that 80 percent of the intended area had been missed. An investigation by the licensee discovered that, in simulating the treatment to be performed on the patient, the oncologist had placed a mark on the patient's chest, in accord with the the ceiling laser position. During treatment, however, the back pointer of the teletherapy unit was positioned on this mark. As the back pointer and ceiling laser result in different angles to the cobalt-60 radiation beam, the tissue volume being treated was medial to the intended treatment site. The oncologist amended the original prescription to include two additional treatment fractions to the appropriate area, bringing the total treatment dose to that area to the intended 3,000 rads.

The patient was notified of the treatment error. The licensee stated that the misadministration should have no adverse effect on the patient. There was a breakdown in communication between the oncologist and therapist during simulation. Either the proper instruction was not given regarding positioning of the patient and which indicator to use, or the instruction was not carried out correctly. The licensee has reviewed the incident with all staff members and communicated by memo to all prescribing physicians an explanation of the different localization methods to be employed. In addition, the licensee's Quality Management Program has been amended to require review of films after the first treatment in a series; this step would not have prevented a misadministration, but could have identified the error prior to the administration of the second treatment.

An NRC inspection was conducted on February 13–14, 1992, reviewing the circumstances of the misadministration. The inspection report was forwarded to the licensee, by letter dated April 6, 1992. Although no violations of NRC requirements were identified, the NRC was concerned that the misadministration was a result of a verbal miscommunication between the oncologist and the therapist. The licensee was requested to describe corrective actions taken to prevent such miscommunications among staff members.

Left Collar Bone Area Treated Rather than Right. On March 16, 1992, the licensee, the Harper Hospital in Detroit, Mich., notified NRC Region III that, on February 24, 1992, a patient with cancer had received a therapeutic radiation dose to the incorrect side of the chest area. The patient was scheduled to receive 28 daily treatments of 180 rads each to the right collar bone area and 90 rads each to tangential areas of the right breast. The treatments began on February 12, 1992, and eight treatments were delivered as prescribed. On February 24, 1992, however, the radiation therapists erroneously treated the left collar bone area. The therapists discovered the error as they prepared to treat the two tangential areas of the left breast. The therapist repositioned the patient to treat the prescribed right breast. The treatment plan was then continued until the balance of the prescribed 28 treatments was completed. The treating physician stated that, in her judgment, the misadministration did not compromise the patient's treatment, either from an underdose to the prescribed site or from the inadvertent dose to the wrong area.

The radiation therapy technologists stated that the error occurred because they confused a leveling tatoo on the left collar bone area with the treatment tatoo on the right collar bone area. They also did not follow the procedures for confirming the accuracy of the treatment site, as specified in the licensee's Quality Management Program. Regarding the lateness in reporting the event to the NRC, the person responsible for reviewing the incident to determine if an NRC report was required used an incorrect draft of the hospital's policy manual, one which contained an error in its definition of a misadministration. The incident was not identified as a misadministration and was, therefore, not reported to the NRC, until March 16, 1992. The remaining treatments in the patient's treatment series were performed by three technologists to assure treatment accuracy. The licensee is now using different tatoos for the treatment area and for leveling.

The licensee implemented a written Quality Management Program on January 27, 1992. The new program requires that, before a treatment is administered, the details of the treatment must be checked for agreement with the prescription and plan of treatment, and the accuracy of the treatment site must also be confirmed. Therapists were provided further instruction on appropriate policies and procedures. The incomplete policy manual has been updated, and personnel have been trained on NRC misadministration reporting requirements.

A special NRC inspection was conducted, on March 26–27, 1992, to review the circumstances associated with the misadministration. On April 22, 1992, the NRC issued a Notice of Violation. Two violations of NRC requirements were cited: (1) failure to follow the instructions of the Quality Management Program, and (2) failure to report the misadministration no later than the next day following its discovery.

Multiple Therapy Miscalculations Cause Underdoses. On March 18, 1992, the consulting teletherapy physicist at the G. Anthony Doener, M.D., Facility in Freehold, N.J., informed NRC Region I of numerous therapeutic misadministrations, which occurred between July 1990 and February 28, 1992. The physicist reported that patients who had received external beam therapy from a Picker Corporation Model 6103 (C-1000) teletherapy unit may have been underdosed by about 15-to-40 percent of the intended doses. The misadministrations appeared to have resulted from an error introduced by the licensee's previous consulting teletherapy physicist into tables of treatment he worked up for various field sizes and treatment depths. The erroneous treatment times were then used by the licensee in treating patients. According to the licensee, approximately 13 patients were involved. One patient was undergoing treatment when the error was identified, on February 28, 1992, and this patient's treatment time was adjusted to correct for the error, prior to completion of treatment. The previous teletherapy physicist was contacted by telephone, on March 18, 1992 and interviewed by NRC Region I, on April 2, 1992. On both occasions, the previous teletherapy physicist stated that in late 1990 he had discovered the error in the treatment time charts he had prepared for January-through-December 1991. He stated that he had mailed corrected time charts for 1991, along with a handwritten note to the licensee, in the first week of January 1991. He did not recall what the note stated, nor did he maintain a copy of the note. He did not send the charts via certified mail, nor did he attempt to contact the licensee by telephone to inform the licensee of the error. He was not aware that a similar error had occurred in charts he provided to the licensee from July 1990-to-December 1990. The authorized user and office manager stated that they had not received corrected time charts for either 1990 or 1991.

The licensee has submitted all required documentation and reports of the misadministrations to the NRC. Based on the licensee's review of patient treatment charts, two patients have received supplemental treatment. Three of the patients are deceased, and the licensee has reported that the remaining eight patients would not be adversely affected. According to the licensee, the patients were notified of the treatment error by phone and in writing. The probable causes are (1) failure of the authorized user to identify the previous physicist's error on treatment time charts through independent verification, and (2) failure of the previous physicist to perform a secondary check of treatment times for charts prepared from July 1990-through-December 1990. Corrected treatment time charts have been provided to the licensee by the current teletherapy physicist. These charts are currently in use by the licensee. The current teletherapy physicist will provide treatment time charts to the licensee on a bimonthly basis. Treatment times will be independently verified by the current teletherapy physicist on a weekly basis, or when treatment times for a patient currently being treated are changed. The licensee has submitted a Quality Management Plan to the NRC, which is under review.

NRC inspections were conducted at the licensee's facility, on March 19 and April 22, 1992. Activities authorized by the licensee were investigated, and actions were taken in response to the NRC's Confirmatory Action Letter (CAL) were reviewed. An NRC inspector confirmed by calculation that the treatment time charts contained errors and that the errors began on the July 1990 time chart. The average error determined by the inspector was 20 percent. The inspector was unable to verify that corrected treatment time charts had been provided to the licensee for 1991. The licensee learned that the misadministrations had occurred on March 13, 1992, but did not report this misadministration to NRC Region I until March 18, 1992. Records of misadministrations were properly maintained by the licensee, as required by 10 CFR Part 35. Corrected treatment time charts provided by the current teletherapy physicist were checked by the inspector, who ascertained that they did contain accurate treatment times. The inspector reviewed treatment charts for patients currently being treated and found that corrected treatment times were being used. The inspector found that seven of eight commitments listed in the CAL had been completed at the time of the inspection. The action not completed by the licensee was to have the teletherapy physicist independently review all patient charts from the date the misadministrations began through December 1991, in order to identify all patients subjected to a misadministration. A letter from the licensee, dated May 1, 1992, stated that patient charts from July 1990-through-December 1991 have been sent to the current teletherapy physicist for review. The CAL is considered closed and authorization was given to the licensee to resume patient treatments.

These misadministrations do not appear to be the result of violations of NRC requirements. However, the inspector identified a number of apparent violations of licensed activities, including: (1) failure to perform a full calibration at intervals not to exceed one year; (2) failure to notify NRC Region I by telephone within 24 hours of a therapeutic misadministration; (3) failure of monthly spot-checks to include a determination of timer on-off error and timer linearity over the range of use; (4) failure of the licensee to require the teletherapy physicist to review teletherapy spot-check results within 15 days; (5) failure to perform an adequate accuracy test of the dose calibrator; and (6) failure to mathematically correct dose calibrator reading for a linearity error exceeding 10 percent. Items 3, 4, and 5 above are repeat violations. A Notice of Violation was issued. The licensee's Quality Management Plan has been submitted to the NRC and is being reviewed. The NRC's medical consultant is reviewing the incident.

Misadministrations Go Unreported to the NRC. During a routine inspection conducted on May 22, 1992, it was discovered that a therapeutic misadministration at the Beth Israel Hospital in Passaic, N.J., as well as an overexposure to a radiation worker's hand, had not been reported to the NRC. On August 23, 1990, a patient was scheduled to undergo an endobronchial implant procedure that involved implanting in the patient two ribbons containing a total of 35 iridium-192 seeds, representing 68.54 millicuries of radiation. One ribbon contained 20 iridium seeds and the other contained 15 iridium seeds. The medical physicist gave the attending physician the wrong end of one of the two ribbons, and the physician proceeded to insert the wrong end into the patient. The other ribbon containing 20 iridium-192 seeds was inserted correctly. The remaining extra lengths of these ribbons were cut off by the physician and given to the medical physicist. The medical physicist, assuming that these pieces of ribbons contained no radioactive material, coiled them and held them in her hand. One of these pieces contained 15 iridium-192 seeds (29.37 millicuries). The medical physicist, following completion of the procedure, discarded these pieces of ribbons into a waste basket, located in a waiting room across from the patient's room, thus creating a radiation dose rate of up to approximately 63 millirems-per-hour in an unrestricted area. This dose rate is well above the regulatory limit of two millirems in any one hour for unrestricted areas. The implant was performed at 2:30 p.m., with the intent of giving the patient a dose of 1,500 rads. The physician decided to remove the ribbons from the patient earlier than planned, because the dose rate was higher than what he normally administers. The ribbons were removed at 8:30 p.m., on August 23, 1990. Neither the medical physicist nor the hospital's radiation safety officer (RSO) was present during the removal procedure.

The following morning the medical physicist inventoried the sources removed from the patient and found that one of the ribbons contained no seeds. She immediately informed the RSO, who conducted a search for the missing radioactive material and found the two pieces of ribbon in the waste basket. The licensee determined that the dose to the hand of the medical physicist was approximately 272 rads, assuming that she held the ribbon containing iridium-192 seeds in her hand for about five minutes. The physician stated that the patient received a dose of approximately 400 rads (which was only about 50 percent of the intended dose). No make-up dose was given to the patient. Neither the therapeutic misadministration nor the overexposure to the physicist's hand was reported to the NRC. Neither the medical physicist nor the physician performed a survey of the ribbons before implanting into the patient. The licensee did not inventory the sources promptly after removal from the patient, and the licensee failed to follow established procedures involving the removal of temporary implants, in that the RSO or his designee was not present during the removal of temporary implants from the patients.

The licensee's corrective actions include a mandatory requirement that the RSO or his designee must be present during all implant and removal of radioactive materials. The management of the licensee is now more deeply involved in the radiological safety affairs and has engaged an independent agent to conduct an audit of its radiation safety program.

NRC Region I inspectors continued their inquiry into the circumstances surrounding this misadministration on June 2, 1992. Numerous apparent violations were identified. A Confirmatory Action Letter was issued on June 5, 1992, and an Enforcement Conference was held with the licensee in Region I, on June 25, 1992, to discuss the violations and the corrective actions proposed and implemented by the licensee.

Incorrect Sources Slip Out of Place to Incorrect Area. On April 8, 1992, the licensee, the Hospital Metropolitano in Rio Piedras, Puerto Rico, informed the NRC that, on March 24–25, 1992, a brachytherapy misadministration occurred involving a patient receiving a therapeutic dose to the wrong part of the body. The misadministration occurred when incorrect cesium-137 sources, devices no longer in use, were placed in a brachytherapy applicator and administered to a patient. Because all of these sources were smaller in diameter than the intended sources, they slipped from the prescribed position and irradiated normal tissue not intended to be irradiated. The applicator was loaded by a technologist who had never performed the procedure. The technologist was supervised by a technologist who had not performed the procedure in eight years, when these kinds of sources had been in common use. The incorrect sources were discovered at the midpoint of the treatment by the licensee's medical physicist, during an unplanned training session for a new physicist. The incorrect sources were promptly removed from the patient and the treatment restarted and completed as directed by the authorized user.

The licensee estimated the dose to normal tissue was approximately 400–500 rads. The licensee advised the

NRC that no adverse effects to the patient are anticipated as a result of the misadministration. The cause of the occurrence was determined to be the licensee's failure: (1) to properly train individuals handling brachytherapy sources, (2) to adequately implement a Quality Management Program (QMP), (3) to develop and implement adequate QMP procedures, and (4) to properly label the storage vault for the brachytherapy sources.

The licensee's corrective actions included revision of the QMP policies and procedures, training of all supervised individuals on brachytherapy procedures and in the revised QMP, arranging safe storage for the sources no longer in use, posting a map of the source storage vault indicating the type of source at each storage point, and improving source accountability practices.

NRC Region II reviewed the circumstances associated with the misadministration and the licensee's immediate corrective actions, during an inspection on April 10, 1992, and a follow-up inspection on April 22 and 23, 1992, which included NRC consultants in the areas of medical physics, oncology and risk assessment.

Whole Body Scan Ordered Instead of Thyroid Uptake Study. On May 20, 1992, the licensee, Baystate Medical Center, Incorporated, in Springfield, Mass., notified the NRC by telephone that a medical misadministration, involving iodine-131 (I-131) radiopharmaceuticals, had occurred at the licensee's facility the previous day, when a therapeutic dose was administered instead of the diagnostic dose intended. A nurse from a referring endocrine clinic called Baystate to make an appointment for a patient to be given a thyroid scan and I-131 uptake study. Baystate's departmental procedure for a thyroid scan and I-131 uptake is to perform the study using 16 microcuries of I-31 and 10 millicuries of technetium-99m. A whole body scan requires that approximately four millicuries of I-131 be given to the patient. Apparently, the order was entered in the patient's scheduling chart as a whole body scan rather than the thyroid scan and I-131 uptake study. Questions were raised on several occasions by licensee personnel, because the patient was diagnosed with an enlarged thyroid and, generally, an I-131 whole body scan is not indicated for this diagnosis. Also, an authorized user was not consulted to review the study and prepare a written directive prior to the administration of greater-than-30-microcuries of I-131, as required by 10 CFR 35.32. A nuclear medicine technologist administered 4.1 millicuries of I-131 for a whole body scan, without following the department's procedures for administration of I-125 or I-131. The licensee evaluated the dose to the patient's thyroid to be approximately 14,300 rads, based on an uptake of 66 percent, and the dose to the whole body to be approximately 6.25 rads.

One of the causes of the misadministration was a miscommunication between staff at both the referring endocrine clinic and Baystate. Other causes were failure of the staff at Baystate to follow regulatory procedures involving radioiodine doses greater than 30 microcuries, which require that an authorized user prepare a written directive prior to the administration. Nuclear Medicine Departmental procedures also require that, when an order for a requested study is unclear or illegible, the referring physician shall be contacted prior to the performance of the study.

The licensee's corrective actions included: (1) instruction of nuclear medicine staff in the department procedures and regulatory requirements for radioiodine studies; (2) preparation, prior to the administration, of a written directive by the Director of Endocrinology, or a designated authorized user before any iodine study using greater than 30 microcuries is performed; (3) prompt transmittal of written requests for nuclear medicine studies from the clinics to the Baystate Medical Center, Nuclear Medicine Division, to compare the request to the computer entry prior to the administration; and (4) review of this patient's progress once every six weeks for three months.

An NRC Region I inspector, on May 27 and 28, 1992, examined and appraised the circumstances associated with this misadministration. An NRC medical consultant worked with the licensee to provide a clinical assessment of the misadministration. Although the medical consultant calculated the thyroid dose to be considerably less than that estimated by the licensee's, his evaluation of the event and consequences to the patient were similar to the licensee's. They were in agreement that, because the patient was diagnosed as having Graves' disease, the ultimate therapy would be treatment with about 10 millicuries of iodine-131 (compared to the approximately four millicuries that were mistakenly administered). Therefore, the patient did not suffer health effects from the misadministration worse than those normally associated with treatment of Graves' disease.

The NRC inspection identified two apparent violations of NRC requirements: (1) failure of authorized user to prepare a written directive, and (2) failure to follow procedures. An enforcement conference was held on June 23, 1992. Enforcement action was pending at the close of the report period.

Radioactive Seeds for Prostate Implanted in Surrounding Tissue. On May 29, 1992, an implant of radiation seeds for treatment of a patient's prostate cancer was performed at the Christ Hospital in Cincinnati, Ohio. The patient had previously received radiation treatment to the prostate, by means of a linear accelerator. The implant treatment plan called for the placement of 58 seeds, each containing 0.31 millicuries of iodine-125. The seeds were to be implanted in the prostate by means of needles guided by an ultrasound image. The implanted seeds were to deliver a dose of 12,000 rads to the prostate. The 58 seeds were implanted, but a subsequent computerized tomographic scan showed that 21 seeds were implanted in tissue surrounding the prostate, rather than in the intended sites. Two seeds were eliminated with the patient's urine. The licensee calculated that the mispositioning of the seeds resulted in the patient's receiving a 5,000-rad dose to the prostate, rather than the intended 12,000-rad dose.

The principal consequence of this misadministration is the potential effect of an underdosage to the prostate. Moreover, the tissue surrounding the prostate received a greater radiation dose than intended. The prescribing physician concluded that the delivered dose from the implanted seeds and from the previous linear accelerator treatment was sufficient.

An NRC medical consultant, retained to evaluate the circumstances and response to the misadministration, noted: "Tumor recurrence is the greatest risk, and it will be monitored closely." The consultant also concluded that there was not a high probability of radiation damage to the rectum, which would be the area of principal concern.

The misadministration resulted from the difficulties in the ultrasound placement technique. The ultrasound image is difficult to interpret in guiding the placement of the seeds with the implanting needles. The prescribing physician, who is the Authorized User in the NRC license, had been trained and certified in the ultrasound guided implant technique, but had not actually performed the procedure. Physicians have recommended several improvements in the implanting technique, including more detailed pre-treatment planning, steps to improve the quality of the ultrasound image, and enhancements to the seed positioning technique.

NRC Region III conducted a special inspection on June 17–18, 1992, to review the circumstances of the misadministration and to evaluate licensee follow-up activities. No violations of NRC requirements associated with the misadministration were cited. The NRC retained a medical consultant to review the case.

Abnormal Occurrences Involving Agreement State Licensees

Non-Hospital Employee Exposed to Cesium Sources. On August 1, 1989, an intra-cavitary procedure was performed on a patient at the San Gabriel Valley Medical Center, San Gabriel, Cal. Two cesium-137 sources, 42.2 millicuries each, were loaded into devices and inserted into the patient for treatment. When the procedure was completed, the physician removed the devices and placed them in a lead container. The container was then transported to the room where the cesium storage safe was located; however, the sources were not removed from the inserts and placed in the safe as they should have been. On September 1, 1989, an employee of the medical center removed the inserts still containing the sources from the lead transport container and, thinking they were empty, placed them in an envelope to be transported to Methodist Hospital of Southern California in Arcadia, Cal., where they were intended to be used. The envelope was placed in the Radiology Department where it was picked up by an employee of a private medical group a few days later. This individual placed the envelope in his private car and drove to Methodist Hospital, which took approximately 25 minutes. When the inserts were received by Methodist Hospital, the envelope was opened immediately and the sources were discovered inside. They were placed in a lead transport container and removed to the storage safe by staff of the hospital.

San Gabriel Valley Medical Center hired a medical physicist to evaluate and determine the extent of exposures that individuals had received as the result of this incident. Extensive time and motion studies were conducted, as well as the processing of personnel monitoring devices, to determine doses received. The individual who had transported the sources from one hospital to the other was a non-radiation worker and, therefore, did not wear a personnel monitoring device. Estimates are that he received about 106 rems to his right hand and 0.168 rem whole-body exposure. All others who came in contact with the sources wore personnel monitoring devices. Estimates of their exposures were within the occupational dose limits specified by the State's Radiation Control Regulations.

The medical center was cited for causing the delivery man to receive 106 rems to his right hand. He was notified in writing by the hospital of the nature and extent of his exposure and was provided a medical review. A medical examination of his hands on the day after the exposure, and three weeks later, did not reveal any evidence of skin changes or other symptoms. Neither did his blood count show any significant abnormalities.

The apparent cause of the exposure was the failure of hospital employees to follow proper procedures for storing of sources following their use. The medical center installed a detector that will alarm if sources are not secured inside the storage safe. A refresher training course was also held for all staff, covering the proper handling of brachytherapy sources.

The State inspection agency cited the medical center licensee for six items of noncompliance.

Exposures of Non-Radiation Workers at Federal Express. On November 2, 1990, the Anaheim Memorial Hospital, Anaheim, Cal., shipped seven cesium-137 sources that had been used for a brachytherapy implant back to the supplier, Therapeutic Nuclides, Inc., Valencia, Cal. The sources were in two 50-millicurie, three 25-millicurie, and two 12-millicurie sizes. The Type 7A package used for shipment consisted of a plastic source retainer, fitted into a lead pig, which was then placed inside a metal can. The metal can was then placed inside a five-gallon metal container, surrounded on all sides by a high-density polyurethane foam. The inside container was secured with a lid and a snap ring. The outside container was secured with a lid and level lock ring.

Federal Express picked up the package on November 2, 1990, and first took it to the Fullerton, Cal., sorting facility and then to the Los Angeles Airport Hub sorting facility. At the airport facility, the package came open while descending eight feet on a 45x-angle conveyor belt. At the bottom of the conveyor, all contents of the package became separated and scattered on the conveyor belt and around the work area. A Federal Express employee noticed that the package had a radioactive label and immediately repacked the five-gallon container; however, he did not realize that the sources had fallen out. The employee reported the incident to his supervisor who called in a hazardous materials specialist to examine the container. The specialist used a survey meter and found no radiation at the surface of the drum. Rather than question why this was so, the specialist assumed that all items inside the package had been properly secured, and he allowed it to continue on to its destination.

The package arrived at Therapeutic Nuclides on Monday, November 5, 1990, but it was not opened until the following day. When the package was opened and found to be empty, the radiation safety officer for Therapeutic Nuclides immediately notified the Los Angeles County Radiation Control office (agency) and an investigation was launched. An agency inspector contacted Federal Express in an attempt to backtrack the route the package took from the time it was picked up at the hospital. She was able to focus her search on the Hub facility at the airport and discovered the sources there as soon as she entered the facility.

The inspector located all seven sources in various places throughout the facility. This inspector interviewed Federal Express personnel who came in contact or worked near where the sources were found. Those individuals who came in close contact with the sources were sent for medical evaluation and follow-up. Dose estimates were established for all workers, and all were notified of their estimated doses. Individual dose estimates for the 24 employees ranged from 10 millirems to 1,810 millirems, whole body. Three individuals who said they had touched the sources were estimated to have received extremity doses ranging from 90-to-260 rems.

The Department of Transportation (DOT) investigated whether the package of sources had been properly secured before pick-up, and concluded that if it had been sealed properly, it would not have spilled its contents.

The State agency cited the licensee for failure to report the incident and for the exposure to personnel in excess of permissible levels. The supplier of the sources, Therapeutic Nuclides, has redesigned their container to prevent this type of spill in the future.

Two Patients with the Same Name Report for Treatment. On May 3, 1991, 15 millicuries of iodine-131 intended for Patient "A" was administered in error to Patient "B," who had the same first and last names as Patient "A." The administration was made at the Northridge Hospital Medical Center, in Northridge, Cal., by the hospital's Certified Nuclear Medicine Technologist; the technologist did this without the responsible physician present, which is a violation of the California Radiation Control Regulations. Patient "B" had reported to the hospital's Outpatient Department for a pre-operational chest X-ray, instead of reporting to her doctor's private office, as expected. Patient "A" was scheduled to receive hyperthyroidism treatment that same morning. When her name was called, Patient "B" answered and signed the consent form. She asked questions of her technologist about thyroid disorders and was given answers. A dose of 15 millicuries was administered.

Later that same day, Patient "A" presented herself for the treatment. It was then that the hospital discovered that they had administered the dose to the wrong patient. Patient "B's" doctor was contacted, and he consulted with the Chief Nuclear Medicine physician. They decided to give Patient "B" 15 drops of a potassium iodine solution three times daily for three days, plus forced fluids, to reduce the uptake of the radioactive iodine. She underwent the previously scheduled surgical procedure three days after the dose was administered, without any regard for the possible exposure of surgical room staff from the patient.

The incident was reported to the wrong unit of California's Department of Health Services by the hospital, five days after it occurred. Not realizing the significance of the error, the unit did not contact Radiologic Health until May 31, 1991, 28 days after the error occurred. The Radiologic Health Unit of the Los Angeles County Health Department, the inspection agency for this licensee, began an investigation. The inspector discovered that the hospital had originally estimated the patient's thyroid dose to be much lower than it actually was. The agency retained a consultant who performed a complete workup of the patient. The patient's dose was established at 3,000 rems to the thyroid, and she was informed of this in writing by the hospital. She was placed into a treatment follow-up program. An evaluation of exposures to the surgical room staff was also made by the consultant. Their exposures were determined to be minimal, and they were also notified by the hospital. The cause of this misadministration was determined to be that the administration had been made by the hospital's Certified Nuclear Medicine Technologist without the responsible physician present.

An enforcement conference was held between members of the hospital administrative staff and representatives of the County and State Radiation Control Program staff. The hospital presented an extensive corrective action plan and explained new controls that would be implemented. Representatives of the State's Radiologic Health Branch accepted the plan, and the case was referred to the city attorney's office to determine whether to file charges.

DIAGNOSTIC EVALUATION PROGRAM

The Diagnostic Evaluation Program (DEP) provides an independent assessment of licensee performance at selected reactor facilities. The DEP 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 through the Systematic Assessment of Licensee Performance (SALP) Program, Performance Indicator (PI) Program, and the routine and special inspections performed by NRC Headquarters and Regional Offices. The program gives greater depth and dimension to information available to the decision-making of senior NRC management in the continuing process of assuring 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 consists of technical staff members from headquarters offices, regional and resident inspectors and contractors, as appropriate. Team members are specifically selected to provide an unbiased and independent assessment of plant performance. Specific emphasis and focus of the DET is dependent on areas of special interest to NRC management. The evaluation process involves observation of plant activity, in-depth technical reviews, 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.

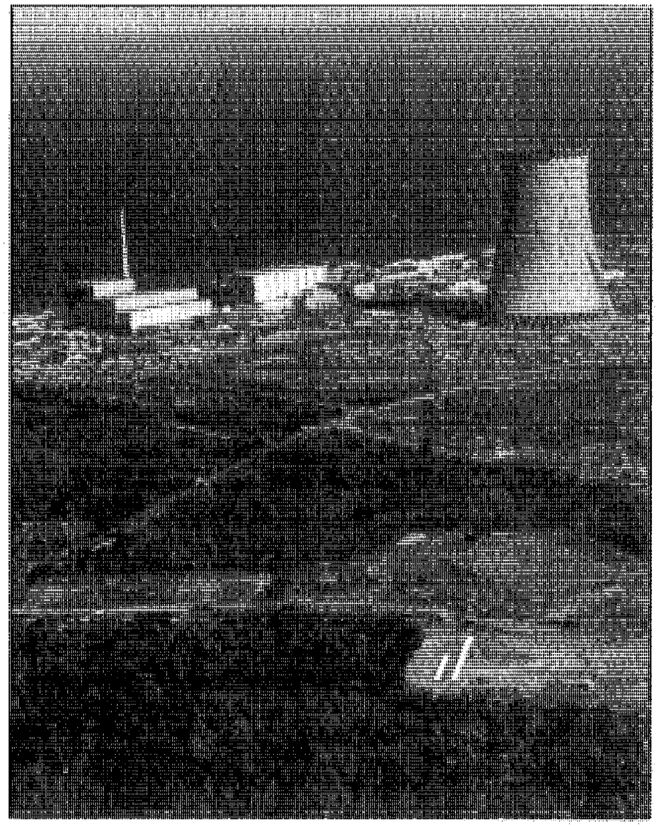
Diagnostic Evaluation of the FitzPatrick Nuclear Power Plant. In June 1991, the EDO directed that a diagnostic evaluation of the FitzPatrick (N.Y.) nuclear power plant be conducted. The evaluation was performed in September and October of 1991. The report of the team's evaluation was issued in December 1991. The recommendation that the evaluation be undertaken was based on an apparent decline in the performance of plant operations, radiological controls, and safety assessment/quality verification. A 17-member team spent approximately three weeks evaluating activities at the FitzPatrick site, and paid visits to the licensee's headquarters and engineering offices in White Plains, N.Y. The areas under scrutiny included operations and training, maintenance and testing, engineering support, and management and organization.

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. Specifically, the team found that management was not aware of many problems; that planning, scheduling, and control of work were ineffective; that plant material condition and housekeeping were poor; and, that the operator requalification training program had not, in many cases, been completed. Furthermore, root cause determinations of equipment failures were judged to be inadequate; deficiencies existed in motor-operated valves; modifications of some safety-related systems had not been adequate; and, headquarters management support and oversight was insufficient.

The DET concluded that the underlying root causes for declining performance were: failure of licensee's headquarters management to adequately and effectively plan for the operational support of the plant; inadequate management oversight and direction; lack of resource allocation and utilization; ineffective use of industry experience; inadequate standards for performance; and, ineffective communications and teamwork between the plant and headquarters.

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 reactors and non-reactor activity licensed by the NRC. The IIP's primary objective is, in general, to ensure that operational events are investigated in a systematic and technically sound manner, and more specifically, to be sure that all available information pertaining to the causes of the



Following the sudden loss, because of transformer failure, of normally available status indications and equipment at the Nine Mile Point (N.Y.) plant in August of 1991, the NRC dispatched an Augmented Inspection Team to the site. The NRC response was upgraded two days later when an Incident Inspection Team was assigned to appraise the event thoroughly for all its safety significance. The Nine Mile Point power plant, shown here, is a two-reactor station on the shores of Lake Ontario.

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 comprises two kinds of investigatory response, based on the safety significance of the operational events. The objectives of both NRC event 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), made up 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 regionally directed team complemented by headquarters personnel and, in some cases, by personnel from other Regions.

In support of the agency's incident investigation capability, an IIT Training Program was developed to provide prospective members of IITs with comprehensive guidance and methodology in conducting systematic and technically sound investigations. The training program was developed by the Office for Analysis and Evaluation of Operational Data (AEOD) following discussions 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.

Since the creation of the IIP, there have been a total of six IIT training courses. An IIT refresher training course was held from June 16 through June 18, 1992, emphasizing training on the newly revised Incident Investigation Manual and Management Directive, the overall incident investigation process, accident investigation techniques, and simulated investigations of reactor and non-reactor incidents. The course included an update on changes to the Incident Investigation Program and covered lessons learned from the previous two IITs.

Of reportable events occurring during fiscal year 1992, none was judged to have a significantly high level of safety significance to warrant an IIT response. However, 14 reactor and non-reactor events resulted in AITs being formed. The IIT investigation report regarding the transformer failure and common-mode loss of instrument power at Nine Mile Point Unit 2 (N.Y.) facility, which occurred late in fiscal year 1991, was not issued until fiscal year 1992.

IIT Investigation of the Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2. On August 13, 1991, an internal failure in the main transformer at the Nine Mile Point Unit 2 (N.Y.) nuclear power plant caused a turbine trip and reactor scram (automatic shutdowns). During the fraction of a second before automatic protective features isolated the transformer, there were depressed voltages on the transmission system and on the in-plant electrical distribution system. Although of very short duration, the degraded voltage resulted in a simultaneous common-mode loss of five "uninterruptible" power supplies (UPS) that powered important control room instrumentation and other plant equipment. Internal deficiencies—common to all five power supplies and unknown to the plant staff—had made them susceptible to failure initiated by degraded voltage.

Automatic reactor protection systems, including the reactor scram, functioned properly. All necessary engineered safety features were available and used as needed. But control rod position indication was lost, and the operators took conservative action, in accordance with procedures, as if there had been a failure to scram. The difficulty experienced by the operators, because of the loss of many normally available plant status indications and equipment, underscored the importance of the lost power supplies.

The NRC initially dispatched a seven-member AIT on August 13, 1991, to investigate the event. However, because of the apparent potential safety significance of the event—and to ensure that any generic technical and operational implications were well understood—the EDO upgraded the response to an IIT, on August 15, 1991.

The event was ultimately judged by the IIT to be of low safety significance, resulting in no actual adverse safety consequences. However, a number of salient findings were reported by the IIT. Among them were these:

- A significant aspect of the event was the challenge that it presented to the operators. The operators managed to deal with the situation, but errors were made.
- The data available from surveillance and maintenance records did not give any warning that a failure of the transformer was imminent.
- The failure of five non-safety-related UPSs was due to a common-mode design deficiency and a common-cause maintenance deficiency. Had either deficiency been corrected, the UPS loss would not have occurred.
- The difficulty that control room operators experienced with loss of rod position indication during a transient had been underestimated.
- The emergency operating procedures (EOPs) guided the operators and generally supported their decision-making process.
- The EOPs did not provide sufficient guidance for stabilizing reactor vessel pressure.
- The scram procedure at Nine Mile Point Unit 2 did not complement the EOPs for ATWS (anticipated transient without scram) conditions. This procedure did not support the operator by specifying priority actions (or immediate actions) to be taken in conjunction with the EOPs for all scrams.
- Lack of certain recovery procedures unnecessarily challenged the operators during the event.
- Licensee actions in response to previous uncontrolled condensate booster pump injections were not effective in preventing their recurrence.

The team also concluded that the NRC had not presented a clear position to the regulated industry concerning control of equipment configuration and treatment of important balance of plant equipment.

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 consists 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 manages 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 the reactor technology full course series, 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. Full course series training was available three times for the GE and Westinghouse designs and twice for the CE and B&W designs. A variety of other stand-alone reactor technology courses have been made available to support other parts of NRC staff qualification programs. Simulator refresher training was provided on numerous occasions to maintain formal qualification. Special technology training was provided in direct support of the reactor engineer intern program.

The specialized technical training curriculum consists of a number of courses in engineering support, health physics, safeguards, and inspection or examination techniques. The TTC provides specialized technical training by means of customized courses developed by the TTC staff or by TTC contractors, by coordination of slots (training opportunities) in courses that are presented by other government agencies, and by identification and promotion of appropriate commercially available courses that NRC personnel can attend as individual training opportunities. For many of the contracted courses, NRC perspectives are provided by specifically designated individuals from within the NRC staff.

During fiscal year 1992, the TTC conducted or coordinated a total of 130 courses in the reactor technology areas and 88 more in the specialized technical training areas. These courses represent a total of 275 course-weeks, 168 of which were associated with reactor technology training and 107 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 training effort represents 96,368 instructional hours, of which 55,265 were associated with reactor technology training and 41,103 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 its technical training in support of various qualification programs for NRC technical staff, the TTC also provided Reactor Concepts Courses in association with the orientation program managed by the Office of Personnel and reactor technology courses in support of the PRA Technology Transfer Program. Reactor technology courses were also presented three times for Oak Ridge National Lab personnel and twice for Illinois Department of Nuclear Safety personnel at their respective facilities. Special training sessions, including simulator demonstrations, were provided for Department of Defense personnel who support NRR. Special EOP training was provided to the Region III staff in preparation for an exercise. A special reactor concepts course was presented to National Security Agency personnel. Special U.S. Power Reactor Overview Courses were presented to a total of 16 international students.

Engineering support training in many forms was provided at various locations and times throughout the year to meet agency needs in this area. Examples of such courses include Motorized Valve Actuators, Motorized Valve Actuator Diagnostic Testing, Fire Protection for Power Plants, Welding Technology and Codes, Nondestructive Examination, Eddy Current Testing, Performance and Aging of RTDs, and Inservice Testing of Nuclear Pumps.

An interagency agreement has been concluded with the Oak Ridge Institute for Science and Education (ORISE) to arrange for technical assistance in the area of radiation protection. The agreement enables the TTC to arrange for attendance of NRC personnel at Oak Ridge's five-week Applied Health Physics Course.

A leading contractor, expert in the field of transportation and waste disposal, was made available to develop and present four Transportation of Radioactive Materials Courses for the NRC and Agreement State personnel. The courses included hands-on exercises and a field trip to the Barnwell (S.C) Waste Disposal Site.

A five-year contract with a leading company in the field of industrial radiography was established to provide Safety Aspects of Industrial Radiography Courses. The number of courses provided was increased from two to three-per-year, to meet the increased needs of NRC staff and Agreement State personnel.

Development and initial presentation of a Fuel Cycle Technology Course were completed during the fiscal year. Significant cooperation between NMSS and TTC personnel ensured that the material addressed relevant issues. The course represents the first step in the training of NRC personnel having inspection responsibility for the various stages of the nuclear fuel cycle. A related course, Criticality Safety, is under development.

The TTC staff, supported by cognizant personnel from a number of NRC Offices (NRR, NMSS, RES, and SP), conducted 10 CFR Part 20 Training Seminars in each Regional Office. The training sessions consisted of an overview of why 10 CFR Part 20 was changed, a side-by-side comparison of the old and revised rules with emphasis on the more significant changes, and workshop activities where samples, examples, calculations, and situations of interest to reactor and materials attenders were discussed. A self-study quiz on the revised 10 CFR Part 20 was distributed to the Regions, NRR, NMSS and RES, helping prepare NRC personnel prior to the 10 CFR 20 training sessions and later provide familiarity with the new 10 CFR Part 20.

Arrangements were made with the DOE Central Training Academy (CTA) for a special presentation of the Sensors Systems Course for NRC personnel. The course was presented at the CTA facility in Albuquerque, NM. Arrangements were also made with CTA to provide a Weapons Familiarity Course for NRC personnel in October 1992. Negotiations will continue during fiscal year 1993 for other courses for NRC personnel.

Techniques courses for operator licensing examiners were provided twice. The courses focused on techniques to be applied during the performance of operating and written examinations for licensed operator candidates.

Inspection techniques training was provided several times through the workshop portion of the Incident Investigation Team (IIT) Refresher Course, multiple presentations of the Inspecting for Performance Course, multiple presentations of Accident/Incident Investigation Workshops as individually requested by several Regions, and multiple presentations of Root Cause—Accident/Incident Investigation Workshops.

A Fundamentals of Inspection Course (FOIC) Work Group was established to review the existing FOIC and revise it to include regulatory impact issues, updating the materials to be consistent with existing policies and practices, and evaluating the best method for presenting the course materials. The work group is also concluding preparations for presentation of the new Fundamentals of Inspection Refresher Course.

There were two meetings of the Training Advisory Group (TAG) during the fiscal 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 TAG meetings covered such diverse training issues as: experience, perspectives, and recommendations on initial implementation of technical training programs for reactor engineer interns; the examination process; cumulative needs and projected distribution of slots for reactor technology full course series; near term implementation of post-course evaluations; establishing a Fundamentals of Inspection Course (FOIC) Work Group and FOIC Refresher Course; Reactor Safety Course status and projected development and implementation schedule; new training requests or identified needs; global training needs and projections based on current FTE counts and hiring patterns; simulator programs; courses being developed or revised; training of foreign regulatory personnel; feedback and status of selected existing courses; and advice on how to proceed in certain course or curriculum areas. The meetings resulted in successful exchanges of information and a deep overall understanding of a number of issues affecting the overall training process.

Revision of the content and structure of technical management courses for all reactor technology areas was effected during the year, in response to concerns raised by senior management. Additional topics (including electrical distribution, emergency operating procedures, and shutdown risks) have been incorporated, and the course length has been extended from three to five days.

Development of the Reactor Safety Course continued, taking new directions based on senior management comments from the Training Advisory Council. Course development is being carried out by RES through Sandia National Laboratory, with AEOD technical input. The course covers historical perspectives, accident sequences, accident progression in the reactor vessel, accident progression in the containment, and radiological releases and consequences respectively. Current projections call for a full dress rehearsal of the course at the TTC, for a specialized audience, in about February 1993. Regular presentations of the course are projected to begin at an appropriate time after that.

A Nuclear Power Workshop for U. S. Congressional Staff members was conducted at the TTC. The workshop included discussions on nuclear processes, BWR and PWR systems, simulator demonstrations of plant operations, transients, and accidents, and NRC technical training, as well as a facility tour.

An International Workshop on the Conduct of Inspections and Inspector Qualification and Training was conducted in Chattanooga, Tenn., in September 1992. The workshop was arranged by the Work Group on Inspection Procedures (WGIP) established by the Committee on Nuclear Regulatory Activities, a technical committee of the Nuclear Energy Agency. NRR was the lead office in arranging and supporting the workshop on behalf of the WGIP. Group discussions involving the various participating countries covered the major topics of conduct of inspections, training and qualification of inspectors, and shutdown risks. The conference was attended by 35 foreign and 20 U.S. representatives.

INCIDENT RESPONSE

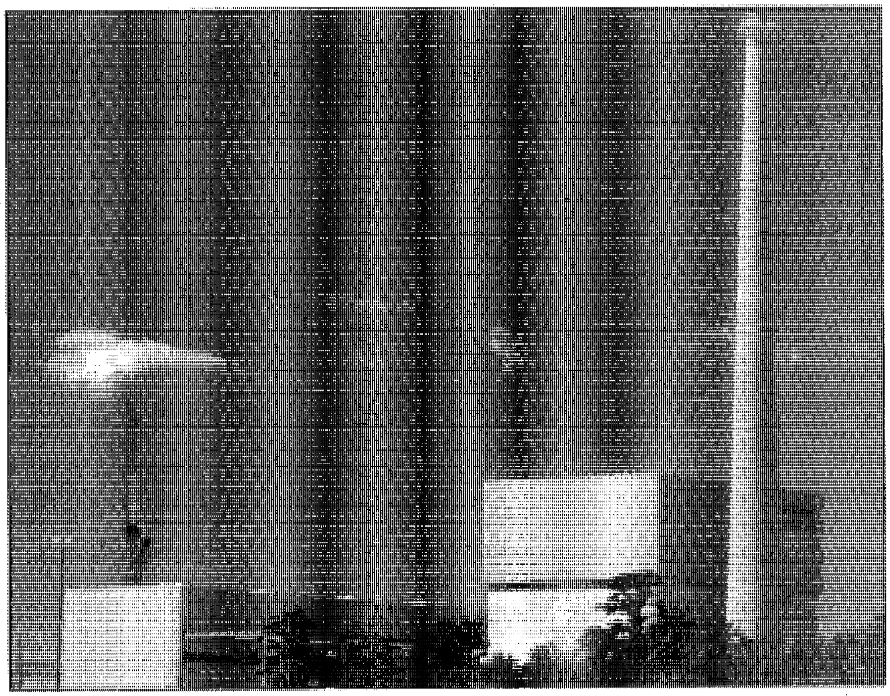
Events Analysis. The NRC maintains a 24 hour-a-day, 365 day-a-year Operations Center in Bethesda, Md. The Operations Center, which is the NRC's center for direct communications, is equipped with dedicated telephone connections with all nuclear plants and certain fuel cycle facilities, providing the NRC with the capacity to receive reports of, and to deal with, significant events occurring at any one of them. The center receives about 3,000 notifications each year from licensees, primarily nuclear power plant operators. During fiscal year 1992, there were more than 2,400 incidents reported to the Operations Center, under the NRC emergency classification system; of these, one was a "Site Area Emergency," 23 were "Alerts," and 147 were "Unusual Events."

The staff at the Operations Center evaluates telephone notifications immediately and, depending on the safety significance of the event, notifies appropriate NRC headquarters 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 a simple recording 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 of off-site organizations, whose immediate responses are defined in their own emergency plans. Each of the 3,000 events reported each year to the Operations Center by licensees is analyzed to determine whether it has any generic implications for other nuclear facilities. Event reports are screened for this purpose early 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 additional 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 System. The International Nuclear Event Scale (INES) is a tool intended to promptly and consistently communicate to the public the safety significance of reported events at nuclear installations. 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.

During fiscal 1992, the NRC defined its position for reporting under the INES and developed procedures for

A new response mode, called "Monitoring," was adopted during the report period, in order to consolidate NRC resources when plant conditions are uncertain, in anticipation of formally activating the emergency response organization into "Standby" or "Activation" status. The new mode was employed four times during the year, once for the Waterford (La.) nuclear power plant, during the Hurricane Andrew alert. The plant, shown above, is a 1,075-megawatt pressurized water reactor facility located near New Orleans.



The NRC Operations Center was activated for seven exercises during fiscal year 1992, including one at the Oyster Creek plant, shown here. This facility, which has been in operation since 1969, is on the Toms River in New Jersey, about 45 miles south of New York City.

this reporting. The NRC's participation, which will begin in 1993, will be limited, in that only events classified at the ALERT level or higher, according to the U.S. event reporting system, will be reported within the INES. Besides that proviso, only events at commercial nuclear power facilities will be considered for INES reporting. And finally, reporting under the INES will only be made after careful consideration of the facts and circumstances surrounding the event. The last step will result in an anticipated one-week delay in making an INES classification and forwarding the Event Rating Form to the IAEA.

Operations Center. A prompt incident response capability entails continuous staffing by well trained individuals with the appropriate resources to receive information, assess the information, and communicate swiftly and reliably with other involved parties. A new "Monitoring" response mode was defined during fiscal year 1992, the purpose of which is to formally consolidate NRC resources during periods of uncertain plant conditions, in anticipation of formally activating the emergency response organization in a "Standby" (or "Activation") status. The "Monitoring" mode was utilized four times during the year. Two of these activations were associated with monitoring Hurricane Andrew and the threats it posed to facilities in Regions II and IV. The hurricane resulted in Alert classifications at the Turkey Point (Fla.), Waterford (La.) and River Bend (La.) nuclear power plants.

The NRC entered the "Standby" response mode once during the year, when the Fort Calhoun (Neb.) nuclear power plant declared an Alert as the result of a small loss-of-coolant accident. However, the center's capabilities were employed in conjunction with several other events, including one reported from the former Soviet Union.

During fiscal year 1992, the Operations Center was activated for seven exercises, including one IAEA annual exercise. These exercises deal with various accident scenarios to confirm and maintain the capabilities of the agency's response personnel. During the year, all of the scenarios involving the Operations Center were concerned with reactor plant incidents. The plants for which exercises were conducted included the Oyster Creek (N.J.), Monticello (Minn.), Washington Nuclear Power 2, River Bend (La.), Quad Cities (Ia., Ill.), Vogtle (Ga.) and Arkansas Nuclear One power plants. Computer generated Nuclear Plant Analyzer accident simulations were also conducted in Regions II, III and V.

Lastly, the telecommunication capabilities of the Operations Center were regularly in use by NRC management for teleconference discussions of events of potential significance, which, as they transpired, did not prove sufficiently serious to warrant staffing of the Operations Center, and also incidents of widespread technical and media interest.

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 during an exercise or event.

New Operations Center. The new Operations Center is being designed for the second NRC office building at Two White Flint North. An Information Management System Plan (IMSP) was adopted in fiscal year 1992 to identify functional requirements inherent to information collection, processing, dissemination, storage, and display, during both normal and emergency response conditions, within the NRC Operations Center. During fiscal year 1993, the IMSP will be completed and a contract to integrate and implement the plan will be issued. The target date for the completion and testing of the new Operations Center is December 1993.

Regional Response Capability. Each Regional Office maintains its own incident response capabilities and its own Incident Response Center to support agency response during licensee events at the Alert (or higher) level, or when the NRC enters the "Monitoring" or "Standby" response modes. These Regional Office responses are based on the pre-defined event classification criteria.

A Regional Base Team and a Regional Site Team are assembled for significant events. Both Headquarters and the Region monitor licensee performance until a decision is made whether to dispatch a team to the site. An initial Site Team of 18–25 specialists, led by the Regional Administrator, can usually be at the site within eight hours from dispatch. After the Site Team has been fully briefed by licensee management and by the Resident Inspector, and is prepared to carry out its assignments, the Chairman of the NRC (or his designee) could 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 an extended NRC response is indicated, the initial Site Team will be augmented by personnel from Headquarters and/or other Regions. Procedures in this area have been revised to allow for coordination at the major response facilities identified in the Federal Radiological Emergency Response Plan and the Federal Response Plan.

Each Region has prepared its own supplement, with specific implementation details, to the NRC Incident Response Plan. 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).

Emergency Response Training. During fiscal year 1992, extensive staff response training was conducted for the 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.
- NRC Headquarters and Regional Office training on the standardized electronic mail system.
- Two-day courses in each Region on the standard technical procedures contained in the Response Technical Manual (RTM–91, NUREG/BR–0150. A one-week advanced course was held in Headquarters on these tools.
- Two courses on the operation of the Federal Radiological Monitoring and Assessment Center (FRMAC) for NRC, and other Federal, State and utility response personnel.
- Discussions of Emergency Response involving Headquarters, Regional Office, EPA, DOE, and HHS. Topics included NRC response procedures and interfaces with other response organizations.

Emergency Response Technical Tool Development. A program has been initiated by the NRC 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.

The focus of the work performed in fiscal year 1992 was on the validation and verification of software code and knowledge base. Development of test procedures and a computerized test procedure check list was started by the University of Maryland. RSAS is also involved as a test case for a formal validation and verification project, jointly sponsored by the NRC and the Electric Power Research Institute (EPRI), for expert systems. Other work completed in fiscal year 1992 included the continued development of necessary computer modeling tools and collection of the necessary plant-specific data for Boiling Water Reactors.

Technical tool development for the protective measures response function centered around the development of a revision to the Response Technical Manual (RTM–91, NUREG/BR–0150), used during an accident for:

- Classification and Core Condition assessments.
- Projection of reactor accident consequences.
- UF_6 Accident Assessments.
- Determination of protective actions for the public.
- Applying EPA/FDA guidance on re-entry and ingestion issues.
- Radiation exposure control for NRC workers.
- Obtaining DOE assistance for monitoring/medical consultation.
- Determining "Extraordinary Nuclear Occurrences."

Work also continued on the RASCAL model, a computer code used to project consequences during accidents. The present version of the code (RASCAL 2.0) has been distributed to the Regional Offices and is scheduled to be available to the public at during fiscal year 1993. Development of a Graphic Image System and of improved electronic mail capabilities also continued.

Emergency Response Data System. The Emergency Response Data System (ERDS) provides for licensee activated transmission of pre-selected plant data from on-site computers to the NRC Operations Center 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 are required to complete ERDS implementation before February 13, 1993. State governments which have expressed an interest 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, Michigan, North Carolina, Ohio, Pennsylvania, and Washington. MOUs are currently being developed with Georgia, New Jersey, New York, Oregon. South Carolina, and Tennessee.

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 for coordinating the Federal response to major emergencies and disasters. To support this, the NRC participated in the Annex Planning Leaders and Catastrophic Disaster Response group meetings for developing implementation procedures for the FRP. The NRC participates on the FEMA-chaired Federal Radiological Preparedness Coordinating Committee (FRPCC) and six subcommittees.

The NRC was also very active in the planning and preparations for the Federal Field Exercise (FFE-3), scheduled for February 1993 at the Susquehanna (Pa.) nuclear power plant. FFE-3 was designed to demonstrate the integrated response of State, local, and Federal agencies to a severe reactor accident. Because of the impact of the multiple natural disasters that occurred in the fall of 1992, (Hurricanes Andrew and Iniki, Typhoon Omar, et al.), and the workload and resource burdens placed on the staff of FEMA, the exercise was canceled. Activities to ensure that all information, interfaces, and lessons are captured and available for future use is being pursued.

During 1992, improvements continued to be made among Federal agencies concerning the coordination necessary during a reactor accident. These were mainly the result of:

- Sponsoring of FRMAC courses (See Emergency Response Training Section).
- Revising all functional procedures to ensure compatibility between the FRP and other Federal agencies and states.
- Developing a course to introduce NRC personnel to the provisions and integration of the FRP with the Federal Radiological Emergency Response Plan (FRERP).
- 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.

State Outreach. During the year, the NRC continued its State Outreach program, designed to increase and improve the NRC's interaction with States during exercises and events. The program emphasizes increased frequency of exercise participation, attempting to exercise with each State on a three-year cycle. The NRC is also working through the Office of State Programs to expand participation in meetings, workshops, and other vehicles that help describe the available NRC assessment tools, response capabilities, and accident assessment training courses. As part of this effort, the NRC participated in four limited exercises with eight states. That brings to 20 the number of States with which the NRC has participated in exercises over the past two years and is consistent with the overall goals of the program.

Along these lines, the NRC worked directly with the States of New Jersey, Vermont, Maine, Iowa, Minnesota, Ohio and Pennsylvania on the Outreach Program, the Federal Response Plan, incident response responsibilities, and the Price Anderson Act. State Liaison Officers in Regions I and III were also briefed on these subjects. In addition, NUREG- 1442/FEMA-REP 17, Revision 1, The Emergency Response Resources Guide, and NUREG-1457, Resources Available for Nuclear Power Plant Emergencies Under the Price Anderson and Stafford Acts, were published to provide guidance to State and local governments on these respective subjects.

Finally, two-day workshops were conducted for State and local response personnel in each of the Regional Offices. This training was on the technical procedures (set forth in the Response Technical Manual) used by the NRC to assess accidents during its response.

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; see Chapter 10). Thus, OI is concerned with the activities of NRC licensees, applicants for licenses, licensee contractors and vendors.

In fiscal year 1992, the process by which suspected wrongdoing matters are referred to OI was revised. The new procedure resulted in OI's becoming involved in potential wrongdoing matters at an earlier stage and increased the number of cases opened significantly. OI opened 196 cases and closed 119 cases in fiscal year 1992. Nineteen cases were referred to the Department of Justice (DOJ) for prosecutorial review. During fiscal year 1992, OI was involved, at any given time, with three-to-six Federal grand juries dealing with criminal referrals. In the fiscal year, referrals by OI to DOJ resulted in eight indictments and two convictions.

During fiscal year 1992, 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 in a subgroup composed of Federal investigative personnel. As noted below, two investigations which resulted in DOJ indictments were jointly conducted with other Federal investigative units. OI also participated in a joint DOJ Task Force in the Philadelphia area in 1992, focusing upon the sale of counterfeit and substandard fasteners to various Federal agencies.

Two cases involving sales of misrepresented or counterfeit valves and circuit breakers were referred to DOJ by OI:Region V during the report period. OI:Region IV referred three cases involving the sale of substandard fasteners. In addition, action was taken by the Justice Department on five cases referred in earlier periods, as follows:

On January 17, 1992, Clifford Ashley and CMA International were sentenced in U.S. District Court, Seattle, Wash. CMA International was placed on probation for five years, ordered to make $213,825.03 in restitution, and given a special assessment of $200. Ashley was ordered to make restitution in the amount of $213,825.03, sentenced to three years' imprisonment, and also assessed $50. The sentencing was the result of an investigation of the sale of counterfeit valves to the Diablo Canyon (Cal.) nuclear power plant.

On April 2, 1992, Thomas Carrol, radiation safety officer, Finlay Testing Laboratory, was indicted in U.S. District Court, Honolulu, Haw., on two counts of conspiracy to defraud the U.S. Government (NRC), regarding the transportation of nuclear material on passenger aircraft and the use of nuclear materials. On April 3, 1992, Carrol pled guilty to both counts and admitted that he conspired with Gordon Finlay, president of Finlay Testing Labs. Sentencing of Carrol will follow completion of the Finlay prosecution.

On September 16, 1992, Gordon Finlay and Finlay Testing Laboratories, Inc., were indicted in U.S. District Court, Honolulu, Haw. on 24 counts, including conspiracy to defraud the U.S. Government (NRC), false statements, mail and wire fraud, and transportation of radioactive material aboard passenger aircraft. Trial has been set for April 20, 1993.

On May 21, 1992, Hector Contreras and HLC Electric Supply, Inc., were indicted in U.S. District Court, Los Angeles, Cal., on 28 counts of fraud, conspiracy and trafficking in counterfeit goods, involving the sale of counterfeit circuit breakers to several nuclear power plants. The indictment was a result of a joint OI, Defense Criminal Investigative Service investigation. Trial has been set for January 18, 1993.

On August 14, 1992, Ricardo Contreras and Molded Case Circuit Breaker, Inc., were indicted in U.S. District Court, Boise, Idaho, on one count of wire fraud, involving the sale of counterfeit circuit breakers to a Department of Energy (DOE) facility. The indictment was a result of a joint DOE Inspector General and OI:Region V investigation.

Another DOJ action involved the Stanford Mining Company (SMC), an NRC materials licensee who operated several coal mines near Indiana, Pa. An investigation by OI:Region I resulted in the indictment of Travis Miller, president of SMC, on May 6, 1992. A Federal Grand Jury in Pittsburgh, Pa., charged Miller, in a four-count indictment, with improperly transferring and disposing of three nuclear weigh scales and making false statements to the NRC concerning the whereabouts of the scales. OI:Region I was able to recover two of the three missing scales. On September 3, 1992, Miller pled guilty to one count of 18 U.S.C. 1001 (False Statements). Miller could be sentenced to five years in prison and fined $10,000, or both. Miller's sentencing is tentatively scheduled for late December 1992. The corporation, SMC, pled guilty and paid a $30,000 fine on similar charges in October 1991.

Enforcement Actions/Civil Penalties

In May 1991, OI:Region I initiated an investigation which substantiated that the Deputy Director for Radiation Control (DDRC), Georgetown University Medical Center, willfully failed to complete the annual review of the radiation safety program for 1990. It was also substantiated that additional NRC regulations and license conditions were violated, although it was not concluded that they were willful violations. The DDRC resigned effective September 1, 1991. A Notice of Violation was issued and a civil penalty of $3,750 was imposed on the licensee.

In April 1992, OI:Region I initiated an investigation to determine whether four radiographers employed by the Grinnell Corporation, Cranston, R.I. (an Agreement State), at the direction or with the knowledge of higher management intentionally performed radiography in Massachusetts on two occasions without complying with NRC reciprocity provisions. Although the radiographers denied any intent to violate NRC regulations, reciprocity violations were committed on two occasions in February 1992. Based on the investigation, the corporation was assessed a $25,000 civil penalty.

An investigation by OI:Region II at Georgia Power Company's (GPC) Vogtle nuclear power plant, Waynesboro, Ga., resulted in the imposition of a $100,000 fine against GPC. The investigation disclosed that senior reactor operators and the assistant general manager for operations at Vogtle deliberately ordered the opening of valves that were known to them to be required to be closed with the plant in its existing status.

As a result of an OI:Region II investigation, the Tennessee Valley Authority (TVA) was fined $75,000 for improper activities at the Sequoyah nuclear power plant, Soddy-Daisy, Tenn. The fine was based, in part, upon the investigative finding that TVA submitted a letter to NRC that contained inaccurate and incomplete information regarding a comparison of cabling at the Sequoyah and Watts Bar (Tenn.) nuclear plants.

OI:Region III investigated several allegations against the owner of Piping Specialists, Inc., a Kansas City radiography firm. Among the allegations were charges that the owner participated in the falsification of records, deliberately lied to the NRC, refused to provide certain of his employees with safety dosimetry devices, and allowed unauthorized individuals to conduct radiography. All of these allegations were substantiated by the investigation, and the NRC revoked the firm's license. A subsequent hearing held before a three-judge Atomic Safety and Licensing Board panel resulted in a unanimous decision to uphold the NRC's revocation action.

OI:Region III investigated an allegation that personnel within the Division of Construction Inspection, City of Columbus, Ohio, were routinely performing unauthorized repairs on moisture density gauges used by their department. The investigation determined that two former radiation safety officers had been knowingly exposing the source rods in their gauges. They had done this during unauthorized cleaning and maintenance, in deliberate violation of the license. The practice had been going on since 1982.

An OI:Region IV investigation determined that Panhandle N.D.T. & Inspection, Inc., deliberately failed to file required forms with the NRC before conducting radiography in NRC's jurisdiction under reciprocity, and also intentionally failed to use ratemeters while performing radiography in NRC's jurisdiction. Based on the OI investigation, the NRC issued an Order on May 18, 1992, suspending Panhandle's general license to conduct radiographic activities in non-Agreement States where NRC maintains jurisdiction.

An OI:Region IV investigation determined that Midwest Industrial X-Ray, Inc., personnel deliberately failed to use ratemeters while performing radiography in NRC's jurisdiction. The investigation further concluded that in 1991, Midwest engaged in licensed activities in NRC's jurisdiction for more than the 180 days a year allowed under NRC regulations. Based on the OI investigation, on September 1, 1992, the NRC issued a Notice of Violation and imposed a civil penalty of $8,000 for violations of NRC requirements by Midwest. Midwest paid the $8,000 fine by check dated September 28, 1992.

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 Violation, 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 causes, 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 1992, the NRC conducted 174 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. Although this program was in place for only a small portion of Fiscal Year 1992, six conferences were open to the public during that year.

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.

In addition to civil penalties, orders may be used to modify, suspend, or revoke licenses. Orders that modify a license may require addition 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.

On several occasions, from April to September 1991, at the request of Region IV, OI assisted inspection efforts at the South Texas Project (STP). OI reviewed numerous internal STP investigative reports regarding possible false documents and/or management integrity issues. As a result of this joint investigative/inspection effort, the NRC issued a Notice of Violation on December 12, 1991, and imposed a civil penalty of $50,000 for several willful violations of NRC requirements that occurred at STP between October 1990 and January 1991. STP paid the $50,000 fine on January 10, 1992.

An OI:Region IV investigation was conducted regarding the loss of a sealed source from a Western Atlas International truck during transport from Oklahoma to Texas. On December 20, 1991, based in part on the OI investigation, the NRC issued a Notice of Violation and imposed a civil penalty of $10,000 for Western's failure to transport radioactive material properly and the resultant loss of a sealed source from a vehicle. Western paid the $10,000 fine on June 11, 1992.

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 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 all other licensees.

Appendix 6 provides a listing and brief summary of the civil penalties proposed, imposed, and/or paid during Fiscal Year 1992; and a listing and brief summary of the 17 orders issued during Fiscal Year 1992. Recognizing that enforcement actions can sometimes span several fiscal years, there were a total of 110 civil penalties acted upon in fiscal year 1992. Of these, 102 cases were proposed, for a total of $4,645,975; 13 were imposed, for a total of $185,455; and 94 were paid (including those for which payments are being made over time), for a total of $3,953,580. In addition, 96 cases were issued as escalated enforcement actions without a civil penalty, for reasons unique to each case.

The NRC Enforcement Policy was modified on February 18, 1992. The modifications: (1) reorganized and expanded the existing organizational structure of the policy itself; (2) reflected the enforcement functions of the two Deputy Executive Directors for Operations and clarified the enforcement functions of the Office of Enforcement and of all offices conducting inspection activities; (3) provided additional guidance and expanded existing guidance regarding severity level categorization; (4) proposed base civil penalties for violations meeting the civil penalty criteria at Severity Level IV and eliminated civil penalties for violations at Severity Level V; (5) reflected modifications to the civil penalty adjustment factors used in developing civil penalties, including additional guidance on when such factors need not be considered; (6) established minimum civil penalties for certain overexposures, loss of licensed material and release of licensed material; (7) provided for expanded use of discretion to either increase or decrease the amount of a proposed civil penalty arrived at after application of the normal guidance, i.e., civil penalty adjustment factors, to ensure that the proposed penalty reflects the appropriate level of concern and conveys the appropriate message; (8) provided for expanded use of discretion to encourage licensee identification and correction of violations. including certain Severity Level II violations and willful violations committed by low level employees, as well as for not issuing enforcement actions for certain licensee-identified and corrected violations involving old design, engineering, or installation failures; (9) provided additional examples in Supplement I (Reactor Operations), Supplement VI (Fuel Cycle and Materials Operations), and Supplement VIII (Emergency Preparedness); (10) substantially revised the examples in Supplement III (Safeguards) to better focus on safety significance; and (11) reflected editorial and other minor changes. These modifications were published in the *Federal Register* on February 18, 1992, 57 FR 5791.

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 regulation of nuclear reactor facilities (covered in Chapters 2 and 3). The NRC conducts materials regulation under three broad programs: fuel cycle and material safety, discussed in this chapter; materials and facilities safeguards, discussed in Chapter 5; and waste management activities, discussed in Chapter 6.

Activities covered in this chapter include licensing, inspection, and other regulatory actions concerned with: (1) the conversion of uranium ore concentrates (after mining and milling) to uranium hexafluoride; (2) enrichment of uranium hexafluoride; (3) conversion of enriched uranium hexafluoride to ceramic uranium dioxide pellets and their subsequent fabrication into light water reactor fuel; (4) production of naval reactor fuel; (5) storage of spent reactor fuel; and (6) production and use of reactor-produced radioisotopes (byproduct material).

Nuclear materials regulation during fiscal year 1992 included:

- Seventy licensing actions involving fuel cycle plants, facilities, and spent fuel issues.
- Approximately 2,700 fuel facility and materials licensee inspections.
- Approximately 6,100 licensing actions on applications for new byproduct materials licenses, amendments and renewals of existing licenses, and reviews of sealed sources and devices.

FUEL CYCLE LICENSING AND INSPECTION

Fuel Cycle Licensing Activities

By the end of fiscal year 1992, the NRC had completed 70 fuel cycle licensing actions. Table 1 shows licensing actions by category.

Sequoyah Fuels Corporation. On October 3, 1991, the NRC issued an Order Modifying License and a Demand for Information to the Sequoyah Fuels Corporation (SFC). The NRC ordered that a manager be removed from supervisory or managerial responsibilities over NRC-regulated activities for the period of one year. The NRC further ordered SFC not to conduct production operations related to NRC-regulated activities for one year. And the NRC further ordered SFC not to conduct production operations until SFC performed certain tasks. SFC was required to submit to the NRC—and obtain NRC approval of—a plan and schedule for outside consultants to review the adequacy of the licensee's health and safety and environmental programs. The licensee was also required to furnish information demonstrating why its license should not be modified to prohibit certain managers from serving in any capacity involving NRC-regulated activities.

In response to the Demand/Order, SFC removed all individuals cited in the Demand/Order, and installed a new management team. SFC also submitted a plan and schedule to review their health and safety and environmental procedures, the review to be performed by outside consultants. After SFC satisfied all the pre-restart conditions, the NRC conducted two team inspections, in December 1991 and January 1992, to evaluate SFC's readiness for restart. A Commission meeting, held in March 1992 to discuss the restart issue, included the licensee, Native Americans for a Clean Environment, and the Cherokee Nation. After careful evaluation and review, the NRC authorized a phased restart, on April 16, 1992, and a final phase for the resumption of uranium hexafluoride production, in June 1992. SFC is continuing to develop and implement performance improvement plans.

Guidance On Fire Protection for Fuel Cycle Facilities. On August 10, 1992, the NRC published, in the *Federal Register* (57 FR 35607–13), guidance on Fire Protection for Fuel Cycle Facilities to applicants and licensees, in the form of a Technical Position (TP). This document gives guidance in the preparation of applications for licenses to conduct operations at fuel cycle facilities. A document of the same title was published, for public comment, in March 1989. After consideration of the comments received and the experience gained in promulgating the earlier draft TP, the NRC revised the document and reissued it in final form.

Table 1. Fuel Cycle Licensing Actions Completed in FY 1992

Category	*No. of Actions*
Uranium Fuel Fabrication	40
Uranium Hexafluoride Production	5
Critical Mass Materials	5
Fuel Research & Development, & Pilot Plants	6
Other Source Materials	2
Waste Processing & Decommissioning	6
Interim Spent Fuel Storage	6
Total:	70

Among the pre-requisites to approval of applications for licenses, as prescribed in 10 CFR Parts 40 and 70, is a determination by the NRC that the applicant's proposed equipment, facilities and procedures are adequate to protect health and minimize danger to life or property. It has been found desirable to supplement this broad requirement with more detailed guidance, in order to inform the applicant of what NRC reviewers consider adequate and to bring about a uniform level of safety in licensed facilities. The TP on fire protection is one of the documents published pursuant to this objective. Previously published TPs cover the areas of Management Controls/Quality Assurance and Requirements for Operations and Chemical Safety.

Nuclear fuel production facilities vary greatly in terms of raw materials handled, processes performed, and end-products produced. Associated fire hazards are similarly varied, and an adequate level of fire protection can be achieved in more than one way. For this reason, the TP is not prescriptive, and the licensee is not precluded from designing and implementing a fire protection program that provides a level of protection equal to or higher than would be achieved by the measures suggested in the TP. Experience gained by the staff in administering the regulatory program and, more specifically, in inspecting facilities for fire safety, indicates that most licensees have little problem meeting the standards detailed in the TP. The standards derive from sound industry practice and are widely employed. A few licensees are in the process of making adjustments in limited areas.

Uranium Enrichment. In January 1990, the Department of Energy (DOE) submitted a plan to Congress for the demonstration and deployment of Atomic Vapor Laser Isotope Separation (AVLIS) uranium enrichment technology. The plan called for submittal of an application for a production facility license 15 months after the demonstration phase. Because of budget cuts in the program, the plan has not proceeded as scheduled. However, the National Energy Policy Act of 1992 created the United States Enrichment Corporation and requires that it determine, within two years, whether AVLIS may be deployed. The NRC staff continues to review the program (on a low-priority basis), to interact with the DOE and/or the corporation, and to familiarize itself with the AVLIS technology and the unique issues related to it.

In January 1991, Louisiana Energy Services 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 would be located in Claiborne Parish, near Homer, La., and would have a capacity of 1.5 million kilograms of separative work-per-year, about 15 percent of the annual enrichment requirements of U.S. nuclear utilities.

In fiscal year 1992, NRC staff continued review of the license application. A public meeting was held in Homer, La., as part of the process leading to preparation of the required environmental impact statement (EIS). The draft EIS and safety evaluation report are scheduled for issuance in fiscal year 1993.

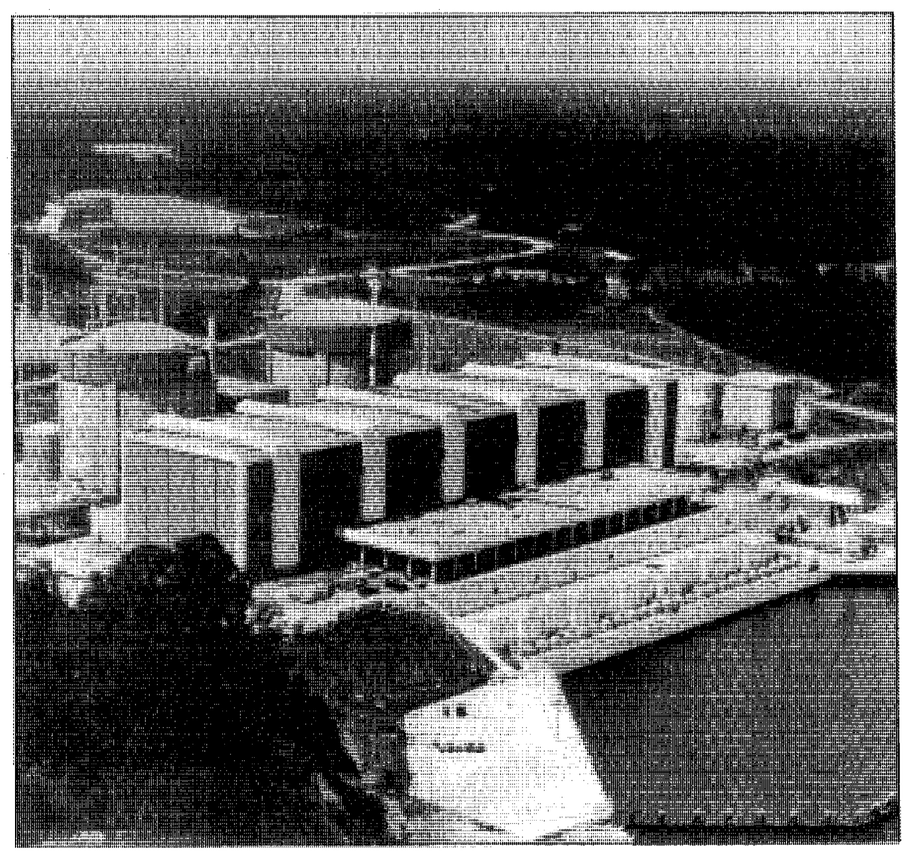
In October 1992, the National Energy Policy Act of 1992 was approved, creating the United States Enrichment Corporation that would own and operate the existing gaseous diffusion (GD) plants of the DOE and any Atomic Vapor Laser Isotope Separation (AVLIS) plant, as noted above. The legislation requires that these plants be certified or, in the case of AVLIS, licensed by the NRC.

West Valley Demonstration Project Oversight. Throughout fiscal year 1992, the NRC staff continued safety oversight at DOE's West Valley Demonstration Project (WVDP), near Buffalo, N.Y. The purpose of the WVDP is to demonstrate the solidification and preparation of high-level radioactive waste from spent nuclear fuel reprocessing, for disposal in a Federal repository. Removal of dissolved cesium from the supernatant (liquid) portion of the waste, begun 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 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 submitted by the DOE. The DOE normally submits a separate Safety Analysis Report for each segment of the waste process, including solidification in glass. The staff reviews each submittal and issues a corresponding Safety Evaluation Report, giving its conclusions regarding the public health and safety implications of that process segment.

In fiscal year 1992, the staff finished its assessment of the safety of the West Valley sludge mobilization and washing system. The DOE began operations, using this system, in July 1992, and will continue the process through 1993. As an agency cooperating in the preparation of an EIS for site decommissioning, the NRC also began 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.

Barnwell Nuclear Fuel Plant. Responding to a licensee request, the NRC found insufficient cause to extend the construction completion date for the Barnwell (S.C.) nuclear fuel plant. The extension had been pursued by the licensee, Allied General Nuclear Services; the finding denying the request for extension was published in the *Federal Register* (57 FR 43989) on September 23, 1992. A separate materials license, issued by the State of South



Interim storage of spent nuclear fuel at the reactor site has enabled utilities to continue operations until a permanent repository for nuclear waste is available. The on-site storage takes place in spent fuel pools, whose capacity can be expanded through reconfiguration of the racks holding the spent fuel assemblies, and also through dry storage in casks or concrete vaults. The NRC gives a thorough safety review to any utility's plans for an Independent Spent Fuel Storage Installation, such as that at the Calvert Cliffs (Md.) nuclear power plant, shown here.

Carolina, will continue in effect. Small amounts of radioactive materials, primarily in the form of natural uranium, remain under control in the facility, as contaminated equipment. The Barnwell plant was originally intended to be used for the reprocessing of spent nuclear fuel from light water nuclear power plants. Reprocessing plans were dropped when President Carter requested that the Commission terminate all activities supportive of a widespread commercial use of plutonium, in order to help prevent the proliferation of potential weapons materials.

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 is available. Utilities are continuing to develop plans for increasing their storage capacity, as they approach the limits of on-site storage pools. Where possible, utilities "re-rack" spent fuel pools, a measure that has successfully expanded 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 1992, the NRC initiated a rulemaking to amend 10 CFR Part 72 of its regulations, to add 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. When these casks are approved, there will be a total of six approved models that any utility may use at its reactor site without a site-specific license. However, the reactor licensee must ensure that there are no unreviewed safety questions, and that no changes to the reactor operating license are needed, before using the casks. Reactor licensees must also conform to conditions set forth in the cask's Certificate of Compliance and develop operating procedures.

In August 1992, the NRC completed its Environmental Assessment of the proposed Independent Spent Fuel Storage Installation (ISFSI) that Northern States Power Company (NSP) plans to build at their Prairie Island site in Minnesota. In September, NSP started to prepare the site for construction, to be completed in the summer of 1993. The utility plans to use the TN-40 cask, built by Transnuclear Incorporated. The ISFSI will provide storage capacity for spent fuel accumulated at Prairie Island until the licenses for Units 1 and 2 expire in 2013 and 2014, respectively.

The NRC staff has completed the safety review for an ISFSI at the Calvert Cliffs (Md.) nuclear power plant. The Commission is expected to approve the license soon, and the utility is planning to load spent fuel by the end of 1992.

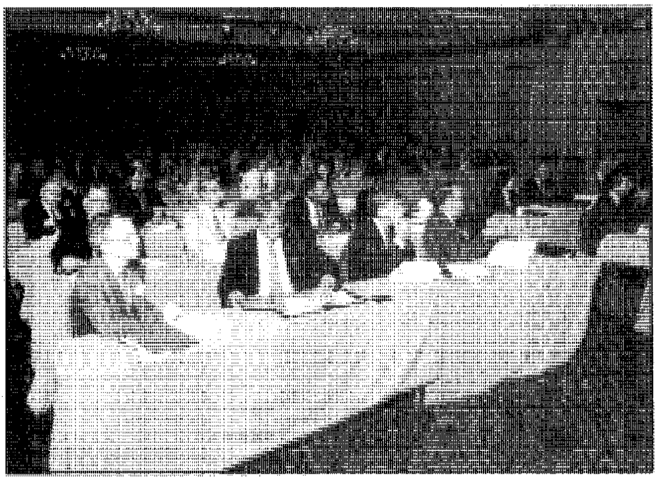
Monitored Retrievable Storage (MRS). The NRC met with the DOE several times during the report period to discuss plans and schedules for a Monitored Retrievable Storage (MRS) facility. The DOE has received applications from 21 Indian Tribes and county governments for "Phase I" grants of $100,000 to study the possibility of being host to the MRS. At the end of the report period, six applications were active, six were under review, and nine were inactive. One applicant, the Mescalero Apache Tribe of New Mexico, had received a $200,000 "Phase IIa" grant to continue exploring the possibility of hosting the MRS facility. In connection with these DOE grants, the NRC met with a number of Indian tribes and county officials from various parts of the country to explain the NRC's role in licensing an MRS. The NRC also provided comment to the DOE on two revisions to an annotated outline for the MRS Safety Analysis Report, which will form the basis for DOE's MRS license application.

Fuel Cycle Inspection Activities

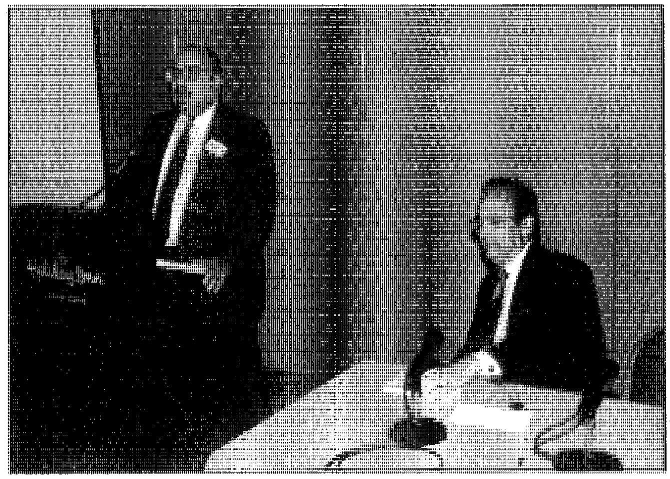
Operational Safety Team Assessments and Expanded Inspections. The NMSS staff continues to conduct operational safety team assessments at major fuel cycle and materials facilities. The team assessments are expanded inspections, with emphasis on all relevant aspects of safety, management organization and controls, chemical process safety, environmental protection, operations, transportation, fire protection, radiation safety, emergency preparedness, safety-related instrumentation and maintenance, and criticality safety. The assessment teams often include representatives from the Regions, NRC Headquarters, and other Federal agencies—such as the Occupational Safety and Health Administration and the Environmental Protection Agency. Since the safety team effort began in 1986, the staff has conducted approximately 40 such assessments. In fiscal year 1992, the NRC conducted operational team assessments at two fuel fabrication facilities—Siemens, at Richland, Wash., and Westinghouse, at Columbia, S.C.

Materials Regulatory Review Task Force. As described in the *1991 NRC Annual Report*, pp. 97 and 98, an accident involving potential criticality (chain reaction), which took place at the General Electric (GE) Nuclear Fuel and Component Manufacturing Facility, near Wilmington N.C., led to the formation of a task force to re-examine the regulatory process for large material facilities and to identify any generic weaknesses that could have contributed to the incident. The task force issued its final report, "Proposed Method for Regulating Major Materials Licensees," NUREG-1324, in February 1992.

The report contained approximately 30 recommendations related to the regulations, licensing, inspection, regulatory guidance, and training. Approximately 90 public comments were received from 10 respondents, including public and industry groups, a State government, licensees and applicants. The industry members responding



The NRC held its fourth Fuel Cycle Workshop, in the fall of 1992, in Bethesda, Md., with presentations by NRC staff, licensees, representatives of the Department of Energy, and others on such topics as integrated safety analysis, open enforcement conferences, lessons learned from past fuel cycle facility incidents, and decommissioning. Shown at left are attenders of the workshop listening to opening remarks from



Robert Bernero, Director of the NRC's Office of Nuclear Material Safety and Safeguards, shown at right at the podium. NRC Chairman Ivan Selin, who also offered opening remarks, is seated to Mr. Bernero's left. The event provided an opportunity for licensees to exchange views among themselves and with NRC staff.

generally agreed with most of the report's recommendations, although there were some concerns about implementation, particularly with regard to costs.

At the close of the report period, the staff was preparing to brief the Commission on an Action Plan drafted to assign priorities, resources, and schedule estimates to the various recommendations.

Regulatory Impact Survey. In May 1992, the staff submitted a plan to the Commission for a regulatory impact survey of fuel facility and materials licensees (SECY–92–166). The plan proposed a three-phased approach designed 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 by NRC requirements and the level of safety achieved. Phase I included a pilot series of nine on-site interviews at selected fuel cycle and major materials facilities. Seven of the nine interviews were completed in August and September 1992, and the others completed in October.

The staff expects to complete its analysis of the Phase I effort in fiscal year 1993, and then to recommend whether or not to proceed with Phases II and III, which would entail a mailed questionnaire to several thousand licensees, and additional site interviews.

Fuel Cycle Workshop. In a continuing effort to improve communication with its licensees, the NRC held its fourth Fuel Cycle Workshop, in September 1992, in Bethesda, Md. The 2+–day workshop emphasized integrated safety analysis. Presentations by the NRC staff, licensees, the DOE, and interest groups were given on such topics as integrated safety analysis ("Proposed Method for Regulating Major Materials Licensees" (NUREG–1324)) and planned staff actions, open enforcement conferences, lessons learned from past fuel cycle facility incidents, and decommissioning. The presentations were followed by open discussions and questions from the audience. The workshop provided a forum for licensees to exchange views among themselves and with NRC staff, to learn from each other, and to discuss the different means by which they are achieving safety objectives.

MATERIALS LICENSING AND INSPECTION

The NRC currently administers approximately 7,200 licenses for the possession and use of nuclear materials in medical and industrial applications. This total represents a reduction of about 600 licenses in the past year, some of which is attributable to the State Agreement reached with Maine (shifting some licensing activity to the State), and to the full-cost recovery license fee rule (causing some licensees to decline renewal). Table 2 shows the distribution of NRC-administered licenses by Region; the 29 Agreement States administer about twice this number.

The program is designed to ensure that activities involving medical and industrial uses of radionuclides do not endanger the public health and safety. NRC regional staff completed approximately 2,700 inspections of materials facilities in fiscal year 1992. 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

Table 2. Regional Distribution of NRC Nuclear Materials Licenses

(as of September 25, 1992)

Region I	2,509
Region II	919
Region III	2,550
Region IV	751
Region V	251
Headquarters	225
Total:	7,205

that extract other metals from ores and slags containing uranium and thorium. These licenses are handled at NRC Headquarters.

The NRC completed over 6,100 licensing actions during the fiscal year. Of this total, over 400 were new license issuances, 4,400 were license amendments, 900 were license renewals, and 400 were sealed source and device design 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 requires an in-depth knowledge of its causes. Human factors evaluations designed to acquire such knowledge—with respect to applications in teletherapy and brachytherapy using remote afterloaders—continued during 1992. Contractors for these projects have completed function and task analyses of the two systems and have collected data on human-machine interfaces, procedures, training, and the organizational policies and practices typical for the systems. At the end of the report period, data were being analyzed to identify and set priorities among human factors problems. This meant identifying tasks with a high potential for the kind of human error that can adversely affect system performance, along with the factors that can contribute to those errors. Alternative means for resolving such problems are also being identified and evaluated.

Human error in the use of medical devices, including devices containing nuclear byproduct material, may be reduced by providing improved human factors engineering guidance to designers. As a member of the Human Engineering Committee of the Association for the Advancement of Medical Instrumentation (AAMI), an NRC staff human factors analyst continued to participate in revision of the document, "Human Factors Engineering Guidelines and Preferred Practices for the Design of Medical Devices."

An NRC project to evaluate information in reports of nuclear medicine misadministrations continued during 1992. The key element of the project is a computerized data base, which now contains information on misadministrations occurring in 1989 and 1990. A preliminary summary of information in the data base was presented at the February 1992 meeting of the American Association for the Advancement of Science.

Industrial Uses

Industrial Radiography. Industrial radiography is a form of non-destructive testing that uses radiation from byproduct material sources (principally, iridium-192 and cobalt-60) to examine the internal structure of materials. At the end of fiscal year 1992, there were a total of 212 radiography licenses in effect. Portable radiography devices can contain radioactive sources of up to 200 curies of iridium-192 or up to 100 curies of cobalt-60; devices at fixed facilities can contain sources of several hundred curies.

Workers in the radiography industry have a high potential for overexposure, and some have accidentally received significant doses of radiation. As a result, the NRC staff has several initiatives under way aimed at reducing these overexposures. One of these involves a rule change, "Safety Requirements for Industrial Radiography Equipment," published in final form in January 1990. A provision of the rule—which covers the design, manufacture, and testing of radiographic equipment, and became effective in January 1992—requires that all newly manufactured radiographic exposure devices and associated equipment acquired by licensees after the effective date comply with certain design, manufacturing, and testing criteria. Over this fiscal year, the NRC staff has continued to evaluate and approve several device systems (exposure detection devices and associated equipment) designed to meet the new requirements.

Another initiative in this area is the development of a certification program for industrial radiographers. As described in the *1989 NRC Annual Report*, pp. 74 and 75, the *1990 NRC Annual Report*, p. 81, and the *1991 NRC Annual Report*, p. 95, the NRC has supported the American Society for Nondestructive Testing (ASNT) in the development and implementation of its "Industrial Radiography Radiation Safety Personnel" (IRRSP) certification program. During this year, the NRC staff continued to work closely with the ASNT, the Conference of Radiation Control Program Directors (CRCPD), the State of Texas and other States to foster cooperation and understanding in the implementation of a mandatory radiographer certification program. In an effort to promote this cooperation and understanding, the NRC sponsored a public workshop with representatives of the States, CRCPD, and ASNT to discuss certification concepts.

A rulemaking on radiographer certification became effective in April 1991. The rule change recognized the ASNT program and was intended to encourage voluntary participation in the IRRSP certification program. However, the voluntary response thus far has been less than expected, with only 246 individuals being certified under the program as of October 1992.

In a related rulemaking, the staff is developing a rule that would mandate radiographer certification. The staff anticipates publishing the proposed rule in early 1993.

Irradiator Rule. On December 4, 1990, the NRC staff published, for public comment, a proposed addition to the regulations (10 CFR Part 36) to specify radiation safety and licensing requirements for the use of large quantities of radioactive material in commercial irradiators. Irradiators usually use gamma radiation from cobalt-60 to change a product's characteristics (e.g., to sterilize disposable medical supplies, such as syringes and gloves. For more information on irradiators, see the *1990 NRC Annual Report*, pp. 82 and 83, and the *1991 NRC Annual Report*, p. 95).

During fiscal year 1992, the staff evaluated information and resolved comments received from the public. This effort involved internal NRC reviews and a staff visit to a facility, Vindicator, a new irradiator in Florida, an Agreement State. Vindicator's principal operation involves irradiation of food to extend shelf life, a measure authorized by Federal agencies, such as the Department of Agriculture and the Food and Drug Administration. The purpose of the NRC visit was to determine the extent to which the State of Florida's experience in licensing a food irradiator needed to be factored into the final version of regulations in 10 CFR Part 36. The staff then prepared a draft final rule, which the Commission was considering at the end of fiscal year 1992. Publication of the final rule is expected during fiscal year 1993.

Source/Device Registration. Manufacturers and distributors of radiation sources and devices containing radiation sources are required to file safety information about their products with the NRC or an Agreement State. The NRC or Agreement State evaluates the information to ensure that the product meets radiation safety requirements and issues a certificate of registration to the vendor. The certificate then is cited 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. The registry includes sources and devices registered by the NRC and the Agreement States; it also includes sources and devices that are listed in the *Radioactive Materials Reference Manual* of the 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, which includes summary information about the sources or devices.

During the fiscal year, the staff completed nearly 400 safety evaluations and generated approximately 200 reports from the computerized registry for NRC Regional Offices, the Agreement States, and foreign countries. The staff also assisted the International Atomic Energy Agency in developing a world-wide registry of sealed sources and devices.

The NRC has developed and issued a draft Radiography Cross-Reference program. Copies of the personal computer-based program have been issued to NRC Regional Offices, Agreement States, foreign countries, radiography equipment manufacturers and major users. The program enables the user to inquire for and ascertain compatibility between radiography exposure devices, sealed source assemblies, and source changers. The NRC will issue a revision of the program in early fiscal year 1993, which will include updates and identification of equipment that meets the requirements of 10 CFR Part 34, Section 20.

The NRC continues to oversee a contract with the Southwest Research Institute to test products for the purpose of determining if testing procedures are adequate to demonstrate acceptable performance under anticipated conditions of use, and if the information submitted to the NRC by vendors is adequate to support licensing, of the products. The contractor has been collecting data and developing procedures for the tests. Actual testing should begin in fiscal year 1993.

General License Program. Under provisions of 10 CFR Part 31, a general license may be issued for possession and use of certain measuring and gauging devices containing nuclear materials. The device generally licensed 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.

As a result of several studies and surveys concerning general licensees and generally licensed devices (see *1991 NRC Annual Report,* pp. 93 and 94), the NRC published a proposed rule that will affect general licensees and distributors of generally licensed devices. The purpose of the proposed rule is to make general licensees more aware of the NRC requirements and to ensure that they are accountable for their generally licensed devices by requiring them to respond to requests by the NRC for information on the devices. The rule also requires 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 those licensees with additional information regarding the possession, use, transfer, and disposal of the devices, and the pertinent regulatory requirements. The staff has evaluated the comments received in response to the proposed rule, and the final rule should be published early in fiscal year 1993.

The NRC prepared and issued a request-for-proposal for a contract to communicate with general licensees about their possession of generally licensed devices. The contractor would keep an inventory of the generally licensed devices and contact the general licensees on a periodic basis, to ensure that the inventory information is correct and that the general licensees are aware of their regulatory responsibilities. The contract is expected to begin in fiscal year 1993.

The NRC also prepared 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 individuals inadvertently subjecting themselves to intersecting radiation beams and thus to unacceptable radiation levels. The proposed rule is scheduled to be published in the *Federal Register* early in fiscal year 1993.

Quality Assurance and Control for Source/Device Vendors. The staff revised its draft Quality Assurance and Control Manual for manufacturers and vendors of sources and devices containing byproduct material. The draft was revised based on information obtained during the pilot evaluation program, in fiscal year 1991. In fiscal year 1992, the staff continued its pilot evaluation program by visiting 11 vendors and manufacturers of sealed sources and devices. The information derived from these visits will be used to develop the draft manual into a Regulatory Guide and possibly a proposed rule in fiscal year 1993.

Sealed Sources Exceeding Part 61, Class C. Licensees possessing certain sealed sources are experiencing problems disposing of them 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 set forth in 10 CFR Part 61.

Under Federal law, ultimate disposal of these wastes is the responsibility of the DOE, and licensees must pay the full cost for disposal. The DOE is in the process of establishing a disposal facility, but the facility may not be available for many years. The NRC and the DOE have discussed the need for the DOE to accept and store such wastes in the interim, and to retrieve and store abandoned radioactive material. Several thousand NRC and Agreement States licensees possess sealed sources that will have to be stored until a disposal facility is available.

The DOE has retrieved and is storing several gauges that were abandoned in the public domain. The NRC staff continues to apprise the DOE of its concerns and has requested that the DOE identify an interim storage facility and establish eligibility criteria for accepting sources for interim storage. The NRC has suggested eligibility criteria that include provisions that prevent sources from being abandoned because of high disposal costs or the lack of disposal sites.

Medical Uses

Advisory Committee on Medical Uses of Isotopes. The Advisory Committee on Medical Uses of Isotopes (ACMUI) met in November of 1991 and in May and July of 1992. Topics discussed at the meetings included the Quality Management Rule; the Interim Final Rule on the Radiopharmacy Petition; the term "supervision," as defined in 10 CFR Part 35; broad scope licensing; abnormal occurrence criteria; training and experience requirements; and the administration of byproduct material to pregnant or nursing women. In July 1992, the ACMUI held its first meeting with the Commission.

At the direction of the Commission, the staff has continued to expand and rotate representation on the Committee. In 1992, two members whose first terms had expired were reappointed to two-year terms. In June 1992, a *Federal Register* notice was published, calling for the nominations of an individual qualified to address medical research, an individual with experience in hospital administration or management, and an oncology physician with experience in teletherapy. The current membership of the Committee is shown in Appendix 2.

Medical Visiting Fellows. In 1990, the NRC created a program for Medical Visiting Fellows, and sought nominees through a *Federal Register* notice dated June 7, 1990. An NRC evaluation panel selected a physician and a radiopharmacist for the first fellowships. (Their experience and credentials were cited in the *1991 NRC Annual Report*, p. 96.) During fiscal year 1992, they were involved in the implementation of the Quality Management Rule, training and experience criteria, the administration of byproduct material to pregnant or nursing women, the resolution of the Radiopharmacy Petition, and the public health and safety consequences of pharmacy-directed departures from package inserts. Throughout the year, both visiting fellows met with staff at NRC's Regional Offices, accompanied inspectors, and participated in NRC workshops and meetings.

Quality Management Rule. On January 27, 1992, regulations became effective requiring licensees to establish and implement a quality management program, in compliance with 10 CFR 35.2 and 35.32. This rule is a performance-based requirement for the development of a quality management program and focuses on the therapeutic uses of radioactive materials. On March 30, 1990, during the proposed rule stage, NRC received approval of the information collection requirements (ICRs) associated with this rule from the Office of Management and Budget (OMB). In December 1991, the NRC was notified that OMB had concerns with the ICR, and on June 26, 1992, OMB disapproved the ICR. This disapproval would have required NRC to conduct further rulemaking to delete the disapproved information collection requirements. However, the Commission overrode the disapproval by a unanimous vote, and on August 21, 1992, the OMB assigned a new control number for a period of three years.

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 an interim final rule addressing two issues raised in the petition, on August 23, 1990 (55 FR 34513), and amended the rule on October 2, 1992 (57 FR 45566). The rule originally permitted physician-directed departures from the manufacturer's instructions for diagnostic reagent kit preparation and generator elution, and from the manufacturer's instructions for preparation and use, and route of administration, for therapeutic radiopharmaceuticals, provided that certain conditions were met and records kept. The 1992 amendment removed the information and record-keeping requirements. Data on the use and the frequency of physician-directed departures—made in accordance with the interim final rule and collected during NRC inspections of medical facilities and commercial nuclear pharmacies—were used as the basis for the amendment. The interim final rule will be in effect until August 23, 1993. The NRC will continue to work closely with the Food and Drug Administration, the nuclear medicine community, and the radiopharmacy community to resolve the remaining issues raised by the petition.

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.

Nuclear Fuel Services. On September 10, 1992, Nuclear Fuel Services reported an explosion and fire in a dissolver tray located in its High-Enriched Uranium Recovery Facility (HEURF). The tray contained about 1,700 grams of uranium, in 22 liters of a liquid classified as a waste product. The fire was extinguished by a Radiation Monitoring Technician in about 10 minutes. No one was in the immediate area at the time, and no one was injured. No one received significant internal or external radiation exposure, and there were no abnormal releases from the building. The damage to the process equipment was small, and there was no damage to the building. The NRC dispatched a special inspection team to the site the same day. When preliminary investigation of the incident pointed to possible weakness in the licensee's procedures and operator training, it was decided to upgrade the NRC response to an Augmented Inspection Team (AIT) investigation.

The AIT determined that the substance being processed in the dissolver tray at the time was inadvertently transferred to the HEURF, and that it probably contained a certain chemical known for its explosive potential. Detailed examination of records and interviewing of facility employees revealed an apparent failure of operators to follow written procedures, certain weaknesses in the procedures, and mislabeling of products as the root causes of the unintended transfer of the substance. The investigation further revealed an apparent failure of the operators at the HEURF to recognize a precursor event—that took place a half hour before the incident—as a danger signal. The licensee management agreed with the team's findings and expeditiously initiated efforts to overhaul its procedures and to strengthen operator training.

Contaminated Steel Fence Parts. As discussed in the *1991 NRC Annual Report*, pp. 96 and 97, cobalt-60 was detected in chain-link fence bars imported from India by two United States importers. The importers identified distributors of the fence products who might have received the contaminated bars, and fence products in the distributors' inventory were surveyed by Federal, State and private health physicists. Contaminated material was segregated and stored against unauthorized removal. The two importers consolidated contaminated material at approved sites in Texas, California and Pennsylvania. One importer has contacted the Indian Government and received approval to return the contaminated material to India. The other importer has transferred the material to an NRC-licensed facility, for temporary storage, and plans to dispose of the material at a low-level radioactive waste disposal site in the United States.

Safeguards and Transportation

Chapter 5

Pursuant to provisions of the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, the Nuclear Regulatory Commission (NRC) regulates safeguards for licensed nuclear materials, facilities, and activities in order to assure protection of the public health and safety and to promote the common defense and security. In this regulatory context, "safeguards" denotes measures that are taken to deter, prevent or respond to the unauthorized possession or use of significant quantities of special nuclear material (SNM), through theft or diversion, and to protect against radiological sabotage. In general, safeguards for licensed nuclear fuel facilities and non-power reactors (NPRs) emphasize protection against theft or diversion of SNM, whereas safeguards associated with power reactors stress protection against radiological sabotage. Similarly, transportation safeguards address protection against theft or diversion of unirradiated SNM and sabotage of irradiated SNM. (SNM and strategic special nuclear material (SSNM) are technical designations for certain types, quantities, and/or isotopic compositions, defined by formula, of various nuclear materials. SSNM means uranium-235 (U-235) contained in uranium enriched to 20 percent or more in the U-235 isotope, uranium-233, or plutonium.)

During fiscal year 1992, NRC safeguards requirements were applied to 111 power reactors, 46 non-power reactors, 13 active nuclear fuel cycle facilities, and several independent spent fuel storage installations (ISFSIs). They were also applied to 31 shipments of irradiated spent reactor fuel; 29 shipments of SNM involving more than one, but less than five, kilograms of high-enriched uranium (HEU); and one shipment of SNM involving five or more kilograms of HEU.

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 SAFEGUARDS AND TRANSPORTATION IN 1992

Reactor Safeguards

Reactor Safeguards Inspection and Licensing. Within the five NRC Regional Offices, a total of 170 safeguards inspections were conducted at licensed nuclear power reactors under NRC safeguards requirements. Approximately 212 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, 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 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. Eleven OSREs were conducted through fiscal year 1992.

Fitness-for-Duty 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 18 months 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 1991" (NUREG/CR-5758, Volume 2). The report indicates that over 262,000 tests for the presence of illegal drugs and alcohol were conducted during calendar year 1991, of which 1,722 were positive. The majority of the positive test results (983) were obtained through pre-access testing (a 0.94 percent positive rate). There were 510 positive tests from random testing (0.33 percent positive rate). The positive rate also varied by worker category. For example, 0.22 percent of random tests of licensee employees were positive; for long term contractors, the rate was 0.31 percent; and for short-term contractors, the rate was 0.59 percent. Except in the case of short-term contractors, positive rates were lower than those reported for calendar year 1990. (The positive rate for short-term contractors in 1990 was 0.58 percent.)

Non-power Reactors (NPRs). NRC conducted 33 safeguards inspections of non-power reactors (NPRs) during fiscal year 1992. Efforts are continuing toward converting 25 NPRs from the use of 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 six reactors have been converted from the use of HEU to LEU fuel. One reactor conversion is fully funded and is expected to be completed by early fiscal year 1993; eight reactors are now partially funded and are expected to be fully funded during fiscal years 1993–95; and one reactor will be partially funded in fiscal year 1993. One government-owned reactor and one university-owned reactor have submitted "unique purpose" applications that are being reviewed by the Commission. There is no suitable replacement fuel for one reactor, and two commercial licensees are not scheduled to receive DOE funding.

Advanced Reactors. Safeguards reviews of advanced light water reactor standard designs continue to be predicated on the Severe Accident Policy Statement provision that "...issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants." The Final Safety Evaluation Report (FSER), issued in fiscal year 1992 for the Electric Power Research Institute (EPRI) "evolutionary" reactor document, and the Draft Safety Evaluation Report (DSER), issued in fiscal year 1992 for the EPRI passive reactor document, address requirements that plant designers analyze for potential sabotage vulnerabilities that could be minimized through design modifications. The FSER, prepared in fiscal year 1992 for the General Electric (GE) Advanced Boiling Water Reactor, addresses design modifications made by GE to remove or reduce specific vulnerabilities. The DSER, prepared in fiscal year 1992 for the Combustion Engineering (CE) evolutionary design, addresses the need for CE to conduct a sabotage vulnerability analysis.

Fuel Cycle Facility Safeguards

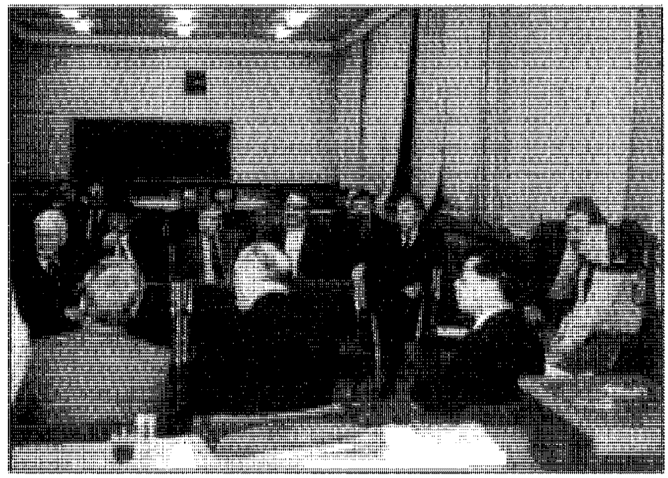
There were 13 active, licensed nuclear fuel cycle facilities subject to NRC comprehensive safeguards requirements during the report period. Of these, eight were major fuel fabrication facilities. Two of the 13 facilities contain significant quantities of HEU, requiring extensive physical security and material control and accounting (MC&A) measures. One of these two facilities, Nuclear Fuel Services (NFS), of Erwin, Tenn., is expected to phase out its naval reactors program work completely by the end of calendar year 1992. However, it has signed an agreement with the Russian Federation for possible involvement in the conversion of HEU, from the former Russian nuclear weapons program, to light water reactor fuel. If the NFS Erwin facility becomes involved in this conversion work, the facility will continue to be operated under NRC license.

The upgraded physical protection requirements established in 1991 were fully implemented at the two facilities cited above as possessing significant quantities of HEU, and performance testing through the use of mandated tactical drills and exercises provided additional assurance that the physical protection at these sites is effective. Besides calling for the physical protection upgrades, the NRC requires licensees possessing SNM to have systems in place to control and account for nuclear material in process and in storage.

Review of the application submitted by Louisiana Energy Services for a license to control and operate a commercial uranium enrichment plant is proceeding under the provisions of Public Law 101-575. Initial operation of the plant is planned for calendar year 1995. The safeguards portion of the environmental assessment has been completed, and the final technical review of the applicant's submittal is under way.

Preliminary meetings with the DOE, on the storage of spent reactor fuel at the Monitored Retrievable Storage (MRS) facility, have provided the foundation for eventual submittal of a license application. These initial meetings occasioned an exchange of scheduling and technical information that apprised the applicant, the DOE, of safeguards measures required by the NRC in the licensing of the MRS. In addition to the facilities noted above, several ISFSIs that are not located on the site of a licensed power reactor were also subject to safeguards requirements.

Support to the Republics of the Commonwealth of Independent States. In response to a national initiative to support the Republics of the Commonwealth of Independent States (CIS), formerly the Soviet Union, in effecting the safe and secure dismantlement of their nuclear weapons and disposition of recovered nuclear material, the NRC has assigned technical experts to the



NRC technical experts were part of the U.S. interagency team created to support the Republics of the Commonwealth of Independent States (CIS), formerly the Soviet Union, in effecting the safe dismantlement of their nuclear weapons and disposition of recovered nuclear material.

Above at left are members of the U.S. team arriving in St. Petersburg, Russia, to take part in a joint seminar with their Russian counterparts. Discussions are under way in photo at right. The seminar included 87 participants from 25 Russian nuclear organizations.

interagency team that formed to coordinate U.S. assistance. The interagency team is responsible for working with the CIS Republics in planning and developing comprehensive bilateral cooperative programs for achieving the common goal of finding peaceful uses for nuclear material.

Fuel Cycle Facility Inspections. Comprehensive physical security and material- control-and-accounting (MC&A) inspections were conducted at the major U.S. fuel fabrication facilities. Newly implemented physical security improvements were thoroughly inspected at the two facilities possessing significant quantities of HEU. Performance-based inspection procedures were followed for both MC&A and physical security inspections.

Transportation

Japanese Plutonium Sea Shipments. The NRC continued to participate on the interagency team reviewing physical protection arrangements for the sea transport of plutonium from Europe to Japan. The new United States-Japan Agreement for Cooperation in the peaceful uses of nuclear energy requires that a classified transportation plan, including contingency plans, be developed for each shipment and approved by the United States. The first shipment is scheduled during the autumn of 1992.

Spent Fuel Shipments. Thirty-one spent fuel shipments were made over approved routes during fiscal year 1992, 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 transfer approximately 1,170 fuel assemblies from other reactors to the Harris pool for storage over a five-year period.

Shipment Route Surveys. The NRC approved three additional transportation routes as acceptable for spent fuel shipments. NRC regional personnel continued to work with local law enforcement agencies in conducting field surveys of routes proposed for shipments of spent fuel. "Public Information Circular for Shipments of Irradiated Reactor Fuel" (NUREG–0725, Revision 8) was updated to include information on shipments through 1991. The report covers 1,114 highway and 100 rail shipments of spent fuel within the United States, subject to NRC safeguards regulations, from 1979 through 1991.

SSNM Shipments. Twenty-nine shipments of less than five but more than one kilogram of HEU were completed during fiscal year 1992. These included seven foreign shipments that entailed transient transport through the United States, and 22 domestic shipments of SSNM. One export shipment of five or more kilograms of HEU also was made during the fiscal year; the domestic portion of this shipment was made by the DOE.

Tracking International Shipments of SNM. NRC regulations require licensees to notify the NRC of international shipments of SNM and natural uranium. During fiscal year 1992, the NRC received about 250 such notifications. When appropriate, these were forwarded to the DOT for notification of international authorities.

Transport Inspection and Enforcement. The NRC continued to conduct safeguards inspections of selected shipments of spent fuel. No significant problems were identified from inspections carried out during the report period. The NRC also continued its transportation-related safety inspection program. The total effort involved approximately 1,400 individual inspections covering byproduct, source, and SNM licensees, and including fuel cycle facilities and shippers of spent reactor fuel.

The Office of Nuclear Material Safety and Safeguards (NMSS) cooperated with the Office of State Programs and the NRC Technical Training Center in conducting two transportation training courses, attended by 60 NRC and Agreement State inspectors.

An inspection program to ensure that transportation containers certified by the NRC are fabricated in accordance with the NRC-approved design and quality assurance programs of the container suppliers continued in fiscal year 1992. Inspections were conducted at eight facilities, representing a broad spectrum of the industry. The container-supplier inspection program includes designers, fabricators and distributors who have NRC-approved quality assurance programs and Certificates of Compliance for transportation packages. The program is structured to provide information as to whether transportation packages are fabricated, procured and maintained in conformance with 10 CFR Part 71 requirements. This quality assurance inspection program also included inspection of spent fuel dry storage casks, licensed under 10 CFR Part 72. Two inspections were conducted to obtain information on the implementation of quality assurance requirements in the fabrication, loading and maintenance of dry storage casks.

Transportation Incidents. NMSS continues to monitor transportation incidents. Seventy-two transportation incidents were reported during calendar year 1991. Of these, 23 were accidents, 6 were handling events, 12 were thefts or loss of packages, and 31 were classified as "other" events. Of those classified as "other," 10 were incidents of contamination. Of the 23 accidents that occurred, 3 involved type B packages, 16 involved type A packages, 1 involved a strong/tight package, and 3 were unidentified. There was no release of contents in any of the accidents involving type B packages. 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.)

Springfield, Mass., Accident. One incident that involved an NRC-certified shipping package occurred on December 16, 1991, when a truck carrying unirradiated (fresh) nuclear fuel was involved in an accident on U.S. Interstate 91, in Springfield, Mass. The accident occurred at approximately 3:15 a.m., when an automobile traveling in the wrong direction on Interstate 91 collided head-on with the oncoming truck. As a result of the accident, the truck and shipping containers carrying the fresh fuel were engulfed in a fire that lasted over three hours. Despite the collision and subsequent fire, there were no deaths or serious injuries, and there was no release of radioactive material. The accident did, however, result in substantial property loss, including the truck, shipping containers, and damaged fuel assemblies.

NMSS published two reports concerning this accident. The first report, "A Highway Accident Involving Unirradiated Nuclear Fuel in Springfield, Massachusetts, on December 16, 1991" (NUREG/CR-5892), was prepared by Lawrence Livermore National Laboratory (LLNL), under contract to NMSS. The report is a technical evaluation of the mechanical and thermal environments experienced by the packages during the accident. The report substantiates that, although the packages were severely damaged, there was no release of radioactive material, and the health and safety of the public was not endangered. The second report, entitled "Emergency Response to a Highway Accident in Springfield, Massachusetts, on December 16, 1991" (NUREG-1458), includes a review of the emergency response information available to personnel responding to the accident and the emergency response measures taken. The review indicates that certain improvements should be made in the nature of the information immediately available to emergency responders and the manner in which it is provided. The NRC is discussing ways to improve emergency response guidance with cognizant Federal agencies. (Copies of these reports are available from the National Technical Information Service, Springfield, Va. 22161.)

Plutonium Air Shipment Criteria Development. Section 5062 of Public Law 100-203 imposes requirements on 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 1992, the NRC continued feasibility studies related to the testing of such packages. This task included performing tests for bench-marking computer codes, in order to verify methodologies developed for meeting the requirements of the law. The feasibility studies and testing were requested and funded by the Power Reactor and Nuclear Fuel Development Corporation, on behalf of the Japanese Government. Contract support for this effort is being provided by Lawrence Livermore National Laboratory.

Incident Response Planning and Threat Assessment

The NRC staff assesses threats to NRC-licensed facilities, materials, and activities and prepares safeguards incident response plans for NRC use in 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, participating in regular interagency meetings of Federal agencies concerned with and prepared to deal with terrorism. Other liaison activity includes briefings and consultations with representatives of other governments regarding NRC threat assessment and incident response activities. During the report period, these activities were expanded to include NRC participation in training provided to other agency threat-analysts, to increase their understanding of nuclear-related matters. As part of these cooperative efforts, the NRC and the Federal Aviation Agency promulgated a revised MOU covering information exchange, incident response, and related mutual support.

In response to continuing tension in the Persian Gulf and in the former country of Yugoslavia, the staff closely monitored and analyzed developments in those areas of the world on a daily basis. The staff also continued to work closely with the DOE, the Federal Bureau of Investigation, and other cognizant agencies, regarding attempts to sell alleged nuclear material.

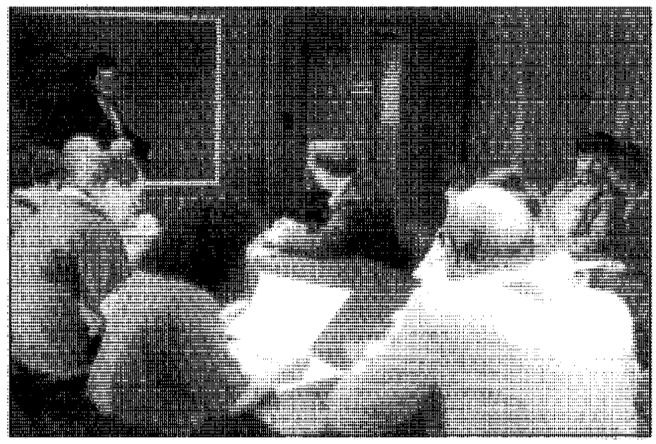
During fiscal year 1992, the staff discerned no significant changes in the threat environment that would warrant modifications in the NRC's current safeguards regulations. Two techniques are employed in assessing reported threats to NRC licensees. Internally, the NRC Information Assessment Team, composed of headquarters and regional personnel, promptly assesses all reported threats and recommends appropriate response actions to NRC management. In addition, the Communicated Threat Credibility Assessment Team, jointly funded by the NRC and the DOE, conducts analyses of written or recorded threats.

During this fiscal year, the fuel cycle safeguards incident response plan was reviewed and updated. In May 1992, specialized training on NRC threat assessment procedures was furnished for NRC Headquarters Duty Officers and, in June 1992, incident response training for safeguards staff was completed. An exercise involving power reactor safeguards was conducted in August 1992.

The staff continued to analyze safeguards events related to threats and incidents, to identify trends, patterns, and anomalies. The "Safeguards Summary Event List" (NUREG–0525, Volumes 1 and 2), a compilation of safeguards events, was revised in July 1992, to include events occurring through December 1991. This document was distributed to the licensed nuclear community, foreign governments, the Congress, and other Federal agencies.

A data base of 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 maintained and used by the agency to perform analyses by which to identify any potential weaknesses in a facility's security system, as well as to characterize generic and facility event precursors. In fiscal year 1992, four reports were issued.

NRC licensees are using the event information and analysis findings to develop and implement long term solutions to equipment failure and human error. The reports allow more efficient use of inspector resources by calling their attention to specific security degradations, causes and trends. The NMSS quarterly reports also include information describing the types of corrective action taken by some licensees, and results that have been successful in reducing degradations, so other licensees can benefit from this experience.



The NRC provides transportation safety inspection training for NRC and State personnel. In the photo above, an instructor conducts classroom training for participants in a transportation inspection course. Below, the trainees perform a radiological survey of a shipment arriving at a low-level waste disposal facility.

International Safeguards

The NRC is responsible for implementation of IAEA safeguards, at licensed nuclear facilities in the United States. During 1992, the IAEA informed the United States that they would not be conducting inspections at

U.S.-licensed facilities, because of a lack of resources.The United States continues to report to the IAEA all accounting information required by the Protocol to the U.S.-IAEA Safeguards Agreement. The IAEA Board of Governors has passed a resolution requesting additional reporting by the United States; this request should be fulfilled during fiscal year 1993.

During 1992, NRC initiated two tasks directly supporting the improvement of IAEA safeguards. One is a technical analysis of enrichment plant safeguards using a computerized analysis program called PASE. The second is application of the NRC-developed Adjusted Running Book Inventory method to the head-end of a reprocessing plant.

The NRC continues to contribute to the total U.S. support of IAEA safeguards through interagency efforts. The NRC participated in the multi-national examination of large scale reprocessing plant (LASCAR) safeguards. The survey was completed in 1992, and the group concluded that techniques are available for effectively safeguarding the large nuclear fuel reprocessing plants expected to be operated on a commercial basis in the 1990s. Two other interagency efforts supported by the NRC are the Action Plan Working Group and oversight of the U.S. Program of Technical Support to Agency Safeguards.

NRC work on U.S. initiatives for strengthening safeguards includes the foreseen changes resulting from certain high-level waste disposal proposals. A new initiative for 1993 will consider the applicability of the U.S.-IAEA Agreement to HEU that may be imported from Russia. The United States has stated, in response to questions related to the President's announcement of the HEU agreement, that this HEU would be subject to the U.S.-IAEA Agreement.

International Physical Protection

In connection with its export licensing program, the NRC participates in an interagency program to visit and to 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 retransfers of U.S.-origin material. During fiscal year 1992, visits for these purposes were made to France, the Federal Republic of Germany, Japan, South Korea, Hungary, and the Czech and Slovak Federal Republic. Similarly, teams from Japan and Australia visited the NRC in the United States.

REGULATORY ACTIVITIES AND ISSUES

Proposed Rules

The following rulemaking actions were initiated during fiscal year 1992:

- Work was initiated on a proposed rulemaking 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 expressed request that this be done. The proposed rule is expected to be published for comment early in fiscal year 1993.
- Work began 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 computer-readable 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 final rule is expected to be published in June 1993.

The following rulemakings continued during fiscal year 1992:

- A proposed rule to amend 10 CFR Part 73 to clarify physical protection requirements was published for public comment in May 1992. No negative comments were received. The amendment would clearly declare 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 Section 73.60(f) would be 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 NPRs licensed to operate at or above a power level of two megawatts thermal. The final rule is expected to be published in early 1993.
- 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 NRC regulations, to make NRC commercial transport protection comparable with that provided by the DOE. The proposed rule is expected to be published in late 1993.

- On December 13, 1991, the NRC published a proposed rule to amend 10 CFR Part 73 to upgrade Weapons Firing Qualification Requirements and Physical Fitness Training and Performance Testing Requirements, for all security personnel at Category I fuel facility licensees. Public comment resolution is expected in early 1993.

- On April 30, 1992, the NRC published a proposed rule that would extend the 10 CFR Part 26 fitness-for-duty rule to licensees who possess, use, or transport Category I (unirradiated formula quantity) SNM. The final rule is expected to be published in December 1992.

Final Rules

The following rulemaking was completed and published in fiscal year 1992:

- On July 29, 1992, the NRC published changes to 10 CFR Parts 70, 72, 73, and 75. These changes: (1) supplement the definitions sections, (2) delete action dates that no longer apply, (3) correct outdated terms and cross-references, (4) clarify wording that is susceptible to differing interpretations, (5) correct typographical errors, and (6) make other minor changes.

- A final rule was published on October 31, 1991, amending 10 CFR Part 74 to establish material-control-and-accounting measures for uranium enrichment facilities that would produce LEU for commercial light-water reactors.

- A final rule was published on January 22, 1992, amending 10 CFR Part 11 to include acceptance of the DOE-L or DOE-Q Reinvestigation Program for NRC-R SNM access authorization renewal requirements.

Nuclear Materials Management and Safeguards System

This project, jointly funded with the DOE, continues the operation and maintenance of the Nuclear Materials Management and Safeguards System (NMMSS). Basically, this 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 continuing 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. A noteworthy step in fiscal year 1992 was the proposed regulatory change making licensee submittal of information in computer-readable form mandatory.

Waste Management

Chapter 6

The Office of Nuclear Material Safety and Safeguards (NMSS) of the 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, decommissioning of nuclear facilities, activities of the Licensing Support System Administrator, and reports of the Advisory Committee on Nuclear Wastes, during fiscal year 1992.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development Activities

During the report period, the NRC continued efforts to ensure that the regulations governing high-level waste (HLW) disposal (10 CFR Part 60) were clear and complete. In particular, the staff analyzed 10 CFR Part 60 to determine if all HLW repository functions related to radiological safety were covered to sufficient depth. As a result of the analysis, the NRC is preparing a proposed rulemaking,"Design Basis Events for the Geologic Repository Operations Area," to clarify the relationship of 10 CFR Part 60 to accident conditions, and to provide consistency among NRC regulations by including a "controlled-use area," and by revising the definition of "important to safety." The rulemaking also addresses a Department of Energy (DOE) April 19, 1990 petition for rulemaking (PRM-60-3). In its petition, the DOE requested that 10 CFR Part 60 be amended to include quantitative dose criteria for a design basis accident. The NRC expected to publish the *Federal Register* notice for public comment by the end of calendar year 1992.

The NRC has also continued working with the Environmental Protection Agency (EPA), as the EPA revised its HLW standards. The NRC staff participated in the EPA's reissuance of its environmental standards for the disposal of HLW by reviewing, preparing comments on, and consulting with the EPA on important features of the draft standard. At the end of fiscal year 1992, legislation was passed mandating National Academy of Sciences (NAS) review of several matters regarding standards for the disposal of HLW at Yucca Mountain in Nevada. The NRC staff will cooperate with the NAS during its assessments, as appropriate. After completion of the NAS review, and issuance of standards by the EPA, the NRC will conduct its own rulemaking to ensure that 10 CFR Part 60 is consistent with the final EPA standards.

Regulatory Guidance Activities

Regulatory guidance issued during the report period included promulgation of two final Staff Technical Positions (STPs). STPs provide guidance to the DOE on selected topics, setting forth criteria by which the NRC staff judges the acceptability of proposed methods of complying with regulations in 10 CFR Part 60. The first STP, "Investigations to Identify Fault Displacement Hazards and Seismic Hazards at a Geologic Repository" (NUREG-1451), provides guidance to the DOE on acceptable investigations that can be used to identify fault displacement hazards and seismic hazards at a geologic repository. The intent is to ensure that the DOE's solutions to actual or potential geologic and seismic effects at a candidate site are based on investigations of sufficient detail and are understood well enough to permit a reliable evaluation of the proposed site. The second STP, "Geologic Repository Operations Area Underground Facility Design—Thermal Loads" (NUREG-1466), is the NRC staff position on a methodology for demonstrating the acceptability of a geologic repository operations area underground facility design, meeting the thermal load design requirements of 10 CFR 60.133(i). The STP states that the methodology 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.

Technical Assessment Capability For Repository Licensing Reviews

During the report period, the NRC staff began preparing the License Application Review Plan (LARP), giving guidance to the NRC staff in its review of the DOE's license application. A table of contents was developed, consistent with the table of contents of the draft Format and Content Regulatory Guide for the License Application, and a standard structure was developed for each of the 102 individual review plans in the LARP. This structure comprises the following sections: applicable 10 CFR Part 60 requirements; review strategy; review method;

acceptance criteria; implementation, and example findings. Applicable 10 CFR Part 60 requirements were identified for each individual review plan, and review strategies completed for 16 individual review plans. Review strategies for the remaining individual review plans will be completed in fiscal year 1993, and the first draft of the LARP will be completed in fiscal year 1994.

Pursuant to preparation of the LARP is the NRC staff's continued development of its independent capability to review the DOE's performance assessments for a geologic HLW repository. These assessments will be used by the DOE, in its license application, to show compliance with 10 CFR Part 60, including, by reference, the EPA radiation protection standard—40 CFR Part 191. For its part, the staff will use its technical assessment capability to review the DOE's performance assessments and other aspects of the DOE HLW program. In May 1992, the staff published a report, "Initial Demonstration of the U.S. Nuclear Regulatory Commission's Capability to Conduct a Performance Assessment for a high-level Waste Repository" (NUREG–1327), documenting the initial (Phase 1) demonstration of its independent performance assessment capability. Also in fiscal year 1992, the staff continued to enhance its performance assessment capability by undertaking a second iteration (Phase 2), using more refined predictive models and treating a more comprehensive set of phenomena and scenarios. Objectives of Phase 2 include the 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. Completion of Phase 2 is planned in fiscal year 1993.

Activity was also initiated in fiscal year 1992 supporting the LARP by seeking to develop various analysis methods. In the area of tectonics, analysis methods were developed in the form of computer-balanced, cross-section tools for the evaluation of alternative tectonic models. In seismology, computer codes were tested for use in the analysis of seismic hazards at a proposed geologic repository site. For the Engineered Barrier System (EBS), work continued on development of an EBS performance assessment computer modeling code.

Yucca Mountain Site Characterization Analysis

Activities at Yucca Mountain, Nev., increased substantially during the fiscal year, following receipt of appropriate permits from the State of Nevada. The DOE began numerous borehole, trenching, and test pit operations, in connection with site characterization purposes. NRC staff participated in several site visits to observe drilling and trenching activities, and to examine data gathered from the boreholes and excavations.

The NRC staff continued to review DOE site characterization activities at Yucca Mountain. In its Site Characterization Analysis (SCA), dated August 1989, the NRC staff identified 198 concerns—classified as objections, comments, or questions—related to the DOE's planned studies for site characterization (See *1991 NRC Annual Report*, p. 108). The DOE continued to make progress toward resolving many of these concerns and, at the end of the fiscal year, one objection and a number of other concerns had been resolved.

The NRC staff has also continued to review DOE site characterization study plans. By the end of fiscal year 1992, the DOE had submitted a total of 43 study plans for the NRC staff's review. To date, the NRC staff has completed 27 study plan reviews, returned eight study plans to the DOE for revisions and resubmittal, and was reviewing eight other plans at the close of the report period. The NRC staff has identified no reasons to object to start-up of activities related to any reviewed study plan, but has conveyed its concerns to the DOE regarding several of them.

The DOE also issued an Early Site Suitability Evaluation (ESSE) report and Site Characterization Progress Report (PR) Number 5, during fiscal year 1992. The NRC staff reviewed and commented on both of these documents. Review of the ESSE focused primarily on whether the DOE's applications and interpretations of siting guidelines in 10 CFR Part 960 were consistent with those concurred in by the Commission in 1984. From its review of the PR, the NRC staff determined that the DOE had been partially responsive to NRC concerns regarding the level of detail in the information provided in the DOE's submission of semiannual PRs.

In June 1992, the Yucca Mountain region underwent a magnitude 5.6 earthquake, located approximately 15 miles southeast of Yucca Mountain, at Little Skull Mountain. NRC Headquarters staff and staff from the On-Site Representative's office closely followed evaluations of the earthquake during the aftershock period. Because of this event, the DOE has expanded its seismic monitoring activities in the region.

Interactions with Affected Governmental Units and Indian Tribes

The State of Nevada and local representatives continued to participate in the technical exchanges and meetings between the NRC and the DOE. State, local and Tribal representatives also continued to receive notification of upcoming NRC/DOE HLW meetings, as well as meetings of the NRC Advisory Committee on Nuclear

Waste (see below). All parties continue to receive all correspondence and publicly available NRC reports regarding the HLW program.

Quality Assurance Activities

During the report period, the staff continued to review quality assurance (QA) plans and procedures (document reviews) of the DOE and of DOE contractors, to evaluate the DOE's effectiveness in auditing its program so as to identify and correct problems in program implementation, and also to evaluate DOE contractor effectiveness in implementing QA programs. A part of this effort for fiscal year 1992 was a review of revisions to those QA plans previously accepted. In carrying out these assessments, the NRC staff observed the DOE audits conducted at all the 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 which indicate that improvements are needed.

In August 1991, the DOE requested that, because of improvements in the QA area, the NRC remove its Site Characterization Analysis (SCA) objection concerning the lack of an acceptable QA program. The staff did lift the SCA objection in March 1992, based on a determination that the DOE Office of Civilian Radioactive Waste Management had demonstrated that it could develop and implement a QA program acceptable to the NRC; could oversee the development of acceptable QA program plans for its participants; and could audit participant QA programs effectively by identifying deficiencies and verifying the effectiveness of corrective actions.

Center For Nuclear Waste Regulatory Analyses

The Center for Nuclear Waste Regulatory Analyses (CNWRA), an NRC contractor, completed its fifth year of operation in October 1992. Its contract has been renewed for another five years. The CNWRA provides the NRC with sustained special expertise in the areas of technical assistance and research, in support of the NRC's HLW program, under the Nuclear Waste Policy Act (NWPA) of 1982, as amended. The CNWRA provides a broad range of services to NMSS and to the Office of Nuclear Regulatory Research, as well as to the Office of the Licensing Support System Administrator. CNWRA staff are located at the Southwest Research Institute campus in San Antonio, Tex., and at the Washington Technical Support Office in Arlington, Va.

The CNWRA, together with the NRC staff, continues to develop and implement a computer-assisted "systems engineering approach," called the Systematic Regulatory Analysis (SRA). 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 to assure that all of the HLW activities under the NWPA are planned, integrated, implemented, documented and managed as thoroughly and effectively as possible. Pursuant to that objective and to reinforce the staff's technical assessment capability, the CNWRA has completed a design for a comprehensive, high-performance computer network, to be made available to the NRC staff over a three-year period, beginning with fiscal year 1992.

The CNWRA's special expertise is useful to NRC staff in their review of study plans and design reports; in NRC/DOE pre-licensing technical exchange meetings; in QA observation audits; in furnishing technical support to NRC rulemaking and regulatory guidance development programs; in the development of analysis methods (e.g., computer codes), and in research. Activities in the research program include studies on the thermodynamic and ion exchange properties of sorbing minerals; studies of geochemical natural analog sites and related laboratory investigations; laboratory and calculational investigations of two-phase flow in heterogeneous fractured porous media; installation of instrumentation for measuring rock mechanical and hydro-geological responses to induced seismic events at an active mine; evaluation of a state-of-the-art seismic rock mechanics computer code; and laboratory investigation of the degradation of nickel- and copper-based alloy container materials.

LOW-LEVEL WASTE MANAGEMENT

The NRC's low-level waste program seeks to ensure the protection of public health and safety, and of the environment, by regulation of the management of low-level radioactive waste (LLW), in conformance with the low-level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA).

Regulations and Guidance

Title Transfer Provision of the Amendments Act. On June 19, 1992, the United States Supreme Court issued a decision in *New York v. United States,* regarding the constitutionality of the LLRWPAA. The Supreme Court held, in a case brought by the State of New York (a non-compact State) and by two of its counties, that the so-called "take-title" provision of the LLRWPAA, which was to take effect on January 1, 1996, is unconstitutional. The court upheld the remainder of the LLRWPAA, including other incentives for the States to assume responsibility for LLW generated within their borders. (The constitutionality of the take-title provision as applied to compact States was not at issue in the case.) Even though this provision of the ACT was held to be unconstitutional, the clear goal of the LLRWPAA remains intact, i.e., to develop new LLW disposal facilities by January 1, 1993, and in no case later than January 1, 1996.

Prior to the Supreme Court decision, the Commission had been considering a rulemaking that would require a licensee to exhaust all other reasonable waste management options before the licensee would be allowed to store LLW on-site, after January 1, 1996. The Commission was considering this proposal because of the health and safety implications of increased reliance on on-site storage of LLW, and also in light of the LLRWPAA goal of developing new disposal capacity. The proposal would require the generator of low-level nuclear waste to request that the State take title to, and possession of, the licensee's LLW as a precondition for on-site storage, after January 1, 1996. And the licensee would be required to attempt to contract, either directly or through the State, for the disposal of its LLW. This proposal was transmitted to the Agreement States for comment. At the close of the report period, the Commission was reviewing the proposal in light of Agreement State comments and of the Supreme Court decision.

"Receipt Back" of LLW by Power Reactors. On February 21, 1992, the staff informed the Commission of its plans to issue for public comment a proposed minor rulemaking amending NRC regulations, in 10 CFR 50.54, concerning "Conditions of Licenses," in order to allow reactor licensees to receive back processed waste that they originally generated. Companies providing nuclear power reactors with off-site LLW processing and volume-reduction services currently transfer treated waste directly to one of three operating commercial LLW disposal facilities. Under the provisions of the LLRWPAA, however, access to disposal facilities may not be available to some waste generators after January 1, 1993. Consequently, instead of shipping LLW directly to the disposal sites, some commercial waste processors will have to return LLW to the generators for interim storage, until regional or State disposal facilities become available. While existing nuclear power reactor licenses do not allow the receipt of processed LLW, the proposed rule would permit it.

The proposed rule was published in the April 24, 1992 *Federal Register*. The NRC received a total of 31 comment letters; 26 commenters endorsed adoption of the rule. All comments received were evaluated in developing the final rule, which was adopted and published in the *Federal Register*, on October 21, 1992.

Standard Review Plan. The low-level Waste Management and Decommissioning staff is developing revisions 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 a low-level radioactive waste disposal facility. The draft revised SRP covers the licensing process (SRP 1.0), as well as surface water hydrology, design of soil cover systems, waste disposal operations, performance assessment and analysis of radioactivity releases, and occupational radiation protection.

Draft SRP revisions were distributed to Agreement States for comment in November 1991, with comments collected from the States by February 1992. NRC staff provided a briefing on the SRP revisions to the Advisory Committee on Nuclear Waste (ACNW) in January 1992. During fiscal year 1993, the NRC plans to give final form to the SRP revisions.

Technical Assistance to the States

During fiscal year 1992, the LLWM staff continued to support the NRC Office of State Programs (SP) in providing technical assistance to the States as they implement their plans for low-level waste disposal facility development and licensing.

The technical assistance included:

- Support to OSP in holding a Regulators' Workshop and a Special Topics Workshop for Agreement State Regulators.
- Support to OSP in conducting program reviews of Agreement State regulatory programs.
- Presentations, written reports, and attendance at public meetings for various States on topics of special interest to them. The staff's objective is to keep the States fully informed of regulatory issues and to respond to their specific requests for guidance on regulatory matters.

Several areas of State interaction are discussed in more detail below.

Review of Draft Environmental Impact Study Plan for Connecticut. In December 1991, the staff completed its review of the Draft Environmental Impact Study Plan (EISP) submitted by the Connecticut Hazardous Waste Management Service (Connecticut Service). The EISP is a generic plan for assessing potential environmental impacts from construction, operation, closure, and monitoring of a proposed LLW disposal facility at three candidate sites. The purpose of the EISP is to identify the resource categories (e.g., Water Resources and Water Quality, Ecology, Land Use, etc.) that will be evaluated and the types of environmental information needed, including environmental impacts, to support in part the State's selection of a preferred site and the preparation of license and permit applications for that site. The staff found that the EISP comprehensively addresses many of the elements expected in a generic environmental studies plan. The staff identified some additional areas required for a complete environmental report under 10 CFR Part 51.

Review of Conceptual Design Report and Quality Assurance Plan for Maine. In September 1992, the NRC staff completed its review of the LLW Disposal Facility Conceptual Design Report (CDR) and Quality Assurance Plan (QAP) for the Maine Low-Level Radioactive Waste Authority. The staff identified technical concerns that would have to be satisfied in the design of the LLW disposal facility, in order to assure that all of the performance objectives of 10 CFR Part 61 will be met. With the exception of several incomplete sections, the staff found that the QAP adequately describes a system of management controls, supported by quality verification and review activities, that should demonstrate the completeness and appropriateness of the level of quality achieved.

Performance Assessment Guidance. The staff has prepared and is carrying out a program for developing low-level waste performance assessment (LLWPA) techniques and for enhancing staff expertise. The program has two primary goals:

(1) To enhance the NRC staff's capability to review and evaluate a LLWPA from a license applicant.

(2) To develop an in-house LLWPA modeling capability that will serve as the basis for development of regulatory guidance.

The program will also improve NRC's ability to provide technical assistance to Agreement States on LLWPA issues.

The principal approach used by the NRC staff to enhance in-house expertise in LLWPA modeling is to develop a hypothetical test case by which to exercise various LLWPA models. The approach to this test case performance assessment (PA) includes: (1) establishing objectives; (2) developing conceptual models for the test case site and its hypothetical facility design; (3) selecting appropriate mathematical treatments, computer codes, and input data bases; (4) conducting iterative PA analyses; (5) integrating sub-modeling results; (6) conducting sensitivity and uncertainty analyses; (7) conducting confirmatory analyses (as needed); and (8) evaluating the results with respect to Part 61 performance objectives. The NRC staff anticipates that assessment of an actual disposal facility would follow a similar approach. Staff recognizes that PA is an iterative process, and that additional site characterization and/or design modifications may be required as part of the PA modeling process.

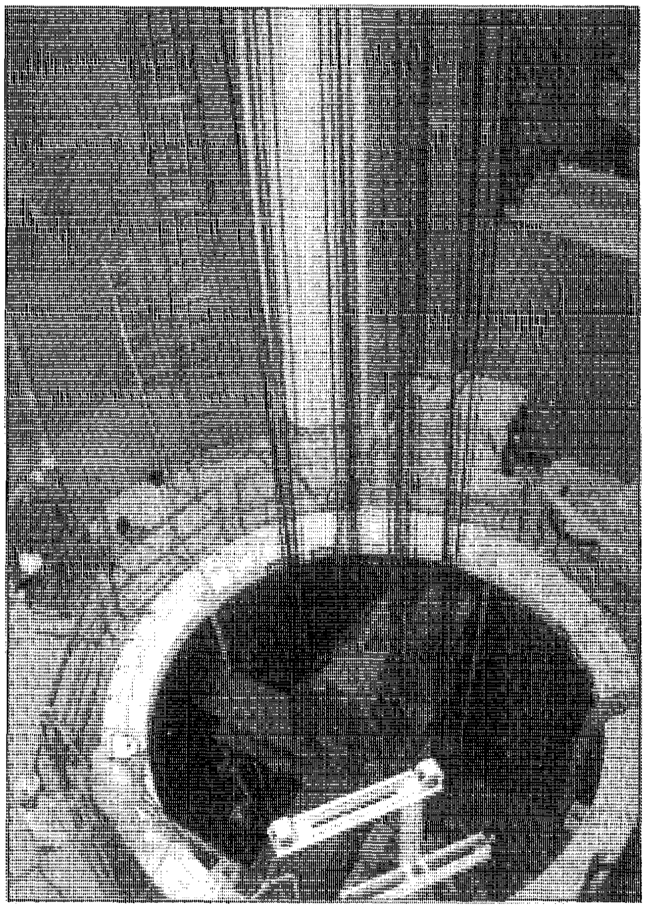
This theoretical exercise is supplemented by review of performance assessments of actual license applications. Currently, these are available principally from Agreement States. The NRC has received PAs for two sites. A PA for a proposed facility in California has been received by the NRC for information, and NRC staff provided guidance for the review of the performance assessment of a proposed disposal facility in the State of Nebraska. The NRC is also actively involved with other Federal agencies in basic research and international efforts addressing PA issues.

In fiscal year 1992, this experience has provided the basis for preparation of a draft Branch Technical Position (BTP) on Performance Assessment. The BTP, still under development, is broken into individual sub-modeling components including: (1) infiltration, (2) engineered barriers, (3) source term, (4) groundwater transport, (5) surface water transport, (6) air transport, and (7) dose modeling. This BTP will provide license applicants with acceptable criteria and technical bases for evaluating the long term performance of a LLW disposal facility.

The low-level waste performance assessment (LLWPA) program, which has been developed jointly by the NRC Office of Nuclear Material Safety and Safeguards (NMSS) and the Office of Nuclear Regulatory Research (RES), involves integrated staff and contractor work, supported by research projects. In order to provide inter-office coordination of LLWPA activities, staff from NMSS and RES have formed a Performance Assessment Working Group, responsible for developing and implementing the LLWPA program.

Working Group staff presented the Performance Assessment Program Plan to the Advisory Committee on Nuclear Waste in October 1991. In February 1992, the staff completed the Plan, forwarded it to the Commission, and provided copies to the public (SECY 92-060). Several members of the Working Group also made presentations at the Agreement State Regulators Workshop, in July 1992, and responded to questions and concerns of the States. Working Group staff provided direct technical assistance to the State of Nebraska in evaluating the performance assessment that was part of a license application for a disposal facility in that State.

Working Group staff personnel have participated in performance assessment activities of the DOE, including the Performance Assessment Task Team (PATT) and the DOE Low-Level Waste Management Program. The staff also participated in international performance assessment activities, through the International Atomic Energy Agency.



The NRC works in close coordination with other Federal agencies across the spectrum of public health and safety concerns. In recent years, the safe and effective decommissioning of nuclear facilities and the secure disposition of former nuclear facility sites have occasioned formal cooperation of the NRC with such agencies as the Environmental Protection Agency and the Department of Energy. The photo shows a work platform inside the reactor vessel of one such facility, the Shoreham nuclear power plant on Long Island, N.Y., which is undergoing decontamination and dismantling. The plant was the subject of long and complex adjudication by the NRC and by the courts before a final decision that it be permanently shut down. The construction permit for the Shoreham plant was first issued in 1973, and an operating license was issued in 1989, but the 820-megawatt boiling water reactor plant never went into commercial operation.

Cooperation With Other Federal Agencies

During 1992, the NRC continued cooperation with other Federal agencies in resolving issues associated with low-level radioactive waste management and disposal, and the safe and effective decommissioning of licensed nuclear facilities and formerly used sites. These efforts have primarily involved the Environmental Protection Agency (EPA) and the Department of Energy (DOE), but they also include other Federal and State regulatory agencies. An important milestone in cooperative efforts with the EPA was the completion of a General Memorandum of Understanding (MOU) providing a framework for interagency cooperation on matters related to the regulation of radionuclides in the environment. The MOU was signed on March 16, 1992, establishing guiding principles and procedures for NRC-EPA interaction and promoting joint exploration of issues. Since completion of the MOU, staff has cooperated with the EPA in evaluating approaches to interagency cooperation and to reconciling risk assessment and risk management practices.

At the staff level, NRC-EPA cooperative activity continued to focus on the resolution of issues associated with the joint regulation of radioactive mixed waste, and with the dual regulation of radionuclide emissions to the air. During fiscal year 1992, a National Profile on the Volumes, Characteristics and Treatability of Commercially Generated Mixed Waste was produced. Sponsored jointly by the NRC and the EPA, the National Profile disclosed that approximately 140,000 ft3 of mixed waste was generated in 1990, and that 75,000 ft3 of mixed waste was in storage as of December 31, 1990. The National Profile also revealed that most mixed waste can be treated by using currently available technologies and capacity, although an additional 12,000 ft3 of treatment capacity is needed to treat the mixed waste generated in 1990 and in storage as of December 31, 1990. In fiscal year 1992, the NRC and the EPA issued, for public comment, a guidance document on the testing of mixed waste and completed the working draft of a guidance document on mixed waste storage.

Regarding emissions of radionuclides to the air, the NRC and the EPA continued to cooperate in determining whether NRC's established regulatory program for air emissions of radionuclides under the Atomic Energy Act of 1954 provides adequate protection of the public with the ample margin of safety provided for under the Clean Air Act (CAA). (Section 112(d)(9) of the CAA states that the EPA need not regulate radionuclide air emissions if it determines that NRC's regulatory program already provides an ample margin of safety.) Cooperation during fiscal year 1992 led to the EPA's completion of a survey of air emissions data from NRC and Agreement State licensees, a staff-level MOU pertaining to rescission of EPA standards for radionuclide emissions from NRC and Agreement State licensees, and a draft regulatory guide on "as low as is reasonably achievable" (ALARA) emission levels for effluents from materials facilities. The staff-level MOU was signed on September 4, 1992, and the EPA was to publish the MOU in the *Federal Register* in November 1992. The survey found that radionuclide emissions from NRC and Agreement State licensees did not exceed eight millirems-per-year effective dose equivalent, and a majority of facilities (over 95 percent) did not exceed one millirem-per-year effective dose equivalent. Based on the survey results, the EPA has tentatively concluded that NRC's program currently protects public health with an ample margin of safety. Completion of the draft ALARA regulatory guide strengthens the EPA's decision to withdraw radionuclide air emission standards on the grounds that NRC's program adequately protects public health.

Cooperation also continued working with the EPA on its standards for radon emissions from uranium mill tailings disposal. In this project, NRC staff focused on fulfilling its commitments, in the October 1991 MOU, to establish enforceable schedules for the timely closure of non-operational tailings impoundments, and on developing proposed amendments to 10 CFR Part 40, Appendix A, to bring them into conformance with the EPA's amendments to 40 CFR Part 192, addressing the timing of compliance with the radon emission standard and measurements to confirm compliance. Current staff initiatives regarding emissions of radionuclides into the air include a proposed rescission of the EPA standards that control radionuclide air emissions from NRC and Agreement State licensees other than nuclear power reactors, and a final rescission of the EPA standards that control radionuclide air emissions from nuclear power reactors, and radon emissions from uranium mill tailings disposal. The agencies also consulted on a variety of other issues across the broad spectrum of shared responsibilities.

Cooperative efforts between the NRC and the Department of Energy (DOE) during the report period were centered primarily on resolving issues associated with the management and disposition of low-level radioactive wastes whose concentrations exceed the upper limits for Class C wastes, as defined in 10 CFR Part 61. Under the low-level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA), the DOE is responsible for disposing of the so-called "Greater-than-Class-C" (GTCC) wastes in an NRC-licensed disposal facility. In the interim—before such a disposal facility is built and begins operations—the DOE is considering the storage of GTCC waste in interim and dedicated storage facilities. The interim storage facility would be used for abandoned and other GTCC material judged to pose a health and safety concern if left in the long term possession of commercial licensees. The NRC and the DOE continued to cooperate in developing procedures and criteria for managing the transfer of GTCC material to such an interim storage facility. At the DOE's request, NRC provided information which defines the needed storage facility capabilities. The NRC also developed, and issued for comment on July 1, 1992, proposed guidance on acceptable encapsulation and concentration averaging practices for LLW. The purpose of the guidance is to encourage uniformity between the Agreement States and the NRC in determining what qualifies as GTCC waste for which the Federal Government has disposal responsibility. During the latter part of 1992, comments on the proposed guidance were being considered and the development of final guidance was being coordinated with Agreement State regulators. In response to a request by the DOE, the staff is also providing limited cooperation in the DOE's Programmatic Environmental Impact Statement on the implementation of an integrated environmental restoration and waste management program.

URANIUM RECOVERY AND MILL TAILINGS

The NRC licenses and regulates uranium mills, commercial in-situ solution mining operations, uranium extraction research and development projects, and disposal of uranium mill tailings and wastes.

The NRC also evaluates and concurs in DOE remedial action projects for inactive uranium mill tailings sites and associated vicinity properties as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA).

Regulatory Development and Guidance

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 ACL's for uranium mills in June 1988. Workshops were held in October 1988 and in December 1990. The staff received comments on the draft technical position from both government and private parties and has submitted the proposed final technical position for Commission approval.

The commingling of low-level and Naturally Occurring Radioactive Material (NORM) waste with uranium mill tailings has been a subject of growing interest in recent years. Uranium and thorium mill tailings and wastes, defined in Section 11e.(2) of the Atomic Energy Act, must be disposed of under a license issued in accordance with 10 CFR Part 40. Radioactive wastes of similar chemical and radiological characteristics (primarily earthen material contaminated with source material)—but not falling within the the definition—must generally be disposed of in low-level waste facilities, unless the NRC authorizes their disposal by some other means. In July 1988, the staff issued guidance on the disposal of such material in uranium mill tailings impoundments. In August 1991, the staff proposed revised guidance to the Commission. In February 1992, the Commission directed the staff to publish the guidance, which had been modified to accommodate Commission comments, in the *Federal Register* for public comment. Another subject of growing interest is the use of uranium mill feed material other than natural uranium ore. In October 1991, the staff proposed guidance to the Commission on the use of alternate feed material that would allow the wastes from the processing of such material to be disposed of in tailings impoundments. The Commission directed the staff to publish that proposed guidance in the same *Federal Register* notice as promulgated the guidance on commingling. The two guidance documents were published in the *Federal Register* in May 1992. Staff received 23 letters in response. After review and analysis of all comments received, the staff prepared revised guidance documents, which will be transmitted to the Commission early in fiscal year 1993.

Licensing and Inspection Activities

In the fall of 1989, the NRC received an application from Envirocare of Utah, Inc., for license to dispose of commercial uranium and thorium mill tailings and wastes received from others at its facility in Clive, Utah. Because of the unique, "first-of-a-kind," nature of the application, the regulatory framework for the staff review had to be established by Commission order. 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 licensing review began in 1991, and the staff completed two acceptance reviews and several rounds of questions to the applicant. The staff began preparation of the Draft Safety Evaluation Report and the Draft Environmental Impact Statement in 1992.

The NRC Uranium Recovery Field Office (URFO) performed 23 inspections of uranium recovery facilities during the fiscal year. In other regulatory action, the URFO staff issued one new license for a commercial *in-situ* solution mining operation, two license renewals, 27 major license amendments, 68 minor license amendments, and one mill tailings reclamation plan amendment; the office terminated three licenses for *in-situ* solution mining pilot projects. In addition, 107 environmental and radiological monitoring report reviews were completed and pre-licensing guidance was provided to two potential applicants.

Of the 28 NRC-licensed uranium recovery facilities, 19 are uranium mills, three are either "heap leach" or other byproduct recovery operations, and six are commercial *in-situ* solution mining facilities. At the close of the fiscal year, three commercial *in-situ* mining operations were in operation, one was in standby, and two were under construction. No conventional uranium mills were in operation, only 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. *In-situ* solution mining facilities are expected to remain moderately active, however, with one currently under licensing review and two more forecast to be applying for licenses during fiscal year 1993. 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. An important aspect of this casework is fulfilling the commitments in the October 1991 staff-level MOU between the NRC, the EPA and the States of Colorado, Texas and Washington with respect to CAA standards for radon emissions from uranium mill tailings. These commitments involve establishing firm and enforceable schedules for the timely closure of non-operational tailings impoundments. Licensing oversight of the *in-situ* mining facilities—including issuance of new licenses for proposed facilities, and inspections of all licensed facilities—will continue for the indefinite future.

Remedial Action at Inactive Sites

There were 24 abandoned uranium mill tailings sites designated under the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA) to receive remedial action by the DOE. UMTRCA requires that the NRC concur with the DOE's selection and performance of remedial action, confirming that the action meets appropriate standards promulgated by the EPA. The DOE has established a Uranium Mill Tailings Remedial Action (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 1992, NRC staff carried out 81 separate reviews at various sites, in meeting responsibilities assigned by UMTRCA. These included 15 Remedial Action Plan (RAP) reviews, 11 inspection plan reviews, 13 RAP modification reviews, 9 other site-specific reviews, 6 Completion/Certification Report reviews, and 15 reviews of generic items. The staff prepared four Technical Evaluation Reports documenting its review of the DOE's remedial action selection of the Gunnison (Colo.), Rifle (Colo.), Falls City (Tex.), and the combined Mexican Hat (Utah)/Monument Valley (Ariz.) sites. The staff also documented the review of the DOE's remedial action completion in the Completion Review Reports for the Green River (Utah), Tuba City (Ariz.), and Spook (Wyo.) sites.

In support of the UMTRA Project casework, the staff visited many of these sites. Inspections of remedial action in progress were conducted at the Lowman (Idaho), Salt Lake (Utah), Grand Junction (Colo.), and Durango (Colo.) sites. NRC technical staff also conducted site visits associated with Remedial Action Plan reviews at the Mexican Hat (Utah)/Monument Valley (Ariz.) and Falls City (Tex.) sites.

At the end of the fiscal year, staff had reviewed and was prepared to formally concur in the DOE's Generic Guidance for Long-Term Surveillance Plans (LTSP). The submittal of a site LTSP to NRC for approval is one of the final actions by DOE prior to the site's coming under the NRC general license, in 10 CFR Part 40.27.

The groundwater remediation phase of the UMTRA Project was started during 1992. The initial phase of this effort includes the development of technical approaches by the DOE and review procedures by the NRC. Groundwater remediation has been deferred by the DOE until after the sites have been reclaimed. DOE contributions from this initial effort will be incorporated into the Programmatic Environmental Impact Statement for this phase of the remedial program. NRC review procedures will be used to assure compliance with uranium mill tailings regulations in 40 CFR 192, Subparts A-to-C.

DECOMMISSIONING OF NUCLEAR FACILITIES

The NRC staff has continued development of 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 nuclear reactors and materials facilities.

Regulatory Development and Guidance

The staff is developing guidance documents for license reviewers and licensees giving needed information on acceptable methods for decommissioning. This guidance includes SRPs for the review of nuclear power plant preliminary and final decommissioning plans. During fiscal year 1991, the staff completed an SRP for the review of decommissioning plans for materials licenses. During fiscal year 1992, the NRC published an Action Plan in the *Federal Register* describing the approach the agency will use to accelerate the cleanup of radiologically-contaminated sites (see summary below under "Materials Decommissioning.") A draft "Manual for Conducting Radiological Surveys in Support of License Termination" (NUREG/CR-5849) was published for comment in June 1992. In July 1992, the staff developed a draft Branch Technical Position (BTP) on Site Characterization for Decommissioning Sites. The draft BTP will be published for comment in the *Federal Register* in fiscal year 1993. The staff also prepared a rulemaking on timeliness of decommissioning that will set a time limit for decommissioning a facility at which operations have ceased. Staff also initiated a rulemaking on record-keeping to ensure that decommissioning records are maintained, including "as built" facility drawings, locations of contamination, and such other documentation as will be needed for decontamination and decommissioning.

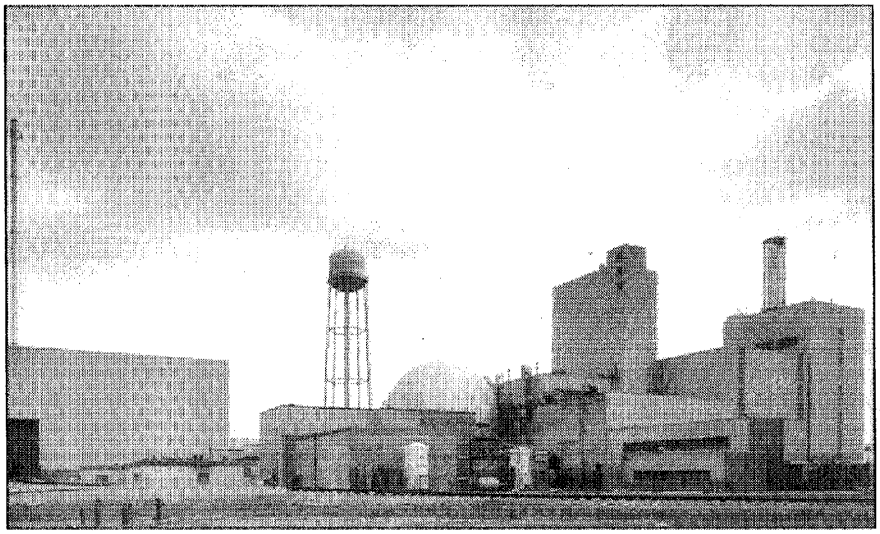
Reactor Decommissioning

The NMSS staff continues to assist the licensing staff of the Office of Nuclear Reactor Regulation (NRR) in reviewing decommissioning plans for power reactors already shut down. The NRC developed and implemented a protocol for the transfer from NRR to NMSS of responsibility for power reactors after approval of a decommissioning plan and issuance of a possession-only license. Since the protocol was initiated, NMSS has assumed responsibility for the Humboldt Bay Unit 3 (Cal.), Vallecitos (Cal.), Fermi Unit 1 (Mich.), Peach Bottom Unit 1 (Pa.), LaCrosse (Wis.), and Shoreham (N.Y.) nuclear power plants. In 1990, the staff approved a dismantlement plan for the Pathfinder (S.D.) nuclear power plant, a 58.5-megawatt facility which has been shut down since 1967. In 1991, the reactor vessel was removed from the containment building in one piece and shipped by rail to the commercial low-level waste disposal site in Hanford, Wash. The staff also reviewed the decommissioning plans for the Fort St. Vrain (Colo.) high-temperature gas-cooled reactor, and the Rancho Seco (Cal.) pressurized water reactor. In 1992, the staff approved the decommissioning plan for Shoreham and the licensee is actively dismantling the facility.

Materials Decommissioning

Several hundred NRC materials licenses are terminated each year. (In 1991, over 600 materials licenses were terminated.) The majority of NRC licensed operations result in little or no contamination of buildings or soil, and decommissioning actions leading to the termination of the 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 the non-routine cases. These are cases that 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 by other radionuclides, presenting varying degrees of radiological hazard, cleanup complexity, and associated cost.

The NRC developed the Site Decommissioning Management Plan (SDMP) in 1990 to guide efforts to identify



Responsibility for regulatory oversight of a facility undergoing decommissioning passes from the Office of Nuclear Reactor Regulation to the Office of Nuclear Material Safety and Safeguards (NMSS) after approval of the licensee's decommissioning plan and issuance of a "possession only" license. Among the plants under the purview of NMSS is the Enrico Fermi Unit 2 plant at Laguna Beach, Mich., shown here. The Fermi plant began operation (1965) as a "breeder" reactor, producing nuclear fuel while generating electricity. The plant has operated as a boiling water reactor unit since 1985.

non-routine decommissioning cases and to ensure 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, issued in May 1992, contains the following guidance.

(1) Criteria for listing a contaminated site in the SDMP (there are currently 46 sites listed in the SDMP).

(2) Priorities of NRC efforts in the oversight of contaminated sites.

(3) Schedules and resources needed for NRC oversight of contaminated site cleanup.

(4) 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. Progress on actual site remediation, however, has been slow. The limited progress prompted the staff to develop the SDMP Action Plan, which was approved by the Commission on April 6, 1992. The Action Plan was released to the public on April 8, 1992, at an NRC press briefing, and was published in the *Federal Register* on April 16, 1992. The Action Plan describes the Commission's position on several issues identified as affecting the timely cleanup of contaminated sites, including:

(1) Interim cleanup criteria.

(2) Finality of NRC decommissioning decisions.

(3) Acceptable timeframes for decommissioning.

(4) Site characterization.

(5) Procedures for compelling timely cleanup.

In general, the release of the Action Plan has been effective in communicating to licensees and the public the Commission's expectation that SDMP sites shall be cleaned up in a timely and effective manner. The plan has incited a number of SDMP site owners and licensees to improve their cooperation and willingness to initiate and conduct cleanup efforts.

Over the last year, the decommissioning of the Allied Signal Aerospace site was completed, and the site was removed from the SDMP list. The Amax and UNC Recovery System sites have completed decommissioning. The licenses for these two sites will be terminated, and the sites removed from the SDMP list, after pending administrative issues are resolved. Ongoing decommissioning activities at other sites include site characterization, development of site decommissioning plans, site remediation, and termination surveys.

THE LICENSING SUPPORT SYSTEM ADMINISTRATOR

The Licensing Support System (LSS) is an information management system established to contain the documentary material generated by the DOE, the NRC, the State of Nevada and other potential parties to the licensing proceeding for the DOE's high-level radioactive waste repository. All potential parties to the proceeding will have electronic access to the system both before and after the hearing on the matter begins.

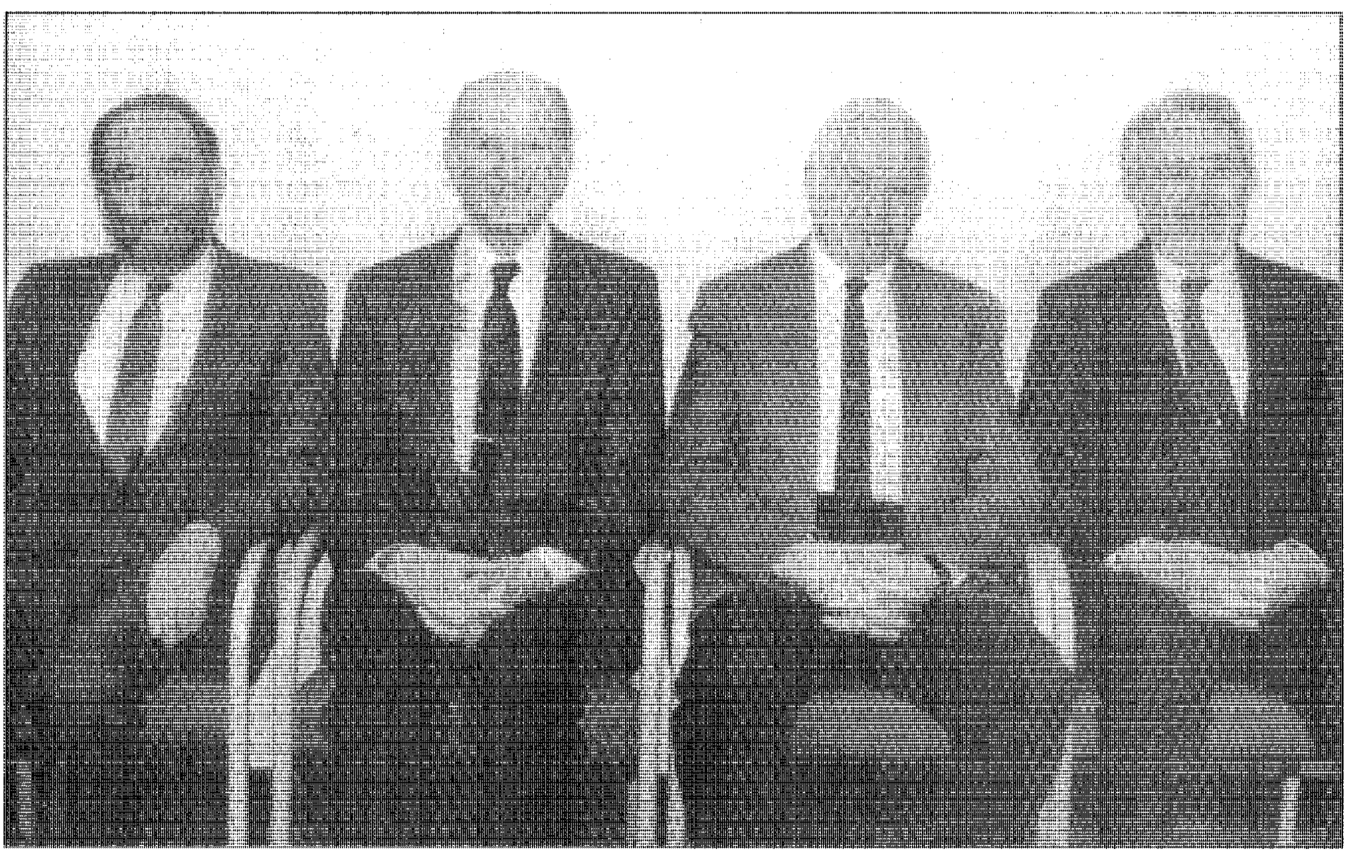
The position of LSS Administrator (LSSA) was established to administer and manage the LSS, to ensure the timely availability of the LSS to all LSS participants, to operate and maintain the LSS, to ensure the integrity of the LSS data base, and to ensure that the LSS meets the requirements of 10 CFR Part 2, Subpart J.

LSSA Activities

No additional work on LSS design and development was undertaken during the report period, while the LSS program and responsibilities for its management were being jointly re-examined by the NRC and the DOE.

Future users of the LSS must have timely and effective access to millions of pages of non-textual data produced during scientific investigations of the candidate repository site. Most of this information—handwritten field notes, maps, photographs, logs, computer tapes, etc.—will require special access procedures through the LSS. During the fiscal year, the LSSA examined numerous issues and explored a variety of alternatives for allowing users to effectively search for and locate such data.

To assure that the LSS is a comprehensive and accurate data source for technical review and litigation support, LSS participants must properly identify, prepare and submit their documentary material to the LSS. In this phase of preparation, the LSSA must evaluate these participants' activities for compliance with the LSS rule. During the report period, the LSSA continued working to create detailed document submission standards, to set realistic document production schedules, to explore the feasibility of setting priorities for document submission, and to develop a cost-effective compliance evaluation program. The LSSA also began developing a concept of, and functional requirements for, a quality assurance operation by which to review the quality of LSS participants' submissions.



The Advisory Committee on Nuclear Waste (ACNW) reports to and advises the Nuclear Regulatory Commission on nuclear waste management, dealing primarily with nuclear waste disposal issues but also such concerns as off-site activities associated with production and utilization facilities—such as the handling, processing, transportation, storage, and safeguarding of nuclear wastes. Committee members (1992), shown above, are, left-to-right: Dr. Paul W. Pomeroy, President, Rondout Associates, Incorporated, Stone Ridge, N.Y.; Dr. Dade W. Moeller (ACNW Chairman), Professor of Engineering in Environmental Health and Associate Dean for Continuing Education, School of Public Health, Harvard University, Boston, Mass.; Dr. Martin J. Steindler (ACNW Vice-Chairman), Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Ill.; and Dr. William J. Hinze, Professor, Department of Earth and Atmospheric Sciences, Purdue University, West Lafayette, Ind.

The LSS Advisory Review Panel (LSSARP), which is administratively supported by LSSA, held no public meetings during the report period. Three additional counties were invited to participate on the LSSARP, as part of the coalition representing local governments adjacent to Yucca Mountain, Nev. (See Appendix 2 for a list of LSSARP members and coalition representatives.)

ADVISORY COMMITTEE ON NUCLEAR WASTE

The Advisory Committee on Nuclear Waste (ACNW) was established by the Nuclear Regulatory Commission in 1988. The ACNW is charged by its charter to, "...report to and advise the Nuclear Regulatory Commission (NRC) on nuclear waste management, as directed by the Commission on the basis of periodic reviews of ACNW proposals. This includes 10 CFR Parts 60, 61, and 72 (as applied to other than the site of production and utilization facilities) 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 will be on disposal but will also include other activities off-site of production and utilization facilities, such as handling, processing, transportation, storage, and safeguarding of nuclear wastes including spent fuel, nuclear wastes mixed with other hazardous substances, and uranium mill tailings. 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 on its own initiative related to those issues directed by the Commission."

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.

During fiscal year 1992, the ACNW reported to the Commission on a variety of issues, including:

- Regulatory Guides for Implementation of the Revised 10 CFR Part 20.
- Performance Assessment for low-level Waste Disposal Facilities.
- Performance Assessment for high-level Waste Disposal Facilities.
- Geologic Dating of Quaternary Volcanic Features and Materials.
- NRC Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Facility.
- Staff Technical Position on "The Identification of Fault Displacement and Seismic Hazards at a Geologic Repository.
- Environmental Protection Agency (EPA) high-level Waste Standards.
- Performance Indicators for Evaluating the Programs for the Management and Disposal of low-level Radioactive Waste.
- Staff Technical Position on Alternate Concentration Limits for Title II Uranium Mills.
- On-site Storage of low-level Radioactive Waste.
- NRC Radioactive Waste Research Programs.
- Proposed Rulemaking on Design Basis Events for Geologic Repository Operations Area.
- Proposed Rulemaking on Emergency Planning Licensing Requirements for Independent Spent Fuel Storage Facilities and Monitored Retrievable Storage Facilities.
- Department Of Energy (DOE) Early Site Suitability Evaluation for the Yucca Mountain Facilities.
- DOE Site Characterization Activities for the Yucca Mountain Facility.
- License Application for the high-level Waste Repository.
- Staff Technical Position on Geologic Repository Operations Area Underground Facility Design-Thermal Loads.
- Draft Regulatory Guide 8013, "ALARA Radiation Protection Program for Effluents from Materials Facilities."

In performing the reviews and preparing the reports cited above, the ACNW holds regular full committee meetings and working group sessions as needed.

Communicating With The Public, The Government, and Other Nations

Chapter 7

The Nuclear Regulatory Commission maintains regular communication with a broad spectrum of governmental entities, domestic and international, as well as with the general public. Several NRC Headquarters Offices and the Regional Offices participate in the dissemination of information about NRC activities. The 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 Federal and State agencies, with Indian Tribes and local community organizations, and with the news media, the Congress and the international community is provided mainly through these four offices of the NRC: the Office of Public Affairs, and the Office of State Programs, the Office of Congressional Affairs, and the Office of International Programs.

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 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.

Transcripts of open meetings and documents released at meetings are available for inspection and copying in the NRC Public Document Room, 2120 L St., N.W., Washington, D.C.

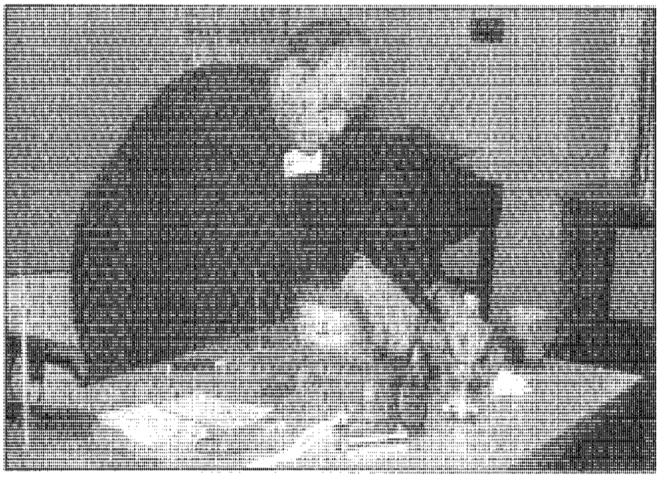
At least one week before a meeting is scheduled, notice of the meeting is published in the *Federal Register*. An announcement is also displayed on a TV-monitor in the lobby of NRC Headquarters and is posted in the Public Document Room. The announcement discloses the time, place and subject matter of 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.

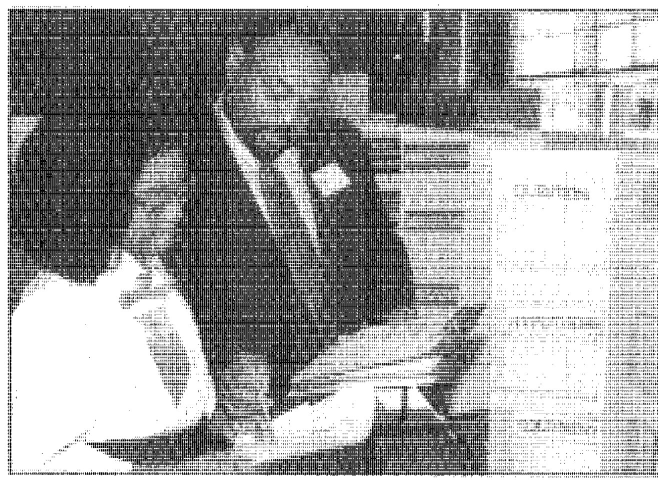
Advisory Committees

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 cross-section of the scientific and technical community, as well as from State and local governmental organizations, and from among private citizens.

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. Committee members provide advice and recommendations to NRC on a broad 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.

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.





NRC volunteers typically visited two-to-three schools a week, and on occasion hosted students at NRC Headquarters, under the National Partnerships in Education program, initiated by the President in 1983. In fiscal year 1992, volunteers responded to over 100 requests from schools in the Washington D.C. Metropolitan Area, reaching almost 5,000 students and faculty, primarily in Montgomery County, Md., public schools, but also in surrounding counties. Shown above, at left, is Gus Giese-Koch of the Office of Nuclear Reactor Regulation, giving a "Mr. Wizard" demonstration of scientific principles for elementary school students. At right, Tim Rollins of the Office of Administration consults with a young participant in a Science Fair. An award was presented to the NRC by Montgomery County for outstanding service to education during the 1991–1992 school year.

Public Information

The five NRC Regional Administrators, assisted by the Office of Public Affairs, conducted periodic news briefings on regional and agency-wide issues, as the Commission continued its policy of expanded openness regarding activities and programs of the agency.

News briefings during the report period were held in Philadelphia, Pa; Syracuse, N.Y.; Cleveland, Ohio; Cambridge, Mass.; Minneapolis, Minn.; Chicago, Ill.; Chattanooga, Tenn.; Charlotte and Wilmington, N.C.; Omaha, Neb.; San Luis Obispo, Cal.; Portland, Ore.; and San Diego, Cal. In some instances, the briefings generated extensive news coverage.

The interests of reporters from newspapers, radio, television and national news services tended to focus on such topics as:

- The performance of specific nuclear power plants, and those on the NRC's list of problem plants.
- The NRC's program for the clean up of contaminated sites.
- Reports of NRC teams sent to investigate nuclear power plant incidents.
- Enforcement actions taken by the agency.

These periodic news briefings were in addition to news conferences held on specific incidents or events and the briefings that follow Commission and/or staff meetings to explain in more detail important NRC rulemakings, policies and programs.

The Office of Public Affairs also keeps the news media and general public informed of agency activities by disseminating news releases, fact sheets, pamphlets and formal orders on the major decisions and actions taken by the Commission and the NRC staff.

Enforcement Conferences. In keeping with NRC Chairman Ivan Selin's emphasis on openness, the NRC initiated a two-year trial program to allow the news media and the public to observe selected enforcement conferences. In some cases, the conferences generated local coverage.

Media Seminar Workshop. Members of the NRC Technical Training Center and regional Public Affairs staff conducted a national seminar on October 14–15, 1991, for reporters from six different cities. The reporters were given the opportunity to operate nuclear power plant simulators at the NRC training center in Chattanooga, Tenn. The simulators duplicate actual plant operations. The reporters also were given a basic orientation in how commercial nuclear power plants are built and licensed, how the NRC regulates them, and how radiation protection is assured.

NRC School Volunteers Program. For the eighth year, NRC volunteers worked with schools throughout the Washington Metropolitan Area, as part of the national Partnerships in Education program, initiated by the President in 1983. During the school year, NRC volunteers typically visited two-to-three schools a week, and on occasion hosted students at NRC Headquarters. In fiscal year 1992, volunteers responded to over 100 requests from schools, reaching almost 5,000 students and faculty, primarily in Montgomery County, Md., public schools, but also in surrounding counties.

An award was presented to the NRC by Montgomery County for outstanding service to education during the 1991–1992 school year.

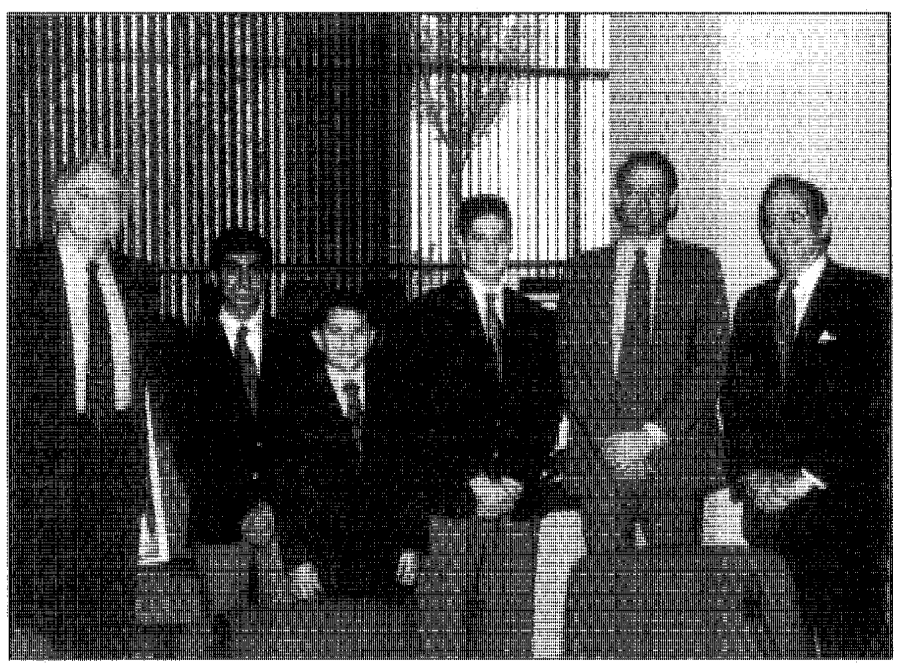
Volunteers worked with grade levels from kindergarten through college (the latter including the Johns Hopkins University and Pennsylvania State University) and with students who ranged from the academically advanced to those at risk of dropping out. Volunteers provided hands-on science demonstrations, academic tutoring, mentoring, assistance on science projects, opportunities for students to shadow them on the job, judging for science and math fairs, assistance to faculty in developing curricula for special study areas, responses to student interviews, lectures on the use of math and science on the job, and career awareness discussions. Volunteer activities were conducted during office hours, as well as in the evenings and weekends, depending on the nature of the schools' activities.

For the second time, the NRC provided special awards for winners at at the annual Montgomery Area Science Fair, with Commissioner James Curtiss making the presentations. In a meeting open to all NRC staff, area students presented their projects to the Commission. Later, one of the Science Fair winners spent a day during the summer at the NRC, with several staff members from the Office of Nuclear Materials Safety and Safeguards, and the Office of the Executive Director for Operations, and met with Commissioners Curtiss and E. Gail de Planque.

During fiscal year 1991, for the first time, NRC organized and participated, together with other area Federal agencies and businesses, in a Science and Technology Program for Educators. As part of this program, 17 secondary science teachers from Montgomery County schools spent a day at the NRC learning about the agency, nuclear power, radiation, nuclear waste, transportation safety, and nuclear power plants in the former Soviet Union and Eastern Europe. Teachers were addressed by Commissioners Kenneth Rogers and Forrest Remick and by Harold Denton, Director, Office of International Programs. Other NRC staff members made presentations which included demonstrations suitable for the teachers to duplicate in their classrooms.

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, on-line 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. PDR users can have documents from the collection, with some exceptions, reproduced for a nominal fee.



For the second time, the NRC provided special awards for winners at the annual Montgomery Area Science Fair. In a meeting open to all NRC staff, area students presented their projects to the Commission. Later, one of the Science Fair winners spent a day during the summer at NRC Headquarters. Shown here are winners of NRC Special Awards from the fair—Ajay Shroff, Ben Nelson, and Justin Ziombra (second, third and fourth from left)—flanked by Commissioner Forrest J. Remick, at left, Commissioner James R. Curtiss, second from right, and Chairman Ivan Selin, at right.

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 Meeting Notices, are posted in the Reading Room at the facility. The PDR does not contain books, journals, trade publications, or documentation of industry standards.

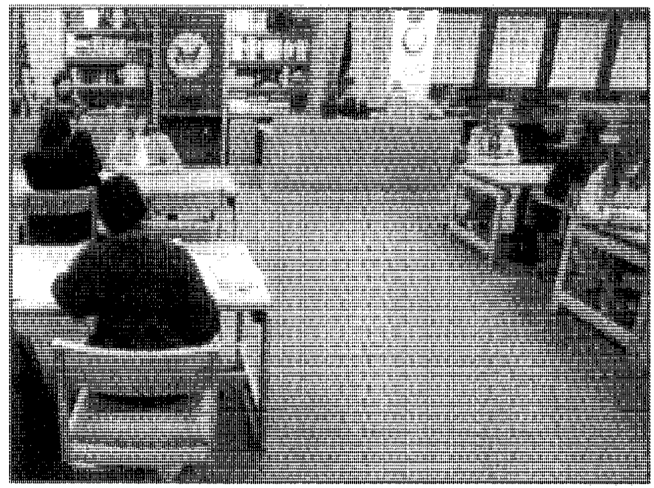
The Headquarters PDR contains more than 1.75 million documents. During a typical month, the PDR serves over 1,300 users. Reference Librarians are available to assist on-site users and those who call or write with information requests. Besides responding to letters and telephone requests, PDR staff make 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 both 1,200 or 2,400 baud, with 9,600 baud planned for 1993) is available for searches 24 hours a day, weekends and holidays included. Training sessions in using the BRS data base may be scheduled by calling the telephone number given below. An on-line tutorial is in development and scheduled to be available some time in 1993.

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, and foreign governments. Foreign contacts with the PDR include users from England, France, Italy, Japan, the Netherlands and Spain.

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 1992, the NRC was maintaining 87 Local Public Document Rooms (LPDRs) throughout the country. These LPDRs house collections of documents related to nuclear power reactors, research reactors, 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 NRC Headquarters Public Document Room, above, contains more than 1.75 million documents related to licensing proceedings and other agency actions; the facility serves over 1,300 users in a typical month.

A primary goal of the LPDR program in fiscal year 1992 was to complete 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. Over 47,000 microfiche were sent to each LPDR library. The NRC's LPDR staff visited 36 LPDRs in fiscal year 1991 and the remaining 43 LPDRs in fiscal year 1992, in order to set up the microfiche files; 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.

Thirty 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 any NRC publicly available record, 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.

A new LPDR was established during the report period for the Chemetron Corporation's decommissioning sites in Ohio. The LPDR is located at the Garfield Heights Library, Garfield Heights, Ohio. The second LPDR for the Callaway (Mo.) nuclear power plant, maintained at Washington University in St. Louis, Mo., was closed.

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 the archives of a number of government agencies, the personal papers of former government officials, and personal interviews with such officials. Drawing on this research, the History Office recently completed the second volume of its detailed history of nuclear regulation. The new volume, *Containing the Atom: Nuclear Regulation in a Changing Environment, 1963–1971*, was published in 1992 by the University of California Press. This study focuses on reactor siting and safety, radiation protection, and environmental issues. It is a sequel to *Controlling the Atom: The Beginnings of Nuclear Regulation, 1946–1962*, published in 1984 by the University of California Press. The two volumes are intended to serve as historical references both for agency staff and for a general readership as well.

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 1992, NRC witnesses testified at 11 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.

In fiscal year 1992, the office obtained the confirmations of Dr. E. Gail de Planque and, for a second term, Dr. Kenneth C. Rogers, as Commissioners of the NRC.

COOPERATION WITH THE STATES AND WITH OTHER FEDERAL AGENCIES

The NRC's program of cooperation with Federal, State and local governments, interstate organizations, and with Indian Tribes, are administered through the Office of State Programs (OSP). The primary goal is to ensure that the NRC has effective relations and communications with these organizations, to promote greater awareness and mutual understanding of the policies, activities and concerns of all parties involved, as they relate to nuclear safety. The office is active in three major and distinct areas: the State Agreements Program; State, Local, and Indian Relations; and the Federal Liaison. The programs are implemented through Headquarters and the Regional Offices.

State Agreements 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 1992, approximately 16,200 radioactive material licenses were administered by the Agreement States, representing about 70 percent of all the radioactive material licenses issued in the United States. An agreement between NRC and the State of Maine became effective during the report period, on April 1, 1992; the agreement does not pertain to low-level waste. The State of Pennsylvania is negotiating a limited agreement with NRC which will give Pennsylvania regulatory authority over the land disposal of byproduct, source and special nuclear material only.

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 examination 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.

Congressional Hearings at Which NRC Witnesses Testified – FY 1992

Date	*Committee*	*Subject*
10/30/91	Committee on Environment and Public Works (Senate)	Commissioner de Planque's Nomination
11/05/91	Committee on Interior and Insular Affairs Subcommittee on Energy and the Environment (House)	License Renewal
11/21/91	Committee on Environment and Public Works Subcommittee on Nuclear Regulation (Senate)	International Nuclear Reactor Safety Standards
01/08/92	Committee on Interior and Insular Affairs Subcommittee on Energy and the Environment (House)	Mill Tailings Disposal (Field hearing in Utah)
01/23/92	Committee on Environment and Public Works Subcommittee on Nuclear Regulation (Senate)	Licensing Reform S. 1220
02/06/92	Committee on Science, Space and Technology Subcommittee on Energy (House)	High-Level Radioactive Waste
02/19/92	Committee on Interior and Insular Affairs Subcommittee on Energy and the Environment (House)	FY 1993 Budget Review
03/12/92	Committee on Appropriations Subcommittee on Energy and Water Development (House)	FY 1993 Appropriations
04/09/92	Committee on Governmental Affairs (Senate)	Radiologically Contaminated Sites
05/06/92	Committee on Environment and Public Works (Senate)	Commissioner Rogers' Renomination

Congressional Hearings at Which NRC Witnesses Testified – FY 1992
(continued)

Date	*Committee*	*Subject*
06/16/92	Committee on Energy and Natural Resources (Senate)	Safety of Nuclear Power Plants in Russia and Eastern Europe

In fiscal year 1992, 13 routine program reviews, 15 review visits, two follow-up reviews, and an orientation visit were carried out. 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. In general, it is the reviewers' judgment that the States are maintaining adequate and compatible programs, in the face of severe budget pressures.

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 I" designation for the more serious concerns. If no significant Category I 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 I 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, by dealing with specialized or unusual radiation applications requiring specialized expertise and knowledge. Specific on-site technical assistance was afforded to six States during the report period.

Training Offered State Personnel by 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 such matters as health physics, industrial radiography safety, well logging, radiation protection engineering, transportation of radioactive and nuclear materials and low-level waste, nuclear medicine, inspection procedures, and materials licensing.

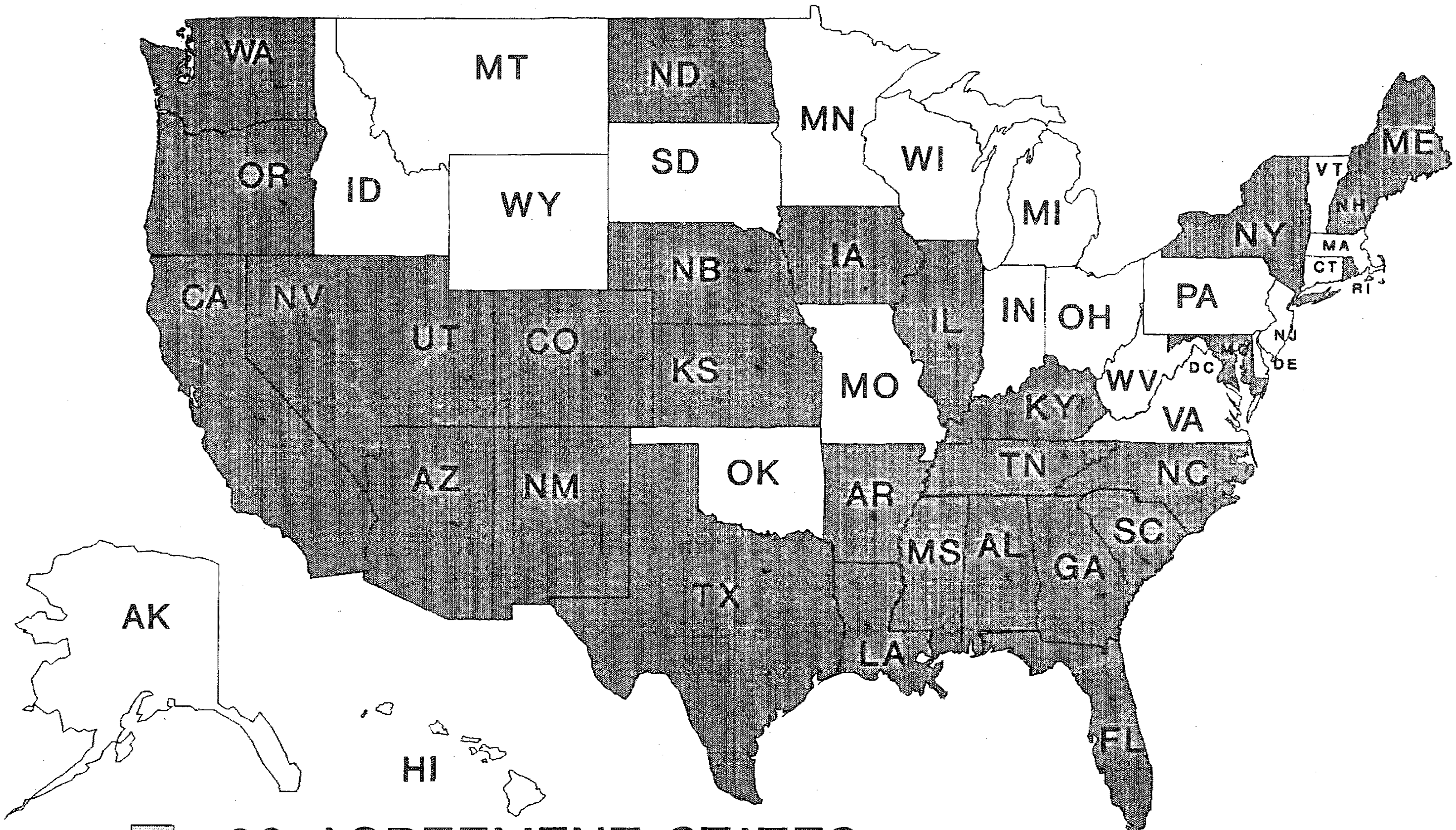
The NRC sponsored 29 such training courses and workshops, attended by 482 State radiation control personnel, during the fiscal year. In addition to State personnel, the sessions were attended by NRC staff and by four military personnel, and also by one individual from the Canadian Atomic Energy Control Board.

Representatives from the 29 Agreement States attended the five special training sessions on the revised Part 20 of Title 10 of the *Code of Federal Regulations,* which were held in January–February 1992 in the NRC Regions.

Total Quality Management Workshops. The Office of State Programs sponsored a pilot program on Total Quality Management (TQM), in Hunt Valley, Md., on January 7–8, 1992. The purpose of the program was to determine the applicability of TQM to Agreement State programs and it was directed toward Agreement State program managers. The program instructor was Jack S. McGurk, a consultant from the California Department of Health Services. Later in the year came a workshop on Total Quality Management for Agreement State managers, on September 2–3, 1992 in Nashville, Tenn., also sponsored by the NRC.

Special Topics Workshops. "Funding Radiation Control Programs with Emphasis on Fee Schedules—

AGREEMENT STATE PROGRAM



29 AGREEMENT STATES

21 NON-AGREEMENT STATES

Effective Strategies for the 1990s" was the topic of an April 28–29, 1992 Special Topics Workshop, held in Bethesda, Md. The workshop objective was to systematically review the budgetary pressures faced by radiation control programs, funding alternatives, and to identify effective strategies for funding these programs in the future. Twenty State personnel participated.

The Office of State Programs sponsored an additional Special Topics Workshop, on September 28–30, 1992 in Houston, Tex. Thirty-seven State personnel, 10 NRC and one Canadian staff members participated in the workshop. The workshop covered a wide range of topics including: implementation of "Part 20"; NRC's memorandum of understanding with the Environmental Protection Agency (EPA); conducting environmental workshops; radiological surveys for license termination; medical waste storage and disposal; contamination in scrap metal; waste compaction; radioactive material in sanitary sewer systems; environmental monitoring; radioactive sludge; regulating irradiators; regulating incinerators; cleaning up contaminated sites; mixed waste; and interim storage of low-level radioactive waste.

Border Monitoring Workshop. The NRC also met with representatives of the U.S. Customs Service, the Mexican Government, and the States of Texas, New Mexico, Arizona and California, in August 1992, in El Paso, Texas. The purpose of the workshop was to discuss radiation monitoring along the Mexican border.

Radiographer Certification Workshop. On May 27–28, 1992, 18 State radiation control program staff and representatives from the Atomic Energy Control Board of Canada, the Conference of Radiation Control Program Directors, Inc, and the American Society for Nondestructive Testing attended a Radiographer Certification Workshop, in Mobile, Ala. The purpose of the workshop was to review and discuss issues associated with third-party radiographer certification rulemaking.

Medical Regulation Workshop. On July 15–16, the NRC sponsored a Medical Issues Workshop, in Atlanta, Ga. Twenty-two representatives from Agreement States participated in the workshop. The purpose of the workshop was to get early Agreement State involvement in the consideration of possible revisions to 10 CFR Part 35, *Medical Use of Byproduct Material.*

Annual Low-Level Waste Regulatory Workshop. The Annual Low-Level Waste Regulatory Workshop was held in Bethesda, Md., in July 1992, 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 August 25–27, 1992, in Denver, Colo. 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, Illinois, Texas, Washington, Utah, and Wyoming participated, along with representatives from NRC program and regional staff.

Annual Agreement States Meeting. The 1991 annual meeting of Agreement States radiation control program directors was held October 27–30, 1991, in Sacramento, Cal. The meeting included panel discussions on low-level waste, material licensing, and materials regulation. This meeting was reported in last year's annual report. The annual meeting for 1992 was held October 26–30, 1992, in Towson, Md., and will be reported in next year's annual report, for fiscal year 1993 (October 1, 1992–September 30, 1993.)

Regulation of Low-Level Waste. The NRC provided technical assistance to the States of Washington, Utah, New York, Nebraska 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, Washington and Nevada continue to participate in the NRC review of several topical reports on high-integrity containers, waste solidification processes, and computer codes to be used in implementing 10 CFR Part 61.

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 reviews, license termination determinations and the conformity of uranium mill regulations with revised NRC regulations. The assistance was provided to the States of Colorado, Texas and Washington.

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 Sacramento, Cal., in October 1991, and in Orlando, Fla., in May 1992, to present plans for future rulemaking. The NRC also consulted with States during development of several other rulemakings.

State, Local and Indian Tribe Liaison Programs

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 forged in an effort to fully address any 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. For example, NRC Chairman Ivan Selin addressed the National Association of Regulatory Utility Commissioners (NARUC) at their annual meeting, in San Antonio, Tex., on November 12, 1991, and Commissioner Kenneth Rogers took an active role in the NARUC proceedings. A delegation from NARUC met with the Commission on March 4, 1992, to discuss economic issues associated with nuclear power plant construction and operations. As a result of that meeting, an ongoing dialogue on issues of mutual interest has been established with NARUC.

The NRC has continued to monitor 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). At its annual meeting in Princeton, N.J., the NGA adopted a policy amendment affirming that licensing and relicensing procedures "must not supplant or interfere with State decisions regarding the need for power, the appropriate energy mix, rate making, land use, or other traditional State responsibilities."

The NRC participated in a meeting of the WGA Waste Task Force, on April 8, 1992, in Denver, Colo. The purpose of the meeting was to discuss procedural and policy issues related to the States' role in local government and State and Indian Tribe applications for monitored retrievable storage (MRS) Phase I and II study grants. Federal agencies were invited to discuss their respective roles and procedures for selection and approval of study grants, evaluating volunteer MRS sites, and the siting and licensing of a MRS facility. State representatives raised other salient issues, including a policy resolution, sponsored by Nevada Governor Robert Miller, regarding the location of an MRS facility.

NRC staff has also actively supported meetings of the National Conference of State Legislatures (NCSL) Working Group on MRS. This forum is intended to provide State legislators with background and information on the various issues related to the MRS program, including State and Tribal relationships and interactions.

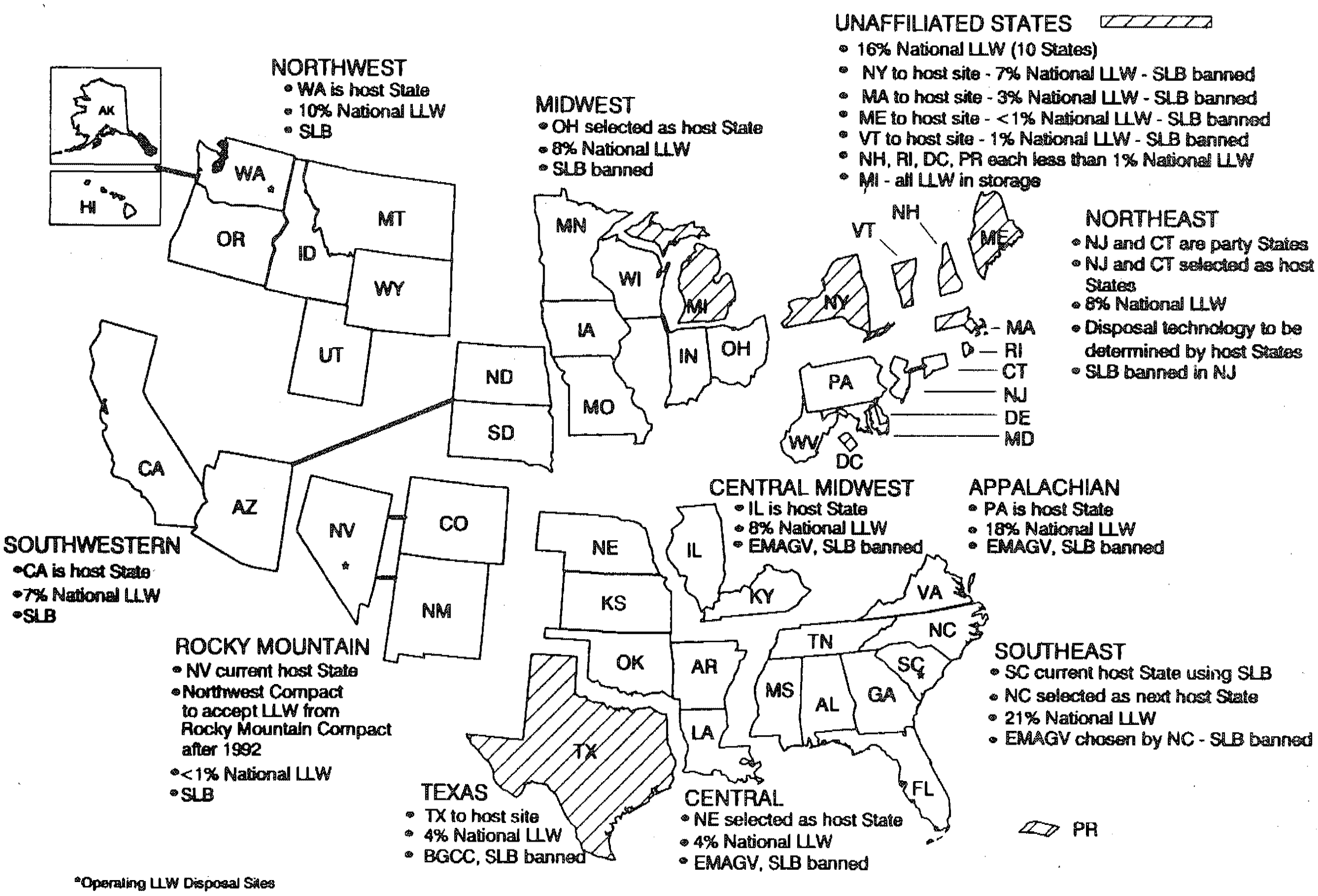
NRC activity with respect to high-level nuclear waste and MRS licensing responsibilities, involving State government and Indian Tribes, has expanded considerably. The agency continues to maintain a good working relationship with the Office of the Nuclear Waste Negotiator and is fulfilling provisions of its Memorandum of Understanding (MOU) with that office; the MOU is generally limited in scope to pre-licensing consultations and discussions, and providing information to potential host States or Indian Tribes.

Cooperation with States. The NRC staff has amended its policy statement on "Cooperation With States at Nuclear Power Plants and Other Nuclear Production or Utilization Facilities (57 FR 6462)." The amendment allows States to observe NRC inspections at reactors which are within the plume exposure pathway emergency planning zone of a nuclear power plant in a neighboring State.

State Liaison Officers 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. SLOs are the NRC's primary point of contact with the States regarding all relevant NRC decisions and actions.

Region III hosted an SLO meeting on February 19–20, 1992, in Glen Ellyn, Ill. Bert Davis and Carl Paperiello, the Region III Regional Administrator and Deputy Administrator, respectively, hosted the meeting, along with Sheldon Schwartz, Deputy Director of the Office of State Programs, and other headquarters and regional staff. The meeting brought together the SLOs, or their representatives, from the Region III States of Illinois, Iowa, Michigan, Minnesota, Missouri, Ohio, Wisconsin and Indiana. Issues discussed included low-level waste (LLW), includ-

LOW-LEVEL RADIOACTIVE WASTE COMPACT STATUS
OCTOBER 1992



Note: National LLW volume for 1991 = 1.4 million cube feet disposed
SLB = shallow land burial
EMAGV = Earth mounded above grade vault
BGCC = below ground concrete canisters

Source: Office of State Programs, NRC

ing interim storage and storage inspections; the enhanced rulemaking process for decommissioning requirements; increased interaction with States on emergency preparedness; Emergency Response Data System (ERDS) update and demonstration; independent spent fuel storage installations (ISFSI), and Minnesota's review of the Prairie Island ISFSI; contaminated sites in Region III; license renewal; and the status of agreements with Region III States.

On May 4–5, 1992, Region I hosted an SLO meeting for the States of Delaware, Maryland, New Jersey, Pennsylvania, New York, Connecticut, Rhode Island, Massachusetts, New Hampshire, Vermont and Maine. The SLOs were particularly concerned with the pace of site decommissioning and compatibility issues. Panel sessions were held on low-level radioactive waste (title transfer and interim storage), emergency preparedness (planning for FFE–3, review of new fuel accident, and State and public outreach initiatives), facility license renewal (10 CFR Part 54, design reconstitution, and the generic environmental impact statement for license renewal), decommissioning activities (participatory rulemaking, site decommissioning management plan, New Jersey site assessment and remediation and other State assessment activities), and NRC and State views on compatibility issues. Special sessions were held with individual regional division directors.

The NRC hosts SLO meetings in the Regional Offices periodically and holds a national meeting at NRC Headquarters every three years. The next national meeting is scheduled for fiscal year 1993.

Emergency Planning. NRC staff from Region III and the Office for Analysis and Evaluation of Operational Data met with emergency response officials from Ohio, on June 4, 1992 in Columbus, Ohio. The purpose of the meeting was to brief State officials on the NRC emergency response program, to include Headquarters and Region III's emergency response organization, the NRC's response to a radiological emergency, the Federal Radiological Emergency Response Plan (FRERP), the Emergency Response Data System, NRC/State liaison during an emergency, and financial assistance. Similar meetings were held with other States in 1992.

Emergency Response Data System (ERDS) MOUs were negotiated with the States of North Carolina, Ohio, Pennsylvania, Michigan, Washington, Alabama and Georgia in 1992. 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. States can have the capability to receive ERDS data during events at power plants, by means of an MOU with the NRC, and these and other States have requested the MOU on ERDS.

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 activity involving State, local government and Indian Tribes. They often attend and participate in State and local meetings, when issues involving the 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 dealing with regional low-level radioactive waste issues and monitor State progress in developing needed capacity for disposal of low-level waste. They also participate in emergency planning exercises involving State and local governments.

Low-level Radioactive Waste Compacts. The Low-Level Radioactive Waste Policy Amendments Act of 1985 set up a series of milestones, penalties and incentives to help ensure that regional compacts and States make adequate progress toward being able to provide for disposal for their low-level radioactive waste by 1993. However, only the Central, Central-Midwest, and Southwestern Compacts met the January 1, 1992 milestone requirement, as their respective "host" States—Nebraska, Illinois, and California—submitted applications for disposal facilities. The State of Texas came into accord with the requirement on March 2, 1992.

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. In support of that decision, the Governor of Illinois ordered a cessation of all efforts to locate a facility at the proposed site at Martinsville. The facility had been scheduled for operation early in 1995. Only two new facilities are now scheduled to be operational by January 1996, one in California and one in North Carolina; the latter will replace the existing Barnwell facility. The host States of Texas, Maine, Massachusetts, Pennsylvania, New Jersey and Vermont are forecast to be operational between the period 1996 and 1999. There are no schedules yet available for the host States of Nebraska, Illinois, Ohio, Connecticut, and New York. The unaffiliated States of Michigan, New Hampshire, Rhode Island, as well as the District of Columbia and Puerto Rico, do not have a disposal site under development. A number of the States believe that they may be able to fulfill their responsibilities through the contracting and/or compact process.

All applications related to mixed waste disposal in the host States are on hold, pending the outcome of consideration by the Department of Energy whether to accept commercial mixed waste for treatment and disposal. The reasons for seeking this alternative solution include the relatively high cost involved, the comparatively small volume of waste, and siting complexities in satisfying both NRC and EPA licensing requirements.

The experience to date regarding the compacts and the States is that schedules or target dates have slipped at all phases of site development for disposal facilities, either because of technical reasons, or litigation, or public or political opposition. Consequently, future target dates cannot be viewed with a high degree of confidence, but rather should be considered the best estimates currently available.

As reported in previous NRC annual reports (most recently in the *1991 NRC Annual Report*, p. 130), New York State, the State of Michigan, and the Concerned Citizens of Nebraska have filed lawsuits seeking to have the 1985 Act declared unconstitutional, because the "take-title provision" exceeds the constitutional limits to Federal imposition on State sovereignty. (The Act provides, in part, that, if a State in which low-level radioactive waste is generated is unable to provide for the disposal of all such waste generated within such State or compact region by January 1, 1996, such State shall, upon the request of the generator or owner of the waste, take title to and become the possessor of the waste, with attendant liabilities.) Defendants in the suits included the United States, the NRC, and the Departments of Energy and Transportation.

On June 19, 1992 the U.S. Supreme Court issued its decision and determination that the "take-title provision" is unconstitutional, but severable from the remainder of the Act. The court determined that monetary and compact exclusion incentives in the Act are constitutional. The lawsuit, which was originally brought in Federal District Court in New York, had been dismissed by both the lower court and the U.S. Court of Appeals for the Second Circuit. Although the short term impact of the Supreme Court decision appears to be minor, the full impact should be felt within the next few years, because the decision removes the last deadline and greatest incentive for the States to develop new disposal facilities.

Action on the Michigan suit was postponed by the Sixth Circuit Court of Appeals pending the outcome of the Supreme Court case for New York. In 1991, the U.S. District Court for the Western District of Michigan granted the Federal Government's motion to dismiss the lawsuit brought against it by Michigan. Besides challenging the constitutionality of the Act, the State included claims brought under the National Environmental Policy Act that the NRC should amend its environmental impact statement for land disposal of low-level radioactive waste (10 CFR Part 61) to take into account the substantial decrease in the amount of waste generated nationwide and the relatively large number, about 13, of sites being developed. The case has been resumed and briefs continued to be filed.

The U.S. Court of Appeals for the Eighth Circuit dismissed, on July 6, 1992, the lawsuit brought by the Concerned Citizens of Nebraska. In a related suit, *Burton v. NRC,* U.S. District Court of Nebraska, June 16, 1992, Diane Burton and Heartland Operation to Protect the Environment, located in Nebraska, complain that the NRC has failed to adopt standards and regulations for land ownership and disposal methods other than shallow-land burial. Briefs continued to be filed by plaintiffs and defendants.

During the fiscal year, the Rocky Mountain Compact Board declared its intention to close the Beatty (Nev.) disposal facility, on January 1, 1993. At the same time, the Northwest Compact Commission intends to exclude out-of-region waste, except for waste from the Rocky Mountain Compact, which will be accepted under contract. The Southeast Compact Commission will allow out-of-region waste to be disposed at the facility at Barnwell, S.C., until July 1, 1994, for an access fee of $220-per-cubic foot. Only generators from States that are currently eligible will be allowed disposal rights initially, and they must sustain progress toward developing alternative facilities. Thus, the States of Michigan, New Hampshire, and Rhode Island, as well as the District of Columbia and Puerto Rico, are ineligible. Other generators may choose to store on-site, based on economic and liability considerations. Because it seems unlikely that there would be no new disposal facilities as of July 1, 1994, widespread storage is expected. Staff estimates that approximately 900 generators (including 48 power reactors), which would normally dispose of their waste, will be faced with on-site storage after this date. State and compact involvement in negotiations for disposal at Barnwell and preparations for storage are adding to the complications surrounding low-level waste disposal.

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, 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. Mailings also include meeting notices, transcripts and letter reports concerning the activities of the NRC's Advisory Committee on Nuclear Waste.

During the past year, NRC staff has met with a number of tribal representatives to hear their concerns and provide information concerning nuclear activities on or near tribal land. Among these were meetings with representatives of the Mescalero Apache Indian Reservation of New Mexico and the Ft. McDermitt Paiute-Shoshone Indian Reservation on the border of Nevada and Oregon regarding the Commission's role in protecting the public health and safety and the program for spent fuel transportation and licensing of a monitored retrievable storage (MRS) facility. Both of these Tribes have received a Department of Energy grant to assess potential interest in becoming host to an MRS.

Interagency meetings are another means by which the NRC keeps up-to-date on American Indian issues. EPA-sponsored quarterly meetings afford the opportunity to exchange new information of potential relevance and importance to Federal and tribal activities. The NRC also maintains liaison with the Department of Interior/Bureau of Indian Affairs in an effort to keep their constituency abreast of nuclear-related issues affecting Indian interests.

Federal Liaison

NRC's Federal Liaison, Maria Lopez-Otin, is responsible for establishing and maintaining effective communications at the policy level between the NRC and other 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 also 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. 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 NRC's Offices of International Programs, of Nuclear Material Safety and Safeguards, and of Nuclear Regulatory Research also serve on various of the FCCSET committees. The Federal Liaison participates in activities of FCCSET committees and subcommittees as either point-of-contact, staff contact, member or alternate. The Federal Liaison reviews and gives input to proposed legislation, rulemakings and correspondence affecting NRC's policy relations with other Federal agencies, and reviews proposed Memoranda of Understanding with them.

COOPERATION WITH OTHER NATIONS

The NRC's international activities, serving the agency's objectives through the Office of International Programs, are intended to:

- Contribute to the safe operation of licensed U.S. reactors and fuel cycle facilities and to the safe use of nuclear materials throughout the world.
- Assist U.S. efforts to restrict U.S. nuclear exports to peaceful uses only.
- Support U.S. foreign policy and national security objectives.
- Improve world-wide cooperation in nuclear safety and radiation protection.

The NRC's international program in nuclear safety involves both bilateral and multilateral regulatory and research cooperation, including extensive interaction with the International Atomic Energy Agency (IAEA), the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA), the European Community (EC), and the G-24 coordinating mechanism for international nuclear safety assistance to the Former Soviet Union (FSU) and Eastern Europe.

Power reactor safety, the primary NRC focus, and materials safety—including radiation protection, waste management, source and by-product materials handling, and international transportation of radioactive materials—are important elements in the NRC international agenda. Nuclear materials safety and safeguards, and export controls on nuclear materials, equipment and technology, are vital aspects of the NRC's statutory responsibilities. Through its international programs, the Commission is continuing bilateral cooperation in nuclear reactor safety with a number of countries, with a special concern currently on Soviet-designed reactors in Eastern Europe and in the FSU.

Key Achievements of Fiscal Year 1992

During the fiscal year, the NRC maintained an active exchange of information and collaborative research with a variety of foreign countries in areas related to safety and security in the civilian uses of nuclear power, to the benefit of both the NRC's regulatory programs and those of the cooperating partners. During the period, the NRC helped develop and implement a significant bilateral program providing assistance in nuclear safety—within the framework of the U.S.–Lisbon Initiative—to Russia, Ukraine and Eastern Europe. The NRC also helped establish an international mechanism in Brussels for the coordination of bilateral and multilateral assistance to nations of the former Soviet Union and of Eastern Europe. The NRC was also a part of U.S. interagency efforts to help Russia and Ukraine develop upgraded systems of accounting and control, as well as physical protection for, nuclear materials. In October 1992, the NRC signed an Information Exchange Arrangement with Indonesia to help that country establish a strong, independent regulatory/safety organization in connection with its projected large-scale nuclear power program. Throughout the fiscal year, the NRC actively supported efforts to develop an International Nuclear Safety Convention, and, after the close of the report period, the NRC received its first high-level visitor from China since the Tienamen Square demonstrations of 1989, to discuss the status of their Protocol on Nuclear Safety, which was subsequently renewed in January 1993.

Highlight Events of Fiscal Year 1992

The following are among the noteworthy accomplishments of the NRC Office of International Programs during the report period.

- Provided support to Chairman Selin as the head of the U.S. delegation to the 36th annual IAEA General Conference in Vienna, Austria. The Chairman delivered the U.S. statement to the General Conference and took advantage of the occasion to hold 13 bilateral meetings with a variety of senior nuclear officials from around the world. Commissioner de Planque, also in attendance, chaired a session of the scientific program for senior regulators. Her session included reports on the nuclear regulatory infrastructure in four newly independent countries.

- Made arrangements for the Chairman to attend a special NEA meeting in Paris of senior regulators from seven major OECD countries (France, Germany, Italy, Japan, Sweden, the United Kingdom and the United States) to discuss developments in affording nuclear safety assistance to Central and Eastern Europe and the FSU, and the formulation of an international nuclear safety convention.

- Planned and coordinated visits by the Commissioners to Belgium, France, Spain, the United Kingdom, Canada, Mexico, Germany, Hungary, Russia, Lithuania, Estonia, Ukraine, South Africa, Japan, Korea, and Taiwan, as well as the EC.

- Continued substantial bilateral cooperation with the FSU through multiple meetings of working groups under the Protocol of the U.S.–U.S.S.R. Joint Coordinating Committee on Civilian Nuclear Reactor Safety.

- Signed an interagency agreement with the U.S. Agency for International Development which made available $3.1 million under the U.S.– Lisbon Initiative to assist the nuclear regulatory bodies in Russia and Ukraine in meeting their responsibilities.

- Contributed to the U.S. effort to work with other G-7 countries to develop an international action plan to enhance the safety of reactors in the FSU and Eastern Europe. The plan served as the basis for the commitment by President Bush and other leaders of the G-7 nations at the July 1992 Munich Economic Summit to assist countries needing such help.

- Signed an Arrangement for the Exchange of Technical Information and Cooperation in Nuclear Safety with Indonesia, a major country in the Far East, which is embarking on a significant nuclear power program.

- Arranged temporary assignments at the NRC for 18 individuals from 13 countries to work alongside NRC staff members in the areas of inspection, technical assessment, emergency response, analysis and evaluation of operational data, accident evaluation, and developing advanced reactor regulatory guidance. Also, for the first time, the NRC placed foreign nationals from the FSU and Eastern Europe (Hungary and the Czech and Slovak Federal Republic) in training assignments at the NRC and sent an NRC staff member for a long term assignment in another country (Japan).

- Participated in several major meetings of experts leading to the development of an international nuclear safety convention.

- Sent 51 participants to IAEA meetings on such nuclear safety issues as radiation protection, waste management, the International Nuclear Event Scale, safety culture, seismic information, software engineering, transport of radioactive materials, decommissioning, aging, future nuclear power plants, research reactor safety, accident management, safety indicators and fire events.

- Coordinated an IAEA Operational Safety Review Team (OSART) mission to the Grand Gulf (Miss.) nuclear power plant. Team members from seven countries and observers from three others participated in the review.

- Provided U.S. experts to participate in six IAEA OSART missions to France (two missions), South Africa, Brazil, Germany and Japan. Sent an NRC expert on an Assessment of Safety Significant Events Team (ASSET) mission to the Fessenheim plant in France.

- Participated in the IAEA's Nuclear Safety Standards Advisory Group meeting in Vienna to review activity related to reactor safety standards, including approval of the document on Fundamentals of Safety for Nuclear Installations.

- Worked closely with the Executive Branch and IAEA in strengthening international safeguards and physical security. U.S. experts visited France, Germany, Japan, South Korea, Hungary, Russia, Ukraine and the Czech and Slovak Federal Republic to exchange information on national physical security programs.

- Participated in the U.S. program to assist Republics of the former Soviet Union in their efforts to improve their national safeguards and physical protection systems.

International Cooperation

U.S.–FSU Civilian Nuclear Safety Cooperation. The NRC inaugurated the planning and development of a new $3.1 million Nuclear Reactor Safety Initiative by hosting a conference of the senior nuclear power regulators of the United States, Russia and Ukraine in July. Proposals were developed and priorities set for assisting the Russian and Ukrainian regulators in training and developing safety standards and procedures. Funding for this program is from the Agency for International Development under the U.S.–Lisbon Initiative to enhance nuclear power safety in the FSU.

With the breakup of the FSU, the NRC has been working with the Department of Energy, the Department of State, and the National Security Council in furtherance of the safe, secure dismantlement of nuclear weapons. The NRC is providing safeguards expertise to Russia and Ukraine, as part of the U.S. program to help ensure the effectiveness of systems for nuclear materials accounting and control and physical security in these countries. The NRC has also provided a list of suggested projects for the new International Science Centers to be set up in Moscow and Kiev to engage former Soviet weapons scientists in non-military pursuits.

Under the Joint Coordinating Committee for Civilian Nuclear Reactor Safety Protocol, the NRC participated in a number of working group meetings in Moscow, Kiev and Washington during fiscal year 1992 to discuss specific nuclear safety issues and to make on-site visits, for the purpose of exchanging operational experience and regulatory information. The aging of components and plant life extension was identified as a dynamic area of mutual cooperation. Since the break-up of the FSU, there has been a dramatic surge of interest in health effects resulting from the Chernobyl accident, and also from earlier accidents—particularly those associated with weapons research and development and nuclear materials production in the Chelyabinsk area.

In late September of 1992, the Chairman visited Russia and met with government and utility representatives, including people from the newly formed utility operating their nuclear power plants (Rosenergoatom), the regulatory body (Gosatomnadzor), the Academy of Sciences, and the Nuclear Safety Institute, to discuss nuclear safety issues and safety assistance to the program. In Ukraine, the Chairman met with the president of the new utility operating their nuclear power plants (Ukratomenergoprom), the Chairman of the State Committee for Nuclear and Radiation Safety, and the Minister of Chernobyl. In both countries he discussed a number of nuclear safety issues, including assistance to their safety and regulatory programs.

Bilateral Information Exchange Arrangements

The NRC participates in a wide range of mutually beneficial programs of information exchange and cooperative safety and research activity with counterparts in the international community. Since 1974, when it formalized the information exchange arrangement program, the NRC has conducted most of its technical regulatory exchanges through a series of 27 general safety cooperation arrangements, signed and renewed over the years, with regulatory authorities in Argentina, Belgium, Brazil, Canada, China, The Czech and Slovak Federal Republic, Denmark, Egypt, Finland, France, Germany, Greece, Hungary, Israel, Italy, Japan, the Republic of Korea, Mexico, The Netherlands, the Philippines, Spain, Sweden, Switzerland, the FSU, the United Kingdom, Yugoslavia (now implemented by Slovenia) and Taiwan.

These arrangements provide for communications channels with foreign nuclear regulatory organizations, in order to ensure prompt reciprocal notification of reactor safety problems that could affect either U.S. or foreign nuclear facilities and which assist 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. These arrangements are usually effective for five years, and they include provision for renewal by mutual written agreement of the parties.

During fiscal year 1992, the NRC renewed its information exchange and cooperation arrangements on nuclear safety matters with The Netherlands and Switzerland and continued active negotiations on the renewal of its arrangements with Germany, Japan, the Philippines and the United Kingdom. At the end of the fiscal year, the agency was completing final arrangements to conclude a first safety agreement with Indonesia.

France. Because of the importance of their respective programs and activities, the NRC and the nuclear establishment of France continued their regular cooperative activities. During the year, Chairman Selin and Commissioners Remick and de Planque made official visits to France to exchange information with key nuclear officials and to visit a number of the nuclear facilities. The Chairman also addressed a meeting of the French Nuclear Society.

For their part, the General Administrator of the Commissariat a l'Energie Atomique, the Chairman of Framatome and an Executive Vice-President of Electricite de France made visits to the NRC for discussions with the Commissioners. There was also a regular exchange of visits at the staff level to discuss current operational issues and plans for advanced reactor designs.

Spain. NRC Commissioner Forrest J. Remick visited Spain in July to make a presentation at the TRAC Computer Code Users Group meeting in Santander. He also met with government, utility and industry representatives for discussions on nuclear safety issues—especially those related to waste management. He met with officials from the Consejo de Seguridad Nuclear (CSN) and ENRESA (the waste management company) in Madrid, and visited the Garona and Vandellos reactor sites, as well as the low-level waste facility in El Cabril.

In March, Dr. Rafael Caro, Commissioner of the CSN, participated in the NRC Aging Research Conference in Washington and later held discussions with Commissioners on the international nuclear safety convention, the acceptability of the new, proposed International Commission on Radiological Protection (ICRP) standards, consensus among European Community countries on resolution of technical problems with nuclear plants in Eastern Europe, Spain's relationship with Eastern European countries, the Spanish nuclear program, upgrades to the Spanish Zorita nuclear plant and the distribution of nuclear regulatory responsibilities within Spain.

United Kingdom. In January, Commissioner Forrest J. Remick made an official visit to the United Kingdom to discuss NRC nuclear safety cooperation and cooperation with Eastern Europe and the FSU. While in England, he also toured the URENCO enrichment facility and the Sizewell-B nuclear power plant.

Sweden. In September, Harold Denton, Director of the NRC Office of International Programs, visited Sweden to discuss cooperative assistance programs with Central and Eastern Europe and other topics. He met with Ambassador Ove Heyman, Head of the Secretariat for Coordination of Relations with Central and Eastern Europe of the Foreign Office, and officials from the Swedish Nuclear Power Inspectorate.

Canada. Chairman Selin visited Canada in September. He met with Canadian officials at the Atomic Energy Control Board, their regulatory agency; Energy, Mines and Resources; External Affairs (equivalent to the State Department); Atomic Energy of Canada Limited; and Ontario Hydro. He also toured the Darlington and Pickering nuclear power plants.

Mexico. In June, Commissioner de Planque was a keynote speaker at the Latin American Nuclear Society meeting in Veracruz, Mexico. During her visit, she also met with senior nuclear officials at the Mexican Ministry of Energy, Mines and Industrial Paraestatals; the Mexican nuclear regulatory organization; the Federal Electricity Commission; the utility operating the Laguna Verde reactor; and ININ, the main nuclear research organization in Mexico. She also visited the Laguna Verde nuclear power plant, which recently completed its second year of operation.

Germany. In December, Commissioner Rogers visited Germany to give the keynote address at the "Plex '91 Berlin" Conference on Nuclear Power Plant Management and Life Extension, sponsored by Nuclear Engineering International. He presented an overview of aging problems in nuclear plants, ways of managing aging, and the approach to license renewal set forth in 10 CFR Part 54.

In April, Dr. Klaus Toepfer, the German Minister for Nuclear Safety and Environment (BMU), and Dr. Walter Hohlefelder, Assistant Secretary for Nuclear Safety (BMU), met with NRC officials in Washington to discuss issues of mutual interest regarding Western nuclear safety assistance to the FSU and Eastern Europe, and to clarify differences of view between U.S. and German positions on several aspects of the international nuclear safety convention.

Eastern Europe. In September 1992, following the IAEA General Conference, Chairman Selin visited Hungary to hold discussions with Hungarian Atomic Energy officials and U.S. Ambassador Thomas. The Chairman also visited the Paks nuclear power facility, where he was briefed on operations there by senior plant officials.

The NRC held trilateral meetings in Washington with Hungary and the former Czech and Slovak Federal Republic (CSFR, subsequently denominated the Czech Republic and the Slovak Republic) in December 1991 to discuss assistance and to identify areas for cooperation, pursuant to information exchange arrangements with the NRC, and in June, in Prague and Budapest, for discussions and exchanges of technical information on those topics.

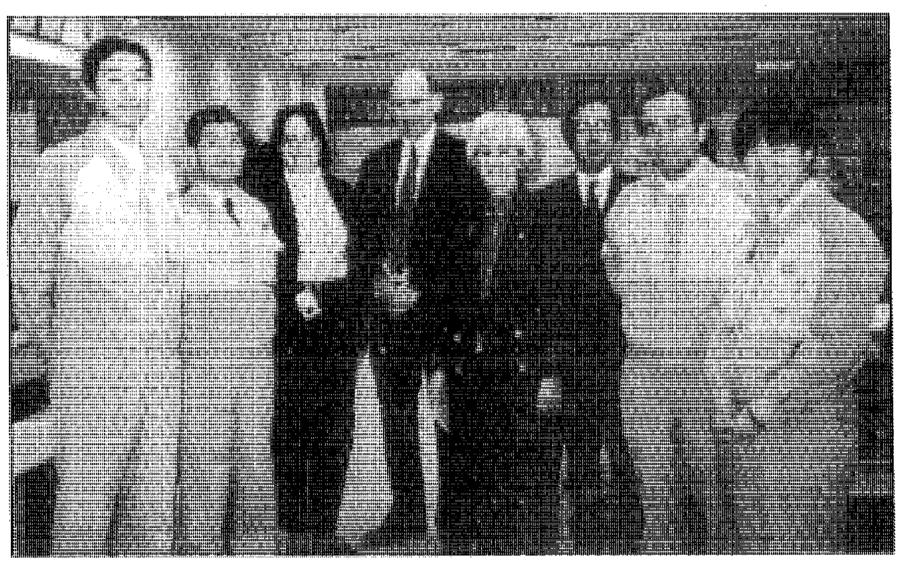
In September, the chief Bulgarian regulator, Dr. Yanko Yanev, met with the Chairman and other Commissioners in Washington to review the operational safety of the Kozloduy plants and to inform them of corrective measures being implemented at the plants. In a separate meeting with the Chairman, Dr. Yanev identified several key safety areas in which NRC assistance is being sought.

In August, Chairman Selin received a request from Dr. Karel Wagner, Chairman of the then CSFR Atomic Energy Commission, for NRC assistance in helping his agency perform safety reviews of the instrumentation and control system and fuel for the VVER-1000 reactors, under construction at Temelin. Westinghouse Corporation has been awarded contracts to supply these systems for the Temelin plant, and the NRC staff has agreed to provide short-term safety assistance in the United States in the specified areas.

In September, Commissioner Kenneth Rogers, OIP Director Harold Denton, and a senior staff member participated in and presented papers at the USAID-sponsored "US Executive Workshop on Nuclear Safety and Power Sector Reform in Eastern and Central Europe." The purpose of the workshop was to clarify nuclear safety

Commissioner E. Gail de Planque visited the French Central Service for Protection Against Ionizing Radiation during the report period. The Commissioner is shown here with her Technical Assistant Joel Lubenau, at left, and Professor Pierre Pellerin, past-Director of the Service, in a gamma-spectrometric railway coach that would be activated in the event of a nuclear emergency.

Here Commissioner de Planque inspects an irradiation facility in Toluca, Mexico. The facility is part of the Instituto Nacional de Investigaciones Nucleares and is used in support of Mexico's national agricultural, industrial and medical programs.



Commissioner de Planque, her Technical Assistant Eileen McKenna (third from left), and Ronald Hauber, NRC International Programs Assistant Director (fourth from left), are touring the control room of the Japan Atomic Company's Tsuruga Unit 2 nuclear power plant. The 1,160-megawatt pressurized water reactor plant began operation in 1987; it is the first power reactor in Japan with a pre-stressed concrete containment vessel.

concerns and initiatives in the area for U.S. business representatives, in the broad context of nuclear power sector restructuring, to enable them to guide the direction of U.S. business activities in Eastern Europe. Other U.S. participants in the workshop included representatives of electric power service and supply companies, electric utilities, and government agencies. East European attenders included representatives of the utilities and nuclear regulatory bodies.

South Africa. In January, Commissioner James R. Curtiss visited South African nuclear officials and made visits to nuclear facilities in the vicinity of Cape Town, Springbok, and Johannesburg. The purpose of the trip was to explain U.S. nuclear safety concerns to South African officials and to signal a new opening in relations between the two countries on nuclear issues, following on South Africa's adherence to the nuclear Non-proliferation Treaty in 1991 and acceptance of full-scope safeguards. Commissioner Curtiss also wished to initiate discussions on nuclear safety, radiation protection, and waste management issues, and especially to convey the basics of the NRC's regulatory approach, its commitment to worldwide nuclear safety, and its willingness to exchange safety information pursuant to these goals.

Japan. In April, Chairman Selin visited Japan where he addressed the 26th annual Japan Atomic Industrial Forum, at which many of the papers focused on the need for assistance to reactor operations in Eastern Europe and the countries of the FSU. The visit included discussions and tours at Tokyo Electric Power Company's Kashiwazaki Kariwa nuclear power plant and the Japan Atomic Energy Research Institute's Rosa IV facility, at Tokai.

Korea. Chairman Selin also visited South Korea in April for discussions and a visit to the Yonggwang nuclear power plant, operated by the Korea Electric Power Company.

Indonesia. Commissioner Remick led a U.S. delegation to the U.S.-Indonesia Joint Steering Committee on Nuclear Energy (JSC) in Indonesia, in February 1992. He held discussions and made site visits to identify and explore possible avenues of safety cooperation with the NRC, with special attention to the development of a strong regulatory regime. Upon his return, Commissioner Remick reported his discussions and key observations to the Commission, which subsequently agreed that the NRC should help Indonesia develop its regulatory program—through information exchanges, training, and on-the-job experience—to the extent possible within established limits of resources and legislative authority. In October 1992, during a visit of high-level Indonesian officials to the NRC, the first Information Exchange Arrangement with that nation was signed, providing a

Chairman Selin, at right, accompanied by International Program staff, discusses operations with control room personnel during a visit to Korea's Yonggwang nuclear power plant in April 1992.

framework for all NRC-Indonesian cooperation in support of nuclear safety.

Taiwan. Commissioner Remick visited Taiwan in April to give one of the keynote addresses at the Eighth Pacific Basin Nuclear Conference and took the opportunity to hold discussions with senior nuclear officials and make site visits to their research facilities and the Kuosheng nuclear power station. While in Taiwan, the Commissioner met with Dr. Y.Y. Hsu of the Atomic Energy Council, Dr. Hsia, Director of the Institute of Nuclear Energy Research, and senior representatives of the RadWaste Administration and the Taiwan Power Company.

Safety Exchange Activities

Regular bilateral exchange meetings and discussions continued throughout the year. A high-level technical staff team, including senior representatives from NRR and NMSS, met with their French counterparts from the Directorate for the Safety of Nuclear Installations and visited French waste facilities. The discussions included a number of current issues dealing with operational reactors, advanced reactor licensing, and high- and low-level waste management safety. Separate bilateral meetings were held with the safety authorities of Germany and Switzerland to discuss instrumentation and control (I&C) issues. An ad hoc multilateral group, consisting of representatives from Canada, Germany, the United Kingdom and the NRC held a meeting to discuss technical safety aspects of advanced I&C systems. Based on the constructive results of this initial meeting, the group decided to meet periodically in the future. Finally, a bilateral meeting was held with the U.K. to exchange information on current safety topics.

Participation in International Organizations and Conferences

IAEA General Conference and Board of Governors Meetings. Chairman Selin, as noted above, headed the U.S. Delegation to the 36th Session of the General Conference (GC) of the IAEA in Vienna in September 1992. While there, he took part in bilateral discussions with delegations from Korea, Bulgaria, Indonesia, Russia, the United Kingdom, Czech and Slovak Federal Republic, Argentina, Ukraine, Japan, Germany, Finland, Brazil and China. Commissioner de Planque was a member of the U.S. delegation and chaired a session of the scientific meeting scheduled for senior regulators. During that session, reports were given by Slovenia, Lithuania, Russia and Ukraine on their regulatory infrastructures. NRC officials also attended both the February and June sessions of the IAEA Board of Governors.

International Nuclear Safety Convention. The NRC participated throughout the year in discussions of an international nuclear safety convention. Such a convention would commit signatory governments to abide by certain basic safety principles, which could improve safety in the countries of the former Soviet Union and Eastern Europe, and also in several new states hoping to use nuclear power to generate electricity. The NRC has been actively involved in meetings taking place during the year in Vienna, to try to reach a consensus on the elements of the convention. The United States has taken the view that such a convention should be limited to civil nuclear power plants and contain general safety principles, along the lines of the IAEA Safety Fundamentals document. Negotiations will continue in 1993.

OSARTS. The NRC arranged for Entergy Operations, Inc. to host an Operational Safety Review Team (OSART), organized by the IAEA, at their Grand Gulf plant in Mississippi from August 3-to-21, 1992. Experts from the IAEA staff, Canada, France, the United Kingdom, Spain, South Africa, Japan, Finland, Sweden, and Germany were on the team. Observers from the Czech and Slovak Federal Republic, Mexico and Bulgaria also accompanied the team. Grand Gulf is a boiling water reactor facility, located near Port Gibson, Miss.

The OSART inspection report advised that the team was greatly impressed with the commitment of management and staff to the achievement of high levels of safety in the operation and maintenance of the plant. The OSART found that the utility (Entergy Operations, Inc.) was well managed and actively supported Grand Gulf operations by providing clear policy direction and adequate resources. The OSART also found the plant management and supervisory staff to be dedicated, the operating and maintenance personnel to be well trained and highly motivated, and good technical support at both the corporate and plant levels.

The OSART report included a number of recommendations for consideration by Entergy management and Grand Gulf operating personnel. The utility will prepare a detailed response to the final report. Conclusions reached by the OSART are in substantial agreement with the NRC assessment of the performance of the Grand Gulf plant and licensee management over the past several years.

An NRC staff person also participated as a maintenance expert on an OSART mission to the Blayais nuclear power plant in France. The NRC arranged to have U.S. utility experts take part in OSART missions to South Africa, Germany, Brazil, Japan, Korea, and a second mission to France.

International Basic Safety Standards. A major effort is under way to update the International Basic Safety Standards, in light of the International Committee on Radiation Protection's Publication 60. The NRC has led the U.S. response to drafts which are being revised for another round of comments and is promoting careful review on a schedule which will allow for careful development of an international consensus.

IAEA Officials Visit NRC. During the report period, the IAEA Deputy Director General for Nuclear Safety, Boris Semenov, visited the NRC Commissioners to discuss the international nuclear safety convention, nuclear regulatory assistance to the former Soviet Union, the International Nuclear Event Scale, and other issues. IAEA Nuclear Safety Division Director, Morris Rosen, who came to the United States for the Grand Gulf OSART (see above), also visited the NRC for consultations with Commissioners and senior managers.

Establishing an NRC Position at the U.S. Mission in Vienna. The Commission endorsed the establishment of a senior nuclear safety position for an NRC representative at the U.S. Mission to International Organizations, in Vienna. The person selected will serve as the Mission's expert on nuclear safety and radiation protection issues arising within the IAEA and other diplomatic missions in Vienna and will provide programmatic and policy oversight, on behalf of the United States, of the IAEA's nuclear safety program.

Nuclear Energy Agency (NEA) Activities. The NRC maintained an active involvement in OECD/NEA activities by serving on key standing committees and working groups and participating in the U.S. delegation to two Steering Committee meetings during 1992. Also during the year, NEA Director General Uematsu visited the United States to discuss matters of mutual concern.

In September, Chairman Selin attended a special NEA meeting, which he had initiated, for the heads of the nuclear safety organizations of seven major OECD countries—France, Germany, Italy, Japan, Sweden, the United Kingdom and the United States—to discuss developments regarding nuclear safety for Central and Eastern Europe and the former Soviet Union, and, in particular, the G-7 Summit Meeting in Munich in July. They also discussed the international nuclear safety convention.

European Community. Mr. Laurens Brinkhorst, Director General, DG XI (Environment and Nuclear Safety), met with Commissioners Rogers and Remick and senior members of the staff in early January, in Washington. Mr. Brinkhorst noted the large financial commitment that the European Community (EC) is making for reactor safety improvements in the former Soviet Union (FSU) and Eastern Europe. The Director General was pleased to hear of the extensive collaboration that the NRC has had with the nuclear organizations in the FSU, and proposed that the EC and NRC work closely together, so that NRC could offer advice on effective uses of the EC's assistance money. Senior NRC staff briefed EC officials in detail regarding NRC's cooperative programs with the FSU.

In September, Commissioner de Planque made the first Commission-level visit to the EC in Brussels, reflecting the increasing importance of NRC cooperation with the EC. During the visit, Commissioner de Planque engaged in discussions with the three Directors General of the Directorates for Energy, Environment and Nuclear Safety, and the Joint Research Center. A particular highlight of the meetings was the expression of common concern about reactor operations in Eastern Europe and the FSU and of each organization's willingness to cooperate in providing safety assistance.

G-7 Summit Nuclear Safety Initiative. The seriousness of the safety concerns regarding Soviet-designed reactors led this year to a major international effort to establish a concrete and effective approach to resolve them. The G-7 Countries (Canada, Japan, Italy, France, Germany, the United Kingdom, and the United States) meeting in Munich in July recommended a program of action developed during a series of meetings of a Nuclear Safety Working Group in Cologne in May. That plan provides for immediate, but relatively low levels of, support during the next two years to bring fast safety fixes on all operating Soviet-designed reactors, including better procedures, operator training, funding of independent regulatory inspectors and fire protection initiatives. It was also recommended that a broader, long-range program for improving the safety of these plants be adopted. Funding is provided by the Lisbon Initiative (announced earlier by the Secretary of State at a conference in Lisbon), and includes a $25 million multilateral nuclear safety initiative, of which approximately $3 million will be devoted to nuclear safety regulatory cooperation.

Export-Import and Non-proliferation Activities

NRC Export License Summary. Under the Atomic Energy Act of 1954, as amended, the NRC is responsible for licensing the export of nuclear-related materials and equipment. This export 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 its responsibilities for exports, the NRC obtains the views and recommendations of other governmental agencies and departments, as needed or required.

In fiscal year 1992, the NRC completed 156 export licensing actions. Of these, 74 involved exports of low-enriched uranium fuel for various power reactors around the world using uranium of U.S. origin or uranium enrichment services of the Department of Energy. Countries using the low-enriched uranium fuel include the EURATOM nations, Japan, South Korea, Sweden, Switzerland and Taiwan.

The NRC also issued two licenses authorizing the export of 105 kilograms of high-enriched uranium for use in research and test reactors in (1) Belgium (32 kilograms for the BR-2 reactor) and (2) Canada (73 kilograms for use over the next three years as target material in the NRX-NRU reactors).

Other significant export action includes the issuance of one license authorizing the export of two pressurized water reactors to South Korea and one license authorizing the export of a TRIGA Mark II research reactor to Morocco.

NRC Consultations with the Executive Branch on Nuclear-Related Export Matters. The NRC consults with the Executive Branch on nuclear-related, dual-use exports licensed by the Department of Commerce, as well as nuclear technology transfers and nuclear material retransfers licensed by the Department of Energy. In fiscal year 1992, there were several transfers of safety-related nuclear power reactor technology to the republics of the former Soviet Union and East European countries.

During the year, the International Nuclear Suppliers Group completed work on the establishment of multilateral export control guidelines for nuclear-related, dual-use items. While the Department of Commerce has licensing responsibility for most of the items on the list, some commodities are licensed by the NRC. The NRC staff is in the process of implementing the new guidelines, which will require some changes to NRC's export licensing regulations in 10 CFR Part 110.

Centrifuge Enrichment Agreement. The United States signed an agreement on July 24 with the governments of The Netherlands, Germany and Great Britain on the establishment, construction and operation of a private uranium enrichment plant in the United States. The agreement addresses safeguards and physical security matters, including controls on classified and other sensitive information related to the proposed Louisiana Energy Services centrifuge enrichment facility, in Louisiana. The technology to be used in the plant was developed in Europe by the URENCO consortium, under the purview of the governments of The Netherlands, Germany and Great Britain.

Non-Proliferation, International Safeguards and Physical Protection. The NRC staff reviews pending export cases in order to confirm the application of IAEA safe-guards and physical security arrangements to the exports by the receiving country. The reviews are performed in conformance with U.S. non-proliferation laws, intended to ensure that U.S. exports will be protected and safeguarded during transit and use in the importing country and that exports will not be used for non-peaceful purposes. To fulfill this function, the NRC staff participates in U.S. Government efforts to assist the IAEA in improving its safeguards system.

The NRC is also part of the Subgroup on Nuclear Export Coordination (SNEC), an interagency body that oversees U.S. nuclear export controls. The SNEC primarily focuses on actions to be taken in response to Department of Commerce license requests for "dual-use" nuclear items (items that may have applications in nuclear reactors and may also be turned to weapons-production purposes). The meetings are chaired by the Department of State and include representatives from the Departments of Commerce, Defense, and Energy, and the Arms Control and Disarmament Agency. The SNEC constitutes a mechanism for the NRC and the Executive Branch agencies to keep abreast of non-proliferation matters and issues and to exchange views on proposed exports to facilitate decision-making.

The U.S. Program for Technical Assistance to IAEA Safeguards (POTAS) furnishes the largest share of voluntary technical support by IAEA member states. In 1992, the NRC provided one staff member to the IAEA Department of Safeguards, to take part in a POTAS-funded research project on the effectiveness of IAEA safeguards. The focus of most POTAS activities during 1992 was on the expansion of the IAEA's international safeguards activities through the application of new methods and techniques, introduced as a complement to 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 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 1992, 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 also participated in an interagency working group to review the effectiveness of international safeguards. This group is charged with generating proposals to strengthen the nuclear non-proliferation regime that go beyond the technical scope of existing working groups.

In support of its review of physical protection arrangements for U.S.-controlled materials in other countries, the NRC participates jointly with other U.S. Government agencies in information exchange trips, for the purpose of discussing national physical protection programs. During fiscal year 1992, U.S. delegations visited France, Germany, Japan, South Korea, Russia, Ukraine, Hungary and The Czech and Slovak Federal Republic. The NRC also participated in an international conference in Vienna to review the Convention on the Physical Protection of Nuclear Material.

Safeguards Support to Republics of the Former Soviet Union. The NRC is participating in the U.S. program to assist Republics of the Former Soviet Union in their efforts to enhance national systems of Material Control and Accounting (MC&A) and physical protection, with the basic goal of developing effective regulatory systems. To date, substantive discussions have been held and the negotiation of both an umbrella agreement and attendant implementation are under way with Ukraine, which had formally requested MC&A and physical protection support in July 1992.

Nuclear Regulatory Research

Chapter 8

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 carry out the rulemaking process, including the issuance of regulatory guides and rules that govern NRC licensed activities. (See "Regulations and Guides" below.)

Regulations issued by the NRC in 1992 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 1992.

The NRC supports the Small Business Innovation Research (SBIR) program, which stimulates technological innovation by small businesses. 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 1992, the NRC was supporting 20 SBIR projects-in-progress.

This chapter summarizes RES activities during fiscal year 1992 under the following major headings: Reactor Licensing Support, Reactor Regulation Support, Nuclear Materials Licensing and Regulation Support, and Assessing the Safety of High-Level Waste Disposal.

Reactor Licensing Support

STANDARD REACTOR DESIGNS

Engineering Issues for Advanced Reactor Designs

Design of Low-Pressure Piping for Intersystem LOCA. Development of criteria for a new advanced light-water reactor (ALWR) design goal—that of the reactor's withstanding reactor coolant pressures and temperatures in low-pressure piping attached to the reactor coolant loop—was initiated in fiscal year 1992. The condition in question, which could follow from multiple valve failures, is important because 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 the causal conditions, the performance goal is to achieve a failure probability in the low-pressure piping of about 10 percent. A probabilistic methodology originally developed to evaluate ISLOCA in operating plants was extended to gauge permissible pipe stresses for both carbon and stainless steel in ALWRs. Work continues to provide criteria for other piping components—including flanges, valves, pumps, heat exchanger tubes, and instrument lines.

Experience-Based Seismic Qualification. The Electric Power Research Institute (EPRI) has proposed, in its Utility Requirements Document (for the ALWR), the use of experience as a method of seismic qualification in ALWRs, as an applicable substitute, case-by-case, for more traditional tests and evaluations. The rigor needed and the equipment categories suitable for this use of experience have not been defined in detail. An expert panel has been established to assess EPRI methods and criteria, as well as the caveats, associated with the use of experience in making these judgments on ALWRs, focusing primarily on equipment and excitation similarity standards.

REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth requirements that must be met by NRC licensees in Title 10, Chapter I, of the *Code of Federal Regulations.*
- Regulatory guides, usually to describe methods acceptable to the NRC staff for implementing specific portions of NRC regulations.

When the NRC proposes new or amended regulations, they are normally published in the *Federal Register* to allow interested persons time for comment on them before they are adopted. This step is required by the Administrative Procedure Act. Following the public comment period, the regulations are revised, where appropriate, to reflect the comments received. Once adopted by the NRC, they are published in the *Federal Register* in final form, with the date on which they become effective. After publication, the regulations are codified and annually incorporated into the *Code of Federal Regulations.*

Some regulatory guides describe techniques used by the NRC staff to evaluate specific situations. Others provide guidance to applicants concerning the information needed by the staff in its review of applications for permits and licenses. Many NRC guides refer to or endorse national standards (also called "consensus standards" or "voluntary standards") that are developed by recognized organizations, often with NRC participation. The NRC makes use of a national standard in the regulatory process only after an independent review by the NRC staff and after review of public comment on the NRC's planned use of the standard.

The NRC encourages comments and suggestions for improvements in regulatory guides and, before staff review is completed, issues them for comment to many individuals and organizations, along with the value/impact statements that set forth the objectives of each guide and both its expected effectiveness and impact, in terms of resources and effort involved.

Containment Performance Goals. In support of the NRC Severe Accident Policy Statement, as it applies to ALWRs, work began on development of design criteria 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 in the Commission's safety goals. Particular attention will be given to evaluating the severe accident design criteria for containments cited in the EPRI Utility Requirements Document.

Qualification of Advanced Instrumentation and Control Systems. Under the auspices of the NRC, the Oak Ridge National Laboratory (ORNL) is conducting a study to identify functional and environmental issues arising from the application of new technologies to instrumentation in the next generation of nuclear power plants. Specifically, the program seeks a thorough understanding of the technical issues involved in evaluating long term properties and performance of "advanced" digital instrumentation and control (I&C) systems proposed for use in ALWRs. Emphasis has been given to identifying vulnerabilities and environmental limitations that could be imposed on microprocessor-based systems in nuclear environments. Initial studies have focused on protection systems and the I&C of engineered safety feature actuation systems. The environmental and functional issues studied thus far are reported in the draft NUREG/CR-5904. In this document an evaluation template is presented which was developed by assembling a reasonably complete configuration of a safety channel instrument string for the ALWR used in the study, then comparing the impact of environmental stressors on that string to their effect on an equivalent instrument string from existing light-water reactors. Functional issues considered in the templates include distribution of function, sources and delivery of electrical power, calibration and testing capabilities, and failure prediction based on environmental monitoring. The application and acceptance of digital computers in reactor protection systems are reviewed in light of current standards.

Valve Operability. The proper operation of valves in existing nuclear power plants is obviously necessary for plant safety. The NRC has concerns about the capability of some of these important components, particularly motor-operated valves, to function as they should when they are needed for safety, and operating nuclear power plants have instituted programs addressing those concerns. The NRC is using valve research results to evaluate the programs at the nuclear power plants; the same effort is being carried over to valve operability in advanced reactor designs.

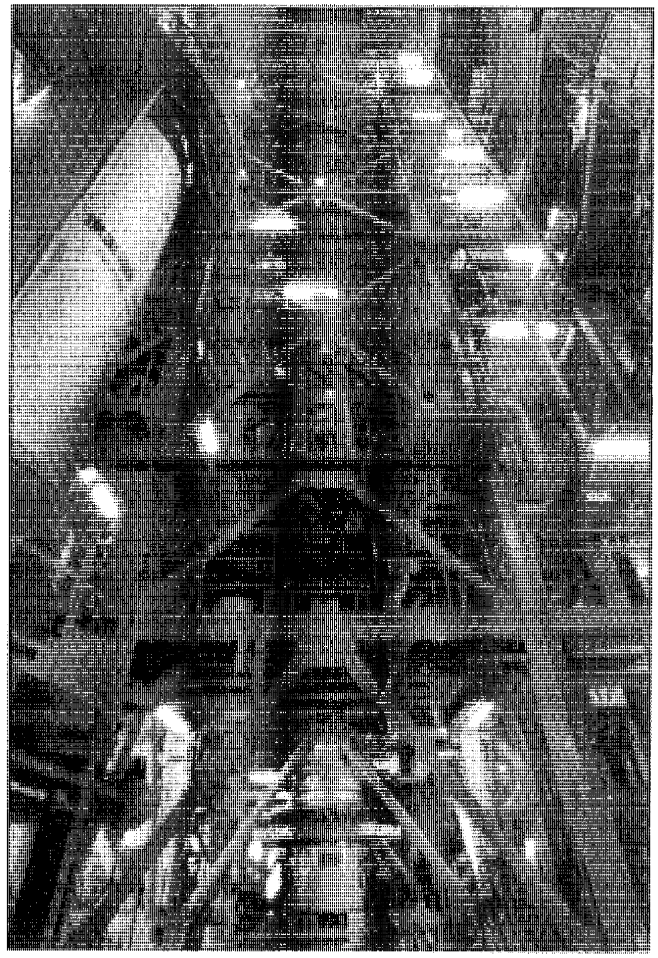
During the past year, a research program was started to identify whether valves expected to be used in the advanced reactors involve any operability problems. Familiarity with system and component functions was gained in meetings with the component designers from nuclear plant vendors. Extensive information, including qualification test data on specific kinds of valves, has been requested from the vendors and will be evaluated upon receipt. This task, which is being conducted to support the NRC licensing office, will continue over the next year.

Systems Performance of ALWRs

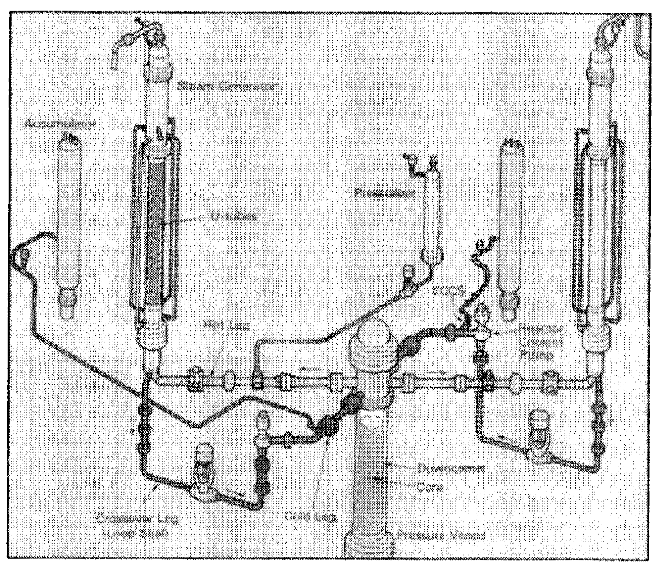
In June 1992, Westinghouse submitted the AP600 advanced reactor design to the NRC for certification, while General Electric followed, in August 1992, with its SBWR design. These designs differ from current plants in that they incorporate passive safety features to deal with incipient accidents. Passive features rely upon natural circulation and gravity to provide coolant makeup and long term core cooling, as distinct from traditional pump-driven systems. Evaluation of such features involves a new application for NRC computer codes, such as RELAP. A program is under way to improve and validate RELAP for use in appraising these designs and, eventually, for quantifying code uncertainty. Areas needing improved modeling have been identified, and code development is in progress and planned for completion in 1993. An overall validation plan was developed which relies primarily on the use of experimental results provided by Westinghouse and General Electric, as well as results from experiments to be performed by the NRC.

In the case of AP600, these experiments will be carried out at the Japan Atomic Energy Research Institute in the ROSA facility. ROSA is an existing large-scale experimental facility modeled after an existing Westinghouse design. The NRC performed an extensive evaluation to decide how best to modify the facility for AP600 testing. The modifications will be carried out in 1993 and the experimental program in 1994.

For SBWR testing, the NRC developed general requirements and objectives to serve as the basis for a competitive contract award to create a new facility. The facility will provide experimental data on the passive decay heat removal features of the SBWR design.



The ROSA-IV Large-Scale Test Facility (LSFT), shown above, will be used to investigate the thermal-hydraulic behavior of pressurized water reactors of advanced design. Below is a schematic of the LSFT, from which data will be developed on small-break loss-of-coolant accidents, on long term cooling by natural circulation, and on techniques for plant recovery under accident conditions.



Systems Performance of Other Advanced Reactors

During fiscal year 1992, the NRC conducted initial research on systems performance of four other advanced reactor designs not directly descended from today's pressurized or boiling water reactors. These designs are (1) General Electric's and the Department of Energy's (DOE) Advanced Liquid Metal Reactor (ALMR), (2) General Atomic's and DOE's Modular High-Temperature Gas-Cooled Reactor (MHTGR), (3) Asea Brown Boveri—Combustion Engineering's Process Inherent Ultimate Safety (PIUS) reactor, and (4) Atomic Energy of Canada Limited's Canadian Deuterium Natural Uranium Model 3 (CANDU 3) reactor. The NRC is currently reviewing pre-application submittals from the four vendors.

As part of the systems performance research for these reactors, systems engineering studies are being conducted to identify important accident sequences and safety systems. More detailed analyses of the accident sequences identified thus far have been initiated, and independent analytic capabilities are being developed for use by the NRC staff in future reviews of these designs for certification. Research is also under way to provide early assessments of severe accidents and source terms, in support of NRC's preliminary safety evaluation reports for each of the four designs. In their entirety, the systems performance studies are helping to identify any additional research and testing needs that may require attention from the NRC and the prospective applicants.

Advanced Reactor Risk Analysis

Passive System Reliability Project. The advanced passive reactors have engineered safeguards systems that maximize the use of passive devices and features, such as nitrogen-powered accumulators, natural circulation flow, and gravity-driven safety injection. They do not rely on active systems, such as a.c. electric-powered equipment, although certain valves may require stored energy (e.g., battery power) to change state. These passive designs are expected by the designers to both increase safety and decrease cost, because of their simplified design. However, in the absence of actual working experience with the designs and because of uncertainties in the modeling of processes such as natural circulation, there are uncertainties in the performance of the engineered safeguard systems.

The purpose of the passive system reliability project is to quantify the uncertainty in predicting core damage frequency; the project is currently focused on the Westinghouse AP600 design. In addition to assessing the effects of the uncertainties in natural processes, the project seeks to identify potential accident initiators that may be unique to the design and to estimate the frequencies of these initiators. Both the reliability analysis model and the analysis of potential accident initiators are expected to be completed in 1993.

Regulatory Application of New Source Terms

The phrase "source terms" refers to the magnitudes of the radioactive materials released from the reactor core to the reactor containment and potentially to the atmosphere, following a postulated severe reactor accident; the phrase encompasses other information, such as the timing involved, needed to calculate off-site consequences of such an accident. Consideration of" source terms" enters the regulatory process through the Commission's reactor site criteria (10 CFR Part 100), which require that an accidental fission product release from the core into the containment should be assumed to occur and that its radiological consequences should be evaluated on the assumption that the containment leaks at its "expected demonstrable leak rate." The criteria for calculating the release into the containment are derived from the 1962 report, TID-14844, which assumed an instantaneous release of fission products. Although this source term is included in the Commission's regulations for siting, it has traditionally affected plant design more than 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 may be affected by changes introduced as a result of this source term and severe accident research. In fiscal year 1992, work continued on a replacement to TID-14844. In July 1992, a draft report, "Accident Source Terms for Light-Water Nuclear Power Plants" (NUREG-1465), was issued for public comment. The following documents, relevant to the subject, were also issued during the fiscal year:

- Draft NUREG/CR-5747, "Estimate of Radionuclide Release Characteristics into Containment Under Severe Accident Conditions," dated January 1992.
- Draft NUREG/CR-5787, "Timing Analysis of PWR Fuel Pin Failures," dated March 1992.
- NUREG/CR-5732, "Iodine Chemical Forms in LWR Severe Accidents," dated April 1992.
- NUREG/CR-5787, "Timing Analysis of PWR Fuel Pin Failures," dated September 1992.

Update of Siting Regulations. In fiscal year 1992, the staff initiated rulemaking to decouple siting appraisals from plant design considerations. This effort more directly incorporates requirements related to acceptable site characteristics into the proposed rule.

Emergency Planning Regulations. In fiscal year 1992, work continued on rulemaking to add emergency planning requirements to 10 CFR Part 72, regarding independent storage of spent nuclear fuel and high-level radioactive waste. It is expected that a proposed rule will be sent to the Commission in early 1993. In January 1992, the staff issued Regulatory Guide 3.67, "Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities," providing guidance to licensees on the information to be included in emergency plans for fuel cycle and material facilities. Also, in August 1992, Revision 3 to Regulatory Guide 1.101, "Emergency Planning and

Preparedness for Nuclear Power Reactors," was issued to revise the approach for the development of Emergency Action Levels.

Advanced Reactor Regulatory Standards

Several major measures were initiated during 1992, and others continued, to ensure that the regulatory framework needed in the near term to license advanced reactor designs will be available when required. These activities included (1) giving guidance for the preparation of acceptable probabilistic risk assessments, needed in support of Part 52 design certification applications; and (2) upgrading of two existing regulatory guides in the quality assurance (QA) area: (a) Regulatory Guide 1.28 endorsing ASME NQA-1 QA program requirements for nuclear facilities, and (b) Regulatory Guide 1.33 endorsing ANS 3.2 administrative controls and QA for the operational phase of nuclear facilities. It is expected that all of these activities will continue in 1993.

REACTOR AGING AND LICENSE RENEWAL

Integrity of Reactor Components

This area of NRC research focuses on reactor plant systems and components to see that they perform as designed and that they continue to do so over the life of the plant. Reactor safety clearly depends on maintaining the integrity of the reactor system pressure boundary, i.e., keeping it free from damage and leaktight. Failure to maintain pressure boundary integrity could compromise the operator's ability to cool the reactor core and could lead to a loss-of-coolant accident accompanied by the release of hazardous fission products.

Reactor Vessel Integrity

Pressure Vessel Safety. The reactor pressure vessel (RPV) is the key element in the primary pressure boundary. It houses and supports the reactor core and provides for channeling the coolant water from the inlet piping, through the core, to the outlet piping. It is also the only component in the primary pressure boundary for which, if it should rupture, the engineered safety systems cannot assure protection from core damage. Because of the importance of the reactor pressure vessel, there is a continuing effort to develop and refine the technical bases for evaluating the vessel and ensuring continued safe operation.

Methods for evaluating the potential for vessel fracture must encompass both normal operating conditions and postulated accident conditions. They must also take into account the full range of material behavior—fully ductile to fully brittle—and the reactor operating environment. In this regard, three areas were given special emphasis in NRC-sponsored research during the report period: fracture evaluation, radiation embrittlement, and surveillance dosimetry.

Fracture Evaluation. The NRC's fracture evaluation effort includes both analytical and experimental efforts. During fiscal year 1992, research continued on evaluating the validity and accuracy of reactor pressure vessel fracture analyses, on analyzing the influence of critical parameters that affect the fracture behavior, and on identifying those requiring additional research. Additional areas of research include developing and refining analysis methods that can be used reliably in predicting reactor pressure vessel fracture and developing the material property data needed as input to these analyses. The results already obtained have contributed to the development of Code Cases being considered by the ASME Boiler and Pressure Vessel Code. If accepted for the Code, it is anticipated that the Code Cases will be endorsed by the staff as acceptable criteria for evaluating low upper- shelf Charpy energy materials used in reactor pressure vessels and for determining low-temperature overpressure protection (LTOP) setpoints for protection against failure during low-temperature operation, such as reactor startup and shutdown.

Considerable achievements were realized during fiscal year 1992 in generating test data and analyses that could be used to evaluate the potential for non-ductile failure of RPVs. Research at the U.S. Navy's David Taylor Research Center (Naval Surface Warfare Center, Annapolis Detachment) and the Oak Ridge National Laboratory (ORNL) produced independent and complementary analyses for pressure vessel fracture and for evaluating the fracture resistance of full-scale RPVs, based on analysis results and test data from small laboratory specimens. These efforts are focused on determining the effects of crack-tip constraint on the operating-load-induced stresses and the material's fracture resistance. Constraint is the result of a triaxial-state of stress near the crack-front. If the stresses are relaxed in any direction (as by an inelastic yielding of the material), constraint is decreased and the apparent fracture resistance of the material increases. The objective of this research is to fully characterize the constraint effects in a handbook format, for use as a design and safety analysis tool for engineers involved with ensuring integrity of reactor primary system components.

As the technology for predicting the fracture behavior of RPVs has matured, the emphasis in NRC's research program has moved from broad-spectrum scoping research to that of specific topics aimed at developing analyses and supporting test data that can eliminate some of the very conservative assumptions built into the early regulatory analyses. A significant effort has been put in the RPV research program on evaluating the increase in fracture resistance of material to shallow flaws. Analyses have shown that the increase in fracture toughness for very shallow cracks can have a significant effect on pressurized thermal shock (PTS) analyses. Tests performed during fiscal years 1991 and 1992 on laboratory specimens have confirmed that shallow flaws are initiated at higher fracture toughness values than are deep flaws. The confirmation has led to an expanded program that seeks to quantify and validate the difference in fracture behavior of shallow-versus-deep flaws in RPVs. This effort is expected to be completed in the next several years. Once completed, the results could have a major impact on pressure vessel safety analyses, significantly influencing the currently perceived degree of risk attaching to potential accidents such as PTS.

Other related areas that will impact PTS fracture analyses include dosimetry and flaw density and location distributions. The work being carried out under the surveillance dosimetry program will provide more accurate values of the neutron fluences for conducting PTS analyses. Evaluations of the spatial distribution, size, and density of fabrication defects were initiated during the last fiscal year and are expected to continue for the next several years. These issues have not been extensively studied in recent PTS analyses but have a strong influence on the results of fracture analyses. The success of the long term effort will be strongly dependent on the availability of pertinent RPV materials for detailed examination.

Besides the research efforts, the fiscal year 1992 program included a major effort to support the Office of Nuclear Reactor Regulation (NRR) in its evaluation of the Yankee-Rowe reactor pressure vessel (RPV), the issuance of the Generic Letter 92-01, "Reactor Vessel Structural Integrity," and the review of the licensees' responses to the generic letter. A significant effort was devoted to performing independent analyses of the vessel failure frequency attributable to PTS transients and to performing analyses for safety issues concerning low Charpy upper-shelf energy RPV material. These efforts drew on expertise in probabilistic and elastic-plastic fracture mechanics, ductile flaw growth and flaw stability, embrittlement effects, flaw size distribution, and inservice inspection techniques. While the regulatory decisions were made in NRR, the research and analysis efforts contributed substantially to the staff decision process on industry proposals.

During this fiscal year, an NRC/industry meeting was held on coordination of RPV integrity issues. This coordination is expected to continue in dealing with specific technical issues related to RPV integrity. The specific technical topics of interest are (in no particular order): fracture mechanics and PTS analyses, evaluation of materials with low Charpy upper-shelf energy, nondestructive examination and inservice inspection "credit," irradiation embrittlement and surveillance, annealing of reactor vessels, dosimetry, and thermal-hydraulic analyses.

Radiation Embrittlement. One of the major concerns in ensuring nuclear power plant safety is the integrity of the reactor pressure vessel (RPV). While the design process and pre-service properties of the RPV materials ensure RPV integrity during early vessel operation, embrittlement of the RPV materials at the vessel beltline, brought about by neutron irradiation, can impose operational restrictions and can reduce safety margins to levels that approach minimum required levels. This research program addresses radiation embrittlement by combining experimental evaluations of embrittlement, determinations of underlying radiation damage mechanisms, correlations of embrittlement with environmental and material parameters, and evaluation of service-degraded material from decommissioned reactors for confirmation of embrittlement prediction methodologies. The results of the research are integrated into the applicable regulatory documents, including the regulations and regulatory guides, for predicting radiation embrittlement.

Additional efforts under way in this task address thermal annealing of the RPV to mitigate the effects of radiation embrittlement and evaluation of the effects of low-temperature, low-flux irradiation on the steels used in RPV supports.

The embrittlement research, coupled with the material properties research, has provided the fracture toughness data base used in the ASME Code, Sections III and XI, in developing the crack initiation and arrest toughness curves. These curves are essential for use in integrity analyses to ensure safe operation of nuclear reactor pressure vessels. Recent results from test reactor irradiations suggest that the ASME Code approach to shifting the fracture toughness curve to account for irradiation damage may not be conservative in every case. It appears that the Code approach may under-predict the actual shift in the fracture toughness curves in some cases, thereby eroding the anticipated margin of safety in regulatory analyses. To assist in evaluating margins of safety, research is under way to study the toughness properties of reactor vessel weld metal from the canceled Midland Unit 1 (Mich.) nuclear plant. The specific weld metal and flux combination in the Midland RPV welds is present in many commercial PWRs, so the results will have wide applicability. In fiscal year 1992, characterization of the material properties of the weld in the unirradiated condition was completed, and test reactor irradiations were initiated.

As the number of variables that could have a significant influence on embrittlement is large and inter-related, an empirical approach cannot completely resolve the issue. Therefore, increasing emphasis has been given to the study of the underlying mechanisms of neutron radiation and the resulting embrittlement. While this work is ongoing and will not be completed for several years, there has been significant progress through the use of high resolution instruments, such as the atom probe field ion microscope (APFIM) and small angle neutron scattering (SANS) methods. This progress has improved confidence in interpreting the empirical (test reactor) results and in defining additional test reactor irradiation programs. The International Group on Radiation Damage Mechanisms, formed to promote international cooperation on addressing these issues, has provided an arena for valuable discussion and interaction on this subject, furthering the efforts to identify the underlying mechanisms controlling neutron irradiation embrittlement.

This research has made substantial progress in identifying mechanisms that appear to control the embrittlement process, thereby facilitating the development of a predictive model that can replace the empirical approach currently used in evaluating irradiation damage. Development of that predictive model is the ultimate goal of this research. While the results of past research have contributed significantly to achieving this goal, the studies have also identified many interactions that must be understood before a truly comprehensive model can be completed. Research has shown that a dominant irradiation embrittlement mechanism for RPV steels is the accelerated formation of extremely small (1–2 nanometers) copper-rich precipitates in the steel microstructure. More recently, using the data made available from SANS and APFIM techniques, a model has been developed to describe the evolution of defects and defect clusters in RPV steels subjected to neutron irradiation. Results from this latter work support the postulate that defect clusters, rather than copper-rich precipitates, may be responsible for creating a greater increase in strength for some combinations of neutron fluence, flux and temperature, depending on the chemistry of the particular material under study.

A comprehensive collection of radiation embrittlement data from surveillance reports and other published reports of commercial power reactors has been compiled in the computerized Power Reactor Embrittlement Data Base (PR–EDB). This data base has proved advantageous in evaluating licensee submittals concerned with RPV embrittlement and will be used to develop improved irradiation embrittlement correlations.

An effort to develop improved irradiation embrittlement correlations was initiated in fiscal year 1992. This is a three-year effort that will use all embrittlement data from the United States and a number of foreign countries to develop correlations of transition temperature shift and upper-shelf decrease as functions of the controlling parameters, principally material chemistry, neutron flux and fluence, and irradiation temperature. The data from foreign sources will be most interesting, because insights will be provided into the influence of different alloy systems and impurities. Results from this program will be evaluated to determine if revisions to the procedures in Regulatory Guide 1.99 are necessary.

A new initiative in fiscal year 1992 is the validation of the embrittlement prediction methodology using material from decommissioned RPVs. Previous work in this area examined material from the decommissioned Gundremmingen reactor in the Federal Republic of Germany. In fiscal year 1992, cooperation was initiated with the Japan Atomic Energy Research Institute (JAERI) to evaluate material from the decommissioned Japan Power Demonstration Reactor. The U.S. part of the collaborative research effort involves material characterization, metallographic examinations, investigation of annealing response, dosimetry studies, and neutron transport calculations. The ultimate goal of the investigations is to validate and improve methods for aging evaluation and life prediction for RPVs using service-degraded material.

The embrittlement program has provided initial data to demonstrate the effectiveness of thermal annealing in recovering degradation in mechanical properties caused by irradiation damage. The results of the annealing work are being augmented by recently initiated industry efforts, and by results from research performed in Russia and exchanged under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS). The Russians have successfully annealed 12 VVER-440 RPVs. The combined results of these efforts provide reasonable assurance that thermal annealing is a practical method for mitigating the effects of neutron irradiation damage. Further work is in progress to improve the ability to predict the rate of annealing recovery and of re-embrittlement. While much more work is needed to provide appropriate regulatory guidance, the principle of the benefits of thermal annealing has been demonstrated.

Surveillance Dosimetry. A necessary aspect of the surveillance program for establishing the degree of neutron embrittlement of the reactor pressure vessel in a nuclear power plant is the surveillance dosimetry program, devised to predict the amount of neutron radiation exposure (neutron fluence) at critical locations of the vessel. These neutron fluences, which cannot be measured directly, are predicted by determining neutron fluences at surveillance locations. The neutron fluence at surveillance locations is determined by a process of (a) dosimetry measurements, (b) transport calculations to compute fluence, and (c) a consolidation of the measurements and calculations to reduce uncertainties of the predictions. These predictions must be accurate in order to ensure that the plant is operating in conformance with NRC safety regulations.

Dosimetry research has led to a new improved set of differential cross-sections of nuclides that have been included in the Evaluated Data Files developed by the National Nuclear Data Center. Use of the new data files is expected to significantly improve the ability to predict neutron fluences in reactor vessels. Work has been initiated to process these files into structured libraries that can be applied to determining light-water reactor vessel fluences. Finally, research continues to improve the methodology and data bases for calculating neutron fluences and fluence rates at critical locations of reactor vessels. Sources of uncertainties are being identified and the sensitivity of these uncertainties to fluence determinations are being established. New approaches to fluence determinations, such as the use of ex-vessel surveillance dosimetry, are being validated. This research has led to the development of a regulatory guide on dosimetry scheduled to be issued for comment in 1993.

Reactor Pressure Vessel Integrity Rules and Regulatory Guides. During fiscal year 1992, a significant effort was devoted to the development of new regulatory guidance and to revising some of the existing regulatory guides and rules pertaining to reactor pressure vessel (RPV) integrity. Evaluation of the Yankee-Rowe vessel—and, in particular, the "lessons learned," as outlined in SECY-92-283—clearly focused attention on needed revisions to and provisions of the regulatory documents. Notable among these are the guidance on evaluation of RPVs with Charpy upper-shelf energy less than 50 ft.-lb., requirements for thermal annealing, and guidance on format and content for assessments of thermal annealing of RPVs. Revisions are being considered for the PTS rule (10 CFR 50.61), Appendices G and H to 10 CFR Part 50, and the regulatory guide on format and content of plant-specific PTS analysis reports for PWRs (Regulatory Guide 1.154). These changes to the regulatory documents affecting RPV integrity are planned to be completed over the next two to four fiscal years and will incorporate the latest technical advances in the RPV integrity field.

Steam Generator Integrity

The emphasis of NRC research on steam generator tube integrity in fiscal year 1992 has been on developing generic guidance for the performance demonstration qualification of eddy current inspection systems. This guidance will be included in a revision of Regulatory Guide 1.83, which covers inservice inspection of steam generator tubing. The need to develop this guidance was one of the conclusions of an NRC-sponsored research program into the reliability of eddy current inspection techniques to detect and measure steam generator tube degradation. The results of this program indicated a need to improve the reliability of steam generator tube inspections. Work was initiated in fiscal year 1991 to develop performance demonstration requirements to ensure that eddy current inspection systems (i.e., personnel, equipment, and procedures) possess adequate capability to identify all the known forms of tube degradation that occur in steam generators. Tests have been designed to screen out eddy current inspection systems that do not possess the capability to detect "significant" flaws a high percentage of the time and accurately measure the through-wall depth of penetration.

The NRC has also been developing additional information on the capability of eddy current inspection systems to detect and gauge crack-type flaws through participation in the international Program for the Inspection of Steel Components (PISC). Crack-type flaws are of great interest to the NRC because they are the most frequent cause of steam generator tube failure; they are the most difficult flaw type to reliably detect and measure; and they significantly decrease tube integrity. The PISC program comprises a round robin, in which seven U.S. teams are participating, on the effectiveness of steam generator tube inspection techniques. In fiscal year 1992, several tube mockups were circulated in the United States. Analysis of the data from this study will be performed in fiscal year 1993, with the conclusion of the program scheduled for fiscal year 1994.

Piping Integrity

Environmentally Assisted Cracking. Fatigue is a potentially significant degradation mechanism in light-water reactor (LWR) primary piping and other portions of the reactor pressure boundary system. Current fatigue design for reactor structural components is based on the ASME Code Section III and its fatigue design curves. The design curves, which were developed more than 20 years ago, were obtained by adding a correction factor to the mean-data curve obtained from room-temperature tests on smooth specimens in air. The correction factor was intended to account for the differences between structural components and the test specimens, and it was intended to account for a variety of factors, including the effect of surface finish, size, and data scatter.

Based on the results obtained in earlier work in the United States and Japan, as well as on the results obtained in the ongoing NRC research program, it is now clear that the Code curves and procedures can significantly overestimate fatigue lives under some reactor loading and environmental conditions. Since no consensus design procedure is available, data from ongoing tests and from the literature and programs in Europe and Japan were evaluated to develop interim design curves that more adequately describe fatigue life in the high-temperature aqueous environments characteristic of LWRs.

The current data on the fatigue crack growth in pressure vessel and piping materials have been obtained almost solely in tests where the ferritic materials have been completely exposed to the simulated reactor coolant environment. In reality, these materials are clad with austenitic stainless steels, and only a very small portion of the material is exposed to the reactor coolant. Tests show that crack growth rates are higher in the clad materials and cracking is easier to initiate and sustain in clad materials, but that the differences are relatively small. It appears that, in most cases, existing predictive methods give an adequate description of crack growth rates; however, some additional testing is being performed to confirm this observation.

Irradiation-assisted stress-corrosion cracking (IASCC) of core internal components fabricated from solution-annealed austenitic stainless steels and high-nickel alloys has been observed in both BWRs and PWRs. Although many of the affected components can be replaced, some safety-significant components such as the top guide, shroud, and core plates in BWRs would be difficult or impractical to replace. Tests have shown that the susceptibility of special high-purity heats of Type 304 stainless steel (HP) materials to IASCC was higher than those of the commercial-purity (CP) Type 304 stainless steel materials in laboratory tests. In crevice or flow-restricted environments, however, susceptibilities of the HP and CP heats appear comparable. Future work will examine the susceptibility of a wide range of compositions to this type of cracking.

Piping Fracture. With the discovery of inservice cracking of nuclear reactor piping came an increased interest in how such "service-degraded" pipe would behave under postulated accident conditions, i.e., whether it would leak or break. The matter of the leak-or-break alternatives had been addressed for years without the emergence of a strong consensus. The NRC and the industry have pursued parallel research efforts in evaluating pipe fracture behavior. The industry's effort has focused on the behavior of stress corrosion cracks, and the NRC has explored the broader questions regarding leak-before-break phenomena for all piping.

Over the last decade, the NRC has funded research into several aspects of pipe fracture, including evaluation of material properties, conduct of full-scale pipe fracture experiments, and development and verification of analysis methods to predict the behavior of piping with flaws. The current program is extending this work to include monotonic loading of piping with short cracks (in terms of length and depth), typical of those that may be found in service and of interest to leak-before-break analyses; seismic loading of fittings; validation of prediction methods under dynamic (water-hammer) loading conditions; and other topics identified in past work as needing attention. These other topics include the fracture behavior of bimetallic welds, the effects of dynamic strain aging on fracture toughness, and the effects of anisotropic material properties.

During fiscal year 1992, a study of several piping-related issues was continued at Battelle in Columbus, Ohio. This experimental and analytical program studies the effects of short cracks (in depth and length) on the fracture behavior of typical nuclear-grade piping materials. Prior experimental and analytical efforts examining the fracture behavior of flawed piping have addressed crack depths and lengths greater than those encountered in service and greater than those of interest in leak-before-break analyses. Therefore, this study is providing experimental data for validating and improving pipe fracture analysis methods. Results to date indicate that the safety margins for short cracks using current ASME Code methods are similar to or even greater than those for longer cracks that had been validated in previous testing. This study will be completed in fiscal year 1994.

During fiscal year 1991, the NRC completed the first International Piping Integrity Research Group (IPIRG–1) program to evaluate the effects of seismic and dynamic loading and other piping integrity issues. The IPIRG–1 was a consortium of nine government and industrial organizations that jointly funded the research. The success of the IPIRG–1 program, and the progress made by the IPIRG participants toward an international consensus on pipe fracture technology, led the participants to form a second jointly funded program, the IPIRG–2 program.

During fiscal year 1992, the inter-governmental agreements were concluded with the participants, and the research contractor—Battelle, in Columbus, Ohio—entered into agreements with several industrial entities for their participation in the IPIRG–2 program. Facility modifications were initiated to permit the seismic loading tests planned for this project. The pipe fracture tests planned for this program build on the work being done in the short cracks in piping and piping welds program, the key difference being that the IPIRG–2 tests will address seismic loading effects. Testing is expected to begin early in calendar year 1993 and to be completed in fiscal year 1995.

Thermal Aging of Cast Stainless Steels. Embrittlement of the ferrite phase in cast duplex stainless steel may occur after 10-to-20 years at reactor operating temperatures. This condition could adversely affect the mechanical response and integrity of pressure boundary components during high strain-rate loading (e.g., seismic events). The problem is of greatest concern in PWRs where slightly higher temperatures are typical and cast stainless piping is widely used.

Research on this subject has been ongoing since 1982. During fiscal year 1992, the procedures and correlations developed to estimate fracture toughness, tensile flow stress, and Charpy impact properties of cast stainless steels in LWR systems were updated and are being used by industry, as well as by NRC staff. Basic study of the effects of thermal aging on embrittlement of cast stainless steels will be complete when the final report is issued early in fiscal year 1993. However, some further assessment of the crack growth rate and fatigue characteristics of aged, embrittled cast stainless steel weld materials will continue.

Inspection Procedures and Technologies

This program includes studies of improved methods for the reliable detection and accurate sizing of flaws during inservice inspection of carbon steel and wrought and cast stainless steel piping and pressure vessels. It includes studies of on-line continuous monitoring techniques, using acoustic emission, for crack growth and leak detection.

Improving the Detection and Sizing of Flaws. An improved method for more reliably detecting flaws and sizing them with greater accuracy in LWR primary circuit components is the Synthetic Aperture Focusing Technique for Ultrasonic Testing (SAFT-UT). The SAFT-UT technology is based on physical principles of ultrasonic wave propagation and uses computers to process the data to produce high-resolution, three-dimensional images of flaws to aid the inspector in locating and sizing them. The SAFT-UT system has been used to inspect a reactor pressure vessel as part of the Program for the Inspection of Steel Components (PISC), assessing the effectiveness of advanced ultrasonic technologies. A final report is being prepared showing the results of this inspection. The SAFT-UT technology has been transferred to General Electric and integrated into their new reactor vessel inspection system. Results will be available when this system is used in fiscal year 1993.

Inservice Inspection System Qualification. Field experience and research, including both national and international studies, over the last several years has shown that inservice inspection (ISI), as currently practiced, is not sufficiently effective or reliable. NRC research indicates a need for qualification of the entire ISI process, including the personnel, equipment, and procedures as described in the *1987 NRC Annual Report*, pp. 115 and 116.

Since Section XI of the ASME Code approved Mandatory Appendix VII on Personnel Training and Qualification (in 1989) and Appendix VIII on Performance Demonstration of Ultrasonic (UT) ISI Systems (in 1990), the NRC and its research contractor have undertaken to review and evaluate industry's plans to implement these appendices. An assessment with proposed revisions to upgrade these two appendices was provided to the appropriate ASME Code groups. Close coordination is maintained with the industry Performance Demonstration Initiative (PDI) group, through the Nuclear Utility Management and Resources Council (NUMARC), to monitor progress and critique industry plans.

Advanced Ultrasonic Imaging Systems. Additional Code requirements were prepared and submitted to the ASME Section V Subcommittee to fulfill a need for Code rules to cover the computerized UT imaging systems that are being used by the industry for examining nuclear power plant components. These proposed Code rules were approved for publication in the 1992 Addenda to the ASME Section V Code on Nondestructive Examination.

Risk-Based Inspection. Improved criteria for ISI planning are being developed using risk-based approaches. Plant-specific pilot studies, centered on the Surry Unit 1 (Va.) nuclear power plant, have shown the risk-based approach to be workable. As a result, ASME Section XI has formed a task group to begin implementation of risk-based methods into Code rules for inservice inspection of pressure vessels and piping. Expected benefits include a reduction of inspections where such inspections are not justified by safety concerns and occupational exposure concerns, and a redirection of inspection efforts to other components with greater safety significance.

Equipment Interaction Matrix. The reliability of ultrasonic inspections of nuclear components is known to vary with human factors, equipment characteristics, procedures, etc. NRC research has shown that changing the equipment parameters in an ultrasonic inspection can greatly affect the results of the inspection. The ASME Code provides tolerance levels for equipment parameters. Results of NRC research have been used to develop and update the tolerances in the ASME Code.

Surface Roughness Evaluation. Currently, there are no ASME Code requirements dealing with surface conditions during ultrasonic inspections. NRC research has shown that ultrasonic inspections are affected by surface conditions in predictable ways using computer models of radiation transfer. Recommendations will be made to the Code to limit the adverse effects of surface conditions.

Coarse-Grained Materials. Improved methods are being evaluated for the reliable and effective inspection of centrifugally cast stainless steel components of PWRs. Cracks in these components of more than a certain size must be detected by an effective inspection system. Low-frequency ultrasonic systems capable of adapting system properties to compensate for changes in the internal structure of these coarse-grained materials are being investigated.

Continuous Monitoring for Crack Growth and Leak Detection. NRC-funded research has produced technology useful in the application of continuous acoustic emission (AE) monitoring, in order to detect the initiation and extension of cracks in nuclear reactor components as they might occur during reactor operation. The technology and application methods have been validated, outside actual reactor operations, in several major tests. An ASTM Standard E 1139 and an ASME Code Case N-471 have been generated and approved to guide and regulate application of the technology. The research program and results obtained therefrom have been summarized in NUREG/CR-5645. The final step in this effort has been to validate the AE technology and methodology on an operating reactor by monitoring a weld flaw indication at the Limerick Unit 1 (Pa.) reactor. AE monitoring of the weld at Limerick demonstrated that continuous AE monitoring can be effectively applied to an operating reactor plant. The AE system was installed, calibrated, operated, and maintained without causing any disruption to the reactor schedule. AE and ultrasonic inspection results agreed with respect to growth of the flaw indication observed in the weld. Both showed limited growth during the first fuel cycle and no growth during a second fuel cycle. The relationship identified between AE and crack growth produced rational results similar in magnitude to those indicated by the ultrasonic inspection. A final report on this effort was in preparation at the close of the report period.

AE technology similar to that just discussed can also provide a very sensitive coolant leak detection capability. This application was developed under NRC sponsorship, with the results presented in "Application of Acoustic Leak Detection Technology for the Detection and Location of Leaks in Light Water Reactors" (NUREG/CR-5134). Benefits from this work include increased safety by detection and evaluation of crack growth as it occurs; improved detection and location of coolant leaks as they begin; and reduced personnel exposure to radiation, by reduced need for manual inspection of reactor components.

Proposed new Code rules were prepared and submitted to the Section V Subgroup on Acoustic Emission. The proposed new SC-V Article will specify requirements for continuous on-line monitoring for various applications. The NRC is focusing on applications involving nuclear reactor systems and components, and input is being solicited from industry to address the other application areas within the scope of the proposed Article.

International Reliability Studies. The NRC has been an active participant in the PISC program (see above), which is assessing the effectiveness of technologies and procedures for ISI of nuclear power plant components. The output from this program will aid regulators and Code bodies in establishing technical bases for improving ISI requirements. The NRC has taken a leadership role in developing PISC program objectives and has funded work at the Pacific Northwest Laboratory (PNL) to produce a design of studies, to fabricate flawed specimens, to implement testing, and to analyze comprehensive data bases. Specific PISC tasks include appraisals of the influence of human factors on inspection reliability; of pressure vessel inspection capability using SAFT-UT; of the inspection of stainless steel piping, nozzles, and dissimilar metal welds; and of the inspection of steam generator tube mockups. These experimental studies are nearing completion, and important results are becoming available. The PISC program is scheduled to be completed by the end of 1993, and the results and analyses of data from all the studies will be released to participants over the next year or so. As the results become available, they will be used to develop and substantiate upgraded inspection requirements.

Specific U.S. activities performed during fiscal year 1992 included definition of the scope of work to be done under contracts funding the participation of UT/ISI teams from the United States in the Wrought Stainless Steel Reliability tests. This reliability study will be conducted in PNL facilities to accommodate the special requirements for this research. All seven U.S. teams completed their inspections of the wrought stainless steel capability specimens, and the six large pipe specimens were returned to Europe for further inspections and/or destructive evaluations. Inspection schedules and preliminary logistics planning have been completed for the PISC-III capability studies for cast/wrought stainless steel and cast/cast stainless steel specimens. When they become available, it is expected that the PISC-III results will give significant impetus to the NRC efforts to upgrade selected ASME Code requirements.

Support to NRC Regulators. NRC research is helpful to NRC Regional and Headquarters Offices by assisting in the training of their staffs to a fuller understanding of the new and developing technologies being applied to inservice inspection. During the past year, major efforts were directed to developing procedures and test blocks for the detailed review and evaluation of computer-based ultrasonic inspection systems. Certain sections of a draft NUREG report entitled Auditing Computer-Based Ultrasonic Inservice Inspection Systems were revised. A computer-based ultrasonic inspection system was rented and subjected to a rigorous review and evaluation, including a hands-on seminar for the NRC staff. Finally, the preliminary results of the review and evaluation were presented to the NRC Technical Advisory Group on Nondestructive Examination. Fabrication of ultrasonic test blocks for the NRC Mobile Nondestructive Evaluation Laboratory continued.

A steam generator tube mockup for on-site evaluations of the effectiveness of eddy current inspection systems has been designed and, during fiscal year 1992, the structural elements fabricated. Two of the flaw types (wastage and fatigue cracks) to be included in the mockup were also produced.

US–CIS Cooperative Agreement

In October 1992, NRC staff and representatives of the Department of Energy and the national laboratories participated in the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Group 12 meeting and workshop on nuclear power plant aging and life extension with members of the Commonwealth of Independent States (CIS), in Moscow, Russia. Maintaining the safety of the aged operating nuclear power plants in the CIS was a major concern of the CIS participants.

The Russian Federation (R.F.), in particular, places significant importance on the activities of Working Group 12. This was evident from the active participation of the representatives of various institutions and organizations and operating plant personnel. The Working Group provides an excellent forum for an integrated and coordinated program for the CIS and serves as an "umbrella" organization to unite fragmented technical activities of a multitude of former Soviet institutions and organizations. Information exchange through the activities of Working Group 12 over the past year has provided a foundation upon which the R.F. and the CIS intend to develop both the near and long term programs for managing aging in their operating nuclear power plants, with a primary emphasis on safety.

Active cooperation and exchange continued this year in JCCCNRS Working Group 3, renamed "Radiation Embrittlement, Structural Integrity, and Life Extension of Reactor Pressure Vessels and Supports." The working group met in St. Petersburg and Moscow in September 1992. Accord was reached on inter-laboratory calibration of test methods for Charpy specimens, and a similar intercalibration is being planned for fracture mechanics testing. Irradiations have been undertaken of U.S. and Russian steels in each other's reactors. One three-month assignment of a Russian scientist in the United States has been completed, and the one-year assignment of another Russian scientist begins in late 1992.

Aging of Reactor Components

Aging Research. Aging is a vital concern with currently operating plants and will clearly be crucial to any assessment of the safety implications of license renewal. Aging affects all reactor structures, systems, and components. If unmitigated, it has the potential to increase risks to public health and safety. There are significant uncertainties about age-related degradation processes and about whether time-related degradation can be detected and managed before safety is impaired. Specifically, there is concern that multiple failures of age-related components could occur during transients or accidents and result in core damage and release of radiation. In the past, failures of safety-related components have occurred because of such degradation processes as corrosion, radiation, and thermally induced embrittlement of electrical insulation, pitting of electrical contacts, surface erosion, metal fatigue, oxidation, creep, binding, and wear. A number of these phenomena also cause deterioration of mechanical components.

The purpose of research into the aging of reactor components is primarily to establish the safety margins of operating plants as they progress through their design life; to define the aging mechanisms; to confirm existing and/or develop recommendations for new detection and mitigation methods, in order to prevent or mitigate the deleterious effects of the aging process; and to ensure that safety systems in nuclear power plants operate reliably. The secondary objectives of the program are to provide data helpful in evaluating the effectiveness of the industry's maintenance programs for reactor components and also to establish the technical bases for criteria to be applied in the processing of the anticipated licensee requests to extend the operating life of reactors past their initial 40-year operating license period.

The Nuclear Plant Aging Research (NPAR) program provides information and technical bases useful in understanding the effects that aging has on the safety function of electrical and mechanical components of commercial nuclear plants. As of the end of fiscal year 1992, the NPAR program consisted of 16 separate but related projects concerned with the study of the effects of aging on 23 individual mechanical and electrical components and 17 systems composed of such components. Also noteworthy are special topic studies applicable to all, and useful to aging, evaluations on a generic basis.

The current NPAR program also consists of individual studies on 11 special topics. They are (1) risk evaluation of significant aging effects; (2) setting of priorities among structures, systems, and components based upon their aging-risk significance; (3) activities of the Joint U.S.–CIS program on aging and life extension; (4) information useful for residual life assessment of major LWR components and structures; (5) development of technical bases for license renewal rulemaking; (6) review of technical specifications from an aging perspective; (7) study of data needs and record-keeping; (8) integration of NPAR results into the inspection process; (9) degradation modeling of component aging; (10) reviews of applicable regulatory instruments useful for license renewal; and (11) reviews of industry-sponsored technical reports for renewed license applications. A phased approach to the research has been adopted to facilitate interim reviews and evaluations and to help arrange for the availability of resources.

In fiscal year 1992, Phase I aging assessments were completed on the following special topics and safety-related components and systems:

(1) Approaches for Age-Dependent Probabilistic Safety Assessment (NUREG/CR-5587, Vol. 1)

(2) Bistables and Switches (NUREG/CR-5844)

(3) Geomagnetic disturbances in Nuclear Power Plants

(4) Standby Liquid Control System Aging Assessment

(5) Nuclear Power Plant Chillers

(6) Aging Assessment of Reactor Instrumentation and Protection System Components (NUREG/CR-5700)

(7) BWR Control Rod Drives (NUREG/CR-5699)

(8) PWR Core Internals Degradation with Age

(9) Turbine Governors and Controls

(10) Heat Exchangers (NUREG/CR-5779)

(11) Aging of Safety Class 1E Transformers (NUREG/CR-5752).

Reports were issued on the above-mentioned Phase I aging assessments to identify degradation sites within the component and system boundary, aging mechanisms, and aging concerns. The reports, which also made recommendations for maintenance and aging mitigation, were reviewed by NUMARC, EPRI, and the various ASME and Institute of Electrical and Electronics Engineers (IEEE) working groups for potential use in revising the corresponding standards.

Phase II aging assessments of components generally involve some combination of (1) tests of naturally aged equipment or equipment with simulated degradation; (2) laboratory or in-plant verification of methods for inspection, monitoring, and surveillance; (3) development of recommendations for inspection or monitoring techniques; (4) verification of methods for evaluating residual service lifetime; (5) identification of effective maintenance practices; (6) in-situ examination and data gathering for operating equipment; and (7) verification of failure causes, using results from in-situ and post-service examinations. Phase II aging assessments were completed on the following components and systems:

(1) Component Cooling Water System (NUREG/CR-5693)

(2) 20-, 40-, and 60-Year Cable Aging Tests (NUREG/CR-5772, Volume 1)

(3) Assessment of Solenoid Valve Diagnostic Methods (NUREG/CR-4819, Volume 2)

(4) Snubbers

(5) Service Water Systems.

Aging Assessment and Mitigation of Major LWR Components. Intrinsic to the general exploration of reactor aging 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 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 to ensure continued safe operation. The studies also included the evaluation of advanced inspection and monitoring methods for characterizing the aging damage. The results will be useful to the NRC's resolution of safety issues associated with LWR aging degradation and development of regulatory guidance and decisions that may safely extend the term of LWR licensed operation. The major components assessed or being assessed are PWR pressure vessels, LWR reinforced and pre-stressed concrete containments, LWR cast stainless steel components, PWR steam generator tubes, and LWR metal containments. Results of these assessments are being documented in a multi-volume report, NUREG/CR-5314.

Technical Bases for License Renewal and Maintenance Effectiveness. Besides a final rule on the subject (10 CFR Part 54), more detailed regulatory guidance addressing the technical bases and safety issues related to aging are being developed, to help implement the rule and to address license renewal application requirements. An interim guidance document will be published for public comment in the first quarter of fiscal year 1993.

PRA-Based Priorities Among Risk Contributions and Maintenance. A second report (revision to NUREG/CR-5587) was issued setting forth priorities based on probabilistic risk assessments (PRA) among aged, active components, according to their risk contributions and maintenance importance. The format and content of the original report has been changed to include the technical bases for identifying the risk-significant components according to PRA-based priorities. For completeness, the second report also describes various approaches for transforming a baseline PRA into an age-dependent PRA, and it provides answers to questions that are likely to arise when applying an aging-related PRA.

One of the major limitations of carrying out risk evaluations of aging in nuclear power plants has been the lack of recorded component aging data. An approach that uses available information, as well as engineering knowledge, was developed to determine generic aging rates for components. Generic aging rates have been determined for all major components. These generic aging rates can be used with standard PRAs to evaluate and assign priorities to the risk effects of aging in nuclear power plants. Maintenance programs can thereby be evaluated for their risk-effectiveness, and individual maintenance tasks can given priorities accordingly. The generic aging rates can also be used to determine risk-effective replacement intervals for components.

Aging of Passive Components. In earlier efforts, a methodology was developed to include the effects of aging on passive components (pipes, structure, and supports) and the resulting 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 substituted into a PRA for the plant that will calculate the effects of this failure on plant risk. This fiscal year, approaches were investigated that can be applied to a large number of passive components that exist in a nuclear plant. A screening approach can be used to identify those components that age and contribute most to risk. Two approaches were investigated, 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 the attributes that have been shown to most affect aging and failure. These approaches, including a screening approach, will be documented in a report in fiscal year 1993.

The draft document (NUREG/CR-5730) that reports the methodology for including the effects of aging on passive components was reviewed internally by other NRC research staff, and responses to their comments have been developed. A revised NUREG/CR-5730 is almost completed and will be recirculated for general internal NRC review. Calculations were also completed to investigate the effects of passive components on the risk of containment failure. An approach was developed to identify the passive components that most contribute to risk.

Regulatory Document Review: Management of Aging of LWR Major Safety-Related Components. Eight selected regulatory documents, e.g., NRC regulatory guides and the Code of Federal Regulations, were reviewed for safety-related information on two additional major LWR components—pumps and valves. The focus of the review was on 25 NPAR-defined safety-related aging issues—including examination, inspection, and maintenance and repair; excessive/harsh testing; and irradiation and thermal embrittlement. It was concluded that the safety-related regulatory documents do provide implicit guidance for aging management, but more explicit guidance would be useful. A final report was prepared and will be published in fiscal year 1993.

Insights Gained from Aging Research. The NPAR program has produced a large data base of operating experience, maintenance activities, and testing information for structures, systems, and components used in nuclear power plants. The goals of this integrated research program are to identify and characterize those plant aging effects that could impair safety; to identify methods of inspection, surveillance, and monitoring that will detect aging effects before system safety function is lost; and to evaluate the effectiveness of maintenance and component replacement practices for mitigating aging degradation.

To facilitate the use and dissemination of NPAR study findings, a document was developed that summarizes the results from all the aging research and presents the essential elements on the effects of aging in nuclear power plants in a comprehensive manner.

The findings from the research and recommendations for improving the performance of selected systems or components are presented in NUREG/CR-5643. This report is divided into two main sections for each study, namely, "Summary of Research Results" and "Aging Assessment Guide." In the first section, the information contains the functional description and background on the subject, aging-related issues, current operating experience in nuclear power plants, recommendations for methods of detecting and mitigating aging effects, and other references related to the subject. The "Aging Assessment Guide" is a more concise document that provides some general observations of the effects of aging on the component or system. The guide lists recommendations associated with the maintenance, operation, design, and testing that the research has shown could be beneficial to understanding and managing the aging of that component or system.

The statements and recommendations made in this document are based on assessments of operating experience, evaluations of materials, testing of naturally aged equipment, and identification of operating and environmental stresses. It is intended that this document will be periodically updated to reflect the latest available research results.

Aging Insights Gained from NRC's Maintenance Team Inspections. A nuclear plant's maintenance program is the principal vehicle through which age-related degradation is managed. From 1988 to 1991, the NRC evaluated the maintenance program of every nuclear power plant in the United States. Forty-four out of a total of 67 of the reports deriving from these in-depth team inspections have been reviewed for insights into the strengths and weaknesses of the programs as related to the need to understand and manage the effects of aging on nuclear plant structures, systems, and components. Relevant information has been extracted from these inspection reports and sorted into several categories, including specific aging insights, preventive maintenance, predictive maintenance and condition monitoring, post-maintenance testing, failure trending, root-cause analysis, and use of PRA in the maintenance process. Specific examples of inspection and monitoring techniques successfully used by utilities to detect degradation attributable to aging have been identified.

The information was also sorted according to systems and components. The systems include auxiliary feedwater, main feedwater, high-pressure injection for both BWRs and PWRs, service water, instrument air, and emergency diesel generator air start systems. The components include emergency diesel generators; electrical components, such as switch gears, breakers, relays, and motor control centers; and motor-operated valves and check valves. The information for systems and components was compared to that obtained from the individual NPAR system- and component-level studies.

Results from this study are presented in NUREG/CR-5812; they indicate that, while some plants appear to adopt a precautionary posture toward aging-related failures of their structures, systems, and components important to safety, others seem to be taking a more passive or reactive stance. The report outlines some of the technical and organizational issues that should be considered for evaluating the plant maintenance activities relevant to understanding, detecting and mitigating the effects of aging.

Standard Technical Specification Aging Assessment. The NPAR program includes evaluation of surveillance requirements (SRs) of nuclear power plant systems and components from the point of view of aging management. SRs are performed on various safety-related components and systems of nuclear power plants to ensure the operability and availability of those components and systems. In the NPAR study, the SRs are being evaluated from three perspectives: (1) their adequacy in considering age-related degradation, (2) their potential contributions to age-related degradation, and (3) their capabilities for detecting and managing age-related degradation. Evaluations were conducted during fiscal year 1992 for an electrical system (Class 1E) and an electrical component (chargers/inverters) and for a mechanical system (residual heat removal system) and a mechanical component (motor-operated valves).

Record-keeping. NPAR studies of the technical issues associated with the role of nuclear plant records systems in understanding and managing aging of nuclear plant systems, structures and components were completed with the preparation of a final report, "Recordkeeping Needs to Mitigate the Impact of Aging Degradation" (NUREG/CR-5848).

Components, Systems, and Facilities

Bistables and Switches. Bistables and switches play a vital role in the instrumentation and control (I&C) logic of a nuclear power plant. They provide control logic inputs, trip signals, and annunciation in essentially every system in the plant—both on nuclear safety systems, such as the reactor protection, and on engineered safety features actuation systems, as well as non-safety or balance-of-plant systems. The use of bistables has increased over the past 25 years and continues to expand today, occasioning the greater emphasis on the importance of this component. The designs of the I&C logic for both of the above-mentioned systems in PWRs originally incorporated transmitters and bistable modules. Most older BWRs have upgraded from the switches originally used in their designs to transmitters and analog trip systems.

The dominant failure mode for the older vintage bistables was called "out of calibration." This failure mode was generally attributed to electronic setpoint drift, resulting from the aging degradation of board-level electronic components, particularly capacitors, potentiometers, and discrete transistors. For the newer units, "out of calibration" was no longer the dominant failure mode, mainly because of improved capacitors and resistors. Instead, loss of function and spurious alarms became the important modes (together with "out of calibration").

For those pressure switches that monitor differential pressure, pressure, and level by means of a diaphragm, Bourdon Tube, or bellows sensing elements, the dominant failure mode was again "out of calibration." Mechanical setpoint drift occurs whenever there is movement of the adjustable parts within the switch from their "as-left" position; a change in the properties of the switch components; or physical or dimensional variations in the switch components. The "out of calibration" condition can result from wear, vibration, fatigue and other aging processes acting upon the internal mechanisms of the switch to produce mechanical setpoint drift. The failure modes most often reported for these switches were loss of function and spurious signal/alarm. Those thermocouples which use electronic bistables to trigger an electrochemical output relay are subject to aging mechanisms related to the degradation of electronics components, similar to what was observed in bistables.

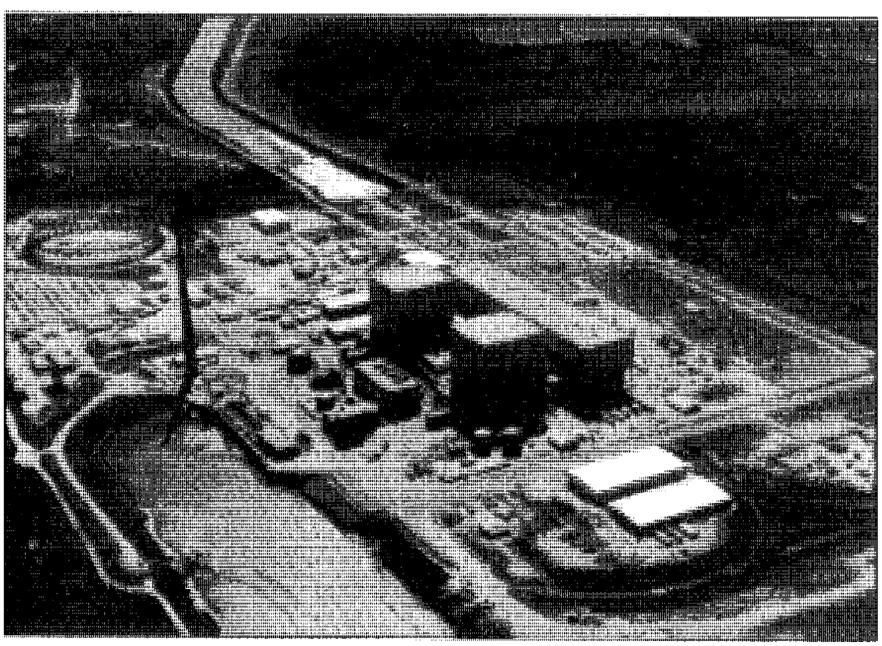
From the foregoing study it was found that trending analysis of required surveillance testing program results can provide valuable information to help optimize surveillance and maintenance intervals, anticipate problems, and monitor the effects of aging degradation.

Containment Cooling Systems. The containment cooling system was selected for study under the NPAR program because of its importance to plant safety during normal, as well as accident, conditions. While the containment cooling function is performed by several different systems, depending on the type and design of the plant, the two systems focused on in this study are the containment spray system and the containment fan cooler system. The data analyzed for this study show that aging is a concern for the containment cooling system and should be addressed. Over 50 percent of the failures reviewed were related to degradation caused by aging. The most commonly failed component in the containment spray system is valves, while in the fan cooler system, it is circuit breakers. These failures typically result in a degraded operating state for the system or a loss of redundancy.

A simplified PRA analysis showed that, for the containment spray design analyzed, a dominant contributor to system unavailability is a non-aging event; namely, a human error involving failure to reposition manual valves following surveillance testing. However, for components that could be affected by aging, pumps and motor-operated valves were found to be important to system unavailability. Failure rates for most of the risk-significant components show a tendency to increase over time. This increasing trend can result in a corresponding increase in system unavailability with age, if the trend is not properly controlled.

Aging Effects on Motor-Operated Valve Performance. An investigation is under way as to whether valve body thinning from erosion will affect the operability of motor-operated valves. A report (EGG–SSRE–10039) was issued on a finite element structural analysis that evaluated the effects of wall thinning. A 16-inch globe valve was analyzed with a wall thinning pattern similar to that observed in the Brunswick (N.C.) plant's residual heat removal system valve, except that significant additional thinning was modeled to simulate a through-wall crack. The results of the analysis indicated that even with this severe thinning the ability of the valve to operate would not be affected. Review of operating history data indicated that corrosion, erosion, and deposition have, in a few cases, contributed to the failure of valves to operate.

Cables. The NRC is currently sponsoring research at Sandia National Laboratories (SNL) to investigate cable condition monitoring methods and cable aging degradation over a 60-year period of plant performance. (See *1991 NRC Annual Report* for background on the research.) During fiscal year 1992, one final report



"Wall thinning" patterns have been observed in valves in various systems of different facilities, such as the residual heat removal system of the Brunswick (N.C.) nuclear power plant. The facility, a boiling water reactor plant located on the Atlantic coast near the North Carolina–South Carolina border, has been in operation since 1974.

(NUREG/CR-5772, Volume 1) was issued, and two more are under final review. Cable connection tests have begun, but no formal results are as yet available.

Snubbers. The Phase II aging assessment of mechanical and hydraulic snubbers was completed with the publication of the final report, "Results of LWR Snubber Aging Research" (NUREG/CR-5870). Recommendations for code revisions to extend snubber life through service life monitoring were made based on this extensive NPAR research.

Service Water Systems. The Phase II service water system aging assessment was completed with the publication of the final report, "Nuclear Plant Service Water System Aging Degradation Assessment" (NUREG/CR-5379).

Engineered Safety Features. An NPAR Phase I aging assessment was completed for high-efficiency particulate air (HEPA) filters and activated carbon gas adsorption units (adsorbers). These key air-treatment system components are affected by stressors that include heat, moisture, radiation, airborne particles, and contaminants. Resulting filter aging mechanisms range from those associated with particle loading to reactions that alter the properties of sealants and gaskets. Aging mechanisms that can lead to impaired adsorber performance include oxidation, as well as the competitive loading of pollutants. The results of this aging assessment will be documented in a Phase I report to be published in fiscal year 1993.

Standby Liquid Control System. An NPAR Phase I study was conducted to determine the effects of age-related degradation on the standby liquid control (SLC) system used in BWRs to provide backup capability for reactivity control in the event of failure of the normal operating systems. The study involved reviews of information on SLC system components and operating experiences, which were obtained from the Nuclear Plant Reliability Data Base System (NPRDS), the Nuclear Document System, Licensee Event Reports (LERs), NRC generic issues, and other data bases. Relatively few SLC component failures were attributed to sodium pentaborate buildup or corrosion. The primary aging concern appears to be setpoint drift in relief valves, which has been discovered during routine surveillance and is thought to be caused by mechanical wear. Degradation was also observed in pump seals and internal valves, which could prevent the pumps from operating as required by the technical specifications. The results of this study have been published in "Phase I Aging Assessment of BWR Standby Liquid Control System" (NUREG/CR-6001).

Chillers. An NPAR Phase I aging assessment was conducted for the chillers that are used in essential safety-related heating, ventilating and air-conditioning systems of nuclear power plants. The primary stressors and aging effects of concern for chillers include vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Other important factors include moisture, non-condensible gases (e.g., air), dirt, and other contamination within the refrigerant containment system and excessive start/stop cycling and underloading of chillers. Aging is also accelerated by corrosion and fouling of the condenser and evaporator tubes. A principal cause of chiller failures is lack of adequate monitoring; a failure to perform scheduled maintenance, as well as human error, is also a factor. The results of this aging assessment will be documented in a Phase I report to be published in fiscal year 1993.

Throttled Valve Cavitation and Erosion. As a result of the erosion of valves used in residual heat removal applications at BWR plants, the Oak Ridge National Laboratory (ORNL) undertook a study to understand the causes of valve body erosion and to identify applications that are most susceptible to erosion. Erosion of valve parts, bodies and adjacent piping was identified to be primarily the result of throttled valve cavitation. The cavitation occurs in throttled valves when the pressure at the minimum flow area of the valve drops below vapor pressure and subsequently recovers in the valve outlet region. This condition results in flashing of the process liquid to steam and subsequent collapse of the vapor back to liquid (or cavitation). The cavitation process causes erosion of the valve body and adjacent piping, particularly when the materials used are not erosion resistant.

Almost three-fourths of the erosion-related, through-wall failures identified from a review of NPRDS data occurred in the service water system. The condensate/feedwater system was also a major contributor. Among the throttled valve applications found to be potentially problematic were:

- Heat exchanger outlet valves, such as those used for component cooling water and residual heat removal exchangers.
- Pressure control valves, such as blowdown or letdown control valves.

In general, throttled butterfly and ball valves are most susceptible to cavitation. Special valve designs, e.g., those with multi-stage trim, can eliminate or minimize cavitation damage. The results of this study are documented in ORNL/NRC/LTR-91/25.

Safety-Related Steam Turbine Pump Drivers. Turbine steam drives for safety-related pumps are used at most of the commercial nuclear power plants in the United States. Turbine-driven pumps are used in PWRs in the auxiliary feedwater system to provide diversity and in crease protection against a common mode failure, such as motor failure, which could cause the loss of the motor-driven pumps. Turbine-driven pumps are also used at BWRs, in the reactor core isolation cooling system and in the high-pressure coolant injection system. During loss of all a.c. power, these pumps provide the only means for supplying emergency cooling water to the systems responsible for decay heat removal and/or keeping the core covered. Evaluation of failure records for this component shows that the turbine governor is the component that is most often involved in reported turbine failures. The data indicated that the majority of failures were discovered during turbine testing. Testing deficiencies were discussed and recommendations for improved testing/surveillance were provided. This information has been published in NUREG/CR-5857, currently under review.

Control Rod Drive Systems for BWR Plants. The BWR control rod drive system study examines and assesses the merits of various methods of managing the effects of aging. Information for this study was acquired from (1) the results of a special control rod drive mechanism (CRDM) aging questionnaire, distributed to each BWR utility in the United States, (2) a "first-of-its-kind" workshop held to discuss control rod drive mechanism aging and maintenance concerns, (3) an analysis of the NPRDS data on failure cases attributed to the control rod drive (CRD) system, and (4) personal information exchange with nuclear industry CRDM maintenance experts.

Utilities evaluate the operability of their CRD systems by performing individual CRDM scram time testing (as required under plant technical specifications) on a weekly-to-monthly basis by step insertion and withdrawal exercises and stall flow testing. When a CRDM fails to meet test timing specifications, begins to show symptoms such as double-notching (erroneously moving two steps instead of one), frequently becomes uncoupled from the control rod blade, exhibits high operational temperatures, or requires excess drive pressure to move, it is usually selected for replacement during a plant refueling or maintenance outage. During an outage, utilities typically replace nearly 16 percent (on average) of a unit's CRDMs with new or rebuilt units.

Nearly 23 percent of the NPRDS CRD system component failure reports were attributed to the CRDM. The CRDM components most often requiring replacement because of normal wear and aging are the Graphitar seals. The predominant causes of aging for these seals are mechanical wear and thermally induced embrittlement. Premature aging of these seals is also caused by excessive amounts of dirt particles, debris, and foreign materials found in the reactor coolant. Some utilities are vacuuming their reactor vessels inside the guide tubes during refueling outages to remove and reduce the amounts of dirt that could travel via the coolant to the CRDM and become entrapped between Graphitar seal sets. This foreign matter creates uneven force distributions at the seal's contact surfaces and causes them to break during scrams (automatic reactor shutdowns).

Throughout the course of this study, it became evident that "as-low-as-reasonably-achievable" (ALARA) dose reduction techniques used during CRD maintenance have become an issue of interest and study to many utilities. CRDM replacement and rebuilding is a procedure involving worker exposure to some of the highest levels of radiation of all intra-plant tasks and is among the most physically demanding and complicated maintenance activities routinely carried out by BWR utilities. Recent innovations in CRDM handling equipment and rebuilding tools have allowed some utilities to make significant reductions in exposures of personnel (as much as 50 percent) during the performance of CRDM maintenance activities.

Heat Exchangers. Heat exchangers are vital components of nuclear power plants, serving as interfaces between both safety-related and non-safety-related systems and components to provide stable conditions during normal operation and the ability to bring the plant to safe shutdown following a design basis event. A review of nuclear plant operating experience by ORNL, documented in NUREG/CR-5779, indicated that inter-fluid leakage caused by corrosion or erosion of tubing is the most commonly identified problem, accounting for approximately 40 percent of the total. External leaks, usually from tube erosion or corrosion in space coolers or from gasket failures, accounted for about 33 percent of the total. In most cases, inter-fluid or external leakage is more of a nuisance that a threat to the operators' ability to bring the plant to a safe shutdown condition. Of more serious consequence is the degradation of the ability of a safety-related heat exchanger to provide design basis cooling. In this category, tube blockage, most often by bivalves or their shells, accounted for approximately 22 percent of the total, and fouling of heat exchange surface by deposit buildup accounted for another 4 percent. These types of problems may not be readily recognized because the exchangers often normally operate at thermal loads that are only a fraction of design loads, and requirements for inservice testing that would indicate degradation have been minimal. NRC's Generic Letter 89-13 requires development of plant-specific inservice testing programs by plant owners. The Operation and Maintenance (O&M) Committee of the American Society of Mechanical Engineers (ASME) standard for inservice testing of heat exchangers, now under development (with ORNL providing data and direct support), should give definitive guidance in detecting degraded capability.

Reactor Core Internals. The NPAR program at ORNL has conducted a Phase I aging assessment study on core internals for LWRs. Westinghouse, Combustion Engineering, and Babcock-Wilcox PWRs, as well as General Electric BWRs, are included in this study. The study identifies potential stressors and aging-related degradation mechanisms associated with the operating environment of core internals. Major stressors and aging degradation mechanisms are selected based on the results of a survey of reported age-related information.

- *Pressurized Water Reactors*. The dominant stressor for PWR internals is oscillatory hydrodynamic forces generated by the reactor primary coolant flow. The associated age-related degradation mechanisms are fatigue, stress corrosion cracking (SCC), and mechanical wear. Major reported age-related failures include thermal shield flow-induced vibration problems, bolting failures in core support structures, fuel assembly damages caused by core baffle water-jet impingement, excessive thinning in flux thimbles and guide tubes, and SCC in control rod guide tube support pins. Uncertainties in the assessment of aging effects on PWR internals include long term neutron irradiation effects and high-cycle fatigue failures in a hostile environment.

- *Boiling Water Reactors*. Major stressors for BWR internals are also related to the reactor primary coolant flow and include oscillatory forces and the presence of dissolved oxygen in the cooling water. Two additional significant age-related degradation mechanisms are SCC and fatigue. Major reported age-related failures include SCC in jet pump holddown beams and feedwater spargers. Questions remain regarding the effectiveness of hydrogen water chemistry programs for mitigating SCC in BWR internals, the long term effects of neutron irradiation, and the effects of high-cycle fatigue in a corrosive environment.

Auxiliary Feedwater System. The Phase I aging assessment found a number of significant auxiliary feedwater system (AFWS) functions that were not tested and certified to be operable by periodic surveillance testing. In addition, the Phase I study identified components actually being degraded by the periodic surveillance tests. Thus, it was decided that the follow-on study would not deal with aging assessments or *in-situ* examination, but would instead focus on the testing omissions and equipment degradation found in Phase I. In the study, the deficiencies found in current practice are categorized and evaluated. Areas of component degradation caused by current practices are discussed. Recommendations are made for improved diagnostic methods and test procedures that will verify operability without degrading equipment. The results will be published in NUREG/CR-5404, Volume 2.

Fire Safety. The NRC is participating in the performance of fire tests in the decommissioned German HDR reactor facility. Recommendations were provided on the test arrangements for the cable fire test that was run in December 1991, involving a large scale cable tray fire in a lower elevation room in the containment building. The NRC provided electric cables and electric relays for installation in the fire room to investigate the effectiveness of cable spatial separation in preventing fire damage and the thermal vulnerability of electrical components to heat and smoke from a fire. Efforts under this program also included participation in an international fire computer code validation comparison using the HDR fire test data. The fire computer code models evaluated include those frequently used in fire risk assessment, such as COMPBRN, for U.S. nuclear power plants.

Reactor Equipment Qualification

Experiments were completed in 1990 to determine whether valves in high-energy pipes will close as they should to prevent leakage during a pipe-break accident outside the containment. The resulting high-velocity flows that develop in the pipe and in the valves must be stopped by the valves. The leakage—if unchecked, and if the valves do not close—can have serious consequences, considering the possible exposure of other emergency equipment to the harsh water and steam environment.

A total of six different valves were tested in 1990, three each having six-inch and 10-inch diameters. The six-inch-diameter valves are typical of those installed in high-energy hot water pipes, while the 10-inch-diameter valves are typical of those installed in high-energy steam pipes. All hardware and fluid environments—flow velocity, pressure, temperature—were selected to simulate actual conditions that would occur in the event of a postulated pipe break accident at some operating nuclear power plants.

Findings from prior years deriving from the six motor-operated valve (MOV) experiments and progress made in evaluating and acting upon the data from the experiments have been covered in past NRC annual reports (e.g., those of 1990 and 1991).

In 1992, the NRC made further progress in MOV technology. Some of the areas where important advances were made include the following:

- An extrapolation methodology that will give the NRC inspectors a way to confirm whether MOV forces measured at low differential pressures on in-situ MOVs can be used to estimate the MOV forces required at high differential pressures was developed. This is important because plants can test many of the MOVs as they are installed in the existing pipes at low differential pressures, where they may not be able to test them at high pressures.

- The NRC has concluded, based on other test data, that accuracy claims for some MOV diagnostic equipment devices that are used by the plants to set up their in-situ MOVs have been overstated. These devices are used to measure important parameters for establishing the setpoints for MOVs.

- The effects of degradation, such as corrosion of internal MOV parts, may significantly affect the free movement of parts internal to MOVs. Steps are being taken to determine whether this effect is significant.

In addition to the sharing of these 1992 research results discussed above, the NRC worked with various industry groups, including EPRI, to seek further assurance of MOV operability. Some EPRI-sponsored research efforts and preliminary results were reported to the NRC during this period, but no firm conclusions have been made at this time. In 1993, the NRC expects to continue research on MOVs to develop the technical bases for evaluating plant MOV capabilities. The NRC will also continue to monitor and evaluate the EPRI research program and results and to exchange technical information on a regular basis. The above results, continuing research and future interaction among interested parties should all contribute to the objective of improving MOV reliability in operating nuclear power plants.

Engineering Standards Support

The national standards program is conducted by the American National Standards Institute (ANSI). ANSI acts as a clearinghouse to coordinate the work of standards development in the private sector.

In 1992, 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. Over 190 NRC staff members serve on working groups organized by technical and professional societies.

Section 50.55a, "Codes and Standards," of the NRC regulations provides a mechanism for integrating into the regulatory process the output of the national codes and standards effort, in particular, the ASME Boiler and Pressure Vessel Code (ASME B&PV Code). During 1992, the NRC published a final rule in the *Federal Register* amending 50.55a to update references to ASME B&PV Code Sections III and XI for the purpose of incorporating improved rules for the construction, inservice inspection, and inservice testing of nuclear power plant components. The final rule expedites implementation of certain new Section XI requirements for examination of the reactor vessel. A proposed rule was initiated in 1992 to update references to later editions and addenda of Sections III and XI and to incorporate by reference, for the first time, the new ASME Operations & Maintenance (O&M) Code, which provides rules for inservice testing of pumps, valves, and snubbers.

ASME Code Cases provide alternatives to the rules specified in the ASME B&PV Code. Regulatory Guides 1.84, 1.85, and 1.147 identify those Code Cases for design and fabrication, materials, and inservice inspection, respectively, that the NRC has found to be acceptable. These regulatory guides, which are updated on a regular basis, were revised and issued in 1992. Although work progressed on revisions to Regulatory Guide 1.36 on nonmetallic insulation and to Regulatory Guide 1.54 on quality assurance of protective coatings to reflect current practices—as identified in new and updated American Society for Testing and Materials (ASTM) standards—the effort was temporarily curtailed at the end of 1992 because of manpower considerations.

Structural Integrity

During fiscal year 1992 nine reports were generated by the structural aging (SAG) research program. These reports describe the program's goals, progress, and accomplishments. They also provide data acquired on aged concrete data, on the effectiveness of field testing for strength of aged concrete structures, on European experience in this area, and on examples of potential applications of the SAG program's information to a variety of structural aging considerations in nuclear power plants. In addition, 11 formal technical presentations were made.

During fiscal year 1992, additional concrete aging data were acquired, and new input was provided to the structural materials aging data base. Work began on the effects of corrosion and cathodic protection systems on concrete reinforcing steel and on steel embedded in concrete. Planning also began on extending the structural aging studies to metal containments and containment liners, which would start in fiscal year 1993.

License Renewal Regulatory Standards

The NRC has been giving extensive consideration to the kinds of requirements that should be placed on nuclear power plants in the event that licenses to operate them beyond the 40-year term of the original license should be granted. Public comments on license renewal requirements have been solicited three times through the *Federal Register*—the first time in connection with seven major license renewal issues (published November 6, 1986), and thereafter as part of an advance notice of proposed rulemaking (published August 29, 1988). The advance notice requested comments on "Regulatory Options for Nuclear Plant License Renewal" (NUREG-1317), issued in August 1988. Comments were summarized and analyzed in "Survey and Analysis of Public Comments on NUREG-1317: Regulatory Options for Nuclear Plant License Renewal" (NUREG/CR-5332), issued in March 1989. A third solicitation of comment occurred when the NRC published the proposed rule for nuclear power plant license renewal, on July 17, 1990 (55 FR 29043). The final rule (10 CFR Part 54), with appropriate supporting documents, was published on December 13, 1991 (56 FR 64943). The NRC, in regular consultation with industry, continued to develop guidance on key license renewal issues throughout fiscal year 1992.

As part of a separate rulemaking, the NRC has undertaken a generic environmental study for the purpose of defining the scope and focus of environmental effects that need to be considered in individual relicensing actions. An advance notice of proposed rulemaking (10 CFR Part 51) was issued on July 23, 1990 (55 FR 29964). Also, a notice of intent to prepare a generic environmental impact statement (GEIS) on the effects of renewing the operating license of individual nuclear power plants was issued (55 FR 29967). The NRC published the proposed rule and draft GEIS for comment on September 17, 1991 (56 FR 47016). Also announced at that time was a public workshop to review the technical basis of the proposed rule. The workshop was held in November 1991. (See the *1991 NRC Annual Report*, pp. 186 and 187, for listing and description of support documents issued with the proposed rule.)

The public comment period closed on March 16, 1992. Extensive comments on the proposed rule and supporting documents were provided by over 120 commenters, dealing with the technical analyses supporting the proposed rule, and also expressing concern that the proposed rule might be too restrictive with respect to allowing public comment on environmental issues at the time of a license renewal review for a particular plant. The NRC continued to work with the Environmental Protection Agency and the Council on Environmental Quality in resolving environmental aspects of the proposed rule. All comments will be taken into account in developing the final Part 51 rule, the GEIS, and other supporting documents. The final rule and supporting documents are expected to be published in 1994.

Reactor Regulation Support

PLANT PERFORMANCE

Reactor Safety Experiments

Thermal-hydraulic computer codes are used to model plant responses to a wide variety of transient and accident events. This provides an evaluation of the adequacy of a plant's engineered safety features to respond safely to such events. Such code analyses are best estimates, that is, they are intended to be an accurate representation of the expected plant response. Since the codes are large and complex, a question remains as to what uncertainty attaches to a single-valued code calculation. An evaluation was performed to quantify the uncertainty of the RELAP5/MOD3 code for calculating a small-break loss-of-coolant accident in a Babcock and Wilcox plant, using Oconee Unit 3 (S.C.) as the example. The analysis was performed for a complete break of a high-pressure injection pipe. A second high-pressure injection pump was assumed to be unavailable, thus leaving only one out of three pumps. The calculation showed that the core remained covered with the worst combination of uncertainties. The key parameter was taken to be the water level in the reactor vessel. The RELAP analysis showed the risk could be calculated accurately and with minimal uncertainty.

Safety Code Development and Maintenance

The International Code Assessment Program (ICAP) was organized by RES and carried out from 1986 until its completion this past year. The program involved approximately 15 nations and sought an assessment of the RES codes RELAP and TRAC. A large number of evaluations were performed by participants, using mostly their own experimental facilities and plant data. Near the midpoint of the program, RES employed the emerging assessment results to identify areas in which models in the codes were deficient. This step led to a code development effort that produced the current code versions RELAP5/MOD3 and TRAC-PF1/MOD2. The ICAP program helped establish RELAP as a predominant world standard safety analysis code.

Operating Reactor Assessments

In support of the NRC's assessment of the continued operation of Yankee-Rowe (Mass.) nuclear plant, RES developed a plant model and performed calculations using the RELAP code. The focus of the analysis was to evaluate the potential for overcooling the reactor vessel, giving rise to potential vessel fracture from pressurized thermal shock. RELAP was used to calculate the overall system response to a small-break, loss-of-coolant accident (SBLOCA), with particular emphasis on system pressure and temperatures. The temperatures calculated by RELAP for the vessel inner surface were provided as a boundary condition to the RES fracture code, VISA. The RELAP analyses confirmed the licensee's evaluation, showing that for this plant a SBLOCA could depressurize the plant sufficiently to allow large coolant injection and rapid cooling of the vessel.

HUMAN RELIABILITY

Through its personnel performance research program, the NRC seeks to improve its understanding of and to maintain effective requirements with respect to the impact of human performance on the safety of nuclear operations and maintenance, whether at power plants or materials facilities.

Personnel Performance

The development of a human factors investigation process was completed during the report period. The process provides a standardized method for investigating events to identify root causes of human errors. Training in the use of the process has been completed at three Regional Offices and Headquarters. Two more training sessions will be conducted as part of the final phase of the research. Thereafter, responsibility for the training will fall to AEOD's Technical Training Center. Work continued on three projects involving a human factors evaluation of operations carried out by materials licensees (industrial radiography, brachytherapy using remote afterloaders, and teletherapy). The projects are intended to identify the causes of human performance problems in these operations.

Personnel performance research continued on the impact of overtime and shift scheduling effects on operator performance, using nuclear power plant data. The laboratory phase of research on the safety implications of routine 12-hour shifts has been completed. Preliminary findings indicate that, although performance of subjects on 12-hour night shift is slightly slower than those on the eight-hour night shift, it is more accurate. Results are currently being incorporated into a final report. Work also continues on the development of a method to assess the effectiveness of training programs at nuclear power plants. The research will develop the necessary measures and supporting documentation for a training effectiveness evaluation method. Research 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—is an ongoing effort. A study on the impact of environmental influences on human performance has been completed. Together with a comprehensive review of the literature in this area, a handbook on the effects of environmental factors on human performance is being prepared for use by nuclear power plant inspectors.

Human-System Interfaces

Human-system interface research entails NRC participation in the Halden Reactor Project of the Organization for Economic Cooperation and Development (OECD) that addresses verification and validation of digital systems, man-machine interaction, surveillance and support systems, and advanced control rooms. Specific NRC research needs on man-machine interface guidelines for advanced control rooms were identified to project participants during a workshop on "Guidelines for Design and Evaluation of Computerized Systems for the Man-Machine Interface."

Following an assessment of the costs and benefits of expanded regulatory guidance on normal and abnormal operating procedures (NUREG/CR-5458), the research turned to the development of guidance for the review of procedures followed during shutdown and low- power nuclear plant operations. Activity continued toward resolution of Generic Issue HF5.1, on local control stations, and on the development of guidance in performing human factors reviews of advanced control and display technology. The draft guideline was reviewed by independent experts, and an international workshop was held to provide for peer review. A study to evaluate the effects of alarm reduction techniques on operator performance and to prepare interim guidance on the safe implementation of computer-based alarms in control room operations continues (Generic Issue HF5.2). Efforts to assess the feasibility of establishing NRC human factors regulatory research facilities is also ongoing.

Initial experimentation to evaluate an indicator of the performance effectiveness of human-machine interfaces has been completed. Among the results, a quantitative memory test was not able to distinguish effectiveness among different displays. The non-quantitative tests showed some promise of useful application, but additional development is needed before such tests could serve as a validated indicator.

Research continued on the identification and documentation of the positive and negative attributes resulting from the use of standards and computer-aided-software-engineering tools in the design, development, evaluation and certification of high integrity software for nuclear power plant safety systems. Research was initiated to develop and test a computer-aided-software-engineering tool for assessing the degree of functional diversity within software performing safety functions. A survey was performed with results used to develop technical bases for regulatory guidance on the design, development, test, and acceptance of computer systems performing safety functions. A project co-sponsored by the Electric Power Research Institute on the verification and validation of expert systems also continues.

Organizational Factors

Research on organizational factors was refocused during the report period to provide for integrated research products that could be more useful in regulatory applications. The development of data-gathering techniques to support inspection and diagnostic evaluation activities was aimed at appraising nuclear power plant organizational effectiveness and performance issues. Field testing of structured interviews and behaviorally anchored rating scales continued at several facilities. Additional research continued to develop alternative quantification methods for incorporating the influence of organizational factors into probabilistic risk assessments (PRAs). "Influence of Organizational Factors on Performance Reliability" (NUREG/CR-5538, Volume 1) was published.

Reliability Assessment

Efforts continued to collect, catalogue and store in a computerized library the estimates of probabilities of operator error and hardware failure. Because one of the largest contributors to the risks assessments is operator cognitive error, two new research projects were instituted to gather data on cognitive performance. A computer simulation has been developed, employing principles of "artificial intelligence," which models the cognitive tasks required of operators during accident scenarios. To validate the model, data were gathered from operating crews responding to a few simulated accident scenarios on training simulators. One product of this research is a list of situations that can make the diagnosis of accident scenarios relatively difficult. The second research project seeks to analyze information from the simulator portion of the NRC-administered operator requalification examinations. This effort should also lead to an improved inventory of situations difficult to diagnose and provide data to initiate validation of previous estimates of the probability of various types of cognitive error, to the extent possible.

Research continued to develop reliability methods for evaluating technical specification changes within the framework of a PRA. These methods are intended to assist in evaluating the risk impact of certain requirements placed on safety systems in technical specifications. The requirements include: (1) surveillance test intervals; (2) allowed outage times; and (3) when an allowed outage time is exceeded, the action statements requiring plant shutdown. These reliability methods support the NRC's program to improve technical specifications for controlling plant risks effectively and efficiently. Related reliability methods analyze the risk impact of scheduling preventive maintenance and are useful in monitoring the unavailability of selected safety systems ("Quantitative Evaluation of Surveillance Test Intervals Including Test-Caused Risks," NUREG/CR-5775).

REACTOR ACCIDENT ANALYSIS

Reactor Risk Analysis

Probabilistic risk analysis is applied by the NRC staff to the resolution of a wide spectrum of reactor regulatory issues. In 1992, these applications encompassed both specific issue-oriented projects and more general work—including development and demonstration of risk analysis methods, and the development of risk-related training and guidance for the NRC staff.

Issue-oriented projects under way in 1992 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 to provide support for the Office of Nuclear Reactor Regulation's (NRR) regulatory analysis and to guide the Phase 2 effort. A significant finding was that the traditional concept of "technical specification" modes of operation does not adequately delineate plant operating boundary conditions (states) 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.

(2) South Texas Risk Analysis. In 1992, the staff completed a review of the South Texas Project risk analysis and documented the results and findings in NUREG/CR-5606. The licensee 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.

Methods development projects performed in 1992 included:

(1) SAPHIRE Computer Tools. A suite of computer codes for the performance of risk analyses has been developed to allow an analyst to perform many of the functions necessary to create, quantify, and evaluate the risk associated with the facility being analyzed. This suite of

codes is called SAPHIRE (System Analysis Programs for Hands-on Integrated Reliability Evaluation). The suite of codes is currently employed by NRC contractors to perform the low-power and shutdown risk analyses described above, to set priorities for the use of agency resources, and to perform regulatory analyses of generic safety issues. During 1992, completed risk analyses for several more licensed nuclear power plants were added to the data base contained in SAPHIRE, bringing the total to 10 plants. Courses have been provided for the NRC staff on the use of these codes. The SAPHIRE codes and user manuals (six NUREG reports) have also been sent to the Energy Science and Technology Software Center at ORNL 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 perform an comparison exercise on probabilistic accident consequence codes. The six codes being evaluated are MACCS (U.S.), COSYMA (Germany), CONDOR (U.K.), OSCAAR (Japan), LENA (Finland), and ARANO (Sweden). The comparison exercise uses a set of standard radioactive accident source terms from which dose consequences such as whole body dose and fatal cancers are calculated with each code. These calculations will be completed in fiscal year 1993 and will provide a data base to judge the performance of the reactor accident consequence codes.

Risk-related training and guidance development in 1992 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 PRA...." The 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.

The group's report, including the guidance on PRA uses and key technical areas, and possible recommendations for training and staffing changes, is planned for completion in April 1993.

(2) Reactor Safety Training Course. In response to a request from the Office of Analysis and Evaluation of Operational Data (AEOD), RES is developing a new course that is intended to treat reactor safety in a broad sense. Topics 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, as well as other NRC staff not generally familiar with these topics. The course will be first offered at NRC's Technical Training Center in Chattanooga in 1993.

Containment Performance

In order to ensure that existing regulations adequately protect the public, the NRC conducts research in a number of areas, among them melt-concrete interactions, direct containment heating, hydrogen combustion, source term, core-melt progression, and fuel-coolant interactions, The overall goals of this research are to develop technical bases for assessing containment performance over the range of risk-significant accidents, 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 used for mitigating the effects of severe accidents.

Melt-Concrete Interactions. In those severe accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity, where it can interact with structural concrete. The consequences of these potential thermal and chemical melt-concrete interactions can have a significant effect on containment stresses, the 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 basemat, the generation of radioactive aerosols and combustible gases, and the overheating of important structures inside 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 is focused on scaled-down experiments simulating a wide variety of concretes used in nuclear power plants in the United States and on consideration of the diverse accident scenarios that may lead to melt-concrete interactions. The analytical research centered on the development of models for studying phenomenological aspects of melt-concrete interactions and included 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 and spans a broad range of melt release conditions, as well as concrete types. More recent experiments on melt-concrete interactions were conducted in the presence of an overlying water pool. The NRC-sponsored WETCOR program—also called the debris coolability program—was designed to address two specific issues: (1) the comparative coolability of oxidic and metallic debris, and (2) the limits of debris coolability in terms of debris composition and depth. A NUREG report describing the WETCOR-1 test, the only integral test conducted under this program, will be published in fiscal year 1993.

The second experimental program on debris coolability, called the MACE program, was developed as an extension of the ACE program under the sponsorship of the NRC, the industry's Electric Power Research Institute (EPRI), and other, largely governmental, agencies in several countries. The MACE program is designed to ascertain the ability of water to quench and fragment prototypic core debris, for a range of debris depth, basemat area, concrete type, and power density, representative of both boiling-water reactors (BWRs) and pressurized-water reactors (PWRs). So far, four tests have been conducted under the MACE program, and at least one test is planned for fiscal year 1993. Except for the last test, which was terminated prematurely, the results from the MACE tests generally support the expectation of crust formation at the melt-coolant interface, with periodic access of water to the melt and partial melt quenching.

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 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 an assessment of the probability of liner failure both with and without an overlying water pool in the drywell—given a core melt accident that proceeds to vessel failure——was completed in 1991 and is documented in NUREG/CR-5423. The results of additional research, performed in fiscal year 1992, in the areas of liner failure criteria, melt superheat, melt spreading phenomena, and melt release conditions confirm the conservatism built into the original study. The NRC plans to issue a final NUREG report on the subject in early fiscal year 1993, with updates from the additional research efforts mentioned above to close the Mark I liner issue.

High-Pressure Melt Ejection—Direct Containment Heating. In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system may remain pressurized. A molten core, if left unmitigated, will slump and collect at the bottom of the reactor vessel. If a breach occurs, the core melt 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 the air or in steam and could generate a large quantity of chemical energy that would further pressurize the containment. This process is called direct containment heating (DCH).

To help develop a data base by which to estimate the containment loads associated with high-pressure core-melt accidents, the NRC has, in fiscal year 1992, completed DCH integral effects testing for a containment configuration simulating that of the Zion (Ill.) PWR plant. A total of 14 integral effects tests were conducted—eight in the 1/10th scale Surtsey facility at the Sandia National Laboratories and the remaining six in the 1/40th scale COREXIT facility at the Argonne National Laboratory. Analysis of the test results is ongoing. In fiscal year 1993, the integral effects tests for a containment configuration modeled on the Surry (Va.) PWR, at two different geometry scales, 1/10th and 1/6th, will be completed at the Sandia National Laboratories.

Besides completing the DCH experiments for the Zion-like and Surry-like containment configurations, appropriate analysis will also be completed in fiscal year 1993 to assess the extrapolation of DCH phenomena to full-size containments using computer models. In addition, a report on DCH issue resolution for PWRs will be completed in 1993. The objective of the report is to document the NRC integrated approach to DCH issue resolution for the Surry and Zion nuclear power plants and to outline the methodology to resolve the issue for other PWRs.

Hydrogen Combustion. Hydrogen combustion research seeks to assess the possible threat to containment and safety-related equipment of hydrogen releases. It is necessary to understand how hydrogen is transported and mixed within the containment and to determine the likelihood of various modes of combustion, i.e., deflagrations, diffusion flames, accelerated flames, transition from deflagration to detonations (DDT), and detonations.

The largest current program in this 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 Center). Under this agreement, a high-temperature hydrogen combustion program related to high-speed combustion modes, i.e., detonations and DDT, has been developed. Two combustion vessels will be used for this research program at Brookhaven National Laboratory. As a result of the cooperative agreement with Japan, the NRC has access to ongoing hydrogen mixing and distribution testing in the latter's Tadotsu facility and the combustion testing in their Takasoga facility. Test results from these facilities will be available in fiscal year 1993 and will provide a greatly expanded and improved data base for the validation of analytical tools. Another joint international program, between the NRC and Germany, involves an evaluation of data from the latter's KfK/HDR hydrogen behavior experiments.

A hydrogen research program is also under way at the Rensselaer Polytechnic Institute to investigate diffusion flame behavior in low-speed hydrogen combustion. Results will be used to help resolve outstanding issues in severe accidents, i.e., hydrogen combustion aspects of DCH, high-temperature combustion phenomena, and detonation initiation by high- temperature steam-hydrogen-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 and validation of computer codes for analyzing severe accident phenomena and for planning accident management strategies. A number of codes (e.g., MELCOR, CONTAIN, SCDAP/RELAP5, CORCON, VICTORIA, COMMIX, HMS) have been developed for examination of various stages of progression in severe accidents for both BWRs and PWRs.

MELCOR

MELCOR is an integrated, engineering-level 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 also 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 U.S. and international user organizations. The focus of the most recent development efforts has been to improve capabilities to handle the phenomena of natural circulation and direct containment heating. These 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. The improvements will be completed in fiscal year 1993. A MELCOR Cooperative Assessment Program (MCAP) is also under way, looking to create an international forum for information exchange on the applicability, limitations, and operational experience of MELCOR.

CONTAIN

CONTAIN is a detailed mechanistic 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 products behavior, flammable gas combustion, melt-concrete interactions, and direct containment heating. The code has the capability to analyze a wide variety of LWR plants, covering a wide variety of accident scenarios and the effects of the operation of engineered safety systems.

One issue currently under extensive investigation is direct containment heating (DCH) and pressurization of the reactor containment atmosphere by molten core materials, ejected as the result of lower head failure of the vessel under pressure. A program to incorporate selected DCH models into the CONTAIN code was initiated in fiscal year 1992. This work will be completed in fiscal year 1993, and the code will be evaluated against the available experimental data. Also, actual plant cases will be run with the updated CONTAIN code to determine the impact of DCH on the containment. Another development is related to containment analyses for advanced light-water reactor (ALWR) designs. Containment designs are being developed by industry for ALWRs that incorporate passive cooling and decay heat removal features for protection against long term containment overpressure in accident situations. Pursuant to this objective, the CONTAIN code was modified in selective areas; it is planned to use the code to evaluate experimental data generated by the industry's research.

COMMIX

COMMIX is a three-dimensional transient single-phase computer program for thermal-hydraulic analysis of single and multi-component engineering systems. The code solves a system of time-dependent and multi-dimensional equations involving mass, momentum, energy and transport. 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 such issues as natural circulation and flow stratification, as well as the effect of non-condensible gas distribution, on local condensation and evaporation. Code upgrades completed in fiscal year 1992 included implementation of the multi-component capability, the development of the liquid film tracking model, incorporation of heat and mass transfer models, and initiation of a code validation effort with the results from the small scale and 1/8th-scale tests on the Westinghouse Passive Containment Cooling System. With these upgrades, the code now has the capability of performing containment analysis of ALWRs for both design basis accidents and severe accidents. COMMIX can also serve as a benchmarking tool for other system codes, such as CONTAIN.

SCDAP/RELAP5

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 has identified several areas of needed modeling improvements. Improvements realized in fiscal year 1992 included (1) reflood oxidation, (2) interaction between fuel rod cladding and Inconel grid spacers, (3) diversion of flow from damaged fuel assemblies, and (4) incorporation of the BWR control-blade/channel-box model developed at the Oak Ridge National Laboratory. These modeling improvements will significantly reduce uncertainties in the code calculation of early phase core-melt progression.

Other SCDAP/RELAP5 research activities initiated in fiscal year 1992 include an extensive peer review effort and model extension to the code, in order to address ALWR issues. The peer review of the code will provide an independent assessment of the technical adequacy of the code. As the late phase core-melt progression data become available, new models will be developed and incorporated into the code. Model assessment and validation efforts will continue, to ensure that SCDAP/RELAP5 meets all design objectives and targeted applications.

CORCON–MOD3

The CORCON code was 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 essentially completed in fiscal year 1992. The 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 sub-cooling 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); incorporation of the VANESA model for aerosol generation and radionuclide release; and inclusion of a solution chemistry model for the calculation of activity coefficients. A topical report has been prepared to describe the phenomenological models and correlations incorporated in the code, as well as to identify accepted limits of validity for the models and correlations. The code was used in fiscal year 1992 to check analyses of melt-concrete interactions involved in calculations of the failure of the Mark I BWR liner. The code was also successfully used to conduct calculations for the International Standard Problems ISP–24 (SURC–4 test at Sandia) and ISP–30 (German Beta Test V5.1), and ACE tests L2 and L6.

VICTORIA

VICTORIA is a computer code designed to analyze fission products behavior within the reactor coolant system (RCS) during a severe accident. The code provides detailed predictions of the release from the fuel and transport in the RCS of radionuclides and non-radioactive materials during core degradation. A new version of the VICTORIA code (VICTORIA–92), completed in fiscal year 1992, includes additional models to account for decay heat structural heatup and its effect on revaporization and possible structural failure; aerosol re-entrainment; deposition of aerosols in sudden contractions and steam dryers; uranium volatilization; enhanced diffusion of fission products within the fuel because of oxidation; permeable flow of fission products through the fuel pores, gaps, and cladding breaches; release of vapors and aerosols during rupture of control rods; mechanisms within the fuel, such as grain boundary sweeping and bubble formation, that affect release from intact fuel; release from degraded fuel geometries (i.e., rubble beds and molten pools); and kinetically limited surface reactions.

HMS

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. Principal code validation activities during the report period included a program to document the development and assessment of the HMS code.

Severe Accident Phenomenology

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 reactor accident. (NRC research in this area is helpful in updating TID–14844, which has been for three decades, in connection with plant siting assessments.)

Research is essentially complete in the development of theoretically based fission product behavior models to predict fission product release and transport in the reactor coolant system (RCS) and the containment. For the RCS, the mechanistic VICTORIA code (see above) has been developed to provide the capability to estimate the quantities of fission products and aerosols released from the reactor core, the extent of their transport through the reactor coolant system, the inventory of radionuclides available for release after core debris is expelled from the reactor vessel, and the extent of fission product revaporization from the reactor coolant system.

The NRC has also entered into an agreement with the Commissariat a l'Energie Atomique of France (CEA) to participate in the PHEBUS–FP program. This program, sponsored by the CEA and the Commission of the European Communities, is centered on studies—in an "in-pile" facility and under sufficient prototypical conditions—of those phenomena that govern the transport, retention and chemistry of fission products in reactor coolant systems and in the containment system, during severe accident conditions. Information on fuel degradation will also be derived from the PHEBUS–FP program. The agreement is of significant benefit to the NRC because, at a relatively modest cost, the NRC can participate in the PHEBUS–FP project over the life of the project. The NRC will be able to obtain integral experimental data to further validate its analytical models for fission product transport in the reactor coolant system and containment and for iodine chemistry in the containment. Information on core-melt progression will also be obtained to supplement data obtained under the NRC Cooperative Severe Accident Research Program (CSARP). This information is supplemental and confirmatory in nature with regard to current efforts to revise the source term assumptions now based on TID–14844. The NRC's PHEBUS–FP facility is undergoing safety review and expects to receive the license to refuel and startup from the funding authorities in January 1993. Six tests are planned in the facility at a rate of one-per-year. The first test is scheduled for March 1993.

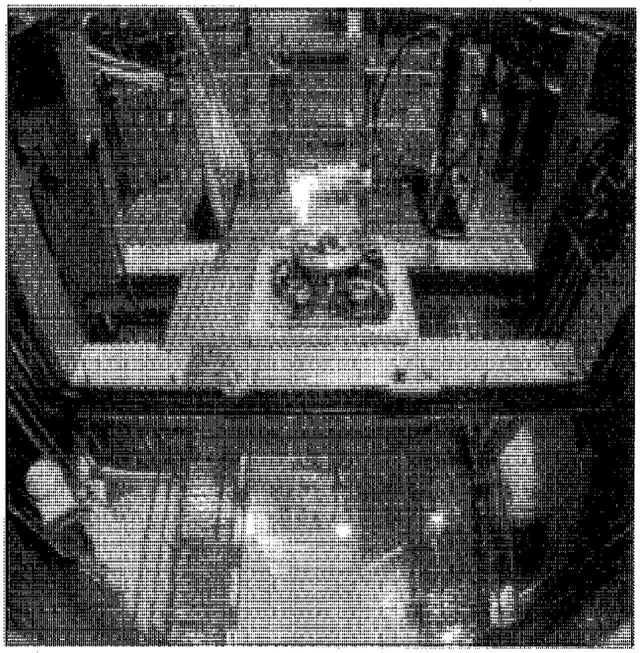
Besides the fission product research cited above, the NRC is participating in an internationally sponsored project called Advanced Containment Experiments (ACE). The project includes four phases of which the first two phases are related to source term research involving large-scale filtration tests and the physical and chemical behavior of iodine in the presence of hygroscopic aerosols, steam and water pools. These phases of the ACE program have been completed.

Core-Melt Progression. "In-vessel core-melt progression" describes the state of an LWR reactor core in a severe accident from core uncovery up to reactor vessel melt-through, in unrecovered accidents, or through temperature stabilization in accidents arrested by core reflooding. Melt progression provides the initial conditions for assessing the loads that may threaten the integrity of the reactor containment. Significant aspects of melt progression are the melt mass, composition, and temperature (superheat), and the rate of release of the melt from the core and later from the reactor vessel at melt-through. 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 also the core conditions for assessing accident management strategies.

Much has been learned about the processes involved in core degradation and in the early phase of melt progression from integral tests in the PBF, ACRR, NRU, PHEBUS, and NSRR reactors, from the LOFT–FP2 test, from tests in the German CORA extra-reactor fuel-damage test facility, and from "separate-effects" experiments on significant phenomena. Most of the available information on late-phase melt progression came from the post-accident examination of the Three Mile Island Unit 2 (TMI–2; Pa.) core, which showed that a debris-supporting metallic blockage formed across the lower core during coolant boildown, from the relocation and freezing of metallic melt. A growing pool of mostly ceramic uranium oxide fuel melt was formed by decay heat in the particulate debris bed and was supported by the metallic core blockage. Eventually, the growing pool melted through the blockage that surrounded it and drained into the water-filled lower plenum of the reactor vessel.

Current NRC research on melt progression is focused on two major issues. The first issue is to determine if there are any accident conditions for BWRs (and possibly PWRs) in which a metallic core blockage similar to that at TMI–2 would not be formed. The second issue concerns the conditions for the melt-through of the growing pool of ceramic (fuel) melt that is supported by the metallic blockage. The melt-through threshold and location determine the mass and other characteristics of the melt released from the core and later from the reactor vessel.

On the issue of blockage of the core by metallic melt, TMI–2 and the results of the experiments cited above



The cooperative research program being carried out at the French PHEBUS facility is designed to study the release, transport and subsequent deposition of fission products under severe accident conditions. The reactor pool at PHEBUS is shown 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 previous experiments for both PWRs and BWRs were performed for these wet core conditions. 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 forming a blocked core, as at TMI-2. This 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. The first of a series of experiments to resolve this question of core blockage under BWR dry core conditions will be performed early in fiscal year 1993.

Preparations were made for the melt progression experiment MP-2, on the process of melt-through of the pool supporting metallic and ceramic crusts by the growing ceramic melt pool in the damaged core, for blocked-core accident sequences like TMI-2. The experiment will be performed in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories, early in fiscal year 1993. Results will be used to assess models of the melt- through process, and these in turn will be used to assess the adequacy of the modeling in severe accident systems analysis codes. Corollary scoping analyses were performed on late-phase melt progression.

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. Under this program, called the TMI-2 Vessel Investigation Project (VIP), test specimens from the lower head of the TMI-2 reactor vessel were removed, in 1990, and initial examinations 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 have provided preliminary 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 critical transformation tempera- ture of the steel. The VIP Management Board decided, in fiscal year 1991, to extend the project until March 1993, in order to perform more detailed testing and examination of the steel samples, in-core instrument tube nozzle penetrations, and in-core instrument guide tubes that were removed from the lower head. Results of examinations performed in fiscal year 1992 have provided additional information on physical properties of the specimens, temperature distributions in the instrument nozzles, and interactions between the molten core material and the vessel. These results were used to perform preliminary scoping calculations of potential reactor vessel failure modes, such as penetration tube failures and global or local failure of the reactor vessel lower head. More detailed analyses of potential failure modes will be performed in fiscal year 1993 to estimate the margin of structural integrity that remained in the TMI-2 reactor pressure vessel. A final report for the TMI-2 VIP will be issued in 1993.

The modes and timing of the reactor vessel lower head failure because of in-vessel melt progression have controlling effects on the subsequent containment loading events in severe accidents. Potential reactor vessel failure modes were under study in fiscal year 1992, both for TMI-2 and for general applications to other reactor vessels. A draft report documenting the results of potential failure modes analysis for a range of debris conditions, lower head designs, and accident scenarios was completed and submitted for peer review in 1992. The failure modes include global "creep-rupture" of the lower head, penetration tube melt- through, tube injection, and ablation by jet impingement of molten core material. A limited vessel wall area may also be heated to a high temperature, as in the TMI-2 accident, resulting in the development of local bulging.

Results of the lower head failure analysis are presented in the draft report in terms of key dimensionless parameters, in order to provide "failure maps" that indicate the relative potential for failure of the lower head, in various failure modes. The report also includes a local bulging analysis (localized "creep-rupture") at high temperature. Such an analysis requires high-temperature material properties data, which were obtained as part of the lower head program for pressure vessel steel and penetration materials. The final report, entitled "Light Water Reactor Lower Head Failure Analysis," will include the failure analysis results and materials properties data mentioned above and will be published early in 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; for example, in NUREG-1150, alpha-mode failure does not seem to be a dominant contributor to early containment failure. In the past, progress in this area has been mainly realized in understanding the 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 fuel-coolant interaction (FCI) research is to provide the appropriate phenomenological and analytical tools for addressing those aspects of FCI which are germane 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, as an integrated module within a mechanistic core degradation code. Work is currently in progress to modify the IFCI module as a stand-alone code for work-station computers. An operational report with examples of runs using the stand-alone version, as well as a code manual, will be completed in fiscal year 1993.

The NRC-funded experimental research program on FCI energetics at the University of California at Santa Barbara (UCSB) has been completed. The objective of this research was to determine under what conditions vapor explosion energetics must be considered and what would be reasonable estimates for the energetic yield. To accomplish the objective, pre-mixing experiments were conducted at UCSB using clouds of hot solid particles. The results from these experiments are expected to provide information of the water depletion phenomenon, as well as a basis for assessing the accuracy of predictions using computational models such as IFCI and PM-ALPHA.

The NRC and the Safety Technology Institute of the Joint Research Center of the Commission of the European Communities (STI-JRC) at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of fuel-coolant interaction experiments at the FARO facility located in Ispra. At the STI-JRC FARO facility, large masses of actual reactor core material can be melted and can interact with different depths of coolant at different temperatures and pressures. At least five molten fuel-coolant interaction experiments will be conducted. The data obtained from FARO will be based on more prototypical conditions and will greatly enhance the existing data base in the United States.

On the matter of water added to a degraded core, the research program on in-vessel flooding in the past year focused primarily on a critical review of past experiments—including PBF, LOFT-FP2, and simulant tests at the Argonne and Brookhaven National Laboratories and at the University of California at Los Angeles. From that review, it became apparent that possible adverse effects of water addition—such as recriticality under certain flooding conditions, in-vessel hydrogen generation and fuel heatup, and the effects of energetic FCI—must be better understood and quantified.

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), in 1984–1989, since it would provide a greater challenge for analytical models. 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, compared to 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, in Albuquerque, N.M. This 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 Implementation

In the 13 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. The NRC then undertook to apply the results of severe accident research directly to the regulatory process while implementing the Commission's Severe Accident Policy Statement. Modification of the Commission's rules or policies regarding siting, emergency planning, containment design, and resolution of severe accident issues are examples of areas in which the results of severe accident research affects the regulatory process.

Containment Performance Improvement. Severe accident research has generated a number of insights concerning containment performance during a severe accident. These insights include both strengths and weaknesses of existing containment designs. In some cases, identified containment weaknesses or uncertainties in containment performance have raised concerns about severe accidents, particularly for BWR Mark I containments. The Containment Performance Improvement (CPI) program systematically examined insights gained from severe accident research, in order to identify containment vulnerabilities and to devise measures to correct them. Because of concerns about Mark I containments, the CPI program initially studied these containments, leading to the recommendation that BWR Mark I licensees backfit a hardened containment vent. Studies of all other types of containments have also been carried out.

The CPI program is closely related and complementary to the individual plant examination (IPE) program (see below). The CPI program examines containments for vulnerabilities on a generic basis and has succeeded in identifying certain features that licensees should evaluate on a plant-specific basis, as part of their IPEs.

All major elements of the CPI program have been completed. Generic letters (GLs) have been issued to licensees to implement the plant-specific backfit of hardened vents for all BWR Mark I containments (GL 89-16, dated September 1, 1989) and requesting that other improvements be considered in the IPE (Supplement 1 to GL 88-20, dated August 29, 1989, for BWR Mark I containments and Supplement 3 to GL 88-20, dated July 6, 1990, for the other containment types). The staff issued a series of technical reports documenting the analyses and evaluations for the various containment types. These reports address the potential vulnerabilities identified (characterization reports), the potential fixes evaluated (enhancement reports), and analyses of the effects of uncertainties (parametrics reports). It is expected that these reports will provide licensees with information they may find useful in assessing their plants as part of the IPE. The last of 12 planned reports was issued in November 1991.

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. 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). The latter activity, discussed below, is referred to as the individual plant examination for external events (IPEEE).

Thirty-one new IPE submittals for internal events were received from licensees in fiscal year 1992, making an overall total of 38 submittals now received. Forty more IPE reports are scheduled for submittal. Staff evaluations were issued on the Seabrook (N.H.) and Millstone Unit 3 (Conn.) submittals, and a draft staff evaluation report was completed on the Turkey Point (Fla.) submittal. Because the Turkey Point IPE submittal was the first not based on a previously reviewed PRA, it became the first submittal selected for the more in-depth review.

In June 1991, the staff issued a generic letter, GL 88–20, Supplement 4, and a guidance document for the IPEEE (NUREG–1407). Licensees' plans and schedules for performing their IPEEEs were received in December 1991.

The staff is also developing a review plan for the IPEEE submittals. The approach for review of the IPEEE will follow closely the plan developed for review of the internal-event IPE submittals. The staff also initiated a procurement process to obtain contractual assistance for the IPEEE reviews.

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 of 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 by which to describe the seismicity in this region and to compare that 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.

With the end of the report period, operational support for the regional networks has all but ended; the function formerly served by the networks is being taken over by the new National Seismographic Network (NSN). (Support for analysis of NSN and regional network data will continue to satisfy NRC's information requirements.) The new network was established through a cooperative agreement with the U.S. Geological Survey (USGS). The NRC has provided the funds for stations of this network in the central and eastern United States and for the satellite receiving station and associated equipment for data processing and storage at the National Earthquake Information Center in Golden, Colo.

The satellite data transmission system and the data receiving and processing facilities at Golden are fully operational. Emplacement of the field stations proceeded more slowly than anticipated, but, at the end of the fiscal year, 18 stations were operational. Installation of new stations is now proceeding rapidly, and completion of the full network is still anticipated during fiscal year 1993. With its dual range, three-component seismometers, the network will carry out the functions of both a micro-seismic and a strong-motion recording network. The NSN is designed for full error-corrected digital data transmission, making data available for rigorous analysis within minutes of the occurrence of an earthquake. The network has the flexibility to incorporate additional stations and regional networks operated by universities and other government agencies. Most of the network components are based on commercially available products, thus minimizing costs and maintenance problems.

In return for NRC's investment in equipment, the USGS will operate the network and provide data to the NRC. A satellite receiving link has been installed at NRC headquarters in Rockville, Md., which will enable the NRC to receive seismic data that the USGS rebroadcasts. In addition, through modem and Internet connections, the NRC staff will have access to data analysis information from the data processing line at Golden.

As noted, most of the support for regional seismographic networks ended at the end of fiscal year 1992. However, because emplacement of NSN stations has taken place more slowly than anticipated, three of the networks on the eastern seaboard will be continued for one year through no-cost extensions to their contracts; they are the Virginia network, operated by the Virginia Polytechnic Institute and State University, and the two New England networks, operated by the Weston Observatory and Massachusetts Institute of Technology.

Northeastern Neotectonics. During fiscal year 1992, investigations were conducted at Newbury, Mass., Moodus, Conn., and Ossippee, N.H., which are areas of historic and ongoing earthquake activity. The effort consisted mainly of screening the areas to identify specific sites meriting detailed investigation because they offer the best prospects of containing seismically induced deformation—such as landslides, rockfalls and slumps, and such liquefaction-prone soil environments as fluvial, glacial outwash, and lacustrine deposits. Extensive trenching mapping and geophysical studies will be carried out in fiscal year 1993 at these specific locations.

Neotectonic Studies in Southeast. A project begun during fiscal year 1991 had the objective of searching for indications of prehistoric earthquakes in the inland areas of South Carolina and in the southern Appalachians. Further investigations of river and lake deposits in South Carolina did not locate any liquefaction features that could be associated with strong earth shaking. While this is a negative result, it does tend to confirm the fact that the Charleston area, where numerous liquefaction features were found in coastal sands, is unique in this region as a locus of repeated strong earthquakes.

The search for indications of strong earth shaking has been continued into the southern Appalachians in Tennessee and Virginia, areas with considerable seismic activity but few deposits that would liquefy in an earthquake. In Giles County, Va., a number of large landslides have been found that may have been seismically induced. In particular, there is strong evidence that Mountain Lake was formed by a catastrophic landslide. Slope stability studies have shown that many of the slide areas should have been stable even under wet conditions and that earthquakes are a likely trigger mechanism in their formation. The next step will be to collect organic soil samples that can be used for age dating with the carbon-14 method.

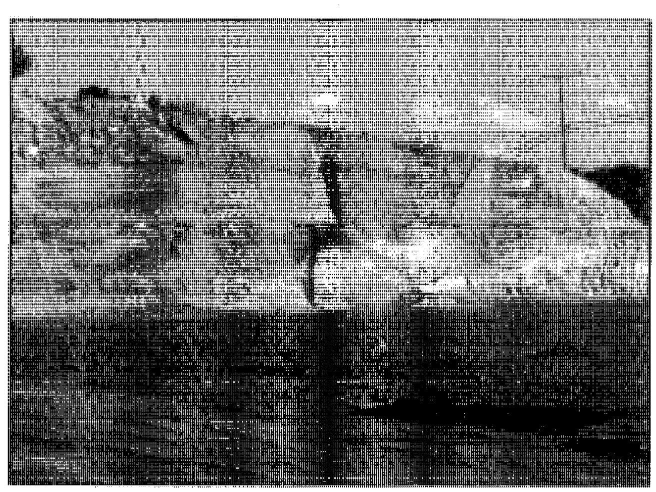
Other candidates for investigation are the numerous caves found in the Appalachians of Virginia and Tennessee. Stalactites and stalagmites that have been broken or deformed may indicate seismic activity. A list of caves with such features has been compiled, and future studies will appraise their usefulness as indicators of seismicity and as a source for radiometric age determinations.

Faulting in Giles County Seismic Zone in Virginia. In June 1992, two faults were discovered at a borrow 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 the 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 70x 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.

Paleoseismicity in Southern Oklahoma. Paleoseismic studies along the Meers fault were completed in October 1989, indicating two left lateral oblique slip surface events during the past 3,200 years, associated with earthquakes ranging in magnitude from 6.75 to 7.25. Another important discovery was that, prior to the 3,200-year-old event, there was a period of quiescence lasting many tens of thousands of years. Surface geologic evidence suggested that the Criner fault, the easternmost member of the Meers-Duncan-Criner fault system, had also experienced late Quaternary displacement. Investigations of the Criner fault were completed in fiscal year 1992, and geologic evidence suggested that, at least at the most diagnostic exposure, displacement of late Quaternary terrace deposits was most likely the result of a late Quaternary landslide and not of tectonic faulting.

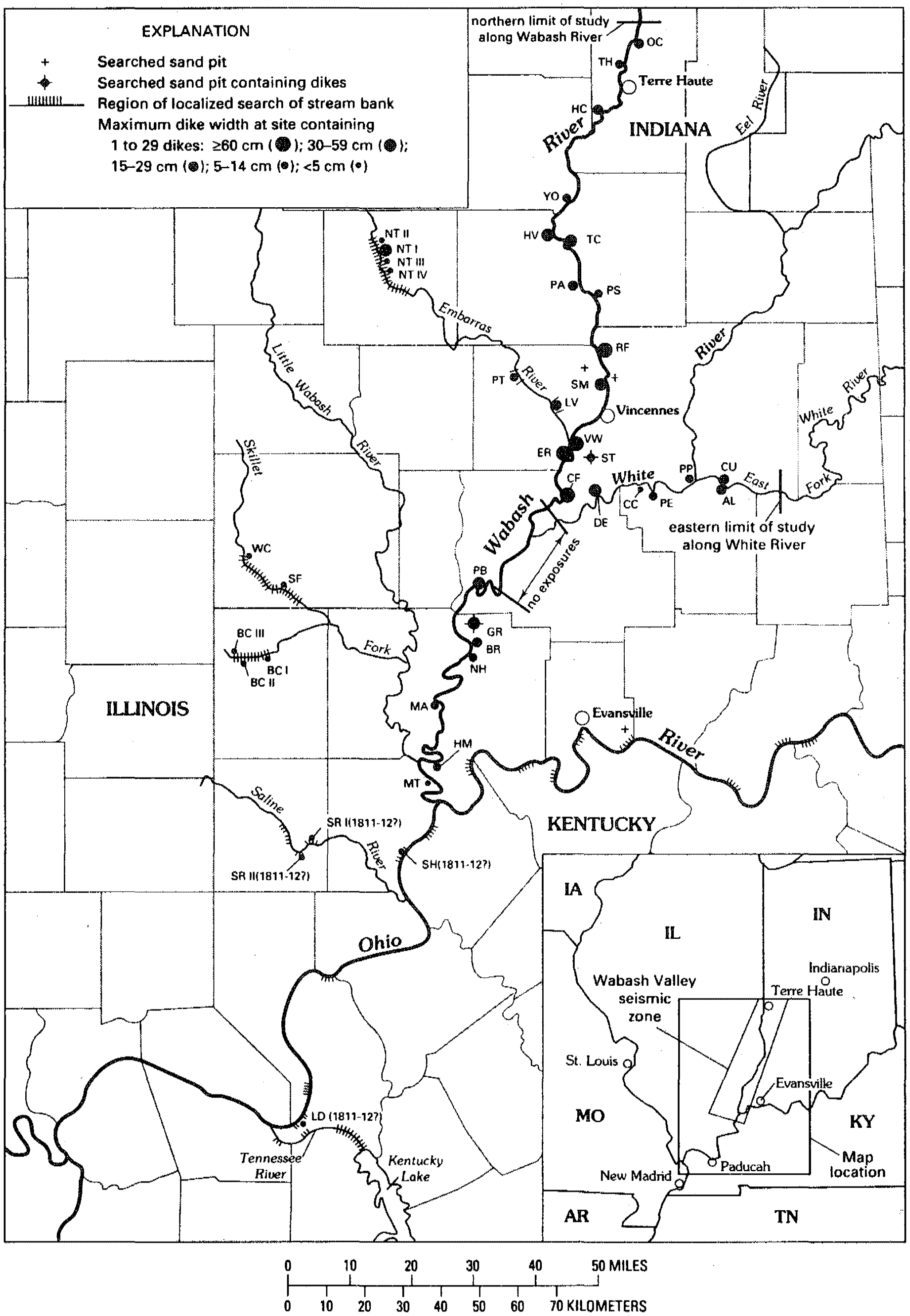
Paleoseismicity in Southern Illinois and Indiana. Investigations began in fiscal year 1991 to identify and analyze paleoseismic evidence along the Wabash River Valley and valleys of its major tributaries. To date, hundreds of planar, nearly vertical, sand-and gravel-filled dikes—most likely caused by 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 being found around Vincennes, Ind., with others, decreasing 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, with an estimated magnitude of about 7.5, that occurred in the Vincennes area between 2,500 and 7,500 years ago.

Pacific Northwest. The Pacific northwest, from southwestern British Columbia to northern California, is underlain by the Cascadia subduction zone, in which three minor oceanic plates—the Explorer, Juan de Fuca, and Gorda plates—are being subducted beneath the North



Above is a fold and graben exposure at Pembroke, Va., in Giles County, where a number of large landslides have been found that may have been seismically induced. (A graben is a depression bounded by faults.)

HOLOCENE EARTHQUAKES IN THE WABASH VALLEY OF SOUTHERN INDIANA AND ILLINOIS



The map shows areas in Illinois and Indiana that were explored by the U.S. Geological Survey for liquefaction features, the sites where liquefaction features were discovered, and maximum dike widths. Sites having dikes are labeled with capital letters; the sizes of the bullets indicate dike width.

American plate. Although geological and geophysical evidence indicates active subduction, there have been no historic large-thrust earthquakes along the plate interface, the phenomenon that characterizes other active subduction zones around the rim of the Pacific Ocean.

The USGS has completed a major 5-year study of the geology and tectonics of the Pacific northwest and continues to sponsor more limited research in the area. The NRC is partially funding several projects under this program, in western Washington and Oregon. These efforts are continuations of investigations that uncovered 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, which have been abruptly terminated by catastrophic subsidence events. 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 are 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. Data available so far could support either hypothesis. Several of the subsidence events in coastal Oregon can be related to deformation along local tectonic structures and others can be attributed to causes other than tectonic.

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 with exposures of liquefaction-susceptible soils along the river banks.

The first positive evidence for seismic shaking that can be ascribed 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, and very numerous, in the vicinity of Astoria, Ore., to fewer in number and smaller in size upstream, running 7-1/2-to-10 centimeters about 30-to-40 kilometers away, and 2-1/2-to-5 centimeters wide about 60 kilometers inland. No features were found beyond 60 kilometers inland from the coast. 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), on younger undisturbed soils, and on the age of the oldest living trees (240 years) unaffected by the event. The evidence for shaking is tentatively correlated with the 300-year-old subsidence event in southwestern Washington.

Geological evidence from excavations at West Point, 10 kilometers northwest of downtown Seattle, is interpreted to indicate that tsunami-like surges of sandy water from Puget Sound covered a tidal marsh that subsided at least one-half meter about 1,100 years ago. Estuarian mud about one-half 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, and 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.

Fault Segmentation Studies. It is well known that faults do not 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. Fault segmentation studies are being carried out to establish a basis for recognizing and identifying the 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 carried out in 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 also 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. That rupture was characterized by strike-slip faulting containing at least three major geometric segments, with echelon steps up to 2.5 kilometers across. Evaluations of these faults will continue through the next fiscal year.

Strong Ground Motion Studies. In 1989, in cooperation with the French Commissariat 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 down-hole accelerometers and a field operable data acquisition system. In 1990, the Electrical Power Research Institute funded the installation of an additional down-hole accelerometer and four surface accelerometers, along with additional 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 6-to-220 meters. The network is located seven kilometers from the San Jacinto fault, at the northern end of the Anza seismic gap on this 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 down-hole 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 them 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 this event was 89 cm/s} 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 encumbered. 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 two 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, including the Miramichi, Nahanni, and Saguenay earthquakes. During the past year the extensive near-field and regional accelerograph recordings from the 1989 Loma Prieta earthquake were analyzed with a view to applying the results to the predicting of strong ground motions in eastern North America.

One of the objectives in the USGS strong ground motion 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 the year, 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.

Another task under way is the prediction of ground motion amplitude as a function of distance, magnitude, and source depth, for earthquakes in central and eastern North America. Research during this report period has consisted of an estimation of ground motions for large earthquakes in the New Madrid and Savannah River regions and an evaluation of source, site, and propagation characteristics for ground motion in eastern North America.

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 of Appendix A to 10 CFR Part 100, 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 industry's Electric Power Research Institute (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 this 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 absolute hazard values will be needed. At the end of fiscal year 1992, an effort was initiated to analyze differences between the LLNL and EPRI seismic hazard methodologies and to arrive at a more unified methodology, one 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—were a cause of 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 discrepancies.

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 active interest in PSHA methods for assessing the numerous critical facilities under its purview. 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 1-1/2-to-2 years.

Plant Responses to Seismic And Other External Events

Besides the earth science research discussed above, the NRC seismic research program encompasses several engineering-oriented programs designed to determine the effect of earthquakes on nuclear plant structures and safety systems.

Revision of Appendix A to 10 CFR Part 100. The staff received permission from the Commission to publish for public comment the proposed revision of Appendix A to 10 CFR Part 100 (see the *1991 NRC Annual Report*, p. 175). The revised criteria will not be applied to existing plants, and the licensing basis for existing plants (Appendix A to Part 100) will remain in the regulations. (The proposed criteria on seismic and geologic siting would be added as a new Appendix B to Part 100; the proposed criteria on earthquake engineering would be added as a new Appendix S to Part 50.)

Several issues were addressed by the staff in conjunction with the proposed revision of the regulations. As noted above, siting criteria remain in Part 100 and engineering criteria have been moved to Part 50; this means, in effect, that siting evaluation has been decoupled from design. In the geosciences area, deterministic and probabilistic seismic hazard evaluations are to be used to assess the adequacy of the Safe Shutdown Earthquake (SSE) ground motion. The proposed regulation (in which the level of detail would be considerably reduced) would establish the requirements, and detailed guidance would be contained in a regulatory guide. In the earthquake engineering area, analysis and design requirements associated with the Operating Basis Earthquake (OBE) ground motion are eliminated provided that the applicant sets the OBE at one-third or less of the SSE. In that case, the OBE serves the function of an inspection and shutdown earthquake, since plant shutdown is required if the OBE is exceeded.

It is expected that the rulemaking notice will be published for public comment early in fiscal year 1993. The total effort consists of the two regulations, four regulatory guides (addressing siting, seismic instrumentation, and post-earthquake plant shutdown and restart), and a standard review plan section revision. During the development of these revisions, the staff had several public meetings with interested industry organizations.

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 (1) determine the influence of relay chatter on circuit breaker tripping, and (2) explain the very large variability in seismic capacity observed when apparently identical specimens of the same relay model are tested, began in fiscal year 1992. 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). The present effort comprises some 3,000 test runs involving various relay models and had its origin in earlier test series where these issues were identified.

Probabilistic Fracture Mechanics Code. A workshop on an NRC-sponsored piping reliability analysis, including seismic events—embodied in the PRAISE code—was given in May 1992, as the development of this piping probabilistic fracture mechanics code was nearing completion. PRAISE includes modeling of both fatigue and intergranular stress corrosion cracking and has been used to determine frequency of initiating events in PRAs, to evaluate repair strategies in cracked piping, to conduct risk- based inspections, to support the application of leak before break, and to assess the consequences of piping design criteria in terms of leak and rupture probabilities of piping. The final phase involves integrating an improved leak rate estimator, developed elsewhere, into the code. PRAISE is known to be operational in Japan, the United Kingdom, Germany, and Finland, as well as at national laboratories, reactor vendors, and utilities in the United States.

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 bigger, more complex 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) program at Hualien, Taiwan, follows the soil-structure interaction (SSI) experiments at Lotung, Taiwan. The planned SSI studies will be performed at a stiff soil site in Hualien, a locale which historically has undergone more destructive earthquakes in the past than Lotung. 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, Korea Power Engineering Co. (KOPEC), and Korea Electric Power Corp.

The duration of the Hualien project is expected to be five years, starting January 1, 1990. The LSST program is moving along according to plan, although obtaining a construction permit for the test model caused some delay. As of the close of the report period, the facility was scheduled for full operation in the fourth quarter of 1992.

Generic Safety Issue Resolution

In December 1983, the Commission approved a priority listing, prepared by the staff at the behest 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 this 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 1992, the NRC identified seven new generic issues, established priorities for 41 issues (Table 1), and resolved seven GSIs (Table 2). Table 3 contains the schedules for resolution of all unresolved GSIs.

Reactor Regulatory Standards

In response to the President's initiative, the NRC in June 1992 (57 FR 27187) published, in a single action, six proposed rulemakings to help relieve the regulatory burden. The rulemakings, which were published as final rules in August 1992 (57 FR 39353), dealt with the following:

(1) *Frequency of Radiological Effluent Reports (10 CFR 50.36(a)).* This action reduces the requirement for the submission of reports concerning the quantity of principal nuclides released to unrestricted areas, in liquid and gaseous effluents, from semiannually to annually.

(2) *Frequency of Final Safety Analysis Report (FSAR) Updates.* This action grants a petition for rulemaking, PRM–50–55, from Yankee Atomic Electric Company (Mass.), which will provide licensees with an option from the current requirements for the annual updating of the Final Safety Analysis Report (FSAR). In lieu of an annual submittal, licensees may choose to provide the required information once each refueling outage. According to the proposed revision, updates to the FSAR may be submitted six months after each refueling outage, provided the interval between successive updates to the FSAR does not exceed 24 months.

(3) *Contamination Monitoring of Packages (10 CFR 20.1906(b)).* This revision clarifies the regulations and reduces the monitoring burden for packages containing radioactive material in the form of a gas or in a special form as defined in 10 CFR 71.4.

Table 1. Issues Prioritized in FY 1992

Number	*Title*	*Priority*
2	Failure of Protective Devices on Essential Equipment	DROP
76	Instrumentation and Control Power Interactions	DROP
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM
89	Stiff Pipe Clamps	MEDIUM
110	Equipment Protective Devices on Engineered Safety Features	DROP
118	Tendon Anchorage Failure	RESOLVED
123	Deficiencies in the Regulations Governing DBA and Single-Failure Criterion Suggested by the Davis-Besse Incident of June 9, 1985	DROP
132	RHR Pumps Inside Containment	DROP
138	Deinerting of BWRs with Mark I and II Containments During Power Operations Upon Discovery of Reactor Cooling System Leakage or a Train of a Safety System Inoperable	LOW
144	Scram Without a Turbine/Generator Trip	LOW
145	Improve Surveillance and Startup Testing Programs	
NEARLY RESOLVED		
147	Fire-Induced Alternate Shutdown Control Room Panel Interactions	Licensing Issue
148	Smoke Control and Manual Fire-Fighting Effectiveness	Licensing Issue
154	Adequacy of Emergency and Essential Lighting	LOW
155.1	More Realistic Source Term Assumptions	NEARLY

Table 1. Issues Prioritized in FY 1992
(continued)

Number	*Title*	*Priority*
RESOLVED		
155.2	Establish Licensing Requirements for Non-Operating Facilities	Regulatory Impact Issue
155.4	Improve Criticality Calculations	DROP
155.5	More Realistic Severe Reactor Accident Scenario	DROP
155.6	Improve Decontamination Regulations	DROP
155.7	Improve Decommissioning Regulations	DROP
156.1.1	Settlement of Foundations and Buried Equipment	Covered in IPEEE
156.1.2	Dam Integrity and Site Flooding	DROP
156.1.3	Site Hydrology and Ability to Withstand Floods	DROP
156.1.4	Industrial Hazards	DROP
156.1.5	Tornado Missiles	DROP
156.1.6	Turbine Missiles	DROP
156.2.1	Severe Weather Effects on Structures	DROP
156.2.2	Design Codes, Criteria, and Load Combinations	DROP
156.2.3	Containment Design and Inspection	DROP
156.2.4	Seismic Design of Structures, Systems, and Components	DROP
156.3.1.1	Shutdown Systems	DROP
156.3.1.2	Electrical Instrumentation and Control	DROP
156.3.2	Service and Cooling Water Systems	DROP
156.3.3	Ventilation Systems	DROP

Table 1. Issues Prioritized in FY 1992
(continued)

RESOLVED (cont.)

Number	*Title*	*Priority*
156.3.4	Isolation of High and Low Pressure Systems	DROP
156.3.5	Automatic ECCS Switchover	Covered in Issue 24
156.3.6.1	Emergency AC Power	Covered in Issue B-56
156.3.8	Shared Systems	DROP
156.4.1	RPS and ESFS Isolation	Covered in Issue 142
156.4.2	Testing of the RPS and ESFS	Covered in Issue 120
157	Containment Performance	Resolved

(4) *Use of Fuel with Zirconium-Based (Other than Zircaloy) Cladding (10 CFR 50.44, 50.46, and Appendix K to Part 50).* This action revised the acceptance criteria in 10 CFR 50.44 and 50.46 to include ZIRLO as an acceptable zirconium-based cladding material. The revision eliminated the need to process recurring exemptions to regulations for the use of ZIRLO.

(5) *Annual Design Change Reports (10 CFR 50.59).* This action revises the requirements for the annual submittal of reports for facility changes under 50.59 (changes, tests, and experiments) to conform with the proposed change for updating the FSAR (see Item 2). Instead of submitting the information annually, the information can be submitted on a refueling cycle basis, provided the interval between successive reports does not exceed 24 months.

(6) *Posting of Rooms Occupied by Diagnostic Nuclear Medicine Patients (10 CFR 20.1903(b)).* This revision reduces the posting requirements for rooms in hospitals occupied by patients administered radioactive materials who might otherwise be released from confinement, under the provisions of 10 CFR 35.75.

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.

Table 2. Generic Safety Issues Resolved in FY 1992

Number	*Title*
29	Bolting Degradation or Failure in Nuclear Power Plants
73	Detached Thermal Sleeves
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown
87	Failure of HPCI Steam Line Without Isolation
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers
121	Hydrogen Control for Large, Dry PWR Containments
151	Reliability of ATWS Recirculation Pump Trip in BWRs

The following chronology outlines progress toward the completion of regulatory guidance to implement the maintenance rule since the publication of the rule on July 10, 1991.

In August 1991, the NRC staff and NUMARC agreed that NUMARC would take the lead in the development of industry guidelines for monitoring the effectiveness of maintenance in nuclear power plants. The NRC staff held a series of public steering group meetings and discussed the planned schedule, the scope, and the criteria for NRC endorsement of the industry guidelines. NRC staff representatives involved in the development of the NRC regulatory guide also visited nuclear power plants where licensees provided presentations on maintenance practices and plans in place.

On January 9 and March 24, 1992, the NRC staff placed drafts of a proposed NRC regulatory guide in the Public Document Room (PDR) and provided it to NUMARC. On March 31, 1992, NUMARC provided a first draft of its industry guideline to NRC for review. On April 22, 1992, the NRC/NUMARC steering group met in a public meeting to discuss the draft industry guideline and the draft NRC regulatory guide. Major NRC staff comments on the industry guideline were that the overall approach and content appeared to emphasize what actions need not be taken. NUMARC agreed to review their document and make changes where needed to clarify their position.

On June 5, 1992, NUMARC submitted a revised industry guideline document to NRC for review. Another public NRC/NUMARC steering group meeting was held on June 12, 1992. The framework of the industry guideline was found by the NRC staff to be acceptable to the NRC staff except for three areas:

(1) Clarification of guidance for performance criteria and related monitoring and annual assessments for the implementation of paragraph (a)(2) of the rule was needed.

(2) Criteria were needed to define "significant reduction" in overall plant safety and acceptable methods to determine risk significance.

(3) More guidance was needed on root cause analysis and determining unacceptable performance for purposes of transferring structures, systems, and components (SSCs) from (a)(2) to (a)(1) of the rule and vice versa.

From June 18 through July 10, 1992, the NRC working group and NUMARC staff working group held a series of nine public meetings to resolve remaining concerns. On July 10, 1992, NUMARC issued the industry guideline, NUMARC 93-01,Rev.2A.

On June 25, 1992, the NRC staff sent SECY-92-229, *Implementing Guidance For The Maintenance Rule*, 10 CFR

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	07/94
105	Interfacing Systems LOCA at LWRs	HIGH	03/93
143	Availability of Chilled Water Systems and Room Cooling	HIGH	01/94
153	Loss of Essential Service Water in LWRs	HIGH	06/93
B-56	Diesel Reliability	HIGH	03/93
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	06/93
HF4.4	Guidelines For Upgrading Other Procedures	HIGH	12/93
HF5.1	Local Control Stations	HIGH	12/92
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	07/93
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	03/94
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	08/93
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM	TBD
106	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	12/92
120	On-Line Testability of Protection Systems	MEDIUM	02/93

Table 3. Generic Safety Issues Scheduled for Resolution
(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
142	Leakage Through Electrical Isolators in Instrumentation Circuits	MEDIUM	08/93
B–17	Criteria for Safety-Related Operator Actions	MEDIUM	04/93
B–55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	06/96
B–61	Allowable ECCS Equipment Outage Periods	MEDIUM	12/93
I.D.3	Safety System Status Monitoring	MEDIUM	12/94
83	Control Room Habitability	NEARLY RESOLVED	05/93
145	Improve Surveillance and Startup Testing Programs	NEARLY RESOLVED	04/93
155.1	More Realistic Source Term Assumptions	NEARLY RESOLVED	03/93
B–64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	10/92
I.D.5(3)	On-Line Reactor Surveillance System	NEARLY RESOLVED	01/93

50.65, to the Commission to outline the staff's progress and current position. The Commission informed the NRC staff that the approach outlined in SECY–92–229 would be acceptable, subject to the resolution of the major issues identified on June 12, 1992.

The NRC-proposed regulatory guide and a regulatory analysis for implementation of the industry guidance document for monitoring the effectiveness of maintenance was scheduled to be issued for public comment in November 1992, as of the close of the report period. The final regulatory guidance is scheduled to be issued by the end of June 1993.

Other Rulemaking Actions. The Commission issued a proposed rulemaking for comment in January 1992 (57 FR 537), 10 CFR Part 50, on training and qualification of nuclear power plant personnel. The proposed rule would amend 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 will consider all modes of operation. The rule would require that the training programs be derived from a systems approach to training, as defined in 10 CFR Part 55. The objectives of the proposed 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* (Pub. L. 97-425). The final rulemaking is expected to be completed in fiscal year 1993.

The Commission issued a denial of a petition for rulemaking (PRM-50-54) from Public Citizens in August 1992 (57 FR 36909). The petitioner requested that the Commission amend its regulations to require the NRC to promulgate rules concerning the licensing of independent power producers (IPPs) and that the rules include specific criteria for financial qualifications for an IPP seeking a construction permit or an operating license for a commercial nuclear power reactor. The petition was denied on the basis that current regulations provide ample authority for the licensing of an IPP, should such an application be submitted, and for a review of the applicant's financial qualifications to construct and operate a commercial nuclear power reactor.

The Commission issued a final policy statement on metrication on September 30, 1992, which was published on October 7, 1992 (57 FR 44202). The statement was in response to the *Omnibus Trade and Competitiveness Act of 1988*. Among other provisions, it specifies that the Commission will publish new regulatory documents in dual units (SI (metrical) first, followed by English units in brackets) and provides for the status of metrication in the U.S. nuclear industry to be evaluated in three years from the date of publication of the policy statement.

Summary of Rulemaking Actions. During fiscal year 1992, 90 rulemaking actions were processed, of which 24 rules were formally published, two were terminated/withdrawn, and 64 are ongoing (see Table 4). Besides the 64 ongoing rulemaking actions, there are 43 potential rulemaking actions, and it is estimated that in fiscal year 1993 there will be approximately 15-to-20 new rulemaking requests requiring review and approval by the Executive Director for Operations.

Regulatory Analysis. The NRC conducts regulatory impact analyses (RIAs) in connection with such regulatory actions as rulemakings, backfits, generic safety issues, regulatory guides, etc. The NRC is currently in the process of updating and revising "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (NUREG/BR-0058, Revision 1), and the "Handbook for Value-Impact Assessment" (NUREG/CR-3568). These documents established NRC's overall guidance and policy regarding the regulatory impact analysis process. These revisions will expand the guidance and structure of the existing operating procedures, the better to integrate backfit analysis requirements and safety goal policy considerations and to update the methods and information bases for performing regulatory impact analyses which will reflect experience gained in recent years. Also, to aid NRC analysts in preparing RIAs, work has begun on updating replacement energy costs and estimating other effects of the long term loss of a plant following an accident. These generic costing methods can be useful in quantifying direct costs to industry and averted on-site costs, both of which are integral components of the value-impact portion of the RIA.

Development of these types of methodologies will continue in an effort to facilitate NRC decision-making in its evaluations of the need for and the effectiveness of a variety of regulatory actions, including rulemaking, standards development, and backfitting safety improvements on nuclear power plants. During the report period, approximately 16 safety-related regulatory impact analyses (both initiated and completed) have been processed.

Radiation Protection and Health Effects

The NRC maintains a program of research and standards development in radiation protection intended to ensure continued protection of workers and the public from radiation and radioactive materials associated with reactor licensed activities. The program is currently focused on improvements in health physics measurements and on the review and dissemination of dose reduction research performed by other Federal agencies and by industry. One goal is to provide acceptable performance standards for the many measurements required of reactor licensees. The program also contributes to monitoring reactor licensee performance in areas such as controlling occupational dose through the use of new dose reduction techniques.

Revision of Part 20 Radiation Standards. The Commission has approved a complete revision to the NRC regulations for radiation protection in 10 CFR Part 20. The final rule was published in the *Federal Register* in May 1991 (56 FR 23360). A rulemaking to amend the Commission's regulation to extend the effective implementation date of this new 10 CFR Part 20 was published for comment in May 1992 (57 FR 21216). The final rulemaking was published in the *Federal Register* in August 1992 (57 FR 38588).

In addition, six regulatory guides critical to the implementation of the revised 10 CFR Part 20 were issued for comment and were published in final form in June 1992. The guides address instructions for recording and reporting occupational radiation exposures; planned special exposures; internal and external occupational doses; air sampling in the work place; dose to embryo/fetus; and medical use programs. Three other regulatory guides supporting Part 20 implementation were also issued for comment.

Table 4. Rulemaking Actions Processed During FY 1992

Rulemaking Activities	*Number*
Final Rulemakings Published	24
Rulemakings Terminated/Withdrawn	2
Ongoing Final Rulemaking Actions	20
Ongoing Proposed Rulemaking Actions	37
Rulemakings on Hold	7
Total Rulemakings	90

The Commission issued a denial of a petition for rulemaking (PRM–20–19) from Betty Schroeder, on behalf of the General Electric Stockholders' Alliance, et al., in August 1992 (57 FR 36611). The petitioner requested that the NRC require a detectable odor to be injected into the emissions of nuclear power plants and other nuclear processes over which the NRC has jurisdiction. The petition was denied on the basis that the proposed action is not necessary because (1) current monitoring and emergency response procedures provide an adequate level of safety; (2) such an action would not result in any increased protection of the public health and safety, and, as a result, would not meet the Commission's "Backfit Rule" (10 CFR 50.109); (3) the proposed action is not technically feasible; and (4) the injection of odors in detectable concentrations over the Emergency Planning Zone for a nuclear power plant or suitable area for other nuclear facility would likely be detrimental to the environment.

Occupational Exposure Data Systems. In 1969, the Atomic Energy Commission began requiring certain licensees to submit reports on occupational radiation dose received by workers. These data are collected and computerized in an NRC system called the Radiation Exposure Information Reporting System (REIRS). The system provides a permanent record of the data and permits expeditious analyses of the two kinds of reports required (annual statistical summaries and individual termination reports). Exposures received as a result of medical procedures are not required to be reported.

A preliminary compilation of summaries of the annual statistical reports for 1990 revealed that about 216,000 persons were monitored, of whom about 52 percent received measurable doses. The workers received a collective dose of approximately 40,230 person-rems or an average annual dose of about 0.36-rem-per-worker among those receiving a measurable dose. The number of workers receiving a measurable dose decreased slightly in 1990. This was offset by a slight increase in the collective dose, which resulted in the average measurable dose remaining the same as for 1989. Of the persons monitored, 90 percent worked in nuclear power plants, and they incurred about 90 percent of the total annual collective dose. After declining for several years, the annual collective dose incurred by nuclear power plant workers appears to have leveled off.

A second kind of exposure report required of certain NRC licensees provides identification and dose data each time a monitored individual terminates work at the licensed facility. Such information is now maintained for some 664,000 persons, most of whom worked at nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories and to analyze the data for trends. The data also assist in the examination of the doses incurred by transient workers as they move from plant to plant. For example, further analysis of the data reported for 77,600 persons terminating employment during 1990 revealed that 3,786 of them had worked at two or more nuclear power facilities and that none of them had received doses in excess of the regulatory limits as a result of their multiple employment.

Brookhaven National Laboratory ALARA Center. The Brookhaven National Laboratory (BNL) ALARA Center, funded by the NRC, continued its surveillance of DOE and industry dose reduction and ALARA research (pertaining to the "as low as reasonably achievable" standard for minimizing radiation exposures) during the report period. BNL has published a series of reports (NUREG/CR-3469) that abstracts 252 national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination. In 1992, BNL focused on making the large dose reduction data base more easily accessible to users and began development of an ALARA landmark.

The center is recognized by the nuclear industry and others as a major source of information on new and effective dose reduction techniques, and its publications are standard references for ALARA planning. The BNL staff is available through the center to the entire NRC organization and to its licensees, for information and advice on all aspects of radiation protection and dose reduction.

In 1992, the BNL ALARA Center worked on an analysis of impacts of implementing new recommendations by the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) for dose limits. The analysis will provide a technical base for future NRC regulatory decisions regarding further changes in worker dose limits. The BNL staff contributed to ALARA assessments of four selected plants resulting in an extensive ALARA checklist that is being published for use by the industry.

Accreditation and Testing of Personnel Dosimetry Processors. An ongoing program that requires accreditation of personnel whole body dosimetry processors became effective in February 1988. Accreditation is acquired through the National Voluntary Laboratory Accreditation Program (NVLAP), operated by the National Institute of Standards and Technology (NIST), and re-accreditation of processors is required every year, with proficiency testing and on-site assessments required every two years. The program goal is to improve and maintain quality assurance and quality control over all aspects of personnel dosimetry processing by requiring all processors to meet the performance requirements of the national consensus standard for processing (ANSI N13.11-1983).

As of April 1992, a total of 77 laboratories, including one in Taiwan, were accredited for processing whole body dosimeters. These include commercial dosimetry processors, military establishments, commercial shipbuilders, nuclear power companies, and other commercial establishments that use radiation measurement techniques.

In the extremity dosimetry areas, a revised standard has been voted on by the Health Physics Society Standards Committee (HPSSC), and it is expected that acceptance by the American National Standards Institute (ANSI) will occur sometime in 1993. Three sets of performance tests against the revised standard (HPSSC P/N 13.32) have been completed, and a final report is being drafted. The tests indicate that the revised standard appears to provide a suitable criterion for testing. Appropriate rulemaking may be initiated soon to require extremity dosimeters to be processed by processors certified under the NVLAP procedures in use at NIST.

New Skin Dose Computer Code. A revised computer code (VARSKIN II) for calculating dose to the skin from radioactive materials on the skin will be published in fiscal year 1993. The code will replace the VARSKIN code, in use since 1986. The revised code will be a great deal more flexible than VARSKIN, allowing for self-absorption of radiation within radioactive particles on the skin and backscattering of radiation, and it will permit the calculation of doses from different shapes of particles and particles separated from the skin by clothing. The code will also calculate the dose from both gamma and beta radiations.

Self-Powered Photon Detector. Research to develop a large area self-powered photon detector (LASPPD)—using a concept similar to that for self-powered neutron detectors (first developed in the Soviet Union in 1961 and improved upon and patented in Canada in 1968)—is complete. A final report on the research will be published soon, as NUREG/CR-4833.

Tissue Equivalent Thermoluminescent Dosimeters. Research is continuing, under contract, to develop a gamma-ray spectrometer/dosimeter. The objective is to demonstrate the feasibility of developing a differential energy absorption spectrometer coupled to a small microcomputer that would have essentially the same response to radiation as that of human tissue, over the energy range of 0.5-10 MeV. Current dosimeters are essentially flat over this range, while tissue response varies by a factor of about eight. Phase I research demonstrated the feasibility of the concept using a four-detector cadmium telluride assembly, but some detector leakage problems arose that prevented low dose measurements. These problems have been corrected. Under Phase II research, construction and testing of a two-detector site monitor has been completed. Work is continuing on the personnel monitor.

"Hot Particles" on Clothing Detector. The rapid detection, measurement, and location of small particulate radioactive material on laundered ("clean") protective clothing is the objective of other research. Under Phase II of this contract, a prototype of a system for surveying clothing was successfully demonstrated. The system has the potential for reducing radiation exposure of personnel who may wear "clean" protective clothing and be unaware that the clothing bears particulate radioactive material. A report of this work was published in September 1992 as NUREG/CR-5868, "Development of Position Sensitive Proportional Counters for Hot Particle Detection in Laundry and Portal Monitors." This report completed work on this project.

Nuclear Materials Licensing And Regulation Support

NUCLEAR MATERIALS

Materials Regulatory Standards

In an action related to the President's initiative to alleviate the regulatory burden, the NRC published for comment in June 1992 (57 FR 24763) a proposed rulemaking (10 CFR Part 35) on "Departures from Manufacturer's Instructions: Recordkeeping Requirements." The final rule was published on October 2, 1992 (57 FR 45566). The revision would eliminate record-keeping requirements related to the justification for, and a precise description of the departure and the number of departures from, the Food and Drug Administration's approved manufacturer's instructions. Upon the effective date of the final rule, licensees would no longer be required to maintain these records.

A final rule (10 CFR Part 71) on modifying NRC's transportation regulations was delayed and is being developed on a schedule consistent with a companion rule to be issued by the Department of Transportation. Public comments on the proposed rulemaking have been evaluated, and the final rule is in preparation, with publication expected in fiscal year 1993. This rule proposes limitations on the shipment of low specific-activity materials and maximizes compatibility between the NRC and International Atomic Energy Agency regulations.

A final rule (10 CFR Part 74) on the material control and accounting requirements for uranium enrichment plants was published in the *Federal Register* in October 1991 (56 FR 55991). The rulemaking followed an accelerated schedule, to accommodate an expeditious review of a license application filed by Louisiana Energy Services for the construction and operation of a gas centrifuge plant that would produce low-enriched uranium for the commercial market. The rule will facilitate the review of that application.

A proposed rule (10 CFR Parts 31 and 32) on requirements for the possession of industrial devices containing byproduct material was published for comment in December 1991 (56 FR 67011). The rule would require general licensees to provide the NRC with specific information about radioactive material used under provisions that establish general domestic licenses for byproduct material. 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 properly accounted for and disposed of. It is expected that the final rulemaking will be completed in fiscal year 1993.

A proposed rule (Appendix H to 10 CFR Part 73) on "day-firing" (of firearms) qualifications and physical fitness programs for security personnel at category I fuel cycle facilities was published for comment in December 1991 (56 FR 65024). These amendments are needed to provide assurance that security force personnel maintain required weapons-handling and marksmanship skills and are physically capable of performing their response duties. The proposed rulemaking would amend the Commission's regulations for day-firing of assigned weapons and require physical fitness training programs, as well as annual performance testing, for specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material. It is expected that the final rulemaking will be completed in fiscal year 1993.

A final rule (10 CFR Parts 70, 72, 73, and 75) amending Commission regulations covering the physical protection of special nuclear material was published in July 1992 (57 FR 33426). The amendments were the result of a systematic review of the NRC's safeguards regulations. These amendments (1) supplemented the definitions section, (2) deleted action dates that no longer apply, (3) corrected outdated terms and cross references, (4) clarified wording that is susceptible to differing interpretations, (5) corrected typographical errors, and (6) made other minor changes.

A proposed rule (10 CFR Parts 26, 70, and 73) on fitness for duty for category I facilities and shipments was published for comment in April 1992 (57 FR 18415). The proposed rule would amend the Commission's regulations for the possession, use, or transport of strategic special nuclear material (SSNM). This action is necessary to ensure that specific employees of licensees who possess, use, or transport SSNM do not have a drug or alcohol problem. It is expected that the final rulemaking will be completed in fiscal year 1993.

A proposed rule (10 CFR 73.40(a) and 73.60) on physical protection requirements at fixed sites was published for comment in May 1992 (57 FR 22670). The proposed rule is necessary to clarify the Commission's regulatory intent that protection against both radiological sabotage and theft of special nuclear material is not required at all facilities. The Commission is also adding a requirement that non-power reactor licensees who operate the reactor at energy levels of two or more megawatts-thermal protect against radiological sabotage, where deemed necessary. It is expected that the final rulemaking will be completed in fiscal year 1993.

The Commission has approved a proposed rule (10 CFR Parts 31 and 32) restricting the accessible air gap of certain generally licensed gauging devices. The rule will be published for comment early in fiscal year 1993. The proposed rule is necessary to amend the Commission's regulations to limit the general license distribution of certain gauging devices that have both an accessible air gap and radiation levels in that gap that exceed a specified value. This action is intended to make it increasingly difficult for personnel to obtain access to the gauge's radiation beam, thereby reducing the frequency and likelihood of unnecessary exposure to radioactivity of plant personnel. It is expected that the final rulemaking will be completed in fiscal year 1993.

A proposed rule (10 CFR 72.214) adding two casks to the list of approved spent fuel storage casks was published for comment in June 1992 (57 FR 28645). The proposed rule would increase the number of spent fuel storage casks that holders of operating licenses for power reactors can choose from to store spent fuel under a general license. It is expected that the final rulemaking will be completed in fiscal year 1993.

A final rule (10 CFR Part 35) specifying a new control number that allows the NRC to collect information from medical-use licensees—in reference to a regulation entitled "Quality Management and Misadministration"—was published in September 1992 (57 FR 41376). Without the reporting and record-keeping requirements, it would not be possible to implement and enforce that regulation effectively. The Commission believes that its requirements for written quality management programs and misadministration reports, if complied with, have a reasonable likelihood of decreasing misadministrations (e.g., administrations of the wrong dose and/or to the wrong patient) with a small incremental cost to licensees. It is expected that the final rulemaking will be completed in fiscal year 1993.

A proposed rule (10 CFR Parts 30, 40, 50, 70, and 72) allowing "self-guarantee" as a means of ensuring the availability of decommissioning funding will be published for comment in November 1992. The Commission is proposing to grant a petition for rulemaking, PRM–30–59, from General Electric and Westinghouse. The petitioners requested that the Commission allow large, financially sound companies (other than utilities) to self-guarantee decommissioning costs, obviating the need for surety bonds or other mechanisms of third-party guarantee. It is expected that the final rulemaking will be completed in fiscal year 1993.

Materials Radiation Protection and Health Effects

Materials Rulemaking. A proposed rule for large irradiators was published for public comment in the *Federal Register* in fiscal year 1990 (55 FR 29043). A two-day public workshop to discuss the proposed rule was held in Rockville, Md. Large irradiators are defined as those capable of delivering a dose of 500 rads in an hour to a person standing one meter from the sources. The final rule will be published early in fiscal year 1993.

A rulemaking to conform the NRC's regulations to a recent amendment of the Atomic Energy Act regarding enrichment facilities was published for comment in September 1991 (56 FR 46739), with issuance of the final rule in April 1992 (57 FR 18388). The rule will amend the Commission's regulations regarding the licensing of uranium enrichment facilities to reflect changes made to the *Atomic Energy Act of 1954*, as amended by the *Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990*. The principal effect of these amendments is that uranium enrichment facilities would be licensed subject to those provisions of the 1954 Act which pertain to source material and special nuclear material, rather than to provisions pertaining to a production facility.

Embryo/Fetal Dose from Maternal Intake. A study to improve understanding of the contribution of radionuclide intake by pregnant women to prenatal radiation exposure of the embryo was continued in fiscal year 1992, with significant progress. The NRC published Revision 1 to NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses," in March 1992. The report provides a method for calculating internal doses to the embryo/fetus with an expanded data base that includes uranium and additional isotopes of previously included elements, such as strontium-89, cesium-134 and plutonium-238. Research that will permit inclusion of other radionuclides—such as technetium, molybdenum, and additional transuranic elements—is planned. The methods and data developed in the project have been adopted by the NRC in preparing Regulatory Guide 8.36, "Radiation Dose to Embryo/Fetus," describing acceptable methods of compliance with Section 20.208 of the revised 10 CFR Part 20. This guide will be revised as new information warrants. The method will also be used to calculate doses in cases of accidental releases of radioactive materials.

Improvement of Health Effects Models. Revision 1 to NUREG/CR-4214, "Health Effects Models for Nuclear

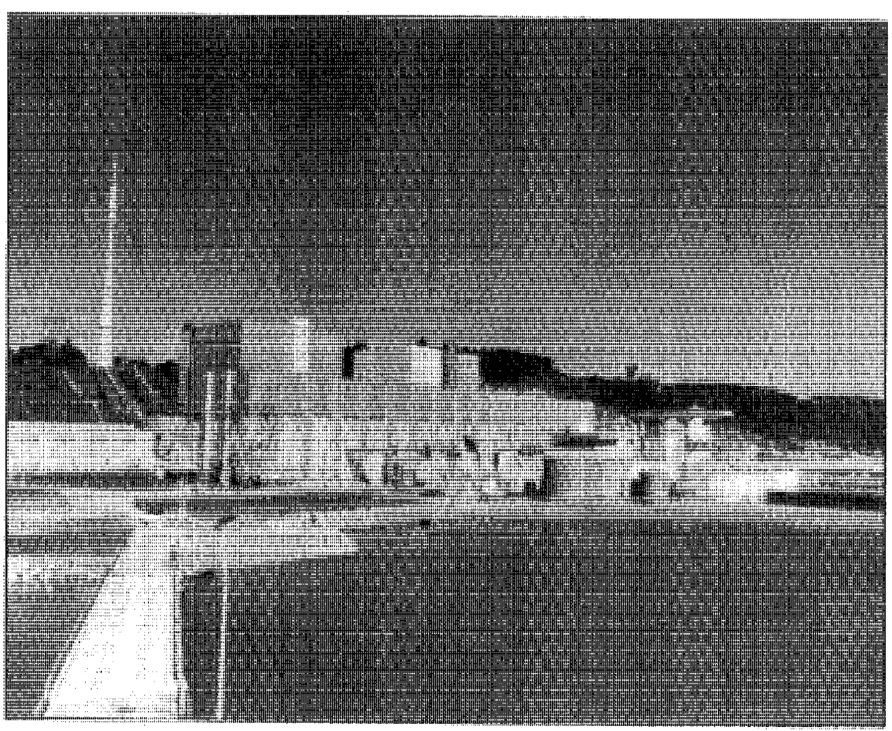
Power Plant Accident Consequences Analysis," published in May 1989, contains health effects models and risk coefficients intended for use in severe accident analyses, probabilistic risk assessments, emergency response planning, and safety goal and cost/benefit analyses. An addendum, "Modification of Models Resulting from Recent Reports of Health Effects of Ionizing Radiation," was published in August 1991. The reports that led to the modification of 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," will be published in fiscal year 1993.

Low-Level Waste Disposal

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 to 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 which were well attended by State representatives—such as "Waste Management '92" and the Annual DOE LLW Management Conference.

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, of which 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 alternative to shallow land burial, while the Idaho National Engineering Laboratory (INEL) has continued to evaluate concrete barriers in limiting radionuclide transport. Results were published in NUREG/CR-5445. 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 superplasticizers that may be used in concrete, in order to reduce its transport properties and improve its strength. Three reports were published in fiscal year 1992: (1) NIST results on modeling transport processes in concrete (NUREG/CR-4269), (2) the diffu-



Decontaminated low-level nuclear waste from the Peach Bottom (Pa.) plant, shown here, was solidified in cement in a process which is under study at the Idaho National Engineering Laboratory. The examination is being carried out to ensure that radionuclide leaching characteristics, as well as the compressive strength of the cement solidified state, are consistent with NRC requirements for waste form stability.

sion of chloride ions in concrete (NUREG/CR-4549), and (3) the use of silicon aggregates for concrete in LLW applications (NUREG/CR-4235). In 1992, a new effort was launched at NIST to develop computer models on the degradation of concrete for LLW performance assessment calculations.

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) that is solidified in cement is being tested at INEL. The studies are aimed at ensuring that the radionuclide and chelating agent leaching characteristics, as well as the compressive strength of the cement solidified waste, are consistent with NRC technical positions, and requirements of 10 CFR Part 61, for waste form stability. Field lysimeter studies of radioactive ion-exchange resins solidified in cement and vinyl ester-styrene are being conducted at the Oak Ridge and Argonne National Laboratories to determine whether radionuclides are released from solidified waste forms under certain environmental conditions. Studies were begun at INEL to investigate biodegradation of solidified LLW by micro-organisms to ensure stability requirements are met as required by 10 CFR Part 61. Measures are under way at the Pacific Northwest Laboratories (PNL) to investigate activated metal and radioactive waste streams for new radionuclides not included in the listing of long-lived radionuclides in 10 CFR Part 61, to determine scaling factors for the assessment of hard-to-measure radionuclides in LLW, and also to obtain activated metals from operating reactors for leaching and field lysimeter research studies.

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. 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, but its long term performance remains to be assessed.

Hydrology and Geochemistry

Radionuclide Migration in Soil. A significant area of uncertainty in the process of predicting site performance is the degree to which soils can retard radionuclide migration. To reduce the uncertainty, the NRC is investigating mechanisms controlling radionuclide movement through soils. The Sandia National Laboratories (SNL) are working to characterize various retardation mechanisms. The University of California at Davis is investigating the mechanisms and kinetics of silicate mineral weathering. PNL is examining the role played by micro-particulates and naturally produced organic complexants. And the University of California at Berkeley is doing a scoping study on the retention of anions in soil. A literature review indicates that soils formed on volcanic parent materials promise to be effective in immobilizing long-lived radioactive isotopes of technetium and iodine, that migrate through soils in an anionic form. Berkeley is also conducting laboratory tests of the anion exchange capacity of soils collected from volcanic terrains in the western United States. These projects began late in fiscal year 1991 and research results are still preliminary.

Hydrology and Contaminant Transport. The NRC continues to sponsor field tests of flow and transport in unsaturated soils at a New Mexico State University field site near Las Cruces, N.M. The program—which includes NRC-sponsored research by PNL, the University of Arizona, and the Massachusetts Institute of Technology (MIT)—is intended to confirm the reliability of unsaturated flow and transport models of LLW disposal facilities. The work is a part of the INTRAVAL international study that deals with model validation of ground-water flow and transport models.

Compliance Assessment and Modeling

Performance Assessment. Research is continuing on a performance assessment methodology, with respect to low-level waste disposal facilities. Emphasis is being give to engineering enhancements to shallow land burial. SNL is assessing the validity of performance assessment models, and INEL is exploring mathematical models for radionuclide transport through concrete. MIT has been investigating the use of stochastic methods for dealing with the large-scale non-uniformity of site hydrologic characteristics. The University of Arizona and New Mexico State University are working in cooperation with MIT by providing a field test at Las Cruces, N.M., of MIT's theoretical work.

LLW Source Term Modeling. Development of the LLW source term code, BLT (breech, leach, transport), continued during fiscal year 1992. The Brookhaven National Laboratory has refined and expanded the transport submodel to consider geochemistry and gas transport. In order to provide greater confidence in the model predictions, the BLT code continues to be appraised against lysimeter experiments of saltstone waste forms at the Savannah River Laboratory and the cement, bitumen, and polymer waste forms at PNL. Results of sensitivity analyses continue to be used to assess radionuclide releases as a function of key parameters. These efforts represents a first attempt at the quantification of source terms for use in performance assessment.

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 for comment in April 1992. The rule would improve the quality and uniformity of disposal of 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, though they had not yet been made final and were subject to the Presidents's regulatory moratorium, as of the close of the report period. It is expected that a final rule will be published in fiscal year 1993.

A proposed rule to amend 10 CFR Part 61 to clarify that requirements related to the performance of land disposal facilities for LLW are applicable to above-ground disposal (i.e., built on the ground without an earthen cover) was published for comment in March 1992. The final rule will be published in fiscal year 1993.

The Commission is considering a petition for rulemaking (PRM-60-4) from the States of Washington and Oregon. The petitioners requested that the Commission establish a process, outlined by them, to regulate the processing and separation of tank wastes at Hanford, Wash. The petitioners also proposed that the Commission change its definitions of "high-level waste (HLW)" and "HLW facility" in 10 CFR Part 60. The staff examined the petition in light of existing regulations and the available facts, including information received during a meeting with DOE held on July 16, 1992. The staff recommendations on the need for rulemaking were sent to the EDO for approval near the end of the report period, on September 21, 1992.

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 low-level 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 will be determined in fiscal year 1993.

Environmental Policy and Decommissioning

A proposed rule (10 CFR Parts 30, 40, 70, and 72) on decommissioning was published for public comment in October 1991 (56 FR 50524). The proposed rule would amend 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 "Synar Committee" hearing on decommissioning and an earlier GAO report. The final rule is expected to be published early in fiscal year 1993.

A final rule (10 CFR Part 20) on disposal of waste oil by incineration at nuclear power plants is expected to be published early in fiscal year 1993. The rulemaking action—responding to a petition for rulemaking, originally filed by the Edison Electric Institute and the Utility Nuclear Waste Management Group (PRM-20-15)—would allow reactor licensees to pursue the option of incineration of waste oils contaminated with small amounts of radioactivity without the need for a specific authorization.

The Commission has approved a proposed rulemaking (10 CFR Parts 30, 40, 70, and 72) on timeliness in decommissioning a materials facility. The proposed rule, which will be published for comment early in fiscal year 1993, would amend the Commission's regulations to establish timeliness criteria for the decommissioning of nuclear internal sites or separate buildings or outdoor areas, following permanent cessation of licensee activities.

The Commission approved an advance notice of proposed rulemaking (ANPRM) which would update Part 40 of the regulations governing source material. This ANPRM was published for comment October 28, 1992 (57 FR 48749). The ANPRM solicits comments on a number of specific issues related to exemptions, general licenses, and specific licenses, and on any issue that may relate to improving the control of source material. In connection with this ANPRM, the Commission will issue NUREG/CR-5881, "An Examination of Source Material Requirements Contained in 10 CFR Part 40," early in fiscal year 1993. This report provides additional background and discussion of options for rulemaking for updating the requirements for source materials.

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 September 1992. The complete report will consist of three volumes and one supplement. This first volume is to provide screening models, mathematical formulations for the screening models, and referenced parameter values for estimating doses (above natural background) to individuals from the residual radioactivity associated with lands and structures of decommissioned licensed facilities. The modeling structure permits the use of either generic or site-specific parameters as screening estimates of radiation doses from multiple environmental pathways.

The staff effort on the development of information on the safety, costs and wastes related to the decommissioning of LWRs is continuing. As stated in SECY–91–164, the staff expects the completion of revised cost estimates for LWRs in October 1993. In addition, the staff is beginning a study to examine the cost of spent fuel management and storage, and its impact on decommissioning costs. Besides the development of information on the safety, costs and wastes related to the decommissioning of LWRs, the staff has a separate study under way on decommissioning costs for non-reactor facilities.

A new initiative on safety and regulatory issues related to the permanent shutdown of nuclear reactor plants before decommissioning has been funded. The staff expects the results of this study to provide some of the technical bases necessary to define the technical and safety criteria to be applied to nuclear power reactors permanently shut down, from the time of permanent core offload to actual decommissioning. The final regulatory guides on standard format and content of plans for reactor decommissioning and associated record-keeping will be coordinated with the results of this study.

The staff is continuing to research technologies for the disposal of radioactive materials. A technical support document, "Evaluation of Exposure Pathways to Man from Disposal of Radioactive Materials Into Sanitary Sewer Systems" (NUREG/CR–5814), was published in May 1992. The report examines potential radiological doses to members of the public should radionuclides be released into sanitary sewer systems in the maximum amounts allowed under 10 CFR Part 20. A follow-up study to examine potential radiological doses to the public from radionuclide releases into sanitary sewers from the excreta of medical patients will be initiated early in 1993.

In June 1992, the NRC issued a draft "Manual for Conducting Radiological Surveys in Support of License Termination" (NUREG/CR–5849) for comment. The final NUREG will be published in fiscal year 1993.

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 related to the management of high-level radioactive waste (HLW). The research combines theoretical study with laboratory and field experiments to improve understanding of the physical processes that control and determine repository performance in the unsaturated volcanic tuff at the Yucca Mountain site (Nevada), currently under consideration by DOE as directed by the Congress in December 1987. The ultimate goal of the NRC's HLW management research is to provide the technical bases for the licensing staff to make independent judgments as to the appropriateness and adequacy of DOE's demonstration of compliance for the HLW repository with NRC requirements and with the Environmental Protection Agency's HLW standard. Key technical issues being addressed include unsaturated flow and transport mechanisms, assessment of the potential for volcanic and seismic events, geochemical processes, and the long term performance of engineered waste isolation systems.

Engineered Systems Research

Stability of Underground Openings. When specifying suitable site conditions for an HLW repository, Federal regulations (10 CFR Part 60) specifically require 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 the ground water levels that could violate established 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 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.

Thermohydrological-mechanical (THM) 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 interaction is bound to be flawed. The NRC is a participant in an international multi-disciplinary and cooperative research effort to study the coupled THM processes, under the acronym DECOVALEX (*DE*velopment of *CO*upled Models and their *VA*lidation against *EX*periments). 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. The fiscal year 1992 effort consisted of formulating THM codes and comparing modeling results against measurements obtained from THM experiments.

Materials Science. An understanding of the materials science aspects of the engineered barriers in high-level nuclear waste disposal systems is necessary in order for the NRC to judge whether test data and models offer reasonable assurance of compliance with regulatory requirements. During 1992, the CNWRA evaluated potential stress corrosion cracking problems for a number of alloys under consideration as HLW package materials, and it performed extensive research on the rates of various possible corrosion processes on these alloys, when exposed to waters that had been in contact with tuff.

The CNWRA also began scoping investigations of material corrosion on ancient Minoan cooper, bronze and lead artifacts that were buried under silicic tuff 3,600 years ago. The artifacts, currently in the possession of the Greek Government, exhibit considerable corrosion from the 3,600-year period in a semi-arid environment under what is presumed to be hydrologically unsaturated conditions.

Geologic Systems Research

Hydro-geology. Since transport by ground water is the most likely path by which most radionuclides from disposed high-level waste might reach the environment, the NRC is actively studying the movement of ground water in partially saturated fractured rock, similar to that currently under consideration by DOE. An experimental site has been located in partially saturated fractured tuff (very similar to that being characterized by the Department of Energy at Yucca Mountain (Nev.) where field and laboratory testing is being conducted by University of Arizona scientists. The objectives of the field study are to (1) determine what types of measurements and instrumentation are needed to characterize flow and transport in fractured rock, and (2) develop analysis strategies and methods for modeling ground-water flow and transport of liquid and vapor phase contaminants in fractured rock. This work currently entails assessing techniques and methods for measuring rock and ground-water properties in place and assessing infiltration, ground-water recharge, and deep movement through fractured rock. The project is using numerical calculations of flow and transport to assess the (1) importance of site features, (2) appropriateness of fracture models, and (3) theories and measurements of flow-controlling properties and processes.

Investigators at the CNWRA in San Antonio, Tex., are examining methods to perform detailed hydrologic analyses for repository-scale ground-water flow systems. The validity of conceptual and numerical models used to describe ground-water flow and radionuclide transport for various hydro-geologic settings is being evaluated in an international project called INTRAVAL. 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 carried out under a cooperative agreement between NAGRA (Switzerland) and the NRC, negotiated during fiscal year 1987, continue to augment the field-testing program cited above.

Geochemistry. Knowledge and application of geochemistry is important to an understanding of many aspects of nuclear waste repository performance, including problems related to waste package corrosion, radionuclide release and transport, and alteration of ground-water flow paths. The NRC has an active research program in geochemistry, as it affects the management of HLW. In 1992, the rates of reaction of several important mineral phases in the repository were explored, and it was found that reaction rates measured early in the experiments were more rapid than those measured after many days. This means that data on reaction rates must be carefully evaluated with respect to experimental design, sample preparation, and the duration of the experiment. Experiments were undertaken on the process, called "ion exchange," that retards the movement of dissolved radionuclides. Results of these experiments, using variable solution compositions, provided confidence in the performance of thermodynamic models to describe and predict ion exchange over a wide range of conditions for a number of radionuclides.

Since 1988, the NRC has been one of five countries participating in the *International Alligator Rivers Analog Project (ARAP)*. The five-year program originally planned for this study of radionuclide transport at a uranium ore body in Australia was completed in 1992. The results of simple transport models have been compared with site data, and more sophisticated transport modeling has been completed. Seventeen final project reports have been prepared to provide an extensive data base for use in evaluating geochemical and transport models. The project showed the need for integration of multi-disciplinary data to describe a complex site in which ground-water flow occurs by both matrix and fracture flow. Geophysical, geological, geochemical and hydrological data were combined to understanding the long term processes controlling radionuclide transport. Hydrological and geochemical modeling appeared to describe and predict the observed conditions and the evolution of the site. Performance assessment models were the subject of considerable scrutiny and a focus for further research and development.

The NRC is sponsoring work by the CNWRA to investigate contaminant transport in an unsaturated tuff at a natural analogue site, in Pena Blanca, Mexico. The site is a tuff-hosted brecciated uranium ore body, which is analogous in many respects to the proposed repository at Yucca Mountain. The site is under study in order to better understand the nature of contaminant transport in a fractured, unsaturated tuff (i.e., to assess the relative roles and interaction of the matrix and fractures in transport and the alteration of uraninite (uranium oxide, $UO2$)) and in an oxidizing environment. Detailed geologic, fracture, and gamma spectroscopy maps have been completed on the cleared, exposed surface of the ore body. Preliminary results indicate the transport of uranium out of the system is slow in comparison to the oxidation of uraninite. Uranium tends to be concentrated along iron-stained fractures, a phenomenon supporting, in general, the findings from the ARAP project on the association of uranium sorption with ferrihydroxides. Migration is generally thought to be fracture controlled; however, extensive samples have been collected both along fractures and across fractures, in seeking to determine the relative mobility of uranium in the fractures, as against the matrix. Uranium series disequilibrium studies are also being conducted to determine the extent and nature of uranium mobility.

Geology. The NRC has started a research project in volcanism, in order to better evaluate the potential for disruption of a nuclear waste repository by igneous activity. The initial work at the CNWRA focused on determining the extent and availability of volcanic, tectonic, and geophysical data from the region surrounding Yucca Mountain. An extensive survey of the literature on the subject was completed during the report period.

Performance Assessment Research

The NRC will assess the claims of compliance made by the HLW licensee, the Department of Energy, 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 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 critical to an assessment of compliance.

The NRC is sponsoring research at the CNWRA to (1) evaluate current models used in performance assessment, (2) develop models for disruptive scenarios, and (3) improve numerical efficiency in the computational tools. The CNWRA scientists will evaluate the technical requirements for assessing an HLW repository and identify important issues related to scenario identification and probabilities and conceptual model formulation, implementation, and solution. This effort will provide a framework for future development of new capabilities and improvement of existing methods.

Proceedings And Litigation

Chapter

9

This chapter covers significant activities, proceedings and decisions of the NRC's Atomic Safety and Licensing Boards, as well as noteworthy decisions of the Commission itself; the chapter includes a judicial review 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

The Atomic Energy Act requires that a hearing precede every issuance of a construction permit for a nuclear power plant or related facility. In addition, the Act and implementing rules provide hearing opportunities for other matters, such as amendments to reactor licenses, antitrust issues, enforcement actions, civil penalties, the licensing of nuclear materials, and special matters the Commission directs to be heard. Hearings provide individuals and organizations an opportunity to voice their concerns before an independent tribunal and provide a means for NRC license holders to contest Commission actions that they dispute. (See "Licensing the Nuclear Power Plant," in Chapter 2.)

Adjudicatory hearings at the Nuclear Regulatory Commission are conducted by administrative judges sitting alone or in three-member Licensing Boards. The judges are drawn from the Atomic Safety and Licensing Board Panel ("the panel"), created by the Commission in 1962 under the authority of Section 191 of the Atomic Energy Act. 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. The Commission or the Chief Administrative Judge assigns individual judges to those particular hearings where their professional expertise will be useful in resolving the technical and legal matters at issue in the proceeding. During fiscal year 1992, the panel comprised 40 administrative judges (16 full-time and 24 part-time). By profession, it included 11 lawyers, 11 public health and environmental scientists, 15 engineers or physicists, and three medical doctors. (See Appendix 2 for a list of the names and disciplines of fiscal year 1992 panel members.)

Responsibilities of Licensing Boards

Licensing Boards consist of three administrative judges, usually one legal member and two technical members. Ordinarily, three-judge boards are used for proceedings involving commercial nuclear reactors and enforcement actions against licenses. 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.

Panel judges conduct both formal and informal proceedings. Formal proceedings follow the traditional procedures used in a non-jury Federal Court case, including pre-trial discovery between the parties and formal trial procedures at the hearing. In informal proceedings (for example, materials license proceedings under 10 C.F.R. Part 2, Subpart L), a hearing is conducted only as to those issues that the administrative judge cannot resolve, based on the parties' written submissions, or on additional information the administrative judge has deemed relevant. Informal proceedings rely heavily on the active involvement of the administrative judge in creating and shaping the record of the proceeding.

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 several topics and multiple issues, the panel sometimes creates separate Licensing Boards and assigns one or more discrete topics to each board. Such parallel adjudications save time and provide for a more precise match of panel members' expertise to the issues to be resolved.

During fiscal year 1992, the panel actively managed its caseload to improve efficiency. Hearings lasted less time than in previous years; the vast majority of proposed contentions were resolved prior to hearing, and a significant number of docketed cases were settled prior to final adjudication.

The panel also continued its leadership role in automating the hearing process. In recent years, the panel had been moving rapidly to achieve an "electronic" office, particularly in managing its voluminous and complex hearing records. To maximize this process in 1992, the panel continued to use INQUIRE, an electronic docket conceived, developed and maintained by the panel. Among other things, panel decisions were entered into INQUIRE the day of issuance and were thus immediately available throughout the agency. In parallel with INQUIRE, the panel also continued research into replacing INQUIRE with a personal computer system with full text data bases for decision writing. Other electronic improvements during the year included the upgrading of personal computer support equipment and completing word processing standardization.

Panel Caseload

During fiscal year 1992, the panel's caseload comprised a total of 38 proceedings. Sixteen involved nuclear power plants or related facilities and 22 involved other Commission licensees. Twenty-two cases were closed and 18 new cases were docketed.

In contrast with the licensing proceedings for power reactors that had dominated the panel's docket over the two previous decades, the fiscal year 1992 caseload primarily involved enforcement actions against licensees, contested license amendment proceedings, and nuclear materials proceedings. The panel expects an infusion in the near future of contested proceedings involving decommissioning, license renewal, reactor licensing, and design certification of new reactors.

Some of the panel's more significant decisions issued during fiscal year 1992 are discussed below.

Antitrust Decisions

A license amendment application to delete the antitrust license conditions for the Perry and Davis-Besse (Ohio) nuclear facilities produced several important rulings. One involved the frequently litigated issue of NRC jurisdiction over antitrust matters. The City of Cleveland, Ohio, opposing the license amendment application, contended that the NRC does not have jurisdiction to delete antitrust license conditions once they are made part of a license. The Licensing Board held that, although NRC antitrust jurisdiction generally ends after an operating license is issued, the NRC has continuing authority over the life of a license to amend antitrust license conditions, if justified, when requested to do so by a licensee. The board reasoned that the narrow authority accorded the Commission under AEA section 105c does not supersede the Commission's more general authority under AEA section 189a to amend a facility license (*Cleveland Electric Illuminating Company, et al.*). (*Perry Davis-Besse Nuclear Facility*), LBP-91-38, 34 NRC 229 (1991), affirmed CLI-92-11 (August 12, 1992).)

In another Perry Davis-Besse ruling, the Licensing Board acted on an untimely petition by the Department of Justice to appear as a party in a license amendment proceeding. The board held that, although the Department has an absolute right to participate in antitrust operating license and construction permit proceedings, it must meet the same standards for intervention as any other party in other NRC proceedings. The board found, however, that in this case the Department met the standards for late filed intervention.

Civil Penalties

Several important decisions were issued during the report period dealing with escalated enforcement actions against nuclear licensees. One was a $6,750 civil penalty against an Oklahoma radiography company. The penalty consisted of nine different violations aggregated into a Severity Level III violation and then escalated by 75 percent. The escalation was based on the company's alleged prior notice of similar events and prior poor regulatory performance. The Licensing Board found this escalation to be excessive, since the past prior notice and prior poor performance cited by NRC staff were related to the non-serious violations (paper work discrepancies and management inattention to detail), while the more-serious violations were first-of-a-kind. Because the staff failed to differentiate between the seriousness of the different violations and based the penalty's escalation on the wrong violation, the board reduced the penalty from 75 percent to 20 percent. (Tulsa Gamma Ray, LBP-91-40, 34 NRC 297 (1991).)

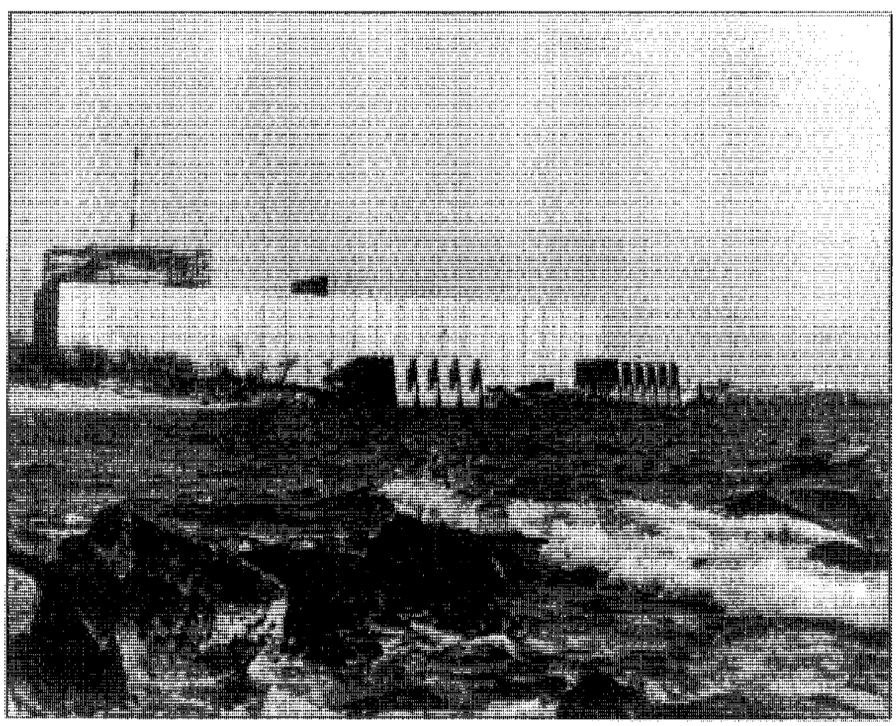
Another case involved the revocation of a radiography company's license because of the dishonesty of its president. This official had lied to NRC inspectors and participated in document falsification. The Licensing Board found the extreme sanction of license revocation to be appropriate because of the need for the NRC to rely on the integrity of licensees. The board also concluded that the NRC is authorized to consider a manager's character and integrity in deciding whether to revoke a license, and that candor and honesty are especially important elements of character. (*Piping Specialists, Inc.*, LBP-92-25 (September 8, 1992).)

Due Process in NRC Enforcement Actions

There were several important decisions during 1992 involving the issue of due process in NRC enforcement actions. One decision concerned the failure of a small radiography company to comply with the prescribed format for NRC adjudicatory proceedings by failing to furnish transcript citations in its proposed findings. The company had not used transcripts because of their high cost. The Licensing Board allowed the findings to be filed, because, as a practical matter, there was no NRC public document room available for the company to obtain free transcripts. The board reasoned that there is a difference between those proceedings where intervenors elect, on their own, to participate, and proceedings where staff acts against a licensee's property interests. According to the board, licensees must be allowed to participate in proceedings to defend vested property interests, and, if they show that they cannot comply with all of the technical requirements, then boards should use their best efforts to understand and rule on the merits of the claims presented. (*Tulsa Gamma Ray, Inc.*, LBP-91-40, 34 NRC 297 (1991).)

Another case involved the revocation of a Missouri radiography company's license. An issue was the burden of proof staff must meet to show that license revocation was justified. Historically, the burden of proof in NRC proceedings has been the "preponderance of the evidence" test, although a "clear and convincing" test has been used on at least one occasion. The radiography company claimed that staff's burden here should be a "clear and convincing" test because an extreme sanction (license revocation) was being imposed. The Licensing Board held that the "preponderance of the evidence" test was appropriate because the public interest weighs against changing the standard of evidence to protect a licensee whose actions could have serious safety repercussions. (*Piping Specialists, Inc.*, LBP-92-25 (September 8, 1992).)

In a third proceeding involving due process, a Licensing Board accepted a settlement agreement between the staff and the New York Power Authority for the Fitzpatrick (N.Y.) nuclear facility. The settlement established a drug testing schedule for an employee who was not admitted as a party to the proceeding, but who was involved in a separate NRC proceeding, brought by the staff against him and involving the same matter. Although not a party, the employee challenged the settlement in this proceeding. The Licensing Board held that it had no authority to alter the provisions of the program agreed to between staff and the licensee. However, the board went on to rule that the employee could try to establish in his own proceeding that a more lenient testing schedule be imposed than that agreed to in the settlement. (*New York Power Authority* (Fitzpatrick (N.Y.) nuclear power plant), LBP-92-1, 35 NRC 11 (1992).)



A local organization successfully argued that it had standing to intervene in a hearing on a license amendment application by licensees for the Millstone (Conn.) nuclear power plant, shown here. The Licensing Board agreed that, even though the application was intended to reduce the capacity of the plant's spent fuel pool and thus did not involve an increase of risk to nearby residents, the petitioner had standing on the grounds that the proposed amendment might not entirely remedy an allegedly existing risk. The Millstone facility comprises three boiling water reactor units.

Intervention in NRC Proceedings

Several significant fiscal year 1992 decisions dealt with intervention in NRC proceedings. In a proceeding involving the Millstone facility, a local organization challenged a license amendment application to reduce the storage capacity of a nuclear reactor's spent fuel pool. The amendment would not have increased the risk to nearby residents because the pool's capacity was being reduced. The petitioning organization claimed the license amendment did not go far enough and that there would still be an adverse safety risk to the public. The licensee, on the other hand, contended that there was no standing to intervene, since standing requires injury in fact, and there was no such injury here because the license amendment was not increasing the risk. The Licensing Board found there is standing to intervene, even if a license amendment does not increase the risk, if the amendment does not entirely remedy the situation. (*Northeast Nuclear Energy Company* (Millstone Unit 2 (Conn.) nuclear power plant), LBP–92–17 (July 29, 1992); LBP–92–28 (September 30, 1992).)

Another standing case during the year involved discretionary standing to intervene. In *Envirocare of Utah, Inc.*, LBP–92–7, 35 NRC 167 (1992), a Licensing Board found that a potential customer of a Utah uranium and thorium disposal site lacked standing on its own right to intervene. The board next considered whether discretionary standing should be granted. Discretionary standing here would have been precedent setting, since such standing had only been granted in the past when there was an ongoing proceeding. Although the board decided that it was not barred from granting discretionary standing, it concluded that some discernible public interest in holding a hearing first must be established. The board found no such public interest in this instance.

A third intervention case pertained to the filing of petitions for intervention and requests for hearings. In a materials license amendment proceeding conducted under the informal rules of 10 C.F.R. Part 2, Subpart L, involving the decommissioning of the Apollo fuel fabrication facility in Pennsylvania, a local organization and several intervenors attempted to supplement their hearing requests. The presiding officer concluded that these petitioners did not have an automatic right to amend or supplement their petitions, as they would in a formal proceeding held under Subpart G of 10 C.F.R. Part 2. He found, however, that he had the discretion under Subpart L to permit supplementation, at least until such time as a final decision was made on the sufficiency of the hearing request. He thus allowed the petitions to be supplemented because it materially aided him in determining whether the petitioners had standing to contest the proposed action and whether they had presented litigable issues. (*Babcock & Wilcox* (Apollo Facility), LBP–92–24 (September 4, 1992).)

COMMISSION DECISIONS

Fiscal year 1992 marks the first year that the Commission operated under its revised appellate structure whereby the Commission exercises all authority for appellate review of decisions of the Atomic Safety and Licensing Board in agency adjudications. The Commission has established an Office of Commission Appellate Adjudication to assist the Commission in the exercise of its adjudicatory responsibilities. Some of the more significant Commission decisions in fiscal year 1992 are discussed below.

Comanche Peak Nuclear Power Plant

In November 1991, Sandra Long Dow and Richard E. Dow sought to reopen the operating license proceedings for the Comanche Peak (Tex.) nuclear power plant, Units 1 and 2. The proceeding had been initiated in 1979, at which time three parties were admitted as intervenors. Neither the Dows nor "Disposable Workers of Comanche Peak," the organization represented by the Dows, were among those parties. Two of the intervenors subsequently withdrew from the operating license proceeding. In July 1988, the Atomic Safety and Licensing Board issued an order dismissing the proceeding pursuant to a settlement agreement among the NRC staff, the licensee, Texas Utilities Electric Company, and Citizens Association for Sound Energy (CASE), the sole remaining intervenor. At the time the Dows filed their motion, Comanche Peak Unit 1 had been licensed to operate, since April 1990; Unit 2 was in the latter stages of construction and pre-operational readiness.

The Dows sought to reopen the record on the basis of allegations of improper payments to plant workers for not testifying before the Licensing Board, false evidence submitted by the licensee to the Licensing Board, and false testimony by the management of the licensee and its principal contractor in Department of Labor proceedings arising out of actions at Comanche Peak. The Dows also alleged that representatives of the licensee, the NRC staff, and CASE perjured themselves or deliberately failed to notify the Licensing Board of relevant information. The Dows relied on selections from various prior pleadings before the NRC or the Department of Labor in support of their motion.

In *Texas Utilities Electric Company* (Comanche Peak nuclear power plant, Units 1 and 2), CLI–92–1, 35 NRC 1 (1992), the Commission denied the Dows' motion to reopen the record, because they were not parties to the Comanche Peak proceeding itself. Although the Unit 1 license had been issued, the Commission noted that a proceeding arguably remained in existence for Unit 2, because no operating license had as yet been issued for that plant. The Dows' request for leave to file for intervention was denied because they did not address the five factors bearing on late intervention specified in 10 CFR 2.714(a)(i)-(v). Moreover, the Commission found that none of the petitioners' arguments satisfied the requirements for reopening the record. The Commission found that none of the evidence on which the petitioners relied was new information that related to a significant safety or environmental issue within the NRC's jurisdiction, or that the documents provided obvious support for the Dows' allegations. For the most part, the Dows relied on arguments made at the public hearing to consider the proposed settlement agreement, and in various motions to reopen the record. The Commission was unconvinced that the allegations were sufficient to void the settlement agreement or to reopen the proceedings.

The Dows again sought leave for late intervention and to reopen the Comanche Peak operating license proceeding and the construction permit amendment proceeding, in a series of motions filed after the Commission's decision in CLI-92-1. In CLI-92-12, 36 NRC _____ (Aug. 12, 1992), the Commission denied the motions with respect to the Unit 1 operating license and construction permit amendment proceedings, because no proceeding remained with respect to Unit 1 once the operating license for Unit 1 was issued. Any challenge to the Unit 1 license must take the form of an enforcement petition under 10 CFR 2.206.

With respect to the Dows' various motions as applicable to Unit 2, the Commission denied the Dows' request for oral argument on their motions, noting that oral argument is essentially discretionary with the Commission, and that the Dows had not shown that the public interest was best served by oral argument, as opposed to a decision based solely on the written public record. The petitioners did not establish that their motion met the standards for late intervention. For the most part, the information on which the Dows relied was already well-known and could not constitute good cause for the lateness of their motion. In the absence of good cause for their lateness, the Commission found that factors favoring the Dows—inability to protect their interest by other means, and lack of any other party to represent their interest—weighed little in the Commission's determination. Moreover, the petitioners had failed to demonstrate that they could contribute to a sound record. The possibility of delay and expansion of the hearings in the absence of any countervailing considerations weighed heavily against the petitioners. The Commission reiterated its earlier decision that the Dows could not seek reopening of the proceeding because they had not been and could not become parties to the Unit 2 operating license proceeding on the record before the Commission.

South Texas Project Investigation

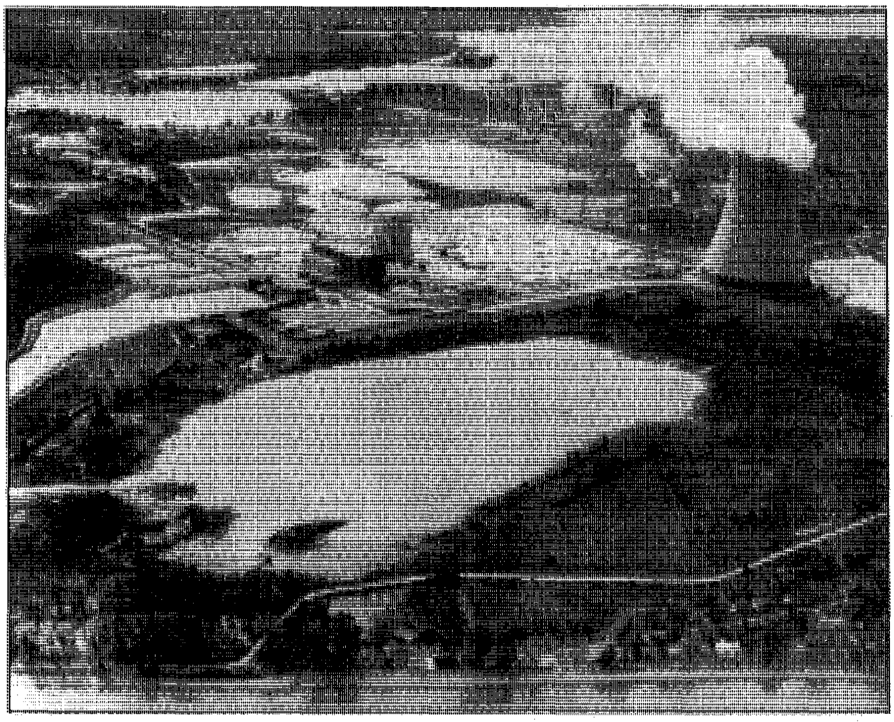
In March 1992, the Administrator of the NRC's Region IV asked the Office of Investigations (OI) to conduct an investigation to determine the facts surrounding the denial of access to Thomas J. Saparito, Jr., a contract instrument control technician, to the South Texas Project. Mr. Saparito contended that he was denied access because he had identified potential violations of regulatory requirements to the NRC. The licensee, Houston Lighting and Power Company, contended that Mr. Saparito's access was denied for his having provided false information on his employment application. The NRC investigator requested interviews, on a non-compulsory basis, with certain licensee personnel who agreed to such only if they would be provided transcripts of their testimony within two weeks after their interview, or on certain other conditions. OI rejected these demands and the impasse led to the issuance of subpoenas to the individuals. The individuals sought to quash the subpoenas.

In *Houston Lighting & Power Company* (South Texas Project, Units 1 and 2), CLI-92-10, 36 NRC _____ (July 2, 1992), the Commission denied the motion to quash. The Commission found provisions of the Administrative Procedure Act, 5 U.S.C. § 555(c), applicable to the question at issue. Under section 555(c), a person compelled to provide testimony is entitled on payment of costs to obtain a copy of his transcribed testimony, but that right may be limited—for good cause, in non-public investigatory proceedings—to inspection of the transcript. The good cause exception is within the agency's discretion, and it may be invoked in a case in which a later prosecution may be brought and it would be detrimental to the due execution of law to permit copies of the transcript to be circulated. Moreover, the agency is not required to make a good cause determination prior to receiving testimony.

Thus, under the circumstances of the South Texas investigation, the Commission found it premature for parties to argue that OI had violated section 555(c) by refusing to guarantee a copy of the transcript as a precondition to the interview. The Commission noted that, at an appropriate time, OI must provide a copy unless, for good cause, the witnesses are limited to inspection of the transcripts.

Perry/Davis-Besse Antitrust Proceeding

Ohio Edison Company, Cleveland Electric Illuminating Company, and Toledo Edison Company have applied for amendments to suspend the effect of the antitrust license conditions in the operating license for the Perry (Ohio) nuclear power plant and the Davis-Besse (Ohio)



Licensees for the Perry and Davis-Besse plants in Ohio (the latter facility shown here), applied for a suspension of antitrust license conditions on operating licenses for those plants, leading to a hearing for which the City of Cleveland sought intervenor status. The status was granted by the Licensing Board and the City raised a number of objections to the application, among them an assertion that the NRC was barred from conducting a post-licensing antitrust review. Details of the multi-phased proceedings are set forth in the text.

nuclear power plant. The NRC staff denied the applications in May 1991, and the applicants sought a hearing on the staff's denial. The City of Cleveland opposed the hearing requests but sought, alternatively, intervenor status in the proceeding. In a prehearing conference order issued on October 7, 1991, the Licensing Board granted the applicants' hearing requests, admitted Cleveland as a party, and also permitted participation in the proceeding by Alabama Electric Cooperative, Inc., American Municipal Power-Ohio, Inc., and the Department of Justice (Ohio Edison Company, LBP–91–38, 34 NRC 229 (1991)). The Licensing Board also admitted issues submitted by Ohio Edison Company, alleging improper Congressional interference and prejudgment, in regard to the staff's decisional process on the amendment applications. With the Licensing Board's encouragement, the parties agreed to formulate and brief a potentially dispositive "bedrock" legal issue or issues in the proceeding.

The Commission took action with respect to two aspects of the Licensing Board's prehearing conference order: the admission of the decisional bias issues and the granting of a hearing to the applicants. The Commission exercised its inherent supervisory authority over adjudications and suspended *sua sponte* consideration of all matters in the proceeding with the exception of the parties' agreed-upon bedrock legal issue and a related estoppel issue (*Ohio Edison Company,* CLI–91–15, 34 NRC 269 (1991)). The Commission took this action in view of the potential disposition of the case on the legal issues and in view of the unusual nature of the decisional bias issues raised by Ohio Edison against the staff.

Ohio Edison sought reconsideration of the Commission's order, arguing that the proceeding could not be fairly resolved, even as to the bedrock legal issue, without reaching the decisional bias issues. In CLI–92–6, the Commission, with Commissioner Curtiss dissenting, denied reconsideration of CLI–91–15. The Commission did not agree that resolution of the decisional bias issues was critical to the determination of the bedrock issue. The Commission noted that the NRC staff does not occupy a favored position in NRC proceedings and, with respect to legal issues like the bedrock issue, the staff's submissions have no more weight than any other party. Ohio Edison had not explained why either the Licensing Board or the Commission was incapable of rendering an independent decision on a question of law, even assuming some bias on the part of the staff.

The City of Cleveland appealed the grant of a hearing to the applicants, in LBP–91–38, to the Commission. Cleveland argued that the NRC was barred from conducting any post-licensing antitrust review and thus was without authority to grant the licensees' request for relief from the license conditions. Cleveland also argued that section 189 of the Atomic Energy Act, 42 U.S.C. § 2239, does not confer hearing rights on license applicants.

In *Ohio Edison Company,* CLI–92–11, 36 NRC _____ (Aug. 12, 1992), the Commission denied Cleveland's appeal of the threshold jurisdictional issues decided in LBP–91–38. The Commission found, as a general rule, that the Commission can amend licenses, and section 189 of the Atomic Energy Act provides an opportunity for the hearing and prescribes procedural requirements that attach to certain specified actions, including proceedings to amend licenses. In accord with a longstanding, unchallenged interpretation of the Act, license applicants and licensees are entitled to a hearing under section 189 upon request, if their interests are adversely affected (e.g., if a license or amendment application is denied or a license is suspended or revoked).

The more difficult question posed by Cleveland was whether the Commission's general authority to amend licenses could be invoked when a license condition involves antitrust matters, or whether post-licensing amendments to an antitrust conditions would be inconsistent with section 105c of the Atomic Energy Act, 42 U.S.C. § 2135(c). The Commission noted that the issue had not been addressed directly by the Congress in the Atomic Energy Act or in its legislative history and had not been squarely addressed in any prior Commission decision. The Commission concluded that the NRC does have jurisdiction under the Atomic Energy Act to consider licensees' requests to amend the antitrust conditions. As the agency empowered to issue nuclear plant licenses, only the Commission can grant relief from the conditions. Otherwise, the antitrust conditions would remain in place for the life of the license without regard to whether they had become unjust over time. Whether any relief is warranted in this case depends on the outcome of further litigation in the proceeding before the Licensing Board. The Commission also left open the question whether other parties could raise the need for additional antitrust conditions if a licensee initiated a proceeding to suspend or modify the existing antitrust conditions.

Rancho Seco Nuclear Power Plant

As part of its implementation of a public referendum directing it to cease operation of the Rancho Seco (Cal.) nuclear power plant, the Sacramento Municipal Utility District sought an amendment to convert its license for the plant into a "possession only" license, which would eliminate the operating authority. In response to a notice of the amendment, the Environmental Conservation Organization (ECO) petitioned for leave to intervene and for hearing in the proceeding. The Licensing Board ultimately denied ECO's petition for lack of standing and lack of a cognizable contention. LBP–91–30, 34 NRC 23 (1991).

ECO appealed the denial of its petition to the Commission, arguing that it had demonstrated standing based on its members' loss of employment at Rancho Seco, and because the NRC's failure to issue an environmental impact statement (EIS) on the amendment had deprived ECO of the opportunity to participate in the EIS process. ECO also argued that the Licensing Board erred in denying admission of its contentions and on other procedural matters.

On appeal, the Commission reached only the standing questions and determined that the Licensing Board had properly denied ECO's petition to intervene and for the hearing (*Sacramento Municipal Utility District* (Rancho Seco nuclear power plant), CLI–92–2, 35 NRC 47 (1992)). The Commission rejected standing based on economic loss to ECO's members, because that loss was occasioned not by the impact the amendment might have on the environment, but by the licensee's decision not to operate Rancho Seco. Thus, ECO could not satisfy the "zone of interests" aspect of the standing test to challenge the agency's action. With respect to its arguments regarding participation in the EIS process, the Commission noted that it had previously rejected the assertion of such "informational" interests as grounds for standing. Participation in proceedings is not an end in itself and, without some specific environmental impact on the petitioner's interests, an interest in disseminating information on environmental issues is insufficient to confer standing.

Seabrook Nuclear Power Plant

The Commission concluded its consideration of adjudicatory matters pertaining to the operation of the Seabrook (N.H.) nuclear power plant in *Public Service Company of New Hampshire* (Seabrook nuclear power plant, Units 1 and 2), CLI–92–8, 35 NRC 145 (1992). The Commission affirmed the Licensing Board's determination in LBP–91–24, 33 NRC 446 (1991), to grant summary disposition in favor of the license applicant on the remaining emergency planning issue requiring resolution. The intervenors, the Massachusetts Attorney General and the New England Coalition on Nuclear Pollution, had questioned whether the New Hampshire Radiological Emergency Response Plan had made sufficient provision for the use of sheltering as a protective action option in an emergency, particularly for persons who frequent the ocean beaches within a two mile radius of the plant.

The Commission upheld the Licensing Board's pivotal finding that the adjudicatory record now showed that New Hampshire emergency planning officials had concluded for all foreseeable circumstances that evacuation, not sheltering, is the planned protective action for the general beach population, in the event of a "general emergency," the highest emergency action level classification. Intervenors failed to demonstrate that a genuine issue of material fact remained with respect to this determination and, thus, the Licensing Board had appropriately granted summary judgment in favor of the license applicant. The Commission also agreed with the board that its determination effectively made moot an earlier direction of the Atomic Safety and Licensing Appeal Board to consider whether State planners had provided sufficient implementing measures for sheltering the beach population.

In a separate decision involving the Seabrook plant, the Commission dismissed the appeal of the Seacoast Anti-Pollution League (SAPL) from the Licensing Board's denial of intervention in an amendment proceeding involving the transfer of ownership of the Seabrook plant from the Public Service Company of New Hampshire (PSNH) to the North Atlantic Energy Corporation, a wholly owned subsidiary of Northeast Utilities. CLI-91-14, 34 NRC 261 (1991). The transfer of ownership amendment was part of a reorganization plan ordered by the bankruptcy court to resolve PSNH's bankruptcy proceedings. SAPL opposed the amendment and averred that a transfer of ownership would create a "material increase in the hazard of operation" to its members, on the basis of alleged harassment and intimidation by Northeast Utilities' management, as evidenced by pending NRC investigations at other Northeast Utilities' plants. The Licensing Board had denied standing to SAPL because the amendment involved only ownership, not management, of the plant and that the mere pendency of an investigation was not sufficient to show particularized harm necessary to confer standing.

Because SAPL had not filed a brief in support of its appeal in a timely manner, the Commission dismissed the appeal. The Commission, however, reviewed the denial of standing and affirmed the result reached by the Licensing Board, although on somewhat different grounds. The Commission determined that SAPL had not satisfied the "injury in fact" aspect of the standing test, because SAPL had not shown—even accepting its claim of injury—that a favorable decision in the proceeding on the ownership amendment would allay the alleged harm to SAPL. Although SAPL challenged the ownership transfer amendment, it had failed to challenge a separately noticed amendment to transfer management and operational responsibility to another Northeast Utilities subsidiary. Even if SAPL were granted relief with respect to the ownership transfer amendment, the purported harm would still occur from an amendment that SAPL left unchallenged. Consequently, the Commission was satisfied that SAPL had not established standing to intervene.

Shoreham Nuclear Power Plant

The Commission also faced a challenge to an ownership transfer amendment in *Long Island Lighting Company* (Shoreham Unit 1 (N.Y.) nuclear power plant), CLI-92-4, 35 NRC 69 (1992).) The NRC staff proposed to issue an immediately effective license amendment authorizing the transfer of ownership of the Shoreham plant from Long Island Lighting Company (LILCO) to the Long Island Power Authority (LIPA). The transfer involved an unprecedented situation in which one utility sought to transfer its license, which had been converted to "possession only " status, for a virtually unused reactor to another entity that intended to decommission and dismantle the facility. This action was opposed by the Shoreham-Wading School District and the Scientists and Engineers for Secure Energy; both petitioners asked for a stay of the amendment, arguing that the Commission could not issue an immediately effective amendment.

Upon consideration of the petitioners' arguments, the Commission determined that, though an opportunity for hearing on a transfer of ownership might be required under section 189 of the Atomic Energy Act, 42 U.S.C. § 2239, the statute did not require a prior hearing before the effectiveness of the amendment. Except in certain circumstances, found not controlling in the instant case, the Commission has not construed section 189 to require a "pre-effectiveness" hearing. The Commission also determined that the petitioners had not raised any matter indicating that a prior hearing was appropriate as a matter of discretion. The Commission, therefore, denied the petitioners' request for a stay of the transfer from LILCO to LIPA, without prejudice to petitioners' rights to a post-effectiveness hearing before the Atomic Safety and Licensing Board.

Vogtle Nuclear Power Plant

Georgians Against Nuclear Energy (GANE) appealed its dismissal from a license amendment proceeding involving changes to the technical specifications for the Vogtle (Ga.) nuclear power plant, in order to permit the licensee to bypass a certain protective shutdown of the emergency diesel generators. The licensee, Georgia Power Company, had proposed the amendment as a safety enhancement to improve the reliability of the diesel generators, in light of difficulties that the licensee had experienced in establishing sustained operation of one of the diesel generators during a serious loss-of-power incident in March 1990. GANE challenged the amendment, sought to establish its standing to intervene, and filed proposed contentions. The licensee offered to undertake an informal exchange of information, to which all parties agreed, in an effort to resolve the issues informally. After this informal exchange, the Licensing Board dismissed GANE's petition to intervene, because GANE failed to set forth any adequate contentions.

GANE appealed the denial of its contentions to the Commission, but failed to file a brief supporting its appeal or otherwise to demonstrate any error in the Licensing Board's decision. Hence, in *Georgia Power Company* (Vogtle Electric generating Plant, Units 1 and 2), CLI-92-3, 35 NRC 63 (1992), the Commission dismissed GANE's appeal for failure to file a brief, as required under NRC practice. The Commission acknowledged, however, that GANE appeared to be seeking relief from the Commission in its broader responsibility for safety. In this context, the Commission determined to seek additional information and explanation from the NRC staff on issues which appeared related, at least in part, to GANE's concerns with the operation of the diesel generators at the Vogtle plant.

JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1992 is summarized below.

Pending Cases

Allied Signal, Inc. v. NRC. (No. 92–1019 (D.C. Cir.).) Petitioner originally filed a lawsuit in the D.C. Circuit (*Allied-Signal, Inc. v. NRC*, No. 91–1407) on August 23, 1991. It challenged the Commission's final rule requiring 100 percent collection of annual fees and charges. (See 56 FR 31472 (July 8, 1991).) This suit challenges the NRC's denial of petitioner's request for an exemption. Petitioner operates a uranium hexafluoride conversion facility at Metropolis, Ill. It contends that its allocation of licensing fees is too high, in comparison to other similar facilities and other sectors of the nuclear industry. The D.C. Circuit has consolidated petitioner's latest suit with four other pending licensing fee cases. Oral argument was heard in November 1992, after the close of the report period.

American Public Power Association v. NRC. (No. 92–1061 (D.C. Cir.).) Twenty-five petitioners, principally municipal power companies, joined in this lawsuit attacking the Commission's license renewal rule. (See 56 FR 64943 (Dec. 13, 1991).) Petitioners argue that the rule is arbitrary and unlawful "because of the Commission's refusal to provide for a review of the antitrust implications of ... license renewal applications" and is contrary to the Atomic Energy Act. This suit is the sole judicial challenge to the Commission's license renewal rule.

The case was being briefed at the close of the report period, and it will be argued orally in March 1993.

Environmental and Resources Conservation Organization v. NRC. (No. 92–70202 (9th Cir.).) This lawsuit challenges the NRC's issuance of a "possession only" license ("POL") to the owner of the Rancho Seco nuclear power plant in California, the Sacramento Municipal Utility District ("SMUD"). SMUD ceased operating the plant in 1989, in compliance with the results of a voter referendum. Petitioner is a group opposing Rancho Seco's shutdown. Petitioner argues that the POL is unlawful on a number of grounds, including an alleged NRC violation of the National Environmental Policy Act. Petitioner sought an emergency stay of the POL pending appeal and requested expedited review. The POL was scheduled to take effect on April 28, 1992. On April 22, 1992, a motions panel of the Ninth Circuit (Farris and Trott) denied petitioner's emergency motion for a stay and request for expedited review. The case has been fully briefed on the merits; oral argument will take place after the close of the report period.

Native Americans for a Clean Environment v. NRC. (No. 92 1167 (D.C. Circuit).) On April 16, 1992, the NRC Staff issued a letter rescinding an October order prohibiting operation of the Sequoyah Fuels Corporation ("SFC") fuel conversion facility in Gore, Okla. That same day, petitioners filed this lawsuit challenging the restart decision. They also filed an emergency motion for a stay, demanding immediate relief. Petitioners claim, as they had in a similar suit they had filed in District Court (dismissed for lack of jurisdiction), that the NRC decision violated the National Environmental Policy Act, because the NRC had not prepared an environmental assessment of the restart.

On April 22, 1992, a motions panel of the Circuit Court for the District of Columbia (Wald, D.H. Ginsburg and Sentelle) denied petitioners' emergency motion for a stay preventing the restart of SFC's facility. The court stated that petitioners had not "demonstrated satisfaction of the stringent standards required for a stay pending court review, particularly in light of the Nuclear Regulatory Commission's representation that it is currently conducting a 'fresh' National Environmental Policy Act study of the Sequoyah facility in connection with Sequoyah's pending license renewal application."

The case has been fully briefed and was orally argued on November 19, 1992, after the close of the report period.

State of Michigan v. United States. (No. 5:90–CV–27.) This is one of three lawsuits brought principally to challenge the constitutional validity of the Low Level Radioactive Waste Policy Act Amendments of 1985. As in the other two cases (brought in Nebraska and New York), the Federal District Court upheld the amendments in full. The Supreme Court, however, resolved the constitutional questions definitively in *State of New York v. United States* (upholding and invalidating the statute in part).

Plaintiffs now argue that the NRC violated its duty under the National Environmental Policy Act (NEPA) to update a 1982 environmental impact statement it had prepared in connection with Part 61 (Licensing Requirements for Land Disposal Facilities of Radioactive Waste). The District Court, however, agreed with the NRC's argument that it, the court, lacked jurisdiction to consider plaintiff's NEPA claim, which amounted to an attack on Part 61 itself, because a challenge to NRC regulations lies only in the Court of Appeals, under the Hobbs Act. The court indicated that plaintiffs must first petition the NRC to alter its regulations, and then it could seek Court of Appeals review of the NRC's action on the petition.

Plaintiffs have appealed the case to the United States Court of Appeals for the Sixth Circuit. The case has been fully briefed and is awaiting oral argument.

Significant Judicial Decisions

American College of Nuclear Physicians v. NRC. (No. 91-1431 (D.C. Cir.).) Petitioners in this case were two organizations of physicians who opposed the NRC's rule for the practice of nuclear medicine, entitled "Quality Management Programs and Misadministration" (the "QM Rule"; see 56 FR 34104 (1991)). To prevent mistakes in identity or dose, the QM Rule mandates various verification procedures in the administration of radiopharmaceuticals. Petitioners argued that the rule was unnecessary because nuclear physicians already followed sound QM practices, which had reduced misadministrations to an "irreducible minimum."

A panel of the District of Columbia Circuit (Edwards, Buckley and Sentelle) heard oral argument on May 12, 1992. Just 10 days later, on May 22, the court issued an unpublished two-page judgment order summarily denying the petition for review, "substantially for the reasons stated by the NRC in its rulemaking." The court held that the NRC had "acted within the bounds of its broad statutory mandate to establish 'such standards ... as the Commission may deem necessary or desirable to ... protect health or to *minimize* danger to life or property.' 42 U.S.C. 2201(b)" (emphasis that of the court).

Critical Mass Energy Project v. NRC. (No. 90-5120 (D.C. Cir., August 21, 1992).) A 7-to-4 majority of the D.C. Circuit, sitting en banc, upheld the NRC's decision under the Freedom of Information Act ("FOIA") not to disclose nuclear industry safety reports that INPO shares voluntarily with the NRC on the condition that the NRC not release them. The majority (Buckley, Silberman, Williams, D.H. Ginsburg, Sentelle, Henderson and Randolph) concluded that "where, as here, the information sought is given to the government voluntarily, it will be treated as confidential under [FOIA] Exemption 4 if it is of the kind that the provider would not customarily make available to the public."

In reaching this conclusion, the en banc court limited the reach of a longstanding D.C. Circuit precedent, *National Parks and Conservation 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, Mikva, Wald and 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."

Local 1245 v. NRC. (966 F.2d 521 (9th Cir. 1992).) The Ninth Circuit (Nelson, J., with Fernandez and Fletcher concurring in a separate opinion) affirmed the NRC's denial of a union's request for an exemption from the agency's fitness for duty (random drug-testing) regulations. The union had claimed that clerical, maintenance and warehouse workers at the Diablo Canyon nuclear plant in California posed an insignificant safety risk and ought not be subject to random drug testing under the Constitution's Fourth Amendment. The NRC argued at the threshold that a party cannot test the constitutionality of regulations in an exemption proceeding but must challenge the regulations themselves in a direct review action, or else seek a new rulemaking. The agency also maintained that, in any event, it was constitutionally permissible to require drug testing of all workers, regardless of their function or job title, with unescorted access to a nuclear plant's "protected area."

The Court of Appeals' two separate opinions considered the merits of the constitutional argument, although it is not clear why (or *if*) the court rejected the NRC's threshold argument that an exemption proceeding was not an appropriate vehicle to litigate a constitutional attack on agency regulations. On the merits, all the judges agreed that the union had not made a sufficient factual showing that its members' jobs were so risk-free that their interest in personal privacy outweighed the government's interest in nuclear safety. The court, therefore, affirmed the NRC's denial of the union's exemption request.

The two concurring judges (Fernandez and Fletcher) did express concern that the NRC regulations may "sweep too broadly" with regard to the clerical workers and indicated they might be willing to strike down the regulations if the union came forward with a more discriminate factual showing on clerical workers' actual duties. The remaining judge expressed a similar view.

Nuclear Information Resource Service v. NRC. (969 F.2d 1169 (D.C. Cir. 1992).) A 6-to-4 majority of the en banc D.C. Circuit upheld the NRC's Part 52 in its entirety. Part 52 substantially revamped the traditional nuclear power plant licensing process, with a view to resolving basic safety questions earlier in the licensing process and to encouraging plant standardization. The court considered and upheld each feature of Part 52: certification of designs by rulemaking, early site approvals, and combined licenses.

The court majority (Sentelle, Silberman, Williams, D.H. Ginsburg, Randolph and Henderson) paid special attention to the problem of late-arising new information that may call into question the safety of a plant after construction but prior to operation. The court found the NRC's petitioning process, set out in 10 C.F.R. 52.103(b)(2)(ii), adequate to deal with this situation. As had been suggested by the NRC, the court analogized the NRC scheme to situations where agencies are asked to rehear or reopen their prior decisions, and do so without reconvening a closed hearing process or revisiting an established rule. The court also held, as the NRC had argued, that the courts could review pre-operational NRC decisions rejecting "new information" petitions.

The dissenters (Wald, Mikva and Edwards, with a separate dissent by Buckley, J.) took issue with the majority on the "new information" issue. In their view material new information that could not have been introduced into the hearing process earlier requires a fresh opportunity for a hearing at the pre-operational stage. This was the same view that Judge Wald took in her opinion for the three-judge panel that originally struck down Part 52 (in part). The *en banc* decision, while endorsing part of Judge Wald's original opinion (on, for example, the validity of combined licenses), supersedes her opinion on the pre-operational hearing question.

Both the majority opinions mentioned the then pending energy legislation (now enacted) that later largely codified Part 52. Judge Wald for the dissenters suggested that the pendency of the legislation was reason to postpone a judicial determination, while Judge Sentelle for the majority indicated that the judges are not "political prognosticators" and must "decide ...cases as they are put."

State of New York, et al. v. United States of America, et al. (112 S. Ct. 2408 (1992).) The State of New York and two counties in New York brought this action in Federal District Court in Syracuse to challenge the constitutionality of the Low Level Radioactive Waste Amendments Act of 1985. Plaintiffs took the position that the Act improperly infringed State sovereignty under the Tenth Amendment, the Eleventh Amendment, the Guaranty Clause (which guarantees the States a "republican form of government"), and "constitutionally protected principles of federalism." Their principal argument was that the Act violated constitutional federalism principles by forcing States either to make arrangements allowing safe disposal of privately generated low-level waste or to take title to it. The main focus of their challenge was the "take title" provision, requiring any State that has not made arrangements by the end of 1995 for disposal of the low-level radioactive waste generated within the State to take title to and possession of such waste, at the request of its generators.

The District Court dismissed the lawsuit on the ground that Supreme Court precedent largely precludes challenges to Federal laws for infringing State sovereignty. An appeal was taken to the Second Circuit. The Court of Appeals affirmed. The plaintiffs then petitioned the Supreme Court for *certiorari*, and the court decided to review the case. Oral argument was held on March 30, 1992, with Deputy Solicitor General Lawrence Wallace representing the government. On June 19, the court issued its decision.

The Supreme Court decided that the "take title" provision is unconstitutional because it *requires* the States to act in accordance with the directives of the Congress. The court held that this is inconsistent with the Federal system of government established by the Constitution. However, contrary to petitioners' urging, the court held that the "take title" provision was severable from the rest of the Act, which was found to be valid. Three Justices (White, Blackmun and Stevens) dissented from the court's decision that the "take title" provision was unconstitutional. In his dissent, Justice Stevens stated that the court's decision should apply only to non-compact States like New York. It is not clear from the court's opinion whether the majority would agree.

Management And Administrative Services

Chapter 10

This chapter deals with internal events and activities of the NRC, such as changes on the Commission itself and in agency organization, consolidation of NRC offices in a single locale, noteworthy aspects and initiatives in personnel management, information resources, license fees levied and collected, activities of the Office of the Inspector General, contracts awarded by the Office of Small and Disadvantaged Business Utilization and Civil Rights, and events carried out under the Federal Women's Program.

Changes Within the Commission

There were two events during fiscal year 1992 affecting Commission membership.

On December 16, 1991, Dr. E. Gail de Planque was sworn in as a member of the Commission, filling the vacancy created with the expiration of Commissioner Thomas Roberts' second term. (The appointment of Dr. de Planque was reported in the *1991 NRC Annual Report*, pp. 1, 207–208.)

Commissioner Kenneth C. Rogers was reappointed by the President—and confirmed by the U.S. Senate, on May 21, 1992—to a second five-year term on the Nuclear Regulatory Commission, beginning July 1. Dr. Rogers, a physicist, had served for 15 years as President of the Stevens Institute of Technology, in Hoboken, N.J., before his initial appointment to the Commission in 1987.

New NRC Component—the Office of Policy Planning

In fiscal year 1992, the NRC Office of Policy Planning (OPP) was created for the purpose of evaluating relevant long-range policy issues from a broad perspective, including consideration of the viewpoints of industry and of public interest groups. OPP serves as the principal advisor to the Commission and to the Executive Director for Operations for policy planning; the Director of OPP is Chairman of the NRC's Steering Committee for Strategic Planning.

Richard H. Vollmer was appointed Director of OPP in May 1992, and office operations commenced in July. Mr. Vollmer has held a number of positions within the NRC, since beginning his service (with the former Atomic Energy Commission) in 1968. In past service to the agency, he has served as Director of the Division of Engineering in the Office of Nuclear Reactor Regulation and later as Deputy Director of the former Office of Inspection and Enforcement. From 1987 to 1992, Mr. Vollmer was Senior Vice President of Tenera, an engineering and management consulting firm.

Since formation of OPP, evaluations have been carried out of policy issues related to the reactor inspection program—particularly in its impact on operational safety—and of the current licensing bases for nuclear power plants. Reports deriving from these analyses were issued after the close of the report period and will be discussed in next year's annual report.

Consolidation of NRC Headquarters

At the close of fiscal year 1991, the first stages of site clearing and excavation for Two White Flint North (TWFN) had begun. By the end of fiscal year 1992, the base-building construction of the 10-story, 364,000 square foot building was nearing completion. Installation of the exterior concrete pre-cast panels and windows had commenced.

Occupancy for more than 1,300 NRC staff in the new building is scheduled for early calendar year 1994. During fiscal year 1992, preliminary space and furniture plans were developed for the twelve offices that will occupy TWFN. Design layouts were generated for the state-of-the-art Emergency Operations Center, central computer facility, multi-purpose auditorium, day care facility, physical fitness center, and an expanded staff training facility.

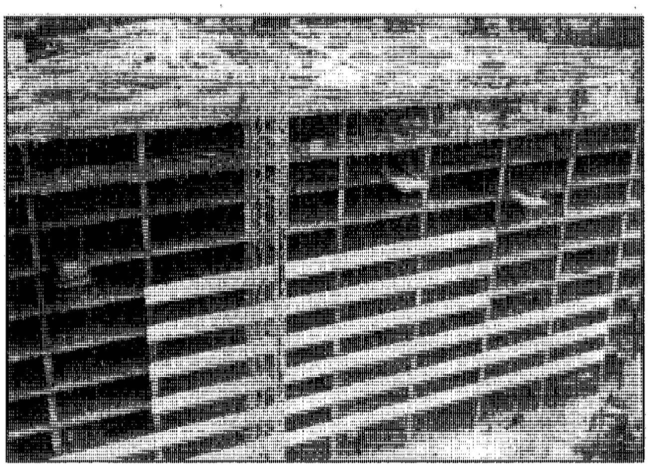
PERSONNEL MANAGEMENT

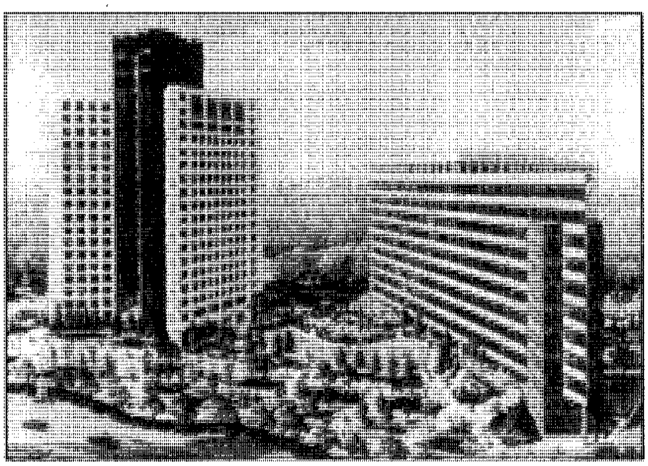
1992 NRC Workload

During fiscal year 1992, the NRC expended a total of 3,396 staff-years in carrying out its mission. Total staff-years included permanent full-time staff, part-time and temporary workers, and consultants.

Recruitment

During the report period, the NRC hired 164 employees and lost 147 permanent full-time employees, the lat





At the start of fiscal year 1992, site excavation for Two White Flint North has just begun, as shown at top. By the end of the fiscal year, base-building construction of the new building was nearing completion. Installation of the exterior, concrete pre-cast panels on one side of the building is shown at center. Above is an artist's rendition of NRC Headquarters, with One White Flint North on the left (fully occupied since 1988) and TWFN on the right. Occupancy of TWFN is scheduled to get under way early in calendar 1994.

ter figure representing an attrition rate of 4.46 percent. During the period, the agency participated in 61 recruitment trips. The recruitment effort generated approximately 1,723 applications for employment. Recruitment during the year was carried out by means of three key mechanisms: advertisements, recruitment trips, and an applicant inventory/tracking system.

Awards and Recognition

In fiscal year 1992, the NRC continued to give full recognition to and commendation of excellent performance on the part of agency staff. At the Annual Awards Ceremony in April, the NRC presented six NRC Distinguished Service Awards, 39 Meritorious Service Awards, and one Equal Employment Opportunity Award. During fiscal year 1992, NRC employees also received 885 Special Achievement Awards, 441 High Quality Performance Salary Increases, six Suggestion Awards, 21 Commendation Awards, and 295 Certificates of Appreciation. Five NRC Executives received Presidential Distinguished Executive Rank Awards, 15 received Presidential Meritorious Executive Rank Awards, 93 received Senior Executive Service (SES) bonuses, and 14 received SES Pay Level Increases. Besides these NRC citations, 11 employees were nominated for awards by outside organizations, and one of the employees received an award.

Labor Relations

The NRC and the National Treasury Employees Union (NTEU) completed negotiation of a new Collective Bargaining Agreement, which became effective on September 30, 1992. Supervisory training in provisions and implications of the new agreement has been completed and copies of the agreement have been distributed to all employees. The Agreement will be in effect for three years, with limited response provisions after 18 months.

Training and Development

During the fiscal year, the Office of Personnel provided more than 80 different on-site courses in the areas of probabilistic risk assessment; end-user computer applications; and executive, management, supervisory and administrative skills. NRC also sponsored a wide variety of training and developmental programs conducted at colleges and universities, at other Government agencies, and in the private sector, in order to improve performance and to assure up-to-date technical proficiency.

The Probabilistic Risk Assessment (PRA) Technology Transfer Program continued to undergo redesign and restructuring. During the fiscal year, 11 courses were offered. Courses substantially revised, or added to the curriculum, include *PRA Basics for Licensing Project Managers, Human Reliability Assessment, and System Analysis and Risk Assessment System Basics.*

Training in NRC's computer capabilities continued to expand. In addition to last year's course offerings, new courses were provided on how to use NRC's new network capabilities and to improve document and graphic programs. Many of these new courses are shorter, modular courses, designed to help employees select the kind of training they need to meet their specific needs and schedules.

The NRC Supervisory and Managerial Development Program offers a wide range of courses for new managers and supervisors. Included in this program are courses in personnel supervision, personnel management practices, and the performance appraisal process.

A training priority during the year was the design and development of several Equal Employment Opportunity (EEO) and cultural diversity courses. The *EEO for Managers and Supervisors* course was redesigned to include a one day-session on cultural diversity. Several sessions of a new half-day course, *Current EEO Issues for Managers and Supervisors*, were held; a half-day program, *EEO at NRC*, was presented; and a course entitled *Cultural Diversity at NRC* was designed and presented approximately every six weeks. Headquarters, regional, and Technical Training Center staff participated in these EEO and cultural diversity training and awareness courses.

During the report period, the NRC Individualized Learning Center continued to provide employees with convenient access to a wide variety of instruction, using the latest in audio/video, computer-based, and multi-media programming. An option for employees to borrow training programs for use in the office, at home, or while commuting has further augmented employees' opportunities to avail themselves of useful training. The Learning Center provides 160 programs in a broad spectrum of subjects including project management, communication, management and supervision, computer skills, secretarial skills, and employee assistance.

In addition to the various courses offered, the NRC sponsored a number of developmental activities and programs during the report period. To help employees clarify their career goals and to improve on-the-job performance, Individual Development Plan workshops were held throughout Headquarters and the Regions, and customized career consultations with a career counselor were made available. Developmental programs sponsored by the agency include: the Certified Professional Secretary 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 Congressional Fellowship Program, and the Graduate Fellowship Program.

Rotational Assignments

During fiscal year 1992, the NRC broadened its use of rotational assignments for the career development of employees and to help meet agency staffing needs. Managers and supervisors were actively involved in identifying candidates for rotational assignments during the period. A revised rotational program went into effect September 30, 1992. The changes in the program add clearer structure and control to the process.

Executive Leadership Development

Members of the Senior Executive Service (SES)—in ongoing efforts to amplify their knowledge of all aspects of the agency's operations and to share ideas on the vital technical and administrative issues facing the agency—attended the fifth annual NRC SES conference for all senior agency managers. A number of the managers also participated in rotational assignments, either within Headquarters, between Regions, or between Headquarters and the Regions, thereby broadening their experience both geographically and technically. During the report period, 22 executives attended the Federal Executive Institute and 19 attended Brookings Institution Education programs.

New Initiatives

The NRC has developed and instituted a Senior Level System which parallels the Senior Executive System and offers an alternative career development path for the agency's non-supervisory technical, legal, and administrative professionals. The NRC also implemented a completely redesigned SES performance appraisal system, whose purpose is to improve the communication of performance expectations and results between senior managers and agency executives.

Voluntary Leave Transfer Program

This program provides income protection to employees affected by a medical emergency, through the voluntary donation of annual leave by other employees. The provisional five-year program will expire on October 31, 1993. In fiscal year 1992, a total of 18 NRC employees received voluntary leave donations from fellow employees.

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

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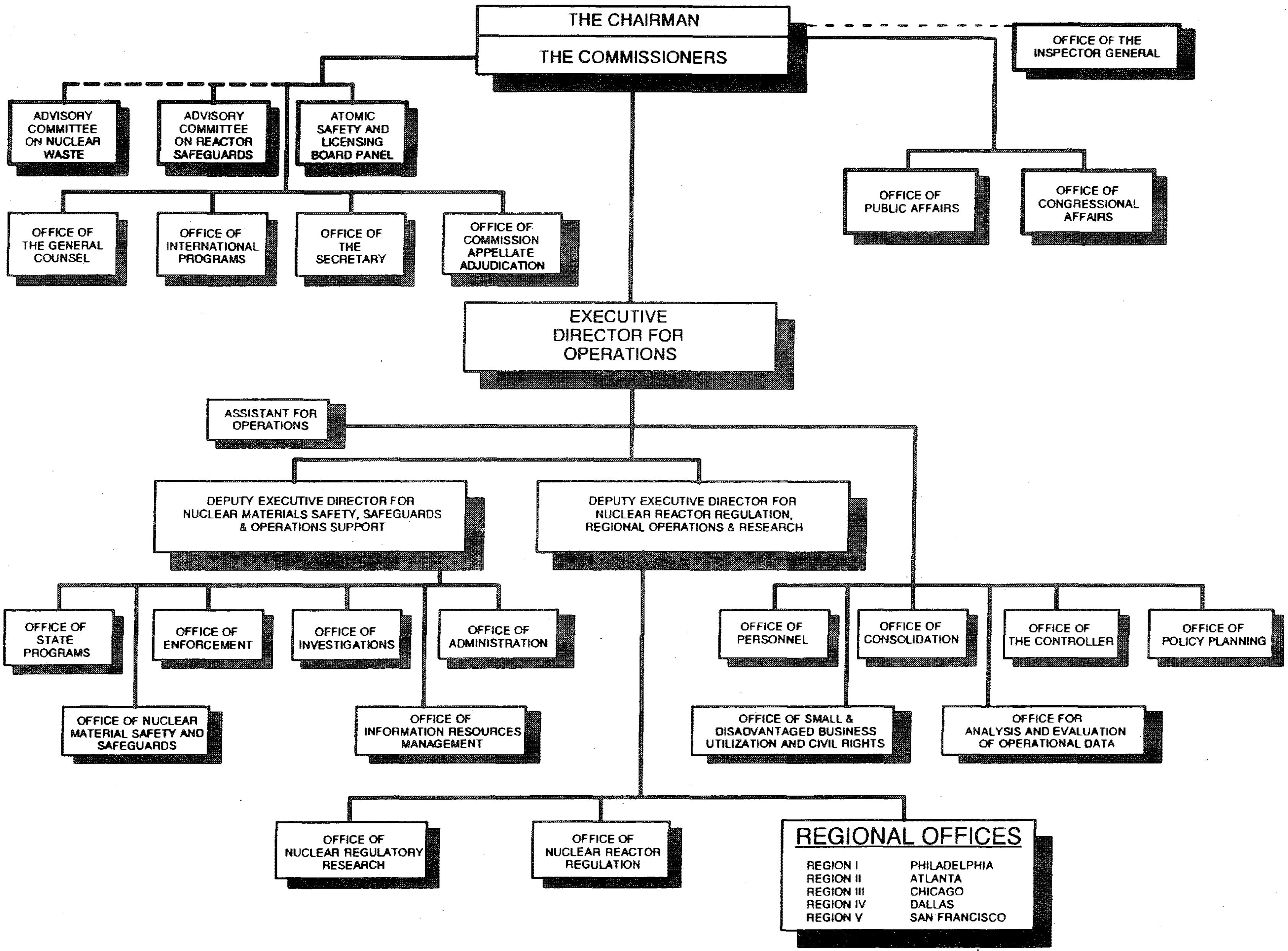


problems as chemical dependency, job stress, chronic illness, and family issues. The agency continued to make EAP services readily accessible to regional and field personnel through its interagency agreement with the Public Health Service. Supervisors were trained in recognizing and confronting troubled employees and referring them to the EAP. The agency conducted programs for agency employees on a variety of substance abuse and wellness topics. The EAP sponsored several smoking-cessation programs.

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; age-40-and-over physical examinations; screening for diabetes, glaucoma, high blood pressure, and cancer; immunizations; and health awareness programs on topics such as tuberculosis, bone marrow and organ transplant, and Lyme disease.

NRC INFORMATION RESOURCES

NRC Office Automation

Work continued during fiscal year 1992 to complete the second year of a three-year project to improve office automation at the NRC. By fiscal year's end, more than 1,900 microcomputers had been successfully connected to the Agency Upgrade of Technology for Office Systems (AUTOS) network. AUTOS, consisting of both local and wide-area networks, provides an important electronic link connecting virtually all NRC employees at Headquarters and in the Regions. Originally intended as a replacement for the outdated IBM 5520 and Displaywriter word-processing equipment, the AUTOS capability constitutes a telecommunications infrastructure to support many of the routine administrative functions carried out daily by the NRC offices. For example, during 1992, both the SES and the non-SES performance appraisal systems were automated and installed on the AUTOS network and used to prepare individual employee work plans for fiscal year 1993. This procedure resulted in significant time savings over the previous, manual method used to prepare the plans. Other form-based automation projects similar to the performance appraisal system are being considered as candidates for conversion to network-based systems. AUTOS also provides networking capability connecting high performance engineering workstations that enable technical staff to share computer codes, data, and other related resources. To date, AUTOS has been a real success and promises to increase individual productivity levels agency-wide when all employees are connected to AUTOS. That goal is expected to be achieved by the end of fiscal year 1993.

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.

A major planning objective has been the redesign of the central processing data entry system, to introduce more modern techniques for document capture and to provide a more readily maintainable computer design. The new system design provides for increased document processing, through the use of a knowledge-based approach to document encoding. The increased throughput gives the capability to meet the demand for high volumes of document processing, anticipated for such agency efforts as relicensing of nuclear plants and in the agency's license-fee management program. The new system design also incorporates specific improvements in the encoding and text edit processes, and provides a document tracking capability.

Software Quality Assurance Activities

Responsibility for providing support to agency computer users, in accord with all Federal and industry computer software standards, belongs to the Information Technology Services Branch of the Office of Information Resources Management. Two Software Quality Assurance Seminars were presented during the report period, and guidelines on ADP software quality assurance for in-house use will soon be published (NUREG/BR–0167). The NRC computer codes collection for distribution was transferred to the new Energy Science and Technology Software Center, at Oak Ridge, Tenn.

NRC Emergency Telecommunications System

Considerable progress has been made toward installing Federal Telecommunications System (FTS) 2000 services at each of the 119 nuclear power plants and emergency operations facilities. By the close of the report period, FTS 2000 telecommunications services had been installed and fully tested at all but six locations. The enhanced telecommunications capability afforded by the FTS 2000 service replaces the aging and obsolete dedicated, single-line, network system that was combined with the public switched network, to satisfy regulatory requirements for emergency communications set forth in 10 CFR 50.47(b)(6) and 10 CFR Part 50, Appendix E, IV.E.9d. The FTS 2000 service, which includes seven lines into each location, provides greater capability than the former emergency telecommunications system and improves the licensee's ability to provide prompt communications among principal emergency response organizations, to emergency personnel, and to the public, should the need arise. At year's end, activities were under way to ensure that FTS 2000 services at the remaining six locations would be installed and fully tested early in fiscal year 1993.

OFFICE OF THE INSPECTOR GENERAL

The NRC's OIG was established as a statutory entity on April 15, 1989, in accordance with the Inspector General Act of 1978, as amended in 1988. It is one of 26 such entities within the Executive Branch. The Inspector General (IG) is appointed by the President of the United States with the advice and consent of the Senate. To ensure the independence of the office, the IG may only be removed by the President, who must communicate the reasons for removal to the Congress. NRC's IG reports to and is under the general supervision of the NRC Chairman, but operates with independent budget authority.

One of the primary goals of the OIG is to assist the NRC in operating more effectively and efficiently by identifying ways to improve the NRC's programs and operations. To accomplish this objective, the OIG, among other things, carries out the following activities:

- Performs audits and makes recommendations concerning the effectiveness and efficiency of NRC programs and operations.
- Conducts, and reports on, investigations and inquiries, as necessary, to ascertain and verify the facts affecting the integrity of all NRC programs and operations.
- Recommends policies to (1) promote economy and efficiency and (2) prevent and detect fraud and abuse in the agency's activities.
- Keeps the Commission and the Congress fully informed of operational and program deficiencies and problems.

The NRC OIG was authorized 41 positions during fiscal year 1993, with a budget of about $4.6 million. During fiscal year 1992, the OIG (1) completed 14 audits of the NRC's operations and programs, (2) reviewed 99 contract audit reports, (3) performed one contract audit, and (4) closed out 90 investigations.

OIG Fiscal Year 1992 Audits

NRC Compliance with the Anti-Lobbying Act; Section 319 of Public Law 101–121. The Anti-Lobbying Act requires that the IG provide an annual report to Congress regarding the NRC's compliance with its provisions. Section 319 of the Act requires each person who requests or receives a Federal contract, a cooperative agreement, a loan, or a Federal commitment to insure or guarantee a loan, to disclose whether they used appropriated funds to lobby for the specific contract. The law applies to awards of contracts, grants, or cooperative agreements exceeding $100,000. Each person who requests consideration for an award or actually receives an award must file a certification or disclosure form regarding any lobbying effort.

The OIG reviewed 65 fiscal year 1992 contract actions that exceeded $100,000, to determine whether the required certifications and contract clauses were contained in each contract package. The OIG found that three of the 65 contract actions did not contain the certifications and contract clauses required by this Act. One possible cause for their omission was that the original estimate of the contract action was less than $100,000, which meant that the certifications and contract clauses required by the Act were not needed. When the negotiations were completed, however, the contract actions exceeded $100,000, and thus required the necessary certifications and clauses. The contract negotiators did not revisit the need for the certifications and clauses and did not put them in.

In response to OIG's 1991 report on contracts exceeding $100,000, the NRC modified the checklist used by its contract administrators to determine which documents or clauses may be required as a result of a contract modification, but it did not modify the contract negotiator's checklist. As a result of the recent OIG review, NRC has modified the negotiator's checklist to correspond to the administrator's checklist.

Survey of NRC's Inspection Programs. The OIG conducted a broad survey of NRC's inspection programs. The inspection programs are one of the primary methods by which NRC ensures that licensees are adhering to regulatory requirements and industry standards. This survey was conducted as part of OIG's strategic audit plan for fiscal year 1992. The results of the survey will be used to identify potential issues for future audits. The survey report describes in a single document how NRC administers and allocates resources among its many inspection programs. This report accomplishes two objectives.

First, it documents how NRC develops and manages its inspection programs. The report describes NRC's inspection programs for power and non-power reactors, vendors, and reactor construction. In addition, it outlines the Office of Nuclear Material Safety and Safeguards' licensee inspection programs for fuel facilities, materials, transportation, safeguards, low-level waste, and uranium recovery.

Second, the report gives detailed information about the allocation and use of inspection resources. This information goes beyond the data presented in either NRC's Five-Year Plan or its formal budget documents by isolating resources for inspection from resources for other activities, such as licensing. This body of work provides the foundation for a series of inspection audits that will be performed during the next several years.

OIG Review of Budget Changes for NRC'S Inspection Programs. The report of the Senate Committee on Appropriations that accompanied the NRC's fiscal year 1992 appropriation expressed concern about yearly increases for NRC's programs. The Committee questioned the need for "across-the-board increases in all mission areas" and directed the Commission to conduct an external review of these increases. The Commission subsequently requested that the OIG review and report on budget changes in NRC's inspection programs.

The OIG reviewed and compared the NRC's fiscal year 1992 estimated budget with the fiscal year 1993 requested budget for inspection programs. The review revealed that budgeted costs for inspection portions are projected to grow $2.3 million (2 percent) from fiscal year 1992 to fiscal year 1993. This increase, however, is for salary adjustments and not for programmatic growth. In fact, both staffing levels and program support (contract) costs are projected to decrease for reactor inspection programs. In NRC's materials and low-level waste inspection programs, staffing levels will decrease slightly, while program support costs will remain constant.

Review of NRC'S Contract Closeout Process. The closeout of a completed contract is basically the process of gathering all essential documents reflecting the completion and satisfaction of a wide list of obligations that are created during the course of the contract. Contract closeout is normally completed by the submission of a final invoice by the contractor and its acceptance and payment by the NRC.

The NRC's backlog of completed but not yet administratively closed contracts increased significantly since the issuance of an October 1987 report on this subject. Although NRC obtained a contractor in April 1990 to assist in reducing the backlog, only limited progress was made. NRC's inventory of completed-but-not-yet-closed contracts increased from 591 in May 1986 to 829 as of October 1991. Many of these contracts had been completed for several years and were valued over $100,000. As a result, the OIG estimated that over $8 million in contract obligations could be made available for use in NRC programs.

The OIG concluded that the backlog would continue to remain unacceptably high unless NRC takes aggressive managerial action to improve its oversight of the contract closeout process. The OIG recommended that NRC strengthen its managerial oversight of this process and made two other recommendations about the deobligation of unexpended balances on completed contracts. The NRC agreed with OIG recommendations to improve the contract closeout process.

Review of NRC'S Allegation Management System. In the early 1980s, the Commission recognized a need for a more systematic and structured process for handling allegations. This was especially important for allegations that were made to the NRC at the end of the licensing approval process. As part of this recognition, the Commission issued policy statements on "late-filed" allegations and matters of confidentiality concerning the individuals making them. The Commission also wanted to ensure that allegations that involved potential wrongdoing issues were effectively tracked and investigated.

In late 1982, responding to the Commission's concerns, NRC managers established the Allegation Tracking System (ATS) as a mechanism to track the large number of allegations that NRC was receiving. Previously, the NRC did not have a system to address and manage the flow of allegations it received. The system was subsequently renamed the Allegation Management System (AMS).

The OIG conducted a review of the allegation management process and concluded that (1) NRC's process for managing allegations met the intent of the Commission's policy; (2) the agency's expectations were clearly documented in NRC guidance, and (3) both headquarters and regional staff had implemented the guidance.

Budget Changes in the Office of Nuclear Regulatory Research. As noted previously, the report of the Senate Committee on Appropriations that accompanied NRC's fiscal year 1992 appropriation expressed concern about NRC's budget increases, including requests for increased research funding. The report directed the agency to conduct an external review of this growth. The Commission subsequently asked the OIG to review and report on increases in NRC's research budget.

The Office of Nuclear Regulatory Research (RES) is one of three NRC offices established by the Energy Reorganization Act of 1974, as amended. The Act directs RES to identify research needs and contract with various organizations to conduct research in support of NRC's licensing and related regulatory functions. Most of RES's $119 million budget for fiscal year 1992 will be used to support about 670 research projects.

The OIG review disclosed that RES's portion of NRC's budget has not grown significantly; in fact, as NRC's budget increased in recent years, RES's portion remained fairly constant, resulting in a decreased share of the agency's overall budget. While the budget for some of RES's various program elements have increased, its 3 percent budget growth rate has been about half that of NRC's overall budget growth rate. The OIG review, however, also noted shifts in RES's budget that could lead to future increases. For example, NRC and the nuclear industry are developing a scale model testing program to support advanced reactor design certification. The testing program could increase NRC costs by more than $13 million through fiscal year 1995.

Improvement Needed in NRC'S Process for Approving Payments to DOE. In December 1991, the OIG initiated a review of NRC's contract management practices related to the services provided by the Department of Energy (DOE). The audit revealed that since 1986, the NRC has paid about $500 million to DOE without employing the required review and approval of DOE cost vouchers and subsequently verifying the accuracy of the payments.

This failure to follow agency policies and procedures is a serious breakdown in internal controls that leaves NRC vulnerable to fraud, waste, and abuse. Until corrected, the OIG believes this condition constitutes a material weakness in the agency's internal controls over disbursements to DOE.

Subsequent to the OIG exit briefing on the results of the review, NRC management (1) directed that corrective actions be made and (2) notified the Office of Management and Budget that the agency will identify the management of agreements with DOE as a material weakness.

OIG Fiscal Year 1992 Investigations

Inspection of NRC Staff's Review and Acceptance of Fire Barrier Material. The NRC received an allegation that questioned the capabilities of a fire barrier material commonly used by the nuclear industry. The material, Thermo-Lag, is installed in a majority of the nation's power plants to satisfy NRC fire protection requirements.

Because of the widespread use of this material since 1981, the OIG conducted an inspection to assess the adequacy of the NRC staff's review and acceptance of Thermo-Lag. In addition, the inspection assessed the staff's response to problems with Thermo-Lag reported to the NRC over approximately a 10-year period.

The OIG inspectors determined that the NRC staff had not conducted an adequate review of the fire protection capabilities of Thermo-Lag and other related information. If the staff had conducted a thorough review of this information and verified test reports submitted by the nuclear industry, a number of problems with the test program and Thermo-Lag would have been discovered. Because the NRC had not conducted the proper reviews and inspections, the staff did not determine until 1991 that Thermo-Lag might not meet NRC requirements. In addition, the NRC did not take substantive action between 1981 and 1991 when NRC received reports of problems with Thermo-Lag.

The OIG inspection disclosed at least seven instances in which NRC staff failed to pursue possible problems with Thermo-Lag. As a result of the inspection, a Commission-directed task force addressed the findings of the OIG report.

Alleged Conflict of Interest With a Sandia National Laboratory Contract. The NRC staff reported a potential conflict of interest related to an NRC research project at the Sandia National Laboratory (SNL), in Albuquerque, N.M. An SNL engineer formerly assigned to the NRC project continued serving as a technical reviewer after being reassigned to a Department of Energy (DOE) project benefiting the nuclear industry. It was also reported that the engineer inappropriately obtained NRC research information to prepare an industry report. OIG determined that the NRC project manager requested the engineer's continued participation in the NRC research program, even though the NRC program was to have been separate and distinct from the DOE project. The project manager failed to obtain a waiver for a conflict of interest, as required by NRC policy.

The OIG has not determined whether the information exchanged by the SNL engineers constituted personal misconduct. However, OIG concluded that SNL created an organizational conflict of interest by allowing an engineer to work simultaneously for the industry and the NRC on similar research projects.

NRC Employees Counseled for Fraternizing with Contractors. The OIG received allegations that certain employees assigned to a program office were involved in inappropriate activities during the course of their work. It was alleged that the employees were guilty of time and attendance abuse, had developed an inappropriate relationship with a contractor, and had failed to exercise proper oversight responsibilities during their management of a major NRC contract.

The investigators confirmed that the employees attended luncheon meetings with contractor employees. Some of these meetings lasted several hours and included the consumption of alcohol. The investigators further determined that inadequate management oversight by these NRC employees allowed the contractor to purchase several types of automated data processing equipment without prior approval of the NRC contracting officer, as mandated in the contract terms.

The employees were counseled by NRC managers, and one received a letter of reprimand and was relieved of supervisory duties. The Division of Contracts and Property Management, in concert with the program office, developed a training curriculum designed to cover acquisition procedures and ethical conduct related to procurement.

Allegations of Sexual Harassment Against an NRC Manager. An NRC program office notified the OIG after a female employee complained of several incidents involving sexual harassment and the handling of the incidents by NRC managers. OIG investigators determined that an NRC manager made repeated advances toward a female employee that she found offensive. Despite her attempts to discourage the manager, his behavior continued. It was also determined that cognizant supervisors failed to appropriately address the issue when it was reported to them. The manager who made the advances was relieved of supervisory duties. Several other managers were counseled with respect to their failure to adequately address the allegations.

Inspection of Radiological Safety Concerns Found Inadequate and Incomplete. The OIG investigated allegations that staff from an NRC Region failed to adequately address safety concerns brought to their attention. A former executive of a nuclear industry consulting firm advised the OIG that NRC inspectors performed an inadequate and incomplete inspection of reported radiological safety violations by his firm, an NRC licensee.

The OIG determined that the alleger's concerns had not been adequately examined during a 1990 NRC inspection. The NRC inspectors did not interview witnesses purportedly capable of corroborating the alleger's claims and did not examine key documents disputing certain assertions the licensee made. The regional staff conducted a second inspection that revealed inaccuracies in the report of the initial inspection and disclosed a number of violations by the licensee.

The OIG investigative report was sent to NRC management, and a task force was convened to address lessons learned in connection with this matter.

Misuse of Frequent-Flyer Mileage by an NRC Engineer. The OIG received information that a senior NRC engineer repeatedly converted "frequent-flyer" mileage accumulated through official Government air travel to personal use. Investigators determined that the engineer inappropriately redeemed frequent-flyer mileage credit on numerous occasions to obtain bonus tickets for personal airline travel, which included foreign travel. The accumulated value of the personal travel was approximately $10,750.

The U.S. Attorney, Baltimore, Md., declined prosecution of the engineer in lieu of administrative action by the NRC.

Alleged Bank Fraud by an NRC Employee. The NRC Division of Security notified the OIG that an NRC employee was under investigation by the FBI for an alleged violation of Title 18, *United States Code*, Section 1344 (bank fraud). At the FBI's request, OIG investigators provided investigative support that aided in the arrest of the employee. Following the employee's arrest, the OIG provided background information about the employee involving a similar prior offense that was not previously available to the pretrial service officer. The employee resigned from the NRC soon after being arrested and is currently awaiting Federal prosecution.

FINANCIAL MANAGEMENT

NRC License and Annual Fees

The Omnibus Budget Reconciliation Act of 1990 (Public Law 101–508) requires that, in fiscal year 1992, 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 1992, a total of $512.5 million was appropriated to the NRC (Public Law 102–104), of which $19,962,000 was derived from the Nuclear Waste Fund. Of the remaining $492,538,000, approximately 99 percent, or $489,265,320, was collected through license fees and annual charges. The net amount appropriated to the NRC in fiscal year 1992 was $3,272,680. Table 1 shows the amounts collected through license and annual fees in fiscal year 1992.

The NRC made two changes to its fee schedules in fiscal year 1992. First, on April 17, 1992, the NRC published in the *Federal Register* two limited changes to 10 CFR Parts 170 and 171. The limited changes became effective May 18, 1992. The limited change to 10 CFR Part 170 allowed the NRC to bill quarterly for those license fees that were previously billed every six months. The limited change to 10 CFR Part 171 adjusted the maximum annual fee of $1,800 assessed a materials licensee who qualifies as a "small entity" under the NRC's size standards. A lower-tier small entity fee of $400 per licensed category was established for small businesses and non-profit organizations with gross annual receipts of less than $250,000 and small governmental jurisdictions with a population of less than 20,000.

Second, on July 23, 1992, the NRC published a final rule in the *Federal Register* that established the licensing, inspection, and annual fees for fiscal year 1992. This revision was made to implement Pubic Law 101–508, enacted by the Congress on November 5, 1990. For fiscal year 1992, the law requires that the NRC recover approximately 100 percent of its budget authority, which is $512.5 million, less the amount appropriated from the Nuclear Waste Fund, by assessing license, inspection and annual fees. The basic methodology used in the fiscal year 1992 rule was unchanged from that used to calculate the 10 CFR Part 170 professional hourly rate, the specific materials licensing and inspection fees in 10 CFR Part 170, and the 10 CFR Part 171 annual fees in the final rule published July 10, 1991 (56 FR 31472).

Major changes in the July 23, 1992 fee regulation are as follows:

Changes in Part 170:

- Amend 10 CFR 170.20 to change the cost per professional-staff hour for all full-cost fees from $115 per hour to $123 per hour.
- Increase all flat fees for radioisotope programs by seven percent, using the increased hourly rate as a basis.
- Add additional categories of fees for export and import licenses.
- Add a definition for nonprofit educational institutions.

Changes in Part 171:

- Increase the Part 171 annual fees assessed to reactor and materials licensees.
- Divide Class I facilities in the uranium recovery class of licensees into two classes. The additional category (Class II) recognizes those licensees who do not generate uranium mill tailings.
- Add a definition for nonprofit educational institutions.
- Amend the exemption provisions of § 171.11 to require that licensees who wish to be considered for an exemption from the annual fees file their respective exemption requests within 90 days from the effective date of the rule establishing the annual fees.

Litigation Concerning Fees. The Commission published a Final Notice of Rulemaking in the *Federal Register* on July 10, 1991, establishing the revised license, inspection, and annual fees for fiscal year 1991. The revisions to 10 CFR Parts 170 and 171 became effective August 9, 1991. Three lawsuits were filed with the U.S. Court of Appeals for the District of Columbia Circuit, petitioning the court to review the final fiscal year 1991 fee regulations. During fiscal year 1992, one of the lawsuits was withdrawn. The two remaining lawsuits were argued on November 5, 1992. Three other lawsuits were filed in the U.S. Court of Appeals for the D.C. Circuit requesting review of the final fiscal year 1992 fee rule. These lawsuits are pending.

Implementing the Chief Financial Officers Act

Enactment of the Chief Financial Officers Act (CFOs Act) of 1990 and subsequent actions of the Office of Management and Budget (OMB) have given increased emphasis to the strengthening of financial management at the NRC.

Office of the Inspector General (OIG) audits and internal financial management assessments have indicated a need for increased awareness of the importance of financial management, improved financial management processes, and enhanced accountability. Specific actions taken in fiscal year 1992 are described under the following financial management functional areas.

Financial Management Organization. The NRC's CFO organization proposal was approved by OMB in fiscal year 1991. The CFO and Deputy CFO (DCFO) were appointed, as required by the statute. Appropriate responsibilities and authorities of the CFO and DCFO were incorporated into management directives.

Financial Management Personnel. The authorized staffing level for the Office of the Controller, the primary financial management organization in the agency, was increased. FTEs were added to prepare auditable financial statements, and to augment staff capabilities for financial systems development, budget execution, and financial accounting. To better enable the OIG to carry out its responsibilities, including the audit of agency financial statements and audits of contracts, the OIG staffing level was increased.

The development of two courses was initiated for allotment financial managers and funds certifying officials in program and regional offices.

To stress the importance of financial management for all NRC managers, financial management was made a mandatory subelement of the management effectiveness element for all Senior Executive Service (SES) performance plans.

Accounting Standards. The NRC remains current with technical developments in accounting and implements changes in accounting standards on a timely basis. In

Table 1. License and Annual Fee Collections – FY 1992

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR Part 170	$93.1 million	$13.4 million	$106.5 million
10 CFR Part 171	$341.4 million	$41.4 million	$382.8 million
TOTAL FEES	$434.5 million	$54.8 million	$489.3 million

preparation for the audit of the fiscal year 1992 financial statements, the NRC is reviewing its accounting practices to determine if they conform to applicable accounting principles, standards and other requirements.

As required by the CFOs Act, the NRC has established a process for the biennial review of fees and other charges imposed by the NRC, to assure that such fees and other charges accurately reflect costs incurred.

Financial Systems. Significant effort has been expended on improving financial systems during fiscal year 1992. In addition to maintaining current systems, much effort has been focused on the replacement of existing systems with more robust financial systems.

The NRC entered into a cross-servicing agreement with the Department of Treasury Financial Management Service to implement the Federal Financial System (FFS). FFS will replace the core financial management system, and was operational October 1, 1992. This system provides for the integration of budget and accounting data and provides for the implementation of the Standard General Ledger at the transaction level. It will automate use of commitment accounting, permit queries of payments and accounting transactions, and consolidate accounting components into one system. FFS conforms to the Government-wide system standards for core financial systems.

The NRC recognizes that the current Payroll System will have to be replaced within the next few years because it is becoming increasingly difficult to make changes to the system. Based on the recommendations of a study conducted in fiscal year 1992, the NRC will enter into a cross-servicing agreement with another Federal agency for an integrated payroll/personnel system.

Improvements were made to the Accounts Receivable System to automate functions such as identification of delinquent accounts, aging of receivables, and calculation of interest and other charges.

Improvements were made to the License Fee Bill Generator System to provide enhanced audit trails and detailed listings for fee billings.

Quarterly budget execution reports for the CFO and Commission were initiated during fiscal year 1992.

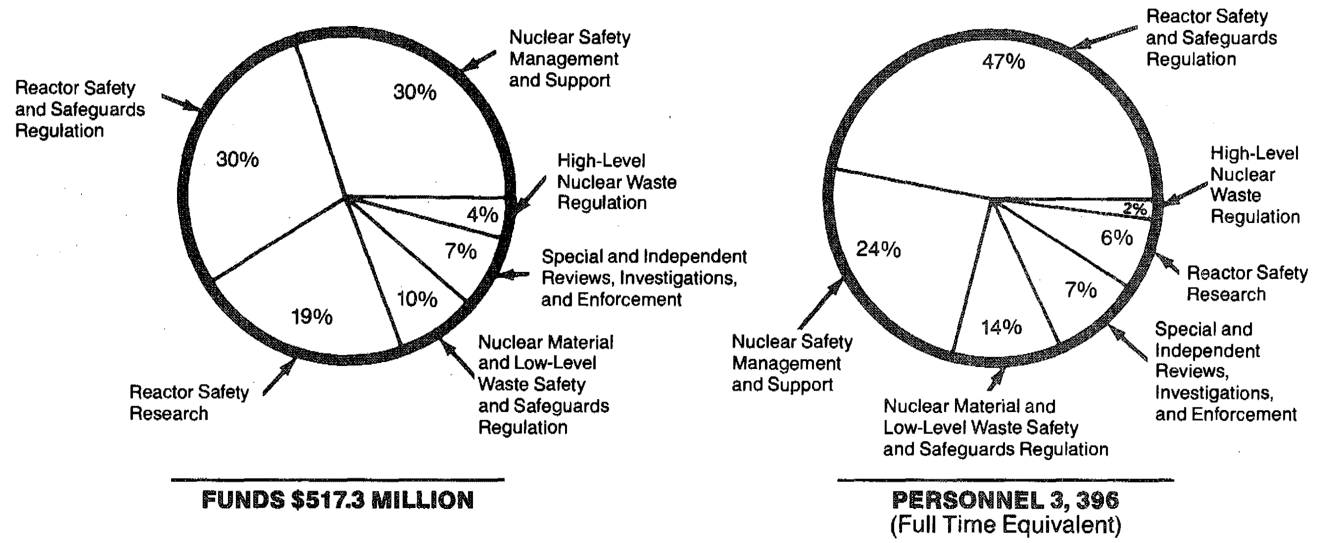
Internal Controls. Internal control reviews were conducted and completed in a timely manner during fiscal year 1992. Material weaknesses were identified, and corrective actions planned.

A quality assurance and monitoring program was established in fiscal year 1992 to ensure that annual financial system reviews are performed as required by OMB guidance, results are evaluated, and corrective actions taken.

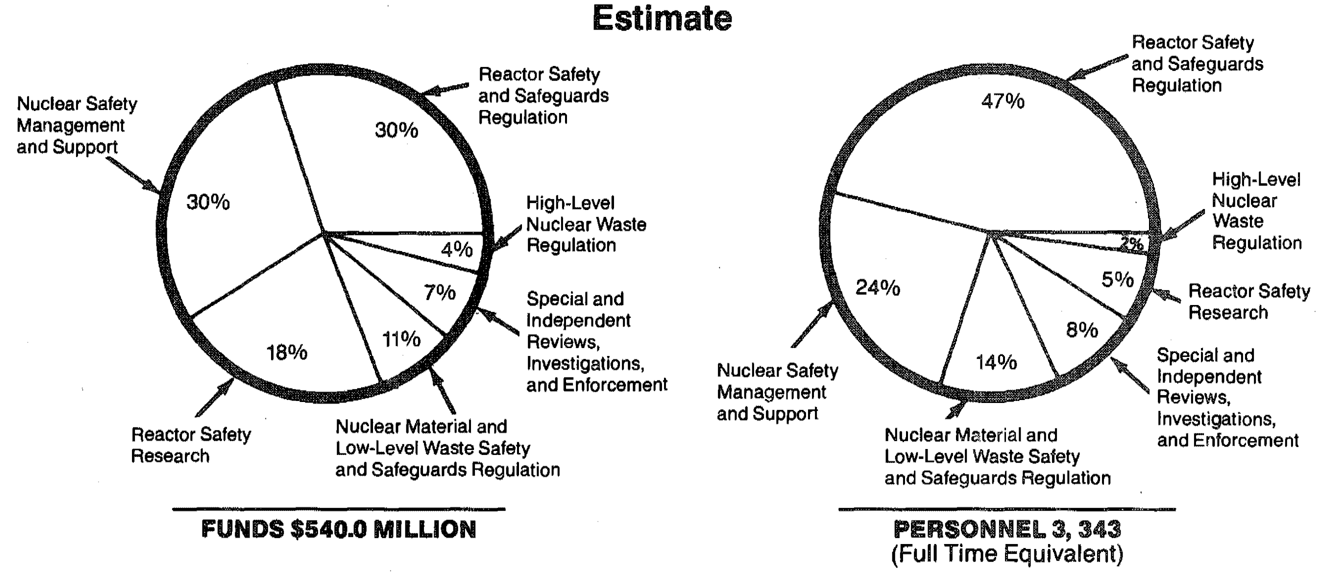
Management resolved all audit recommendations made in OIG audit reports issued through July 1992.

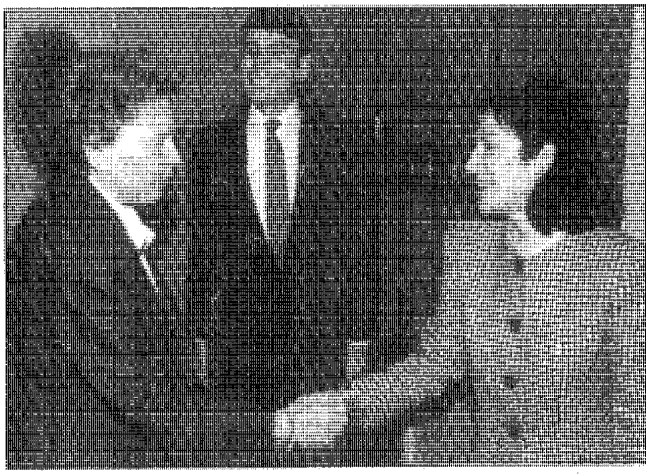
Asset Management. Efforts to improve the timely payment of invoices subject to the Prompt Payment Act have consistently resulted in over 90 percent of payments being made on time; fiscal year 1991 Government-wide average was approximately 88 percent.

NRC RESOURCES
FY 1992
Actual



NRC RESOURCES
FY 1993
Estimate





The NRC celebrated the Hispanic Heritage Observance of the 500th Anniversary of the Discovery of America in October 1992 with an event featuring as guest speaker the Honorable Jeane J. Kirkpatrick, Former U.S. Ambassador to the United Nations. Shown at left, welcoming Dr. Kirkpatrick, is Maria Lopez-Otin, NRC's Federal Liaison (see Chapter 7). At right is Ambassador Kirkpatrick addressing the NRC audience; also on the dais are, left-to-right, Ms. Lopez-Otin, Chairman Ivan Selin, Commissioner Forrest J. Remick, and Deputy Executive Director for Operations Hugh L. Thompson, Jr. The occasion was sponsored by the newly created NRC Hispanic Employment Program Advisory Committee and the Office of Small and Disadvantaged Business Utilization and Civil Rights.

NRC imprest funds were reduced by approximately 50 percent in fiscal year 1992 due to the increased use of third-party drafts, for the payment of travel vouchers.

Use of electronic collections and payments has been expanded in order to improve cash management.

A process was formalized in fiscal year 1992 to handle delinquent fee-related debts, and a debt collection contractor was retained to assist in the collection of debts. License revocation orders have been issued due to non-payment of fee-related debt.

Audited Financial Reporting and Performance Information. During fiscal year 1992, the NRC has taken necessary actions to prepare and audit the fiscal year 1992 financial statements in conformance with OMB requirements and guidance. An auditability survey of fiscal year 1991 financial statements was completed. Contractor support has been obtained to assist both in the preparation and the audit of the financial statements.

In September 1992, the NRC submitted to OMB and congressional oversight committees a list of program and financial performance measures that will be reported in the NRC's fiscal year 1992 financial statements.

Federal Managers' Financial Integrity Act Report. The Chairman submitted a 1991 report to the President on the agency's internal accounting and administrative controls. The report, prepared pursuant to the Federal Managers' Financial Integrity Act of 1982 (FMFIA), indicated that there were neither pending material internal control weaknesses nor material nonconformances with OMB financial policies and objectives. However, in May 1992, the NRC advised OMB that it will identify the weakness associated with the management of agreements with DOE as a material weakness in the 1992 FMFIA report.

OFFICE OF SMALL AND DISADVANTAGED BUSINESS UTILIZATION AND CIVIL RIGHTS

Small and Disadvantaged Business Utilization Program

The Small and Disadvantaged Business Utilization Program annually establishes procurement preference goals, in conformance with provisions of Public Law 95–507, amending the Small Business Investment Act of 1957. The following is a summary of estimated and actual contract awards during fiscal year 1992.

- It was estimated that $70,000,000 in total prime contracts would be awarded during fiscal year 1992. The actual total for prime contract awards was $80,687,568.

- 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 $37,439,167, or 46.40 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

1992. Awards to "8(a) firms" were actually $14,999,684, or 18.59 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 $987,980, or 1.22 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 $1,007,423, or 1.25 percent of the total dollar amount of all prime contracts, regardless of dollar value.

- 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 $2,950,000, or 79.19 percent of total subcontracts awarded. The NRC's total subcontract goal in fiscal year 1992 was $3,150,000. The NRC's actual subcontract dollar awards were $3,725,168.

- The goal for subcontract awards to small disadvantaged businesses was $410,000, or 13.02 percent. Subcontracting awards to small disadvantaged businesses totaled $500,000, or 13.42 percent of total subcontract dollars awarded.

During the report period, 140 interviews were conducted with firms wanting to do business with the NRC, and 48 follow-up meetings were arranged with NRC technical personnel. The staff of the Office of Small and Disadvantaged Business Utilization and Civil Rights also participated in five major small business conferences. Most noteworthy among these were the Small Business Development Week, in May 1992, and the Minority Enterprise Development Week, in September 1992.

Civil Rights Program

During the report period, the Commission was briefed on January 16, 1992 and July 29, 1992, concerning NRC's EEO and Affirmative Employment programs, goals, and accomplishments.

The annual accomplishment report for the NRC's Multi-Year Affirmative Employment Program Plan was signed by the Chairman and submitted to the Equal employment Opportunity Commission.

The Director of the Office of Small and Disadvantaged Business Utilization and Civil Rights, William B. Kerr, continues to serve as a non-voting, *ex-officio* member of the SES Performance Review Board.

The Civil Rights Program staff sponsored a three-day training seminar for Equal Employment Opportunity counselors from NRC Headquarters and the Regional Offices. The event, which was held at Hunt Valley, Md., was well attended and well received.

Two new Equal Employment Opportunity committees—the Hispanic Employment and the Asian/Pacific American Advisory committees—were created during the fiscal year.

Federal Women's Program

National Women's History Month was observed throughout NRC during March 1992. Various programs were held with outstanding speakers, receptions, exhibits and presentation of awards. A special program was held in NRC Headquarters featuring Maryland Congresswoman Constance Morella as the Guest Speaker and Commissioner E. Gail de Planque as the Keynote Speaker.

Women continued to make gains in attaining grades GG–13 and above in the NRC. The number of women in grade-13 increased by 5.2 percent, in grade-14 by 11 percent, and in grade 15 by more than 17.6 percent. Women represent 39.8 percent of the total NRC work-force.

The Annual Training and Planning Conference of the Federal Women's Program took place during the fiscal year in Cincinnati, Ohio, in conjunction with the Federally Employed Women's National Training Conference.

Appendix 1

NRC Organization

(As of December 31, 1992)

COMMISSIONERS

Ivan Selin, Chairman
Kenneth C. Rogers
James R. Curtiss
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, Harold R. Denton, Director
Office of the Licensing Support System Administrator, Lloyd J. Donnelly, Administrator
Office of Public Affairs, Joseph J. Fouchard, Director
Secretary of the Commission, Samuel J. Chilk

Other Offices

Advisory Committee on Nuclear Waste, Dade W. Moeller, Chairman
Advisory Committee on Reactor Safeguards, Paul G. Shewmon, 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 and Disadvantaged Business Utilization/Civil Rights, William B. Kerr, Director
Office of State Programs, Carlton Kammerer, 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., A. Bert Davis, Regional Administrator
Region IV—Dallas, Tex., Robert D. Martin, Regional Administrator
Region V—San Francisco, Cal., John B. Martin, 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 regulatory 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 directs and coordinates the Commission's operational and administrative activities among the program and support staff offices described below and also coordinates the development of policy options for Commission consideration. The Executive Director for Operations reports directly to the Chairman.

The Office of Nuclear Material Safety and Safeguards is responsible for the licensing, inspection, and regulation of facilities and materials associated with the processing, transport and handling of nuclear materials, and with the disposal of nuclear waste; the office also regulates uranium recovery facilities. The safeguards staff of the office reviews and assesses protections against potential threats, thefts and sabotage for licensed facilities, including reactors, 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 carries out the licensing and inspection of nuclear power reactors, test reactors, and research reactors. Reactor licensing is a two-phase process. A construction permit is granted before facility construction can begin and an operating license is issued before fuel can be loaded. The office reviews license applications to assure that each proposed facility can be built and operated without undue risk to the health and safety of the public and with minimal impact on the environment. The office monitors operating reactor facilities during their lifetime through decommissioning.

The Office of Nuclear Regulatory Research plans and conducts the comprehensive research and standards program that is deemed necessary for the performance of the Commission's licensing and regulatory functions and that is responsive to current and future NRC needs. The program covers such areas as facility operation, engineering technology, accident evaluation, probabilistic risk analysis, siting, health, and waste management.

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 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 is responsible for conducting investigations and audits which are directed principally toward improving program management, assuring the integrity of the NRC's regulatory program, and preventing and identifying fraud or misuse of agency funds by agency employees.

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 the Licensing Support System Administrator is responsible for ensuring that the NRC's Licensing Support System (LSS) meets the requirements of 10 CFR Part 2 related to the use of the LSS in the Commission's high-level waste licensing proceedings; advising the Department of Energy (DOE) on the design, development, testing and any necessary redesign of the LSS; providing for the operation and maintenance of the LSS to include the entry of documentary material into the LSS and access to the System by LSS participants and the public; maintaining the integrity and security of the LSS data base; and reviewing compliance of LSS participants with the applicable LSS rules, including DOE compliance with the document submission requirements in 10 CFR 2.1003. (This office, as such, was discontinued in fiscal year 1993, and the function has been restructured.)

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 for and records Commission meetings; manages the Commission staff paper and COMSECY systems; codifies Commission decisions in memoranda directing staff action and monitors pending staff actions; 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; administers the NRC Historical Program; directs and administers the NRC Public Document Room; 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 long-term objective of consolidating all of the NRC's Headquarters operations at a single location; consolidation has begun and is expected to require several years to reach completion.

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 is responsible for developing, providing and administering information resources throughout the agency in the areas of computer operations, telecommunications, and similar centralized information services, including data base management, office automation, computer hardware and software, systems development, nationwide telecommunications equipment and services, an Information Technology Services Support Center, and user training.

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 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 Personnel plans and implements NRC policies, programs, and services to provide for the effective organization, staffing, utilization and development of the agency's human resources.

The Office of Small and Disadvantaged Business Utilization 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 labor surplus area firms and women-owned 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.

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 was established by the Nuclear Regulatory Commission in 1988 to advise the Commission on all aspects of nuclear waste management within the purview of NRC responsibility.

Advisory Committee on Medical Uses of Isotopes, established in July 1958, is composed of qualified physicians and scientists who consider medical questions referred to them by the NRC staff and give expert opinions on the medical uses of radioisotopes. The Committee also advises the NRC staff, as required, on matters of policy. Members are employed under yearly personal services contracts.

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 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 1992.)

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. PAUL G. SHEWMON, Professor Emeritus, Material Science and Engineering Department, Ohio State University, Columbus, Ohio.

VICE-CHAIRMAN: MR. JAMES C. CARROLL, retired Manager, Nuclear Operations Support Department, Pacific Gas & Electric, San Francisco, Cal.

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.

DR. 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. J. ERNEST WILKINS, JR., Distinguished Professor of Applied Mathematics and Mathematical Physics, Clark Atlanta University, Atlanta, Ga.

MR. CHARLES J. WYLIE, retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.

Atomic Safety and Licensing Board Panel
(See Chapter 9; membership as of September 1992.)

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, 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, Oak Ridge, Tenn.

JUDGE THOMAS E. 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, Minneapolis, Minn.

JUDGE RICHARD F. FOSTER, Environmental Scientist, Sunriver, Ore.

JUDGE CADET H. HAND, JR., Marine Biologist, University of California, Bodega Bay, Cal.

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, Datona 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.

ELVA W. LEINS, Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JACK G. WHETSTINE, Assistant to the Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Licensing Support System Advisory Review Panel
(Membership as of September 1992.)

The Licensing Support System Advisory Review Panel (LSSARP) was established in 1989 to advise the NRC's Licensing Support System Administrator and the Department of Energy on selected aspects of the design, development and operation of the Licensing Support System.

CHAIRMAN JOHN C. HOYLE, U.S. Nuclear Regulatory Commission.

BOYD ALEXANDER, U.S. Patent and Trademarks Office.

KIRK BALCOM, State of Nevada.

DENNIS BECHTEL, Clark County, Nevada, Comprehensive Planning Department.

STEVE BRADHURST, Nye County, Nevada, Board of Commissioners.

BARBARA CERNY, U.S. Department of Energy.

DAVID COPENHAFER, U.S. Securities and Exchange Commission.

PETER CUMMINGS, Las Vegas, Nevada, City Manager's Office.

BILL ELQUIST, Lander County, Nevada.

PETE GOICOECHEA, Eureka County, Nevada, Commissioner.

CHRISTOPHER HENKEL, Edison Electric Institute.

ELGIE HOLSTEIN, Nye County, Nevada, Board of Commissioners.

FELIX KILLAR, U.S. Council for Energy Awareness.

STEVEN KRAFT, Edison Electric Institute.

JOHN LAMPROS, White Pine County, Nevada.

ANTHONY LESSARD, Mineral County, Nevada.

CORINNE MACALUSO, U.S. Department of Energy.

LORETTA METOXEN, National Congress of American Indians.

MALACHY MURPHY, Nye County, Nevada, Board of Commissioners.

JAMES REGAN, Churchill County, Nevada.

JAY SILBERG, Utility Nuclear Waste Management Group.

LENARD SMITH, Lincoln County, Nevada, Commissioner.

HARRY SWAINSTON, State of Nevada.

OTHER NRC ADVISORY GROUPS

Advisory Committee on the Medical Uses of Isotopes
(Membership as of September 1992.)

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 Nuclear Regulatory Commission (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.

CAPT. WILLIAM H. BRINER, Associate Professor of Radiology, Duke University Medical Center, Durham, N.C.

JUDITH I. BROWN, Health Policy Consultant for American Association of Retired Persons, Washington, D.C.

STEVEN C. COLLINS, Chief of Division of Radioactive Materials, Department of Nuclear Safety, State of Illinois, Springfield, Ill.

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, Center for Drug Evaluation and Research, 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. GERALD M. POHOST, Director, Division of Cardiovascular Disease, University of Alabama, Birmingham, Ala.

DR. EDWARD W. WEBSTER, Director, Division of Radiological Science, Massachusetts General Hospital, Boston, Mass.

Advisory Committee on Nuclear Waste
(Membership as of September 1992.)

The Advisory Committee on Nuclear Waste reports to and advises the Nuclear Regulatory Commission on nuclear waste management. The primary emphasis is on disposal but also includes other activities off-site of production and utilization facilities, such as handling, processing, transportation, storage, and safeguarding of nuclear wastes including spent fuel, nuclear wastes mixed with other hazardous substances, and uranium mill tailings.

CHAIRMAN: DR. DADE W. MOELLER, Professor of Engineering in Environmental Health and Associate Dean for Continuing Education, School of Public Health, Harvard University, Boston, Mass.

VICE-CHAIRMAN: DR. MARTIN J. STEINDLER, Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Ill.

DR. WILLIAM J. HINZE, Professor, Department of Earth and Atmospheric Sciences, Purdue University, West Lafayette, Ind.

DR. PAUL W. POMEROY, President, Rondout Associates, Incorporated, Stone Ridge, N.Y.

Advisory Panel for the Decontamination of Three Mile Island Unit 2
(Membership as of September 1992.)

The Advisory Panel for the Decontamination of Three Mile Island Unit 2 (Pa.) was established in October 1980. Its purpose is to obtain the views and perspectives of residents of the Three Mile Island area near Harrisburg, Pa., and to afford 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 the following members representing agencies of the Commonwealth of Pennsylvania, local government, the scientific community, and persons having their principal place of residence in the vicinity of the Three Mile Island nuclear power plant.

CHAIRMAN: ARTHUR E. MORRIS, Resident and former mayor of Lancaster, Pa.

JOHN LUETZELSCHWAB, 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. Hersey Medical Center, Hersey, Pa.

FREDERICK S. RICE, Resident of Harrisburg, Pa.

GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.

JOEL ROTH, Resident of Harrisburg, Pa.

THOMAS SMITHGALL, Resident of Lancaster, Pa.

ANN TRUNK, Resident of Middletown, Pa.

NEIL WALD, Professor, Department of Environmental and Occupational Health, University of Pittsburgh, Pittsburgh, Pa.

Nuclear Safety Research Review Committee
(Membership as of December 31, 1992.)

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. 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., Parsippaly, 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. DONALD L. TURCOTTE, Chairman, Department of Geological Sciences and Maxwell Upson Professor of Engineering, Cornell University, Ithaca, N.Y.

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.

DR. DAVID D. WOODS, Associate Professor, Department of Industrial and Systems Engineering, Ohio State University, Columbus, Ohio.

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. Ann Kuntzman, 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

- Mr. Richard Kraus
  West Los Angeles Regional Library
  11360 Santa Monica Boulevard
  Los Angeles, Cal. 90025
  UCLA Training Reactor

- 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
  Thames Valley State Technical College
  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. Roxanne Frey
  Library Director
  Government Documents Collection
  Wilmington Public Library
  201 South Kankakee Street
  Wilmington, Ill. 60481
  Braidwood nuclear plant

- Ms. Sandy Sherwood
  Reference Librarian
  Waukegan Public Library
  128 N. County Street
  Waukegan, Ill. 60085
  Zion nuclear plant

- Ms. Ann Bergstrom, Library Assistant
  West Chicago Public Library
  332. E. 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

- Dr. Carol Juth, 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

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

- Mr. David Robrock
  Special Collections Librarian
  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. Laura Given
  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

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  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
  2801West Bancroft Avenue
  Toledo, Ohio 43606
  Davis-Besse nuclear plant

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- 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
  701 South Cooper
  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
  Assistant Librarian
  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 1992

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Salary Offset Procedures for Collecting Debts Owed by Federal Employees to the Federal Government—Parts 15 and 16

On October 16, 1991 (56 FR 51829), the NRC published an amendment to its regulations establishing procedures to collect certain debts owed by Federal employees to the NRC and other Federal agencies by deduction(s) from their pay. This amendment, effective November 15, 1991, is necessary to conform NRC regulations to the Debt Collection Act of 1982 which requires each agency to establish a salary offset program for the collection of these debts.

Material Control and Accounting Requirements for Uranium Enrichment Facilities Producing Special Nuclear Material of Low Strategic Significance—Parts 2, 40, 70 and 74

On October 31, 1991 (56 FR 55991), the NRC published an amendment to its regulations to include performance-based material control and accounting requirements that will apply to uranium enrichment facility licensees who produce significant quantities of special nuclear material of low strategic significance. This amendment, effective December 2, 1991, is necessary to ensure that enrichment facilities produce enriched uranium of low strategic significance only as authorized. This amendment applies to all applicants who build or operate enrichment facilities.

Revision of Fee Schedules; 100 Percent Fee Recovery; Clarification of Size Standards—Part 171

On November 13, 1991 (56 FR 57587), the NRC published an amendment to its regulations, effective immediately, concerning the payment of annual fees, to clarify the provisions that identify the size standards used to determine whether an NRC licensee would qualify as a "small entity" under the Regulatory Flexibility Act, for the purpose of paying a reduced annual fee.

Nuclear Power Plant License Renewal—Parts 2, 50, 54 and 140

On December 13, 1991 (56 FR 64943), the NRC published an amendment to its regulations that establishes the requirements that an applicant for renewal of a nuclear power plant operating license must meet, the information that must be submitted to the NRC for review so that the agency can determine whether those requirements have in fact been met, and the application procedures. This amendment, effective January 13, 1992, is necessary to provide the regulatory requirements for extending nuclear power plant operating licenses beyond 40 years.

Exclusion of Attorneys From Interviews Under Subpoena—Part 19

On December 19, 1991 (56 FR 65948), the NRC published an amendment, effective January 21, 1992, revoking its regulations pertaining to exclusion of attorneys from interviews under subpoena. These regulations were vacated upon judicial review by the United States Court of Appeals for the District of Columbia Circuit.

Reorganization of the Office of Governmental and Public Affairs—Part 1

On January 15, 1992 (57 FR 1638), the NRC published an amendment to its regulations, effective immediately, to reflect the Commission's decision to abolish the Office of Governmental and Public Affairs and to reassign its subordinate offices and functions.

DOE-L or DOE-Q Reinvestigation Program for NRC-R Access Authorization Renewal Requirements—Part 11

On January 22, 1992 (57 FR 2441), the NRC published an amendment to its regulations to allow an exception to NRC-R access authorization renewal requirements. This amendment, effective February 21, 1992, is intended to reduce administrative and investigative costs to affected licensees and administrative costs to the Federal government.

NRC Licensee Reinvestigation Program—Part 25

On January 31, 1992 (57 FR 3719), the NRC published an amendment to its regulations, effective March 31, 1992, to require a reinvestigation program for NRC licensee personnel with "Q" and "L" access authorizations and to amend the fee schedule to recover the investigative costs. The reinvestigation program is consistent with the Department of Energy's program for its contractors and with NRC's policy of reinvestigating its own employees, consultants, contractors, experts, and panel members.

Revisions to Procedures to Issue Orders—Parts 2 and 15

On February 4, 1992 (57 FR 4152), the NRC published an amendment to its regulations, effective immediately, to conform several sections in 10 CFR Parts 2 and 15 to the changes in Part 2 contained in the final rule "Revisions to Procedures to Issue Orders; Deliberate Misconduct by Unlicensed Persons," which was effective September 16, 1991 (56 FR 40678; August 15, 1991).

Fingerprint Cards: Resubmittal Procedure Change—Part 73

On March 4, 1992 (57 FR 7645), the NRC published an amendment to its regulations to conform to new procedures adopted by the Federal Bureau of Investigation. This amendment, effective April 3, 1992, reflects an administrative change pertaining to the resubmittal of rejected fingerprint cards associated with granting access to Safeguards Information or for granting unescorted access to an operating nuclear power plant as required by Public Law 93–399.

Limited Revision of Fee Schedules—Parts 170 and 171

On April 17, 1992 (57 FR 13625), the NRC published an amendment to its regulations to make two limited changes to its assessment of license and annual fees. The amendment assesses license fees, which are based on the full-cost method, quarterly instead of semiannually and establishes a lower tier small entity annual fee for those licensees that are small entities with relatively low annual gross receipts or supporting populations. This amendment, effective May 18, 1992, is intended to improve NRC financial management and further mitigate the impact of the annual fee on small licensees with relatively low annual gross receipts or supporting populations.

Uranium Enrichment Regulations—Parts 2, 40, 50, 51, 70, 75, 110, 140, 150 and 170

On April 30, 1992 (57 FR 18388), the NRC published an amendment to its regulations, effective June 1, 1992, concerning the licensing of uranium enrichment facilities to reflect changes made to the Atomic Energy Act of 1954, as amended by the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990.

Revisions to Procedures to Issue Orders: Challenges to Orders That Are Made Immediately Effective—Part 2

On May 12, 1992 (57 FR 20194), the NRC published an amendment to its regulations, effective June 11, 1992, to allow challenges to the immediate effectiveness of an order to be made at the outset of a proceeding and provide procedures for the expedited consideration and disposition of these challenges. The amendment also requires that challenges to the merits of an immediately effective order be heard expeditiously, except where good cause exists for delay.

Acquisition Regulation (NRCAR): Debarment—48 CFR Chapter 20

On July 1, 1992 (57 FR 29220), the NRC published an amendment to its regulations, effective July 31, 1992, establishing the Nuclear Regulatory Commission Acquisition Regulation (NRCAR). The NRCAR is intended to implement and supplement the government-wide Federal Acquisition Regulation. This final rule contains only the agency's debarment, suspension, and ineligibility procedures.

Decommissioning Funding for Prematurely Shut Down Power Reactors—Part 50

On July 9, 1992 (57 FR 30383), the NRC published an amendment to its regulations, effective August 10, 1992, on the timing of the collection of funds for decommissioning for those nuclear power reactors that have shut down before the expected ends of their operating lives. This amendment requires that the NRC evaluate decommissioning funding plans for power reactors that shut down prematurely on a case-by-case basis. The NRC's evaluation would take into account the specific safety and financial situations at each nuclear power plant.

Revision of Fee Schedules; 100% Fee Recovery, FY 1992—Parts 170 and 171

On July 23, 1992 (57 FR 32691), the NRC published an amendment to its regulations, effective August 24, 1992, to amend the licensing, inspection, and annual fees charged to its applicants and licensees. The amendments are necessary to implement Public Law 101–508, signed into law on November 5, 1990, which mandates that the NRC recover approximately 100 percent of its budget authority in Fiscal Year (FY) 1992 less amounts appropriated from the Nuclear Waste Fund. The amount to be recovered for FY 1992 is approximately $492.5 million.

Minor Amendments to the Physical Protection Requirements—Parts 70, 72, 73 and 75

On July 29, 1992 (57 FR 33426), the NRC published an amendment to its regulations that covers the physical protection of special nuclear material. This amendment, effective on August 28, 1992, (1) supplements the definitions section, (2) deletes action dates that no longer apply, (3) corrects outdated terms and cross references, (4) clarifies wording that is susceptible to differing interpretations, (5) corrects typographical errors, and (6) makes other minor changes.

Codes and Standards for Nuclear Power Plants—Part 50

On August 6, 1992 (57 FR 34666), the NRC published an amendment to its regulations, effective September 8, 1992, that incorporates by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and the 1986 Addenda, 1987 Addenda, 1988 Addenda and 1989 Edition of Section XI, Division 1, of the ASME Code.

Standards for Protection Against Radiation; Extension of Implementation Date—Parts 19 and 20

On August 26, 1992 (57 FR 38588), the NRC published an amendment to its regulations extending the implementation date for its revised standards for protection against radiation and making a conforming change to its regulations. This amendment, effective September 25, 1992, extends the date by which the NRC licensees are required to implement the revised standards for protection against radiation to January 1, 1994. The 1-year extension provides licensees additional time to examine and implement the regulatory guidance developed to support the rule. It also establishes a concurrent implementation date for the NRC licensees and Agreement State licensees.

Reducing the Regulatory Burden on Nuclear Licensees—Parts 20 and 50

On August 31, 1992 (57 FR 39353), the NRC published an amendment to its regulations, effective October 1, 1992, to reduce the regulatory burden on nuclear licensees. This action reflects an initiative undertaken by the Commission in response to a Presidential memorandum requesting that selected Federal agencies review and modify regulations that would eliminate any unnecessary burden of governmental regulation and ensure that the regulated community is not subject to duplicative or inconsistent regulation. The NRC's Committee to Review Generic Requirements identified eight areas where regulations could be revised to reduce the regulatory burden on licensees without reducing the protection for the public health and safety or the common defense and security. The final amendments address unnecessary regulatory requirements related to the frequency of reporting information, analysis of emergency core cooling systems for operating power reactors, and clarification and update of regulations affecting certain material licensees.

Access Authorization Fee Schedule for Licensee Personnel—Parts 11 and 25

On September 10, 1992 (57 FR 41375), the NRC published an amendment to its regulations, effective October 13, 1992, to revise 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 comply with current regulations that provide that the 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. This rule also inserts full identification (NRC Form number and name) of several forms used in the NRC personnel security process.

Quality Management Program and Misadministrations; NRC Override of OMB Disapproval of NRC Information Collection Request—Part 35)

On September 10, 1992 (57 FR 41376), the NRC published an amendment to its regulations, effective immediately, announcing the Commission's vote to override the Office of Management and Budget (OMB) disapproval of the information collection requirements imposed in the final rule entitled "Quality Management Program and Misadministrations" (July 25, 1991; 56 FR 34104). This amendment reflects OMB's assignment of a new control number to these information collection requirements.

Minor Modifications to Nuclear Power Reactor Event Reporting Requirements—Part 50

On September 10, 1992 (57 FR 41378), the NRC published an amendment to its regulations to make minor modifications to the current nuclear power reactor event reporting requirements. The final rule, effective October 13, 1992, reduces the industry's reporting burden and the NRC's response burden in event review and assessment.

REGULATIONS AND AMENDMENTS PROPOSED

Decommissioning Recordkeeping and License Termination: Documentation Additions—Parts 30, 40, 70, and 72

On October 7, 1991 (56 FR 50524), the NRC published an amendment to its regulations that would 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 radioactive waste to prepare and maintain additional documentation identifying areas where licensed materials and equipment were stored or used outside restricted areas, areas where spills have occurred, locations and contents of current and previous burial areas within the site, and equipment involved in the licensing activity that will remain on site at the time of termination of the license. This amendment would provide greater assurance that decontamination and decommissioning of licensed facilities are carried out in accordance with the Commission's regulations.

Physical Fitness Programs and Day Firing Qualifications for Security Personnel at Category I Licensee Fuel Cycle Facilities—Part 73

On December 13, 1991 (56 FR 65024), the NRC published an amendment to its regulations that would amend security personnel performance regulations for fuel cycle facilities possessing formula quantities of strategic special nuclear material (Category I licensees).

Exclusion of Attorneys From Interviews Under Subpoena—Part 19

On December 19, 1991 (56 FR 65949), the NRC published an amendment to its regulations that would provide for the exclusion of counsel from interviews of a subpoenaed witness when that counsel represents multiple interests and there is concrete evidence that such representation would obstruct and impede the investigation. The proposed amendments are designed to ensure the integrity and efficacy of the investigative and inspection process. Concurrently, the NRC published a final rule (December 19, 1991; 56 FR 65948), effective January 21, 1992, revoking its previously published attorney exclusion regulations. Those regulations were vacated upon judicial review.

Requirements for the Possession of Industrial Devices Containing Byproduct Material—Parts 31 and 32

On December 27, 1991 (56 FR 67011), the NRC published an amendment to its regulations that would govern the safe use of byproduct material in certain measuring, gauging, or controlling devices. The proposed changes are intended to ensure that general licensees are aware of and understand the requirements for the possession of devices containing byproduct material.

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 January 3, 1992 (57 FR 222), the NRC published an amendment to its regulations that would clarify the applicability of the criminal penalty provisions of the Atomic Energy Act of 1954, as amended, to certain regulations. The proposed rule would identify more clearly those regulations which may subject the violator to criminal penalties for willful violation, attempted violation, or conspiracy to violate.

Training and Qualification of Nuclear Power Plant Personnel—Parts 50 and 52

On January 7, 1992 (57 FR 537), the NRC published an amendment to its regulations that would require 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. The amendment is being proposed to meet the directives of Section 306 of the Nuclear Waste Policy Act of 1982.

Limited Revision of Fee Schedules—Parts 170 and 171

On January 9, 1992 (57 FR 847), the NRC published an amendment to its regulations that would govern the assessment on license and annual fees. The proposed amendments would improve NRC financial management and further mitigate the impact of the annual fee on small licensees with relatively low annual gross receipts or supporting populations.

Licensing Requirements for Land Disposal of Radioactive Wastes—Part 61

On March 6, 1992 (57 FR 8093), the NRC published an amendment to its regulations containing licensing requirements for low-level radioactive waste (LLW) disposal facilities. The proposed amendment would simplify LLW disposal facility licensing interactions for NRC, the NRC Agreement States, and potential applicants for LLW disposal licenses.

Low-Level Waste Shipment Manifest Information and Reporting—Parts 20 and 61

On April 21, 1992 (57 FR 14500), the NRC published an amendment to its regulations that would improve low-level waste (LLW) manifest information and reporting. The proposed amendment would (1) Improve the quality and uniformity of information contained in manifests which are required to control transfers of LLW intended for disposal at a land disposal facility; (2) Establish a set of forms to serve as a national Uniform Low-Level Radioactive WAste Manifest, in response to requests by Compacts and States; (3) Require the use of one of these forms as a mandatory shipping paper for LLW transport; (4) Require LLW disposal site operators to electronically store the information contained in the Uniform Manifest documents for each container; and (5) Require the disposal site operators to report the Uniform Manifest information on a machine-readable medium (e.g., magnetic disks or tapes).

Loss of All Alternating Current Power—Part 50

On April 21, 1992 (57 FR 14514), the NRC published an amendment to its regulations regarding the reliability of onsite alternating current sources for light-water-cooled nuclear power plants. The proposed amendment would require licensees to test and monitor emergency diesel generators (EDG) against criteria that indicate possible degradation from the EDG target levels selected for determining the specified station blackout duration.

Receipt of Byproduct and Special Nuclear Material—Part 50

On April 24, 1992 (57 FR 15034), the NRC published an amendment to its regulations that would allow a reactor licensee to receive back byproduct and special nuclear material that is produced by operating the reactor after that waste has been sent offsite to be reduced in volume by compaction or incineration. The proposed amendment would ensure that licensees have adequate short-term on-site storage capacity for self-generated LLW until permanent disposal capacity is available.

Import and Export of Radioactive Wastes—Part 110

On April 28, 1992 (57 FR 17859), the NRC published an amendment to its regulations that would reflect the September 1990 decision of the General Conference of the International Atomic Energy Agency approving a voluntary Code of Practice to guide Nation States in the development and harmonization of policies and laws on the international transboundary movement of radioactive waste. The proposed amendment would conform U.S. policies with these international recommendations.

Revision of Fee Schedules; 100% Fee Recovery, FY 1992—Parts 170 and 171

On April 29, 1992 (57 FR 18095), 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 amendment would implement Public Law 101–508, signed into law on November 5, 1990, which mandates that the NRC recover approximately 100 percent of its budget authority in Fiscal Year 1992, less amounts appropriated from the Nuclear Waste Fund.

Fitness-for-Duty Requirements for Licensees Who Possess, Use, or Transport Category I Material—Parts 26, 70 and 73

On April 30, 1992 (57 FR 18415), the NRC published an amendment to its regulations that would establish fitness-for-duty requirements for licensees authorized to possess, use, or transport unirradiated formula quantity of strategic special nuclear material.

Standards for Protection Against Radiation; Extension of Implementation Date—Parts 19 and 20

On May 19, 1992 (57 FR 21216), the NRC published an amendment to its regulations that would extend the date by which NRC licensees are required to implement the revised standards for protection against radiation to January 1, 1994. The proposed amendment would also establish a concurrent implementation date for NRC licensees and Agreement State licensees by eliminating the 1-year period during which Agreement States could continue to enforce the existing Part 20 while the NRC would be enforcing the revised standards.

Clarification of Physical Protection Requirements at Fixed Sites—Part 73

On May 29, 1992 (57 FR 22670), the NRC published an amendment to its regulations that would make clear that the Commission's regulations do not require protection against both radiological sabotage and theft of special nuclear material at all facilities. The proposed amendment would also add a requirement that nonpower reactor licensees who operate at or above 2 megawatts thermal protect against radiological sabotage.

Departures From Manufacturer's Instructions; Elimination of Recordkeeping Requirements—Parts 30 and 35

On June 11, 1992 (57 FR 24763), the NRC published an amendment to its regulations that would eliminate certain recordkeeping requirements related to the preparation and use of radiopharmaceuticals. The proposed rule would eliminate recordkeeping requirements related to the justification for and a precise description of the departure, and the number of departures from the Food and Drug Administration's approved manufacturer's instructions.

Reducing the Regulatory Burden on Nuclear Licensees—Parts 20 and 50

On June 18, 1992 (57 FR 27187), the NRC published an amendment to its regulations that would reduce the regulatory burden on all licensees. This proposed rule reflects an initiative undertaken by the Commission in response to a Presidential memorandum requesting that selected Federal agencies review and modify regulations that will reduce the burden of governmental regulation to ensure that the regulated community is not subject to duplicative or inconsistent regulation. The NRC's Committee to Review Generic Requirements (CRGR) identified regulations in eight areas that could be amended to reduce the regulatory burden on licensees without in any way reducing the protection for the public health and safety or the common defense and security. The proposed rule would address the frequency of reporting information and emergency core cooling system analysis for operating power reactors, clarify and update regulations affecting certain material licensees, and remove unnecessary regulatory requirements.

Minor Modifications to Nuclear Power Reactor Event Reporting Requirements—Part 50

On June 26, 1992 (57 FR 28642), the NRC published an amendment to its regulations that would make minor modifications to the current nuclear power reactor event reporting requirements. The proposed amendment would reduce the industry's reporting burden and the NRC's response burden in event review and assessment.

List of Approved Spent Fuel Storage Casks: Additions—Part 72

On June 26, 1992 (57 FR 28645), the NRC published an amendment to its regulations that would approve two additional spent fuel storage casks (TN-24 and VSC-24). These casks would be added to the "List of Approved Spent Fuel Storage Casks."

Acquisition Regulation (NRCAR); Organizational Conflicts of Interest—48 CFR Chapter 20

On August 18, 1992 (57 FR 37140), the NRC published an amendment to its proposed Nuclear Regulatory Commission Acquisition Regulation (NRCAR) concerning organizational conflicts of interest. The proposed amendment would modify a section of the conflicts of interest regulations relating to work for others during the period that work is being performed for the NRC.

ADVANCE NOTICES OF PROPOSED RULEMAKING

Acceptability of Plant Performance for Severe Accidents; Scope of Consideration in Safety Regulations—Part 50

On September 28, 1992 (57 FR 44513), the NRC published an advance notice of proposed rulemaking indicating its consideration of an amendment to its regulations which would add provisions for the design of the plant structures to withstand certain challenges from phenomena associated with severe core damage accidents beyond the current "design basis accidents."

Appendix 5

Regulatory Guides—Fiscal Year 1992

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, 1991, to September 30, 1992.

Division 1—Power Reactor Guides

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 28)

1.85 Materials Code Case Acceptability—ASME Section III, Division (Revision 28)

1.101 Emergency Planning and Preparedness for Nuclear Power Reactors (Revision 3)

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 9)

Division 2—Research and Test Reactor Guides

None

Division 3—Fuels and Materials Facilities Guides

3.67 Standard Format and Content for Emergency Plans for Fuel Cycle and Materials Facilities

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

8.7 Instructions for Recording and Reporting Occupational Radiation Exposure Data (Revision 1)

8.25 Air Sampling in the Workplace (Revision 1)

8.33 Quality Management Program

8.34 Monitoring Criteria and Methods To Calculate Occupational Radiation Doses

8.35 Planned Special Exposures

8.36 Radiation Dose to the Embryo/Fetus

Division 9—Antitrust and Financial Review Guides

None

Division 10—General Guides

10.8 Appendix X to Regulatory Guide 10.8, Guidance on Complying with New Part 20 Requirements

DRAFT GUIDES

Division 1

DG–1021 Proposed Revision 3 to Regulatory Guide 1.9, Selection, Design, Qualification, Testing, and Reliability of Emergency Diesel Generator Units Used as Class IE Onsite Electric Power Systems at Nuclear Power Plants

DG–1022 Proposed Revision 3 to Regulatory Guide 1.101, Emergency Planning and Preparedness for Nuclear Power Reactors

Division 8

DG–8004 Radiation Protection Programs for Nuclear Power Plants

DG–8005 Assessing External Radiation Doses from Airborne Radioactive Materials

DG–8006 Control of Access to High and Very High Radiation Areas in Nuclear Power Plants

DG–8007 Proposed Revision 1 to Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data

DG–8008 Planned Special Exposures

DG–8009 Proposed Revision 1 to Regulatory Guide 8.9, Interpretation of Bioassay Measurements

DG–8010 Monitoring Criteria and Methods To Calculate Occupational Radiation Doses

DG–8011 Radiation Dose to the Embryo/Fetus

Division 10

DG–0002 Appendix X to Regulatory Guide 10.8, Guidance on Complying with New Part 20 Requirements

Appendix 6

Civil Penalties And Orders - -Fiscal Year 1992

CIVIL PENALTIES PROPOSED, IMPOSED AND/OR PAID IN FISCAL YEAR 1992
(Listed according to Enforcement Action (EA) numbers)

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Alabama Power Company (Farley) (EA 88-040)	$450,000 proposed in FY88; $450,000 imposed FY90; $150,000 paid in FY92	Violations relating to equipment qualification.
Certified Testing Laboratories, Inc. Bordentown, NJ (EA 89-079)	$8,000 proposed and imposed in FY90; paid in FY92	Falsification of audit reports and providing false information to the NRC.
Tulsa Gamma Ray, Inc. Tulsa, OK (EA 89-223)	$7,500 proposed in FY90; $6,750 imposed in FY90; $4,275 paid in FY92	Breakdown in control of licensed activities.
P.X. Engineering, Inc. Boston, MA (EA 90-065)	$7,500 proposed in FY91; $7,500 imposed and paid FY92	Inaccurate information, failure to adequately supervise.
Lafayette Clinic Detroit, MI (EA 91-017)	$11,500 proposed in FY92, $7,500 imposed in FY92, pending	Discrimination against the Radiation Safety Officer.
Copley Hospital Morrisville, VT (EA 91-031)	$2,500 proposed and paid in FY92	Diagnostic misadministration, breakdown in control of licensed activities.
V.A. Hospital Albany, NY (EA 91-050)	$2,500 proposed and paid in FY92	Falsification of inventory records.
Houston Lighting & Power Company (South Texas) (EA 91-055)	$50,000 proposed and paid in FY92	Falsification of preventive maintenance records for safety related valves.
Chemetron Corporation Newburgh Heights, OH (EA 91-060)	$7,500 proposed in FY91, imposed and paid in FY92	Failure to maintain control of licensed material.
Rutgers University New Brunswick, NJ (EA 91-070)	$6,250 proposed in FY 91, $5,535 imposed and paid in FY92	Breakdown in control of licensed activities.
University of Cincinnati Cincinnati, OH (EA 91-071)	$2,000 proposed and paid in FY92	Incomplete and inaccurate information concerning leak test and inventory cards for sealed sources

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Humana Hospital Greenbrier Valley, WV (EA 91-082)	$21,500 proposed and paid in FY92	Careless disregard for NRC requirements and inaccurate statements.
Univ. of Puerto Rico San Juan, PR (EA 91-089)	$6,250 proposed in FY91, $5,830 imposed and paid in FY92	Breakdown in control of licensed activities.
V.A. Hospital Houston, TX (EA 91-096)	$25,000 proposed, imposed, and paid in FY92	Breakdown in control of licensed activities and repetitive violations.
Duquesne Light Company (Beaver Valley) (EA 91-098)	$25,000 proposed and paid in FY92	Degraded control room habitability and welds not in in-service inspection program.
Alabama Power Company (Farley) (EA 91-102)	$25,000 proposed in FY91; paid in FY92	Startup of Unit 1 with an auxiliary feedwater pump flow path inoperable.
Northeast Nuclear Company (Millstone) (EA 91-107)	$50,000 proposed and paid in FY92	Inadequate corrective Energy action for fouling of service lines with mussels.
University of Missouri Missouri (EA 91-113)	$1,875 proposed and paid in FY92	Improper shipping and Columbia, transfer of licensed material.
Virginia Electric Power Company (Surry) (EA 91-114)	$125,000 proposed and paid in FY92	Emergency diesel generator and charging pump inoperability.
Dag Hammarskjold Cancer Treatment Center Beckley, WV (EA 91-116)	$1,250 proposed, $1,040 imposed and paid in FY92	Breakdown in control of licensed activities.
V.A. Hospital Dallas, TX (EA 91-117)	$6,250 proposed and paid in FY92	Breakdown in control of licensed activities.
Cleveland Electric Illuminating Company (Perry) (EA 91-118)	$100,000 proposed and paid in FY92	Failure to fully verify and validate emergency operating procedures; inadequate or nonexistent procedures for certain situations.
Alt & Witzig Engineering, Inc. Indianapolis, IN (EA 91-119)	$3,700 proposed and paid in FY92	Unauthorized possession and use of licensed material.
Western Atlas International, Inc. Houston, TX (EA 91-121)	$10,000 proposed, imposed, and paid in FY9	Loss of licensed material; failure to block and brace during transport.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Winona Memorial Hosp. Indianapolis, IN (EA 91-124)	$1,250 proposed and paid in FY92	Breakdown in control of licensed activities.
Consumers Power Company (Palisades) (EA 91-125)	$100,000 proposed and paid in FY92	Piping and pipe support design deficiencies during steam generator replacement.
Consumers Power Company (Palisades) (EA 91-126)	$50,000 proposed and paid in FY92	Containment spray pump inoperable due to inadequate startup procedures.
St. Joseph's Community Hospital Paterson, NJ (EA 91-128)	$10,250 proposed and paid in FY92	False statement to NRC; device moved without license amendment.
George S. Wineburgh Association, Ltd. Philadelphia, PA (EA 91-129)	$1,500 proposed and imposed in FY92, civil penalty is being paid over time	Improper disposal of licensed material.
Gulf States Utilities (River Bend) (EA 91-132)	$75,000 proposed and paid in FY92	Inoperable hydrogen mixing system.
Allied Inspection Service, Inc. St. Clair, MI (EA 91-135)	$5,000 proposed in FY 92, civil penalty is being paid over time	Performing radiography without alarm rate meters.
Lancaster General Hospital Lancaster, PA (EA 91-137)	$6,250 proposed and paid in FY92	Lack of management oversight; multiple examples of unsecured materials.
Watertown Memorial Hospital Watertown, WI (EA 91-138)	$6,250 proposed and paid in FY92	Breakdown in control of licensed activities.
Material Testing Laboratories, Inc. Norfolk, VA (EA 91-139)	$12,500 proposed in FY92, pending	Failure to use alarming dosimeters and other miscellaneous violations.
Westinghouse Environmental and Geotechnical Richmond, VA (EA 91-140)	$1,750 proposed and paid in FY92	Lack of management oversight and control; operations conducted at unauthorized site.
Georgia Power Company (Vogtle) (EA 91-141)	$100,000 proposed, imposed, and paid in FY92	Opening valves on reactor makeup water storage tank in violation of technical specifications.

Licensee, Facility and EA Number	Civil Penalties Proposed, Imposed and/or Paid in FY 92	Summary
Carolina Power & Light Company (Robinson) (EA 91-142)	$37,500 proposed and paid in FY92	Inadequate design controls; small break loss of coolant accident and over-temperature/ delta-temperature modifications.
Public Service Company of New Hampshire (Seabrook) (EA 91-144)	$100,000 proposed and paid in FY92	Insufficient radiography weld records; failure to meet ASME requirements.
Ketchikan General Hospital Ketchikan, AK (EA 91-146)	$2,500 proposed, imposed, and paid in FY92	Breakdown in control of licensed activities.
Wisconsin Electric Power Company (Point Beach) (EA 91-149)	$150,000 proposed and paid in FY92	Failure to identify and correct main steam isola tion valve deficiencies resulting in inoperable valves.
Commonwealth Edison (Dresden) (EA 91-152)	$25,000 proposed and paid in FY92	Breakdown in control of Company licensed activities.
Carolina Power & Light Company (Brunswick) (EA 91-158)	$125,000 proposed and paid in FY92	Inadequate work controls for safety related equipment.
Wolf Creek Nuclear Operating Corporation (Wolf Creek) (EA 91-161)	$150,000 proposed and paid in FY92	Failure to take prompt corrective action for safety-related motor operated valve deficiencies and contractor identified program weaknesses.
Overlook Hospital Summit, NJ (EA 91-163)	$3,125 proposed and paid in FY92	Failure to follow procedures; misadministration.
Commonwealth Edison Company (Dresden) (EA 91-164)	$75,000 proposed and paid in FY92	Containment integrity; isolation valve not tested after maintenance; valve left partially open.
Commonwealth Edison Company (Dresden) (EA 91-165)	$112,500 proposed and paid in FY92	Operations program breakdown evidenced by four events with 10 violations involving procedural adherence.
Duke Power Company (Oconee) (EA 91-167)	$125,000 proposed and paid in FY92	Loss of decay heat removal, safety injection overpressure
Monmouth Medical Center Monmouth, NJ (EA 91-174)	$3,125 proposed and paid in FY92	Breakdown in control of licensed activities.
St. Joseph's Community Hospital Paterson, NJ (EA 91-175)	$6,250 proposed and paid in FY92	Therapeutic misadministration not timely reported.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Curwood, Inc. Oshkosh, WI (EA 91–177)	$250 proposed and paid in FY92	Failure to control gauge which resulted in its loss.
Triad Engineering, Inc. Winchester, VA (EA 91–178)	$500 proposed and paid in FY92	Unattended gauge damaged at construction site.
Portland General Company (Trojan) (EA 91–181)	$50,000 proposed and paid in FY92 in fire protection.	Failure to implement adequate corrective actions
Arizona Public Service Company (Palo Verde) (EA 91–182)	$162,500 proposed and paid in FY92	Loss of control of refueling operation and loss of offsite power.
Washington Public Power Supply System (Washington Nuclear 2) (EA 91–183)	$25,000 proposed and paid in FY92	Inoperable hydrogen recombiners.
General Electric Company Wilmington, NC (EA 91–185)	$20,000 proposed and paid in FY92	May 29, 1991, event related to ineffective process and mass limit controls which created the potential for an inadvertent criticality.
Texas Utilities Electric (Comanche Peak) (EA 91–189)	$25,000 proposed and paid in FY92	Improper residual heat Company removal and auxiliary feedwater alignment.
Portland General Electric (Trojan) (EA 91–190)	$50,000 proposed and paid in FY92	Inadequate corrective action involving valve maintenance, radiation protection, and fire protection.
Duke Power Company (Catawba) (EA 91–191)	$15,000 proposed and paid in FY92	Repetitive violations involving configuration control and independent verification problems.
Lone Pine Coal Company Danville, WV (EA 91–192)	$2,375 proposed and paid in FY92	Unauthorized removal of Texas Nuclear gauge by unauthorized person and failure to secure material.
Southern California Edison Company (San Onofre) (EA 91–198)	$50,000 proposed and paid in FY92	Failure to maintain fire protection system operable; failure to provide complete and accurate information in Licensee Event Report.
Philadelphia Electric Company (Peach Bottom) (EA 92–001)	$285,000 proposed and paid in FY92	Automatic depressurization system valves inoperable due to improperly installed insulation; inadequate corrective actions in that condition was not identified by licensee field inspection of other unit.
Florida Power Corporation (Crystal River) (EA 92–002)	$50,000 proposed and paid in FY92	High pressure injection system disabled during an event due to operator error.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Thomas Jefferson Univ. Philadelphia, PA (EA 92-004)	$8,750 proposed and paid in FY92	Failure to secure and control radioactive material which resulted in loss.
Alonso & Carus Iron Works Catano, PR (EA 92-012)	$2,500 proposed and paid in FY92	Failure to perform surveys after radiographic exposures.
Georgetown University Medical Center Washington, DC (EA 92-016)	$3,750 proposed and paid in FY92	Breakdown in control of licensed activities.
Tennessee Valley Authority (Sequoyah) (EAs 92-021 and 92-092)	$150,000 proposed and paid in FY92	Inoperable containment spray suction valves; inaccurate and incomplete information and failure to take corrective action.
Carolina Power & Light Company (Brunswick) (EA 92-024)	$100,000 proposed and paid in FY92 f	Emergency diesel generator inoperable due to immobile uel injection arms after cleaning due to maintenance error.
Shared Medical Technology Rice Lake, WI (EA 92-026)	$2,500 proposed and paid in FY92	Loss of two Technicium—99 packages.
District of Columbia Washington, DC (EA 92-027)	$7,500 proposed, $6,550 imposed and paid in FY92	Breakdown in control of licensed activities.
Hoechst Celanese Corporation Portsmouth, VA (EA 92-032)	$250 proposed and paid in FY92	Removal and reinstallation of Omart gauge by unqualfied personnel.
Power Authority of the State of New York (Fitzpatrick) (EA 92-033)	$500,000 proposed in FY92; pending	Inadequate corrective actions in fire protection program; inadequate design control and post-modification testing of reactor protection system relays; incomplete and inaccurate information.
Power Authority of the State of New York (Indian Point) (EA 92-034)	$225,000 proposed and paid in FY92	Inoperable emergency electrical bus and boric acid heat tracing system.
Hospital de Damas Ponce, PR (EA 92-038)	$3,750 proposed and paid in FY92	Breakdown in control of licensed activities.
Oakland University Rochester, MI (EA 92-042)	$5,000 proposed, imposed, and paid in FY92	Breakdown in control of licensed activities.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Niagara Mohawk Power Corporation (Nine Mile Point) (EA 92-048)	$200,000 proposed and paid in FY92	Loss of ultimate heatsink and inoperable reactor protection system instrumentation channels.
ATEC Associates, Inc. Indianapolis, IN (EA 92-051)	$2,375 proposed and paid in FY92	Breakdown in control of licensed activities; failure to control access to licensed material.
Iowa Electric Light & Power Company (Duane Arnold) (EA 92-056)	$12,500 proposed and paid in FY92	Substantial potential for overexposure during in-service inspection of recirculation system in an area with a dose rate approximately 15 rad/hour.
Taylor Hospital Ridley Park, PA (EA 92-064)	$1,250 proposed and paid in FY92	Unauthorized disposal of fourteen Americium sources.
Northern States Power Company (Prairie Island) (EA 92-067)	$12,500 proposed and paid in FY92	Loss of decay heat removal due to allowing the reactor coolant system water level to drop below the level necessary for continued operation of the inservice residual heat removal pump.
Globe X-Ray Service, Inc. Tulsa, OK (EA 92-068)	$2,500 proposed and paid in FY92	Failure to secure radio-graphic device.
Consumers Power Company (Palisades) (EA 92-074)	$75,000 proposed and paid in FY92 qualification issues.	Failure to take corrective actions on environmental
Sibley Memorial Hospital Washington, DC (EA 92-080)	$2,500 proposed and paid in FY92.	Breakdown in control of licensed activities
Ashford Presbyterian Community Hospital San Juan, PR (EA 92-082)	$3,750 proposed and paid in FY92	Breakdown in nuclear medicine program; management oversight.
Hospital Metropolitano San Juan, PR (EA 92-083)	$2,500 proposed and paid in FY92	Quality management violations resulting in a misadministration.
Duquesne Light Company (Beaver Valley) (EA 92-085)	$75,000 proposed and paid in FY92	Inadequate design and control for vendor-recommended changes to emergency diesel generator sequences resulting in an internal electrical configura tion that was not qualified.
Midwest Industrial X-Ray Fargo, ND (EA 92-091)	$8,000 proposed in FY92, pending	Willful failure to use alarm rate meters.
Virginia Electric Power Company (Surry) (EA 92-093)	$50,000 proposed and paid in FY92	Inadequate configuration control of charging pump power supply alignment.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Baltimore Gas & Electric Company (Calvert Cliffs) (EA 92-095)	$50,000 proposed and paid in FY92	Potential for emergency diesel generators to load in a manner not previously analyzed during certain loss of coolant accidents coincident with a loss of offsite power.
Cardi Corporation Warwick, RI (EA 92-099)	$1,250 proposed and paid in FY92	Failure to file form 241 for work in NRC jurisdiction.
Sequoyah Fuels Corporation Gore, OK (EA 92-100)	$12,500 proposed in FY92, pending	In-plant release of uranium hexafluoride; inadequate response to plant alarms.
Metals Evaluation & Testing, Inc. camera. (EA 92-105)	$7,500 proposed in FY92, pending	Failure to survey, improper dosimetry, incorrect Oakland, CA
Texas Utilities Electric Company (Comanche Peak) (EA 92-107)	$125,000 proposed and paid in FY92	Operator errors resulting in isolation of spent fuel pool cooling system.
WESTEX Company, Inc. Oxnard, CA (EA 92-111)	$7,500 proposed in FY92, pending	Failure to submit NRC Form 241 for work in NRC jurisdiction.
Beth Israel Hospital Passaic, NJ (EA 92-113)	$13,500 proposed and paid in FY92	Misadministration; employee overexposure.
Baystate Medical Center Inc. Springfield, MA (EA 92-114)	$2,000 proposed in FY92, pending	Failure to implement the quality management program which lead to a patient dose administration.
Arizona Public Service Company (Palo Verde) (EA 92-119)	$100,000 proposed and paid in FY92	Inoperable check valve and improper maintenance of reactor trip breakers.
Wisconsin Electric Power Company (Point Beach) (EA 92-120)	$50,000 proposed and paid in FY92	Exceeded reactor coolant system cooldown rate during steam generator crevice cleaning due to inadequate procedures.
Frances Mahon Deaconess Hospital Glasgow, MT (EA 92-121)	$5,625 proposed and paid in FY92	Multiple violations and several repetitive violations indicating a breakdown in control of licensed activities.
University of Michigan Ann Arbor, MI (EA 92-123)	$1,250 proposed and paid in FY92	Fuel movement while reactor was critical.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Western Stress, Inc. Houston, TX (EA 92-125)	$7,500 proposed and paid in FY92	Failure to use alarm rate meters during radiography.
Missouri Dept. of Highways Jefferson City, MO (EA 92-126)	$1,250 proposed and paid in FY92	Damaged moisture density gauge; failure to secure gauge against unauthorized use.
CTI Incorporated Martinez, CA (EA 92-127)	$12,500 proposed in FY 92, pending	Failure to use alarm rate meters and willful failure to post high radiation areas.
Nucletron Corporation Columbia, MD (EA 92-128)	$2,400 proposed and paid in FY92	Careless disregard associated with failure to file NRC Form 241.
Power Authority of the State of New York (Indian Point)	$100,000 proposed in FY92, pending	Service water system violations, including non-ASME code repairs to pipe leaks, (EA 92-134) failure to promptly correct identified deficiencies, or perform adequate safety evaluations for temporary modifications.
Eastern Testing & Inspection, Inc. Thorofare, NJ (EA 92-136)	$7,500 proposed in FY92, pending	Transportation and numerous program violations due to careless disregard for NRC requirements.
Arizona Public Service Inc. (Palo Verde) (EA 92-139)	$130,000 proposed in FY92, pending	Discrimination violations involving a "hostile work environment".
Grinnell Corporation Cranston, RI (EA 92-141)	$25,000 proposed in FY92, pending	Multiple occurrences of failure to file NRC Form 241 and other radiography violations.
Howard Needles Tammen Bergendoff Indianapolis, IN (EA 92-144)	$875 proposed in FY92, pending	Failure to control gauge which caused it to be damaged, breakdown of control of licensed activities.
Consolidated Engineering Laboratory Pleasanton, CA (EA 92-154)	$5,000 proposed in FY92, pending	Failure to use alarm rate meter, inadequate surveillance of high radiation area.
Tennessee Valley Authority (Sequoyah) (EA 92-155)	$62,500 proposed in FY92, pending	Inoperable safety injection pump.
Sequoyah Fuels Corporation Gore, OK (EA 91-067)	Order Modifying License issued October 3, 1991	Demand for Information regarding whether the NRC can have confidence that certain managers will carry out the responsibilities defined in the license; Order confirmed removal of a manager.

Licensee, Facility and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 92*	*Summary*
Patrick K. C. Chun Tulsa, OK (EA 91-104)	Order Modifying License issued November 12, 1991	False statements made to NRC in a license application.
Lafayette Clinic Detroit, MI (EA 91-130)	Order Modifying License issued October 3, 1991, appeal is pending	Order removed individual who discriminated against an employee.
Piping Specialists Kansas City, MO (EA 91-136)	Order Suspending License issued October 17, 1991	False statements and falsification of documents.
Randall C. Orem Dayton, OH (EA 91-154)	Order Revoking License issued November 29, 1991	Misrepresentation on license application.
V.A. Hospital Houston, TX (EA 91-157)	Confirmatory Order issued November 15, 1991	Program breakdown and repetitive violations.
St. Joseph's Community Hospital Paterson, NJ (EA 91-168)	Order Modifying License issued December 3, 1991	False statements to NRC, device moved without license amendment.
Alonso & Carus Iron Works, Inc. Catano, PR (EA 91-171)	Order Modifying License issued December 13, 1991	Order restricted a radiographer from performing work as a radiographer, assistant radiation safety officer, or supervisor of radiography operations.

ORDERS ISSUED IN FISCAL YEAR 1992 (Listed according to Enforcement Action (EA) numbers.)

Licensee, Facility, and EA Number	*Date of Issuance*	*Summary*
Sequoyah Fuels Corporation Gore, OK (EA 91–196)	Confirmatory Order issued on January 13, 1992	Confirmed licensee's decision on personnel action discussed in October 3, 1991, Order, Demand for Information.
St. Joseph's Community Hospital Paterson, NJ (EA 92–013) Safety Committee.	Confirmatory Order Modifying License issued February 10, 1992	Confirmed licensee's commitments with respect to the role of the Chairman of the Radiation
Mayaguez Medical Center Mayaguez, PR (EA 92–039)	Order Modifying License issued April 22, 1992	Multiple violations of radiation safety program.
Sequoyah Fuels Corporation Gore, OK (EA 92–045)	Order Modifying License issued March 13, 1992	Order changed license reporting requirements and Demand for Information regarding confidence that Vice President for Regulatory Affairs will provide complete and accurate information to NRC.
Piping Specialists Kansas City, MO (EA 92–054)	Order Modifying License issued on April 22, 1992	False statements and falsified documents.
American Inspection Company Itasca, IL (EA 92–058)	Order Suspending License issued April 30, 1992	Multiple willful violations including false statements and records, unauthorized use.
Sequoyah Fuels Corporation Gore, OK (EA 92–059)	Order Modifying License issued April 3, 1992	Order modifying license concerning reporting requirements.
Panhandle NDT & Inspection Borger, TX (EA 92–077)	Order Suspending License issued May 18, 1992	Willful failure to file NRC Form 241 while in NRC jurisdiction.
Aircraft Components, Incorporated Branford, CT (EA 92–171)	Order Modifying License issued September 21, 1992	Willful violation of financial assurance requirements, possession of greater than 45.5 kg thorium without surety bond.

Appendix 7

Nuclear Electric Generating Units in Operation or Under Construction

(As of December 31, 1992)

The following is a listing of the 118 nuclear power reactor electrical generating units which were in operation or under construction in the United States as of December 31, 1992, representing a total capacity of approximately 109,000 MWe, of which about 10,000 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 118 reactor units listed, 80 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, 1992. At that time, there were 110 commercial nuclear reactors in the United States with operating licenses and operating; these units had been operating for a cumulative 1,440 reactor-years (an additional 153 reactor-years had been accumulated by reactors now permanently shut down). At the end of 1992, there were eight 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
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 nuclear power plant	836	PWR	OL 1974	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One nuclear power plant	858	PWR	OL 1978	Arkansas Power & Light Co.	1980
CALIFORNIA						
San Clemente	San Onofre Unit 1 nuclear power plant	436	PWR	OL 1967	So. Calif. Ed. & San Diego Gas & Electric Co.	1968
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

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

ILLINOIS (continued)

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
KANSAS						
Burlington nuclear power plant	Wolf Creek	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	810	PWR	OL 1972	Maine Yankee Atomic Power Co.	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

MICHIGAN (continued)						
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
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.
OREGON						
Prescott	Trojan Unit 1 nuclear power plant	1,080	PWR	OL 1975	Portland General Electric Co.	1976

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

TENNESSEE (continued)

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	1988
Glen Rose	Comanche Peak Unit 2 nuclear power plant	1,150	PWR	CP 1974	Texas Utilities	1989
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	North Anna Unit 1 nuclear power plant	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 Indef. Supply System	
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 Indef. Supply System	
WISCONSIN						
Two Creeks	Point Beach Unit 1 nuclear power plant	495	PWR	OL 1970	Wisconsin Electric	1970 Power Co.

WISCONSIN (continued)

Two Creeks	Point Beach Unit 2 nuclear power plant	495	PWR	OL 1971	Wisconsin Electric Power Co.	1972
Kewaunee	Kewaunee nuclearpower 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.
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 1 (Cal.) = pwr, 436, 3/64, 3/67.
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.
Trojan (Ore.) = pwr, 1095, 2/71, 11/75.
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 12/31/92 = 110.

Reactor projects for which construction permits were in effect* as of 12/31/92 (cp date shown):

Bellefonte 1 (Ala.) = pwr, 1235, 12/74.
Bellefonte 2 = pwr, 1235, 12/74.
Comanche Peak 2 (Tex.) = pwr, 1150, 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 12/31/92 = 8.

*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.

INDEX

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