ANNUAL REPORT 1991

United States Nuclear Regulatory Commission

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July 31, 1992

The President
The White House
Washington, DC 20500

Dear Mr. President:

This Annual Report for 1991 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 1991, with additional treatment of events after that period where circumstances warranted.

Respectfully,

Ivan Selin
Chairman

ANNUAL REPORT 1991

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

The *1991 NRC Annual Report*, NUREG–1145, Vol. 8, is available from

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Washington, D.C. 20013-7082

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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.)

1991 Highlights/Licensing and Inspection Summary

Chapter 1

This is the 17th annual report of the U.S. Nuclear Regulatory Commission (NRC), covering events and activities occurring in fiscal year 1991 (the year ending September 30, 1991), with some treatment of events from the last quarter of calendar year 1991.

The NRC came into being 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 or fuel cycle plants, or in medical, industrial or 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 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 1991 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 reports Commission and senior staff changes taking place during the report period and provides a summary of licensing and inspection activity, as well as the status of agency consolidation.

Changes Within Commission and Senior Staff

On July 2, 1991, Dr. Ivan Selin was sworn in as the new Chairman of the NRC, succeeding Kenneth M. Carr, whose term expired at the end of June. Prior to his appointment and confirmation, Chairman Selin held a number of major positions in both the private and public spheres. He has served as Chairman of the Military Economic Advisory Panel to the Director of Central Intelligence (1978 to 1989); as a member (1979 to 1989) and as Chairman (1988 to 1989) of the United Nations Association of the United States of America; as a member of the Advisory Board on the U.S.S.R. and Eastern Europe, at the National Academy of Sciences (1986 to 1988); and as a member of the Council on Foreign Relations (1979 to 1989). Dr. Selin became Under Secretary of State for Management in May 1989, serving as the principal advisor to the Secretary of State on all matters involving the allocation of State Department resources, in support of the President's foreign policy objectives.

After the close of the report period, on December 16, 1991, E. Gail de Planque was sworn in as an NRC Commissioner, filling the vacancy created when Commissioner Thomas Roberts completed his second term on the Commission, on June 30, 1990, and bringing the Commission to its full complement of five. Commissioner de Planque was most recently Director of the Environmental Measurements Laboratory, operated by the Department of Energy in New York City.

Power Reactor Licensing in Fiscal Year 1991

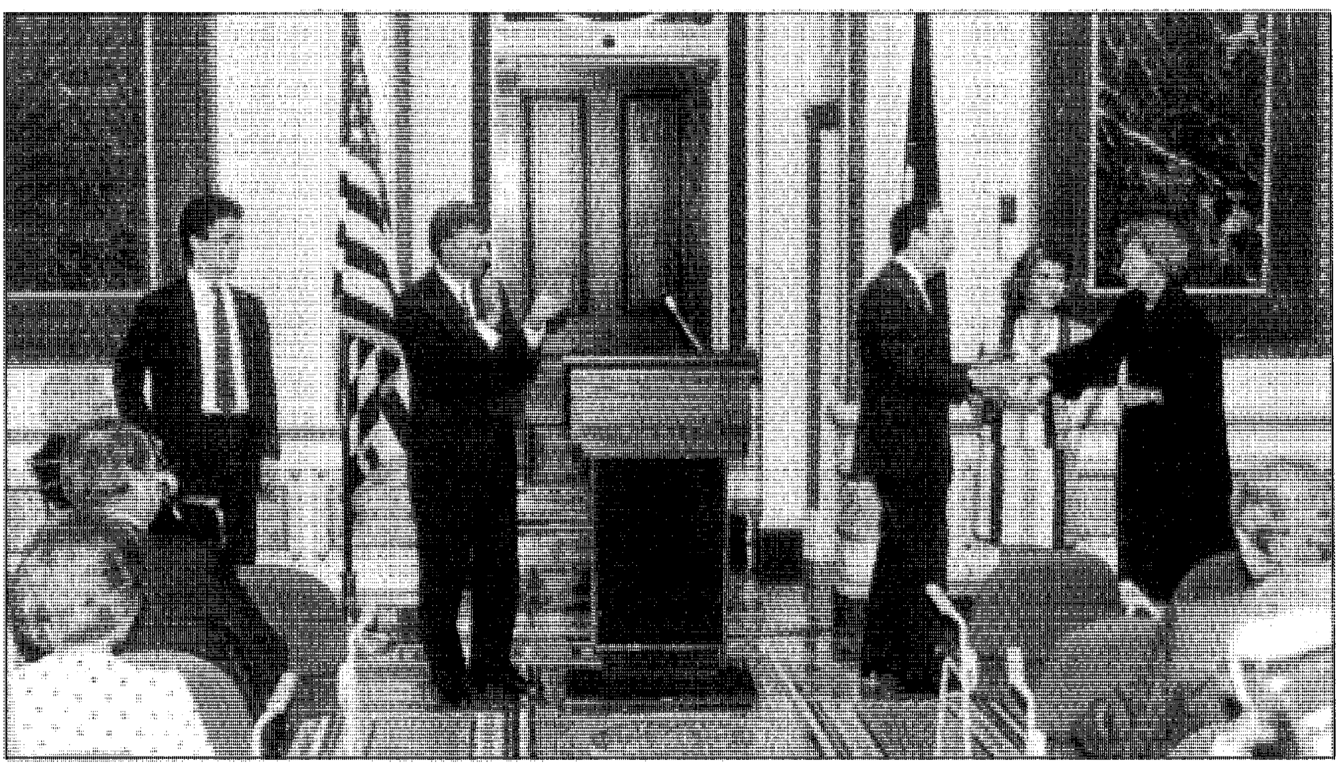
During the fiscal year, the NRC issued no new operating licenses. The status of one plant (Shoreham (N.Y.)) was changed from indefinitely shut down to permanently shut down. That action brings the number of reactors licensed to operate at full power in the United States to 112, excluding plants licensed but permanently shut down, as of September 30, 1991. (Included in the 112 facilities is one unit that is indefinitely shut down and defueled; this unit would require NRC approval to reload fuel.) The construction permit for one plant was cancelled at the request of the applicant, resulting in a total of eight plants, as of September 30, 1991, for which construction permits have been issued. Most of these are projects which have been halted and/or deferred. (See Appendix 7.) There were no new applications for operating licenses or construction permits during the period, and no construction permits or manufacturing licenses were issued. (See Chapter 2.)

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 1991, the Office of Nuclear Reactor Regulation (NRR) completed about 1,568 licensing actions. About 84 percent of these actions were plant-specific and predominantly licensee-initiated. The balance were multi-plant actions deriving from the imposition of NRC requirements. The total licensing action inventory has decreased from about 1,570 to 1,411 licensing actions under review. (See Chapter 2.)

Implementation Status of Safety Issues At Nuclear Power Plants

Annually, the NRC publishes a document presenting the status of implementation and verification of safety issues in major NRC requirement areas. Volume 1 of this document, which addresses the status of implementation and verification of Three Mile Island (TMI) Action Plan Requirements, was published in March 1991. Volume 2, giving the status of implementation and verification of unresolved safety issues (USIs), was published in May 1991. Volume 3, describing the status of implementation and verification of generic safety issues (GSIs) was published in June 1991. These reports constituted the basis for the combined annual report update due to the Commission in the last quarter of calendar 1991. As reported in Volume 1, approximately 99 percent of the TMI Action Plan items have been implemented at the licensed nuclear power plants. Of the 13,527 applicable items, 13,404 have been completed or closed, and only 123 remain open. About 50 percent of the remaining 123 open items



On July 2, 1991, Ivan Selin was sworn in as the eighth Chairman of the U.S. Nuclear Regulatory Commission, succeeding Chairman Kenneth M. Carr, whose term expired on June 30. The swearing-in ceremony, shown above, was carried out in the Indian Treaty Room of the Old Executive Office Building, adjacent to the White House. Supreme Court Justice Sandra Day O'Connor administered the oath of office to Dr. Selin, at the right, with Mrs. Selin looking on. On the left are the Chairman and Mrs. Selin's son, Douglas, and Vice President J. Danforth Quayle. Chairman Selin, who holds doctoral degrees from Yale University and the University of Paris, comes to the NRC from the U.S. Department of State, where he had served as Under Secretary of State for Management since May of 1989. In previous years, Dr. Selin served in the Office of the Assistant Secretary of Defense (from 1965–to–1970); founded American Management Systems, Inc. (1970), a computer services and consulting firm he headed for 19 years; was Chairman of the Military Economic Advisory Panel to the Director of Central Intelligence (1978–to–1989); was a member of the Advisory Board on the U.S.S.R. and Eastern Europe at the National Academy of Sciences (1986–to–1988); and served on the Council on Foreign Relations (1979–to–1989).

Dr. E. Gail de Planque became a member of the Nuclear Regulatory Commission, on December 16, 1991, filling the vacancy created when Commissioner Thomas Roberts completed his second term, on June 30, 1990, and bringing the Commission to its full complement of five. At right, the newly sworn Commissioner receives congratulations; from left-to-right are Dr. de Planque; the Commissioner's father, Martin W. de Planque; Chairman Ivan Selin; and Commissioner Forrest J. Remick. Commissioner de Planque previously served as Director of the Department of Energy's Environmental Measurements Laboratory, in New York City, where she had earlier served as a physicist in the Radiation Physics Division. Dr. de Planque holds degrees from Immaculata College and the New Jersey Institute of Technology, and took her Ph.D. in Environmental Health Sciences from New York University. She is a past president of the American Nuclear Society and a charter signatory to the International Nuclear Societies Council, established in 1990.

are projected to be implemented by the end of calendar year 1992.

Progress continues to be made in reducing the number of USI items yet to be implemented. Approximately 85 percent of the USI items have been implemented at licensed reactor plants. Of the 1,819 applicable items, 1,545 have been completed and 274 remain open. Progress also continues in reducing the number of unimplemented GSI items. Approximately 86 percent of the applicable items associated with GSIs have been implemented at licensed nuclear power plants.

Safety Performance Improving At Licensed Operations

According to a 1990 survey and report by the NRC Office for Analysis and Evaluation of Operational Data (AEOD), the safety performance of power reactor operations in the United States shows a continuing trend of overall improvement. The conclusion is based on analysis of a set of "performance indicators," on reported "abnormal occurrences," and on "accident sequence precursors." (See Chapter 3.)

AEOD also reported on the approximately 7,800 non-reactor licensees authorized by the NRC to possess and use radioactive materials the majority of them for applications such as radiography, gauges and well logging. (About 2,400 licensees are authorized to administer radioactive materials or radiation from those materials to individuals for medical diagnosis or therapy.)

The dominant health concern associated with these uses of NRC-licensed radioactive materials is the possible damage that can occur from overexposure to radiation. In this regard, for 1990, there were (1) 24 non-reactor events reported to the NRC, in which 30 licensee individuals received exposures that were greater than those permitted by NEC regulations (compared to 28 events and 40 licensee employees in 1989); and (2) 467 medical misadministrations (443 diagnostic treatments and 24 therapy administrations)—about three times the average number of therapy misadministrations and an increase of about 10 percent in the average number of diagnostic misadministrations reported in the prior nine years. However, since the NRC staff estimates there are about seven million diagnostic and 180,000 therapy procedures performed every year in this country (about 40 percent by NRC licensees and the remainder by Agreement State licensees), the error rate for all types of misadministrations remained very low.

Renewal of Operating Licenses

The first operating license of a currently active plant will expire in the year 2000, and more than 50 percent of all currently operating plants will expire by 2013. Because some of the licensees for these plants may submit an application to renew their operating licenses, the NRC has placed a high priority on defining the requirements that will have to be met by a utility 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 rigorous and systematic review of systems, structures and components in the 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 issued a rule (10 CFR Part 52) addressing the matter. The focus of the rule is design certification, a regulatory instrument that would permit the early resolution of many licensing issues. Areas 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 was reviewing safety analysis reports, and a number of other documents pertaining to standardized designs, at the close of the report period.

Power Plant Maintenance

The proper maintenance of equipment is essential to nuclear power plant safety. After extensive reviews of various utility maintenance programs, the NRC issued a rule, on July 10, 1991, requiring licensees to monitor the performance or condition of certain systems, structures and components to ensure their ability to perform their intended functions. At the end of the report period, regulatory guidance was in preparation to describe acceptable methods for implementing the rule and inspection procedures were being developed by which to verify satisfactory implementation.

Special Reactor Plant Inspections

During fiscal year 1991, NRC headquarters and regional staffs performed 49 on-site special team inspections, each involving 8-to-10 inspectors and requiring 2-to-4 weeks to complete. The objective of the special inspections is to determine whether, when called upon to do so in an emergency, the nuclear plant systems and personnel will perform their safety functions as set forth in the facility's Safety Analysis Report.

A new type of special team inspection—the Electrical Distribution System Functional Inspection—was developed in 1990. After testing at five plants and subsequent further development, it was decided that the inspection should be conducted at every plant in the country. As of the end of fiscal year 1991, the inspection had been carried out at 24 plant sites (of a total 74); by current planning, this special inspection will have been conducted at all sites by early 1993.

During fiscal year 1991, the staff completed its long term program of improving Emergency Operating Procedures (EOPs). The staff had begun an accelerated inspection of EOPs in fiscal year 1988, in order to determine whether licensees' EOPs were technically correct; could in fact be carried out, given considerations of locale, accessibility of resources, and relevant physical factors, during an emergency; and could be carried out by available personnel with the requisite knowledge and ability. The EOP inspection has been completed at all operating plants. Improvements were noted from inspection results from 1989 through 1991, but some problems persist. EOP follow-up inspections will continue.

A total of 33 vendor inspections were conducted during the report period by the the Office of Nuclear Reactor Regulation (NRR), plus several others involving NRR support to activities of the NRC Office of Investigations (see Chapter 3). Several inspections grew out of allegations of falsified records, defective materials, and suspect piping. The program also includes inspection of foreign vendors who supply components for use in U.S. nuclear power plants, such as steam generators, pipe supports, hydraulic snubbers, etc. (See Chapter 2.)

Nuclear Materials Licensing/Inspection

As of the end of fiscal year 1991, the NRC was administering about 7,800 licenses for the possession and use of nuclear materials for medical and industrial applications. The 29 Agreement States administer an additional 16,000 licenses. NRC regional staff completed approximately 3,000 inspections of material facilities during the report period (the NRC Regional Offices administer all materials licenses, with the exception of certain exempt distribution licenses and "sealed source" and "device design" evaluations, which are handled at NRC Headquarters).

Licensing actions related to nuclear materials at fuel cycle plants and facilities came to 73 during the fiscal year, and there approximately 5,600 licensing actions on applications for new byproduct materials licenses (about 500), amendments (about 3,900), and renewals (about 900) of existing licenses. Headquarters staff carried out about 300 sealed source and device reviews.

The staff continued its operational safety team assessments at major fuel cycle and materials facilities during the period. There have been about 40 such assessments, since the program began in 1986, by teams made up of representatives from NRC Regional Offices, NRC Headquarters, and other Federal participation, such as personnel from the Occupational Safety and Health Administration and the Environmental Protection Agency. In fiscal year 1991, the NRC conducted operational team assessments at two fuel facilities, and expanded inspections at another six sites. (See chapter 4.)

Table 1. License and Annual Fee Collections FY 1991

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR Part 170	$75.5 million	$7.6 million	$83.1 million
10 CFR Part 171	$321.1 million	$34.4 million	$355.5 million
Total Fees	$396.6 million	$42.0 million	$438.6 million

Safeguards Inspections

During fiscal year 1991, a total of 170 inspections of safeguards at all of the nation's nuclear plant sites were performed by NRC regional staff. ("Safeguards," in this context, refers to measures taken to deter, prevent or respond to the theft or diversion of nuclear material or sabotage of nuclear facilities.) Headquarters and regional staff appraised and approved about 228 revisions to licensee security, contingency and guard–training plans at reactor plants.

Also during the report period, the NRC staff, in conjunction with U.S. Army Special Forces personnel, completed the Regulatory Effectiveness Review program, in May 1991. This was an assessment of reactor safeguards regulations and of the practical effectiveness of licensee's safeguards programs for the protection of vital equipment at reactor plants. Beginning in 1981, the program has now reached every operating reactor site; in general, the reviews tend to confirm the soundness of safeguards regulations, and they have contributed to the realization of over 500 significant improvements in safeguards protections. Interdisciplinary team reviews will continue, focusing on licensees' contingency response capability and the interaction between operations and security staffs.

The NRC conducted 29 safeguards inspections of non-power reactor facilities during the report period, as efforts continue to convert non-power reactors from the use of high-enriched uranium to low-enriched uranium fuel.

Comprehensive physical security and material control and accounting inspections were performed at the nine major fuel fabrication facilities, during fiscal year 1991. Special teams also inspected the new physical security upgrades at two facilities possessing "formula quantities" of strategic special nuclear materials. The NRC also continued its safeguards inspections of selected shipments of spent fuel during the period and its transportation–related safety inspection program. The latter effort entailed more than 1,400 individual inspections, covering byproduct, source and special nuclear materials licensees, and including fuel cycle facilities and shippers of spent reactor fuel. (See Chapter 5.)

NRC License and Annual Fees

The Omnibus Budget Reconciliation Act of 1990 (Public Law 101–508) requires that, in fiscal year 1991, 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 1991, a total of $465 million was appropriated to the NRC (Public Law 101–514), of which $19,650,000 was derived from the Nuclear Waste Fund. Of the remaining $445,350,000, approximately 98 percent, or $438,610,118, was collected through license fees and annual charges. The net amount appropriated to the NRC in fiscal year 1991 was $6,739,882. Table 1 shows the amounts collected through license and annual fees in fiscal year 1991.

Consolidation of NRC Headquarters

During the first half of fiscal year 1991, the Government and the developer agreed on lease terms and conditions for the construction of the second building of the two-building NRC headquarters complex, in Rockville, Md. Agreement also was reached among the parties concerning Montgomery County (Md.) restrictions on the

The consolidation of NRC staff into a single headquarters venue, delayed for various reasons during 1990 and 1991, is now scheduled for early 1994, with the completion of a second office building, Two White Flint North, at 11545 Rockville Pike in North Bethesda, Md. One White Flint North, the building at left in the artist's depiction shown here, has been fully occupied since 1988 and houses the Commission and most of the headquarters staff.

site plan and traffic management. At the close of the report period, the first stages of site clearing and excavation for the second building had begun. The first building was purchased in 1986 and occupied in 1988.

The ten-storey, 364,000-square-foot, second building will be constructed over a 27-month period, beginning in mid–September 1991, with occupancy scheduled in early calendar year 1994. Designated Two White Flint North, the building will house the Offices of Nuclear Material Safety and Safeguards, Nuclear Regulatory Research, Analysis and Evaluation of Operational Data, Licensing Support System Administrator, Controller, Administration, Information Resources Management, Personnel, Small and Disadvantaged Business Utilization and Civil Rights, Inspector General, the Advisory Committees on Reactor Safeguards and Nuclear Waste, and the Atomic Safety and Licensing Board Panel.

Nuclear Reactor Regulation

Chapter 2

The Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC) is responsible for regulating the safe operation of the nation's operating nuclear reactors and for evaluating all applications to construct and operate new reactors. The latter include both nuclear power reactors operated by the electric utilities and non-power research reactors, such as those operated by the 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 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 these facilities. NRR also provides direction to, and oversight of, 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 throughout a facility's eventual operating lifetime—leading to issuance of permits or licenses, and licensing actions taken thereafter. (See "Improving the Licensing Process," 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 112 nuclear power plants licensed for operation in the United States, as of the close of fiscal year 1991. 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 1991 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.

Regulatory Impact Survey

In the fall of 1989, the NRC staff initiated a regulatory impact survey, consisting of three separate and distinct inquiries, to ascertain the views of the regulated utilities as to the effect of the large number of NRC regulatory initiatives and requirements imposed in the wake of the 1979 accident at Three Mile Island Unit 2 (Pa.). Following a similar survey in 1981, the NRC made a number of changes in its organization and regulatory practices.

The surveys were performed to gain an understanding of the perceptions of industry, and of the regulatory staff as well, regarding the effect of the NRC's current activity in assuring the safe operation of nuclear power plants, and also to determine whether modifications are called for in NRC regulatory programs.

In the fall of 1989, teams of senior NRC managers conducted the first survey on the particular effects of NRC regulatory activity at 13 specific facilities located at a variety of sites throughout the country. The staff compiled and summarized the results of the discussions in a draft document, "Industry Perceptions of the Impact of the U.S. Nuclear Regulatory Commission on Nuclear Power Plant Activities" (draft NUREG–1395).

LICENSING THE NUCLEAR POWER PLANT

The first step in the nuclear power plant licensing process is the submission by a utility of an application to the NRC 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 1990

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

A second phase of the survey was instituted when the staff issued Generic Letter No. 90–01, "Request for Voluntary Participation in NRC Regulatory Impact Survey." The Generic Letter asked licensees to give estimates of the time spent at six levels of management in responding to 11 different kinds of inspections and audits, including those conducted by the utility itself, by the NRC, and by other outside organizations.

In the third part of the survey, begun in January 1990, teams of staff members from NRR conducted interviews of regional and headquarters staff. The interviews covered a broad spectrum of the staff—ranging from engineers to middle managers, Associate Directors, and Regional Administrators. The interviews were designed to elicit NRC staff perceptions of the impact of NRC licensing and inspection activities on licensees' ability to safely operate licensed facilities. Evaluation of the results, along with the surveys of licensees, was forwarded to the Commission, with proposed actions, in SECY–90–347, "Regulatory Impact Survey Report."

The three studies comprising the survey were completed in 1990. The staff completed its evaluation of the survey and reported resultant improvements to the Commission in SECY–91–172, "Regulatory Impact Survey Report—Final." NUREG–1395 will be updated by the addition of this Commission Paper and will be issued during fiscal year 1992.

Reactor Engineer Intern Program

The Reactor Engineer Intern Program (formerly the Technical Intern Program) was initiated in 1988 to recruit and train new talent to meet the agency's future work force requirements. Approximately half of the 45 interns now in the program are based in Headquarters and half among the Regions.

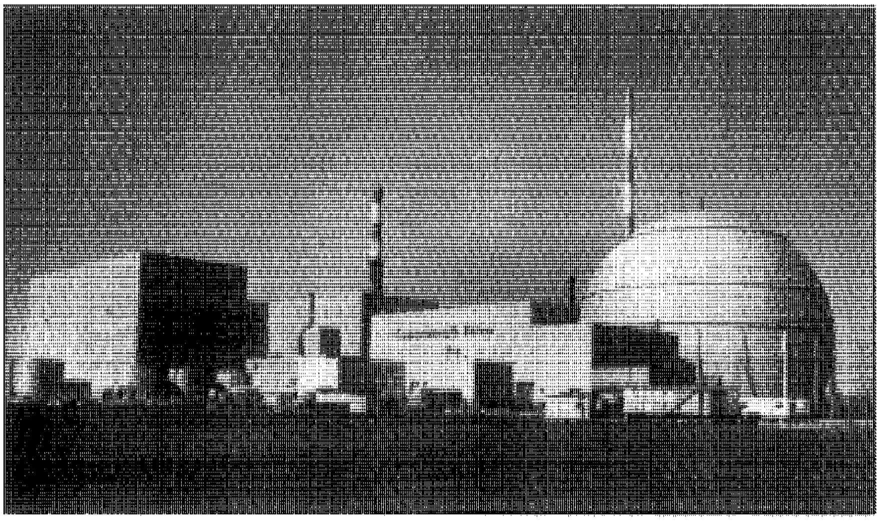
Interns spend from 2-to-3 years in a series of developmental assignments at Headquarters, Regional Offices and plant sites. Interns are also expected to complete at least 17 weeks of formal training in reactor technology, in addition to extensive reading assignments and other training courses. Monthly seminars and developmental trips are arranged to give interns a broad understanding of the NRC and its regulatory mission.

At the conclusion of the program, some interns become regional or resident inspectors, and some are placed in permanent positions in NRR. Nine interns have been graduated from the program since its inception.

STATUS OF LICENSING

License Applications and Issuances

During fiscal year 1991, the NRC issued no new operating licenses. The status of one plant (Shoreham (N.Y.)) was changed from "indefinitely" to "permanently" shut down. This brings the number of reactors licensed to operate at full power in the United States to 112—excluding several plants still licensed to operate but permanently shut down—as of September 30, 1991. (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 permits during the report period, and no construction permits or manufacturing licenses were issued. The construction permit for one plant (Grand Gulf Unit 2 (Miss.)) was cancelled at the request of the applicant. As a result, as of the close of fiscal year 1991, there were eight nuclear plants still technically under construction in the United States, although some of them are delayed indefinitely.



Over 20 years ago, the Atomic Energy Commission ended the option of issuing provisional operating licenses to nuclear plants (to test certain operating parameters against actual performance). But nothing was decided at the time regarding licensees then holding provisional licenses, each of which filed for renewal of their licenses within 30 days of expiration, thus qualifying for indefinite operation until issuance of full term licenses. One of the four plants still operating under provisional licenses, the Dresden (Ill.) facility, is shown here. All four were granted full term licenses during fiscal year 1991.

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 activity affecting reactor operations includes 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 regulations requiring fiscal year 1991, NRR completed about 1,568 licensing actions. About 84 percent of these were plant-specific and predominantly licensee-initiated. The balance were multi-plant actions that result from NRC-imposed requirements. The total licensing action inventory has decreased from about 1,570 to 1,411 licensing actions under review.

Conversion of Provisional Operating Licenses to Full-Term Licenses

From 1959 to 1971, the Atomic Energy Commission (AEC) issued Provisional Operating Licenses to 15 power reactor licensees for periods up to 18 months, as an intermediate measure prior to issuance of a full-term operating license. The provisional license was issued to provide for an interim period of routine operation during which the licensee and AEC staff could gauge actual plant operating parameters and assess performance against predicted values, as well as resolve generic concerns identified during the licensing process. In March 1970, a rule change went into effect which ended the option of issuing provisional operating licenses, but no provision was made in the rule for converting previously issued provisional licenses. Pursuant to 10 CFR 2.109, the provisional license would not be deemed to have expired so long as the licensee filed an application for renewal at least 30 days before the expiration date. Since each of the provisional licensees did in fact submit a timely action for full-term license, these provisional licenses could continue indefinitely until the Commission completed its licensing action. Notwithstanding the silence of regulations on a conversion process, the NRC policy has been to proceed with the provisional license conversion reviews. During the fiscal year, all four of the plants still operating under provisional licenses were converted to full-term operating licenses. Dresden Unit 2 (Ill.), Palisades (Mich.), Oyster Creek (N.J.) and San Onofre Unit 1 (Cal.) were issued full-term licenses on February 20, February 21, July 2 and September 26, 1991, respectively.

Special Cases

Calvert Cliffs. In December 1988, the Calvert Cliffs (Md.) nuclear power plant was placed on the NRC's list of plants calling for close monitoring because of regulatory concerns about declining performance. Consequently, the licensee was asked to submit its comprehensive plan to address the identified performance problems.

In March 1989, Unit 2 at the facility was shut down for routine refueling. Following shutdown, leaks were discovered in the Unit 2 pressurizer heater sleeve welds. Unit 1 was shut down in May 1989 for inspection to ensure that similar leaks did not exist in that unit's pressurizer. The licensee agreed not to restart either unit until the pressurizer leakage problem was fully understood and resolved, with better controls over work activity and procedural compliance in place. The licensee, Baltimore Gas and Electric Company, implemented both short and long term corrective action, including organizational and management changes. The licensee also developed a long term Performance Improvement Plan to correct identified deficiencies in operations, maintenance and other processes. Subsequently, in June 1989, the NRC set up a

Calvert Cliffs Assessment Panel to evaluate the plan and its implementation and to decide whether it adequately addressed the identified performance issues.

An NRC assessment team conducting inspections in late 1989 found improved performance levels in most areas, although several deficiencies remained. In April 1990, the NRC staff concluded that the licensee had implemented the short term corrective action necessary to address the identified problems and to warrant startup and operation of Unit 1. On April 13, 1990, the licensee started the Unit 1 reactor and operated at power until April 23, 1990, when the reactor was shut down for a planned maintenance outage. The short initial operational period was planned in advance to permit the licensee to confirm the adequacy of changes to both hardware and to the management process that had been implemented during the prolonged outage. The NRC conducted a special startup inspection to assess the licensee's performance during plant operations. The inspection included continuous 24-hour coverage during significant plant operations. In general, the NRC determined that the licensee performed satisfactorily in most areas during this period. Following the maintenance outage in September 1990, Unit 1 was returned to operation.

Unit 2 remained in an extended maintenance and refueling outage. Restart of Unit 2 required approval of the Regional Administrator, Region I, based on the NRC staff's evaluation of the physical readiness of Unit 2 and its assessment of the licensee's ability to safely operate both units. In February of 1991, an NRC team conducted inspections and concluded that Unit 2 was physically ready for restart and sufficient controls were in place to support safe simultaneous operation of both units. In early April of 1991, the Regional Administrator released Unit 2 for restart and the Unit 2 reactor was restarted in the latter part of the month. Inspections similar to those conducted prior to restart of Unit 1 were carried out and the Unit 2 startup was deemed generally satisfactory. The Unit 2 reactor was shut down on October 18, 1991, for a scheduled 32-day surveillance and maintenance outage.

The NRC assessment panel continued its efforts to ensure that the implementation of the improvement plan was effective in the near term and that procedures were in place to assure its effectiveness in the future. An NRC team inspection was scheduled for December of 1991 to assess the licensee's overall performance with respect to control of those operations and activities necessary to assure the continued safe operation of both units.

Nine Mile Point Unit 1. The Nine Mile Point Unit 1 (N.Y.) nuclear power plant is a boiling water reactor owned and operated by the Niagara Mohawk Power Corporation, the licensee. The plant was shut down by the utility on December 19, 1987, following the failure of a feedwater system flow control valve. The outage was extended in order to deal with problems uncovered in the inservice inspection program and also to refuel the reactor. Shortly thereafter, deficiencies in the program for the maintenance of operator licenses led to the NRC's issuance, on March 28, 1988, of Confirmatory Action Letter 88–13, confirming the licensee's agreement that corrective action for the deficiencies in the licensed operator retraining and in the continued training program would be completed prior to further operation of the unit.

During this period, deficiencies in other programs were found by the licensee and by NRC staff. Problems were discovered in, though not limited to, the feedwater system, the inservice inspection program, the maintenance of operator licenses, control of commercial grade items, fire barrier penetrations, and operator knowledge of emergency operating procedures. In June 1988, Nine Mile Point Unit 1 was categorized by NRC senior management as a facility requiring close monitoring from both NRC Headquarters and the Regional Office. Because of the many problems identified, the NRC staff issued Confirmatory Action Letter (CAL) No. 88–17 on July 24, 1988. The letter dealt with issues addressed by the earlier letter and documented the licensee's agreement not to restart Unit 1 until certain corrective action had been carried out. The agreement required the licensee to determine why the problems had not been recognized and remedied earlier, to prepare a restart action plan identifying actions needed to address the identified causes, and to provide a report substantiating the readiness of the unit for restart.

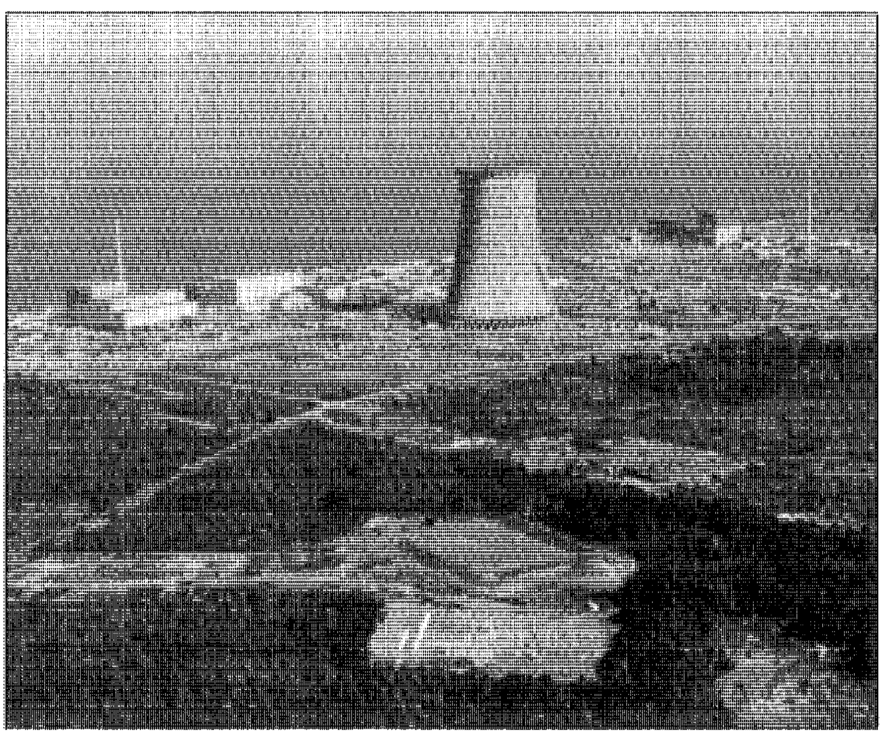
The utility's Restart Action Plan was submitted on December 21, 1988. After several revisions, in response to the NRC staff review, it was approved by the NRC staff on September 28, 1989. A Restart Readiness Report was submitted on September 8, 1989, indicating that, pending completion of certain specified items, the unit could be restarted. Following extensive further inspection and review of the licensee's activity by the NRC staff, the staff issued Supplement No. 1 to CAL 88–17, on July 27, 1990. In the supplement, the staff indicated that—based on its review of licensee's actions to resolve its problems over the prior two-and-one-half years—the NRC staff had concluded that the facility, its management and its staff were ready to restart the plant. The letter also documented the NRC staff's understanding that the utility would conduct its own assessments of operations throughout the power ascension program and would review the results with the NRC staff prior to proceeding with successive phases of the power ascension program.

Nine Mile Point Unit 1 resumed operation on July 29, 1990, and the licensee proceeded successfully with the power ascension program. On February 11, 1991, the NRC staff closed out Supplement 1 to CAL 88–17 by letter to the licensee. In the NRC staff's Systematic Assessment of Licensee Performance (SALP) Report for the period March 1, 1990, through March 31, 1991, substantially improved overall performance was noted. The improvement was attributed to increased corporate and plant management involvement in day-to-day activity, broad implementation and employee participation in self-assessment activity, and greater adherence to procedures and attention to detail. The NRC staff found overall performance to be good for the period subsequent to the SALP report. Licensee management continued to show significant involvement in day-to-day activity, reinforcing the need for personnel accountability and performance improvement. On June 18, 1991, the licensee was informed by the NRC that Nine Mile Point Unit 1 had demonstrated sustained improvement sufficient to warrant removal from the category of plants that require increased attention from both NRC Headquarters and the Regional Office.

Seabrook. Having attained commercial operation in August 1990, the Seabrook (N.H.) nuclear power plant continued operation through its first full refueling cycle; it was shut down for refueling on July 25, 1991. In the period of operation prior to refueling, the plant operated well, experiencing no major outages and a high operating capacity factor of about 85 percent. During 1991, the licensee undertook and completed a records reverification program for all field welds made by the Pullman-Higgins Company during initial construction in the early 1980s. A number of weld radiographs were re-shot, but no physical weld deficiences were found. The NRC staff inspected the results of this effort and concluded that the licensee's records now meet the ASME Code, as required.

Yankee-Rowe. The Yankee-Rowe nuclear power plant, located in Rowe, Mass., is a 185-megawatt (electric), four-loop, pressurized water reactor (PWR) owned and operated by Yankee Atomic Electric Company. The plant is a one-of-a-kind design by the Westinghouse Electric Company. Licensed for operation in 1960, it was the first commercial nuclear electric generating facility in the United States. The operating license for the plant was amended in 1988 and will expire in the year 2000. The plant had been the lead PWR in the license renewal (extension) program, but its involvement in that effort has been discontinued.

The NRC staff and the licensee have been engaged in a major effort to study and appraise reactor vessel embrittlement at Yankee-Rowe. In August 1990, the NRC staff issued a safety evaluation report (SER) which concluded that the probability of reactor vessel failure from a pressurized thermal shock (PTS) event at the facility was acceptably low, even though vessel conditions exceeded the screening criteria set out in 10 CFR 50.61. In June 1991, the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution filed a 10 CFR 2.206 petition challenging the staff's conclusion and calling for an immediate shutdown of the Yankee-Rowe plant.



Going back to 1988, the Nine Mile Point Unit 1 (N.Y.) nuclear power plant came under close monitoring, for various reasons, by the NRC. In June 1991, the licensee, Niagara Mohawk Power Corporation, was notified that the NRC considered improvements in plant operations and performance sufficient to warrant removal of it from the category of plants requiring intensified attention. The two-reactor Nine Mile Point facility is located on the shores of Lake Ontario in upstate New York.

On July 31, 1991, the Commission issued a memorandum and order responding to the petition. In the order, the Commission required Yankee-Rowe management to evaluate possible modifications to operating conditions which would reduce the likelihood of vessel failure from a PTS event by a factor of 5-to-10. The proposed modifications and supporting analyses were submitted on August 26, 1991. From its review of the submittal, the NRC staff was unable to conclude that the required reduction in vessel failure could probably be met. Subsequently, on September 30, 1991, the staff recommended to the Commission that the plant be shut down until the NRC could satisfy itself that the reactor vessel provides adequate margins against PTS events. The NRC staff based its recommendation on the following considerations: (1) the pressure-temperature profiles identified from a re-analysis of the small break loss-of-coolant accident—following the licensee's changes in the thermal-hydraulics models—substantially reduced the staff's confidence that the previous calculations of the likelihood of vessel failure were conservative; and (2) the fact that the vessel plates contribute significantly to the vessel failure probability reduced the staff's confidence that conservative bounding values had been used throughout the analyses. The licensee elected to voluntarily shut down the plant on October 1, 1991, in response to NRC concerns. The licensee affirmed, in a letter dated October 25, 1991, that it would not seek restart of the plant prior to an April 1992 outage.

During the April 1992 outage, the licensee plans to implement its weld material sample removal and vessel inspection program to obtain information which could reduce uncertainties with the reactor vessel material characteristics. Meanwhile, plant management was to undertake accelerated irradiation testing at the University of Michigan, as part of the program to determine the embrittlement effects of the plates. The licensee's overall program will be evaluated by the NRC staff before any decision as to whether a return to power operations should be authorized.

Comanche Peak Unit 2. Texas Utilities (TU) Electric, the licensee for Comanche Peak Unit 1 (Tex.), which was licensed for operation in fiscal year 1990, is actively involved in the completion of the second unit at this site. TU Electric initiated engineering activity in June 1990 and resumed construction activity in January 1991 for Comanche Peak Unit 2. This unit was more than 90 percent complete at the end of fiscal year 1991 and is scheduled for completion by December 1992. The licensee has incorporated into the Unit 2 design and construction effort lessons learned from the Unit 1 effort, including the experiences derived from an extensive corrective action program to identify and correct design deficiencies. Hot functional testing of the plant's systems is scheduled to begin in June 1992 and the licensee expects to be ready to load fuel by December 1, 1992.

Zion. The Zion (Ill.) nuclear power plant, which is owned and operated by the Commonwealth Edison Company, comprises two four-loop Westinghouse 1,040 megawatts (electric) pressurized water reactors. In January 1991, the Zion plant was added to the list of plants that are authorized to operate but warrant increased NRC attention because of regulatory concerns about declining performance.

The NRC is closely monitoring the licensee's corrective action programs and efforts to improve performance at Zion. In addition to the increased inspection by the resident and region-based inspectors, a Zion Review Team, consisting of Headquarters and Region III management and staff, periodically evaluates Zion's performance. The Review Team provides recommendations to senior NRC officials regarding future regulatory activity at the site. After several visits in 1991, the team determined that the utility has made decisions and committed resources in ways that can significantly improve the Zion plant's performance. All levels of the facility staff have demonstrated a positive attitude and a clear commitment to making needed improvements. The NRC staff believes that performance at Zion—with respect to both the physical plant and its operation—is improving, but close monitoring will continue until the NRC staff is fully satisfied with the overall level of performance at the facility.

TVA Projects

In 1985, the NRC staff issued a letter to the Chairman of the Board of Directors of the Tennessee Valley Authority (TVA) indicating that there were significant continuing weaknesses in TVA performance and that management of the TVA nuclear program was ineffective. By that time, the TVA had taken the Browns Ferry (Ala.) and Sequoyah (Tenn.) facilities into a cold shutdown status and had made commitments to the NRC that the plants would not be restarted without NRC concurrence. The number and complexity of the issues were not limited to 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.

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 an operational incident. In March of 1985, TVA volunteered to keep all three units in a shutdown condition until corrective action could be effected to resolve serious NRC concerns regarding TVA's ability to safely operate and manage the Browns Ferry facility.

In January 1991, the staff published a "Safety Evaluation Report on Tennessee Valley Authority: Browns Ferry Nuclear Performance Plan" (NUREG-1232, Volume 3, Supplement 2). The SER concluded that TVA's corrective action programs and commitments satisfied prior NRC staff concerns and, when fully implemented, would justify a restart of Unit 2. TVA had decided to focus its efforts on restoring Unit 2 first, with Units 1 and 3 to follow. By letters dated April 12 and May 30, 1991, NRR issued the results of its Operational Readiness Assessment Team inspections of Unit 2. These inspections confirmed that TVA was prepared to safely restart and operate Unit 2. On May 2, 1991, the Commission voted unanimously to approve restart of Unit 2.

After nearly seven years, Browns Ferry Unit 2 achieved criticality on May 24, 1991. TVA completed the Unit 2 Power Ascension Test program on August 6, 1991. On August 13, 1991, Unit 2 was returned to normal full-power commercial operation.

Following the successful restart of Browns Ferry Unit 2, TVA began the process of returning Unit 3 to service. Unit 3 was scheduled for restart sometime in the fall of 1993. There was no restoration schedule for Unit 1 at the close of the report period, but, since work on Unit 1 will not begin until after restart of Unit 3, it seems unlikely Unit 1 would be ready before 1995.

Watts Bar. The TVA had announced that its priorities for the startup of its facilities would be in this order—Sequoyah, Browns Ferry, and Watts Bar. Having restarted Sequoyah and Browns Ferry Unit 2, the TVA stepped up activity at Watts Bar and established a September 1991 target date for Unit 1 fuel loading. Subsequent problems at Watts Bar have postponed fuel loading until early 1994. No completion schedule had been adopted for Unit 2 at the close of the report period. In May 1989, the TVA had submitted the Watts Bar Nuclear Performance Plan (WBNPP), describing the actions taken and corrective actions planned to qualify for licensing at Watts Bar Unit 1. An independent Watts Bar Program Team, made up of TVA personnel assisted by nationally recognized nuclear power experts, which was formed in November 1987, was responsible for development of the WBNPP and for defining the scope of necessary corrective actions and of the special programs. In June 1988, the NRC staff approved the approach taken by the TVA in identifying new corrective actions and/or modifying existing programs.

Although Unit 1 was virtually complete in 1985, significant corrective activity was required to resolve deficiencies identified through allegations, employee concerns, inspections and independent reviews. In January 1990, the NRC staff issued a SER approving the WBNPP and all but two of the 29 specific program plans. The remaining program plans and the extensive revisions to the Watts Bar Final Safety Analysis Report have been addressed in subsequent SERs. During fiscal year 1990, the staff re-initiated the licensing review for Watts Bar. The NRC staff has prepared the master licensing and inspection plan and is closely monitoring the TVA's implementation of corrective activity.

Bellefonte. In July 1988, the 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 a result of 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. The TVA identified various activities that have continued during the deferral period, and the NRC staff is performing periodic inspections at the Bellefonte site.

PLANT LICENSE RENEWAL

In calendar year 1990, about 20 percent of the nation's electricity was produced by nuclear energy (about 100,000 megawatts); the Department of Energy has projected an increase in consumption of electricity of approximately another 100,000 megawatts in the next decade. In light of the anticipated demand, the electric utility industry has urged the NRC to expedite its license renewal preparations. Utilities require some 10-to-12 years prior to reactor license expiration—whether or not the licenses are to be renewed—in order to allow for timely system planning decisions regarding such matters as replacement power alternatives capital acquisition.

The prospect of renewing operating licenses for nuclear power plants has long been considered a top priority by the NRC and the nuclear industry. Within the next 20 years, many commercial nuclear power plants will have reached the 40-year term of their operating licenses, a figure imposed 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 the process to be followed; thus an immediate focus of NRC effort is to better define the process for review of licensee renewal applications.

In order to help maintain an adequate energy supply for the nation into the early 21st century, some utilities are reviewing the actions which would be necessary to extend the useful life of their nuclear power plants beyond 40 years. In response to this very important issue, the License Renewal Project Directorate is working closely with industry, represented by the Nuclear Management and Resources Council (NUMARC), in this area. Furthermore, 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.

The first currently active operating license will expire in the year 2000, and the licenses of more than 40 percent of all currently operating plants in the United States will expire by the end of the year 2010. The NRC staff estimates that approximately 3-to-5 years will be necessary to complete the review of the initial license renewal applications. Assuming that licensees apply for and are granted revised license expiration dates as a result of construction recapture, and using a five-year estimate and the industry-stated planning need for 10-years advance notice, license renewal applications would be filed with the NRC as follows: 1992—four plants; 1994—four plants; 1995—four plants; 1997—two plants; 1998—six plants; and 1999—six plants.

Rulemaking

The NRC published a proposed license renewal rule (10 CFR Part 54), in the *Federal Register,* July 17, 1990. It is a basic premise of the rule that, since existing nuclear power plants currently operate at an adequate level of safety, new requirements are not indicated other than those necessary to manage potential age-related degradation. License renewal regulations in the proposed rule focus on such age-related degradation issues, and licensees will be required to demonstrate, as part of their license renewal applications, that they have established programs which effectively manage age-related degradation of those plant systems, structures and components important to license renewal. The final rule was published in December 1991.

The NRC is also pursuing two environmental initiatives, related to requirements of the National Environmental Policy Act (NEPA). First, it is preparing an environmental assessment in connection with the Part 54 license renewal rulemaking, and also a proposed revision of the NRC's environmental regulations contained in 10 CFR Part 51. Second, a generic environmental impact statement (GEIS) has been published for public comment, in September 1991, in support of the issuance of a revision, related to license renewal, to 10 CFR Part 51, "Environmental Protection Regulations For Domestic Licensing and Related Regulatory Functions." A workshop on the draft 10 CFR Part 51 revision was held in November 1991. Publication of the GEIS and the revised rule was scheduled for the summer of 1992.

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), which are proceeding in parallel with (1) the renewal rulemaking, (2) reviews of industry technical reports, and (3) "lead plant" review. The draft regulatory guide and the draft SRP–LR were published for comment in December 1990. The staff plans to publish an interim regulatory guide and SRP–LR six months after the final rule is issued. The staff anticipates completion of the final regulatory guide and SRP–LR after the "lead plant" review is completed.

Similarly, the NRC has published for public comment a draft regulatory guide and a draft Environmental Standard Review Plan (ESRP–LR) for license renewal, in September 1991. The staff anticipates completion of the final regulatory guide and ESRP–LR in the summer of 1992. (See discussion under "License Renewal," in Chapter 8.)

Industry Technical Report Reviews

The Nuclear Management and Resources Council (NUMARC) has prepared 11 technical industry reports (IRs) for NRC review that focus on the potential age-related degradation mechanisms associated with a given system or component, and that identify the preventive, corrective or mitigative actions necessary to any program for renewing plant licenses. The NRC has completed the first round review of all 11 IRs and is awaiting the revised IRs to be submitted by NUMARC. When the reviews are completed, the staff will prepare safety evaluation reports, so that each licensee requesting license renewal will have the option of referencing these reports in renewal applications. The NRC staff anticipates completion of all IR reviews in the spring of 1993.

Lead Plant Reviews

The Yankee-Rowe (Mass.) plant of the Yankee Atomic Electric Company and the Monticello (Minn.) plant of the Northern States Power Company are the "lead plants" in the license renewal program. However, Yankee Atomic has recently stated that it would postpone making a decision concerning submittal of a renewal application for Yankee-Rowe until late 1992. The Monticello license renewal application was scheduled to be submitted in December 1991. The staff expects to complete its review of the Monticello application approximately 3-to-5 years from the date the application is received.

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 public 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. This rule reflects the understanding the agency has acquired in its review of standard designs, of the applicable provisions of the Commission's "Severe Accident Policy Statement," and of the proposed standardization legislation, as well as views of the Commission and the industry. The focus of the rule is design certification, a regulatory instrument that would bring about early resolution of licensing issues. Subpart B of this rule provides a regulatory framework for certification through rulemaking of standard plant designs. (Also, the requirements of 10 CFR Part 50, Appendices M, N, O, and Q have been moved to 10 CFR Part 52.)

The NRC staff continues to work with NUMARC and the individual vendors to develop the procedures and practices for implementation of 10 CFR Part 52. The areas under development include defining the content of a design certification rule, and identifying the inspections, tests, analyses, and acceptance criteria (ITAAC) to verify that the facility was built and will operate in accordance with the design certification.

Future Reactor Designs

EPRI Advanced Light Water Reactor Program. The NRC continues to work with the Electric Power Research Institute (EPRI) on an advanced "evolutionary" light water-reactor (LWR) standard plant program. EPRI has submitted for NRC review a utility document defining utility-proposed licensing basis requirements, investment protection requirements, and risk performance requirements, under which advanced LWRs could be designed and constructed using proven technology. This requirements document also proposes resolutions of all applicable unresolved safety issues and generic safety issues and delineates ways of complying with 10 CFR Part 52 and the Commission's severe accident and safety goal policy statements. The NRC staff has issued draft Safety Evaluation Reports deriving from its review of this document and was preparing final reports at the close of the report period.

In fiscal year 1990, EPRI also submitted parallel chapters applicable to a "passive plant," i.e., one designed to minimize or eliminate the need for active intervention to correct off-normal conditions. The NRC staff is developing draft Safety Evaluation Reports based on its review of this document.

GE Advanced BWR. The General Electric Company (GE), 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 incorporate diverse means of controlling rod motion, as well as special features to prevent and mitigate severe accidents. The ABWR is expected to be the first standard design to conform to the EPRI requirements document (see above).

The NRC is continuing its review of all chapters of the ABWR standard safety analysis report through Amendment 18. The staff has issued its draft safety evaluation and plans to issue its final safety evaluation next year.

Westinghouse RESAR SP/90. The NRC completed its review of the Westinghouse Electric Corporation's application for preliminary design approval of its reference safety analysis report SP/90. The SP/90 design was developed independently of the EPRI requirements document. The NRC staff completed its review in April 1991 and issued a Safety Evaluation Report (NUREG–1413) discussing the staff's review of the design.

CESSAR-DC, SYSTEM 80+. In March 1989, Combustion Engineering (CE) submitted an application for final design approval and design certification (FDA/DC) of the System 80+ nuclear power plant design. The NRC staff found the application sufficiently complete to docket and commence review in May 1991. Requests for additional information by the staff are virtually complete. The staff's draft safety evaluation report is scheduled for August 1992, with the final safety evaluation and final design approval scheduled for November 1993. The design certification rulemaking process then follows.

Passive ALWRs. The NRC staff continued discussions with Westinghouse Electric Corporation and General Electric Company (GE), in order to familiarize itself with the Westinghouse AP600 and the GE SBWR designs, and to obtain testing and experimentation information. These are 600-megawatt (electric) designs that will employ passive safety features. Although formal applications for design certification will not be received until 1992, early communication with the vendors enables NRC to assure that appropriate testing of unique features is performed before the designs are approved and employed.

MHTGR. The Modular High Temperature Gas Reactor (MHTGR) design 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 provides for early Commission review and interaction with potential applicants proposing advanced designs. The MHTGR concept features a helium-cooled, graphite moderated 350-megawatt (thermal) standard reactor module. One design objective is to meet the accident dose limits at the exclusion area boundary—set forth in the Protective Action Guideline of the Environmental Protection Agency—with minimal reliance on active systems and without operator actions. The MHTGR design may not include a conventional low-leakage containment building. 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 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.

At the close of the report period, the review of this design was focused on the issuance of the final Pre-application Safety Evaluation Report in fiscal year 1993. A draft report was issued in March 1989.

PRISM. The Power Reactor Innovative Small Module (PRISM) design concept has also been submitted by the Department of Energy (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 using 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 to produce 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 additional amendments to their Preliminary Safety Information Document, in response to open issues identified in the draft PSER. The staff is reviewing the two additional amendments and dealing with the issues opened as a result of these reviews. The final PSER is scheduled for issuance in fiscal year 1993.

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. In late 1990, responsibility for this review was transferred from the Office Nuclear Regulatory Research (RES) to the Office of Nuclear Reactor Regulation (NRR). Since that time, NRR staff has had a number of meetings with AECL Technologies and one meeting with the Atomic Energy Control Board, the Canadian regulatory body, to discuss features of the design. The staff has also planned and initiated review activity with a view to issuance of a Pre-application Safety Evaluation Report in 1993. AECL Technologies expects to make a design certification decision sometime in 1996.

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-watermoderator 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.

PIUS. In October 1989, ABB Atom asked that the NRC perform a review of their Process Inherent Ultimate Safety (PIUS) Preliminary Safety Information Document (PSID), under provisions of the Advanced Reactor Policy Statement, for the purpose of determining whether the design could be licensed. In December 1990, the primary responsibility for review of the PIUS PSID was transferred from RES to NRR. 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. RES is participating in leading the computer code development and modeling effort. The PSER is scheduled to be issued in fiscal year 1993.

PIUS is an advanced pressurized water reactor (PWR) design that employs certain phenomena of physics to accomplish control and safety functions 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, which is intended both for core cooling and 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 stationary interface. During certain transient conditions, the density difference is overcome and the borated water flows into the core to shut down the reactor.

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 (site suitability, environmental protection, and emergency planning). In fiscal year 1991, the NRC developed and started implementing a program plan to enhance its capability to review an application for an early site permit, including the Department of Energy (DOE) co-funded demonstration project. The plan included clarifying regulatory issues, updating technical guidance, and developing staff expertise, in order to provide stability and predictability in the licensing process.

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 bring the Standard Review Plan (SRP) for review of future power reactor applications up to date. A "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" has been prepared to provide acceptance criteria for staff members who carry out safety reviews of applications to construct or operate nuclear plants. The principal purpose of the SRP is to ensure that a predictable, rational and creditable regulatory foundation is in place for dealing with reactors of the future.

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. The SRP has not been revised to reflect all these changes and consequently requires updating to reflect current requirements and regulations. Revision of the SRP is also necessary to expand its coverage to address design certification, the issuance of early site permits, and the licensing reviews of new types of nuclear power reactors, currently under consideration. The updated SRP will reflect existing agency requirements and guidance and will have new review criteria to accommodate the unique technology or the unique application of existing technology in future reactor designs.

The NRC staff will revise regulatory guidance and the SRP in parallel with its review of the evolutionary and passive advanced light water reactor (ALWR) design certification applications. SRP–UDP activity includes program planning, developing procedures for updating the SRP, reviewing generic regulatory documents for revised regulatory positions, revising and developing SRP sections, and converting the SRP, including reference documents and the rationale for each SRP section, into electronic media. A major element of the program involves the evaluation of those industry codes and standards that are presently referenced in NRC regulatory documents. The codes and standards evaluation will include (1) judgments as to whether the NRC should reference the current version of the code in those cases in which a previous version is currently referenced, (2) identification and assessment of potentially relevant industry codes and standards that are not currently referenced by the NRC, (3) identification of the need to develop new industry codes and standards to address unique aspects of future reactor designs, and (4) an assessment of the codes and standards referenced in the EPRI's ALWR Requirements Document.

In fiscal year 1991, the staff developed a draft of the governing procedures for updating the SRP, completed a pilot effort on the review of generic regulatory documents, initiated the evaluation of codes and standards, and developed initial versions of the program's management and information computer data base. The NRC has entered into a technical assistance contract with Battelle Pacific Northwest Laboratories to support the SRP-UDP effort and will also obtain technical assistance from other national laboratories and commercial contractors in the development of the revised SRP.

Technical Specifications Improvements

The Technical Specifications Improvement Program (TSIP) comprises three major tasks—the development of new Standard Technical Specifications (STS), the development of line-item improvements to Technical Specifications, and the performance of related activity to fully implement the interim policy statement on improving Technical Specifications for nuclear power plants, issued by the Commission on February 10, 1987. The development of new STS was undertaken to enhance safety by making the Technical Specifications clearer, easier to use and more focused on safety concerns. The effort is based on NRC review and approval of the vendor owners groups' proposals. In response to the approved guidance of the NRC, the vendor owners groups submitted proposed new STS, during the second quarter of 1989, which included about 4,000 changes. From May 1989 to January 1991, the NRC reviewed the submittals and held about 90 public meetings with the respective vendor owners groups to discuss the proposed new STS, as the NRC review progressed. The submittals were reviewed by numerous organizational elements throughout the agency, and several national laboratories. In January 1991, the NRC issued draft sets of the new STS for the different vendor owners groups, including Babcock & Wilcox, Combustion Engineering, General Electric and Westinghouse. The following improvements have been incorporated into the new STS: (1) specifications are presented using a tabular format, based on human factors principles, rather than a narrative format; (2) sections have been added to provide guidance on the use and applicability of the new STS; (3) improved bases clarify the relationship between requirements and safety concerns; (4) there is greater consistency between the vendor owners groups' new STS. Following issuance of the draft new STS, the public, industry and the NRC staff were given an opportunity to comment. The comment period, which ended July 31, 1991, generated about 17,000 pages of comments. The NRC has developed a formal process for dealing with the comments and anticipates issuing the new STS by the summer of 1992.

The NRC is continuing its work on specific line-item improvements to the existing Technical Specifications. The improvements may be adopted by licensees who submit license amendment requests, and they have also been incorporated into the new STS. Examples of line-item improvements include extending surveillance intervals and outage times for reactor protection system and engineered safety features actuation system instrumentation, or transferring the contents of specifications that address administrative matters to other, more appropriate, licensee-controlled documents.

The implementation of the policy statement will lead to a transfer of existing Technical Specifications to licensee-controlled documents, such as the final safety analysis report. The Commission has therefore directed the staff to assure that an adequate program exists for implementing the requirements of 10 CFR 50.59, requiring a licensee to evaluate whether proposed changes to the facility involve an "unreviewed safety question" requiring prior NRC review and approval. The industry, with input from the NRC staff, has developed a document that provides guidance on the implementation of this regulation. The staff is continuing to evaluate the application of the guidance document by the industry to determine whether further changes may be indicated before the document is completed and endorsed by the NRC.

Other areas of NRC staff effort in support of the policy statement include improving surveillance testing practices, in order to reduce personnel exposure to radiation, to reduce wear on equipment, to reduce reactor trips and other transients that challenge safety systems, and to enable operators to focus on safety concerns. The NRC is also continuing to develop the use of risk insights to improve Technical Specifications that address plant system and equipment configuration management, low-power and shutdown operations, and maintenance scheduling. The staff has also continued to develop risk-based Technical Specifications, which could potentially play an important role in overall risk management for nuclear plants in the future.

Improving NRC Analytical Capability

The nuclear industry has been using 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 the actual facility to the conditions of concern. In the case of reactor accidents or phenomena such as earthquakes, actual tests may be impractical or physically impossible. Models are the only practical means available to examine the response of a facility.

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

Although computer codes are important tools and have contributed significantly to evaluating the safety of nuclear facilities, this use constitutes only one aspect of their importance to the regulatory process. While computer codes are used regularly to check the calculations of an applicant or licensee, their most important application is in helping 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 the emergence of very powerful computer workstations within the agency, NRC management has decided that significant benefits can accrue from a strengthening of analytical capability within the staff. Several new projects were begun, therefore, in 1991, to reinvigorate technical expertise in several offices, which would then serve as exemplars for efforts to improve the analytical capabilities of the entire technical staff. It is expected that, with the introduction of a high-performance computing environment throughout the agency, every technical reviewer should eventually have access to a wide range of sophisticated and powerful computer codes, as well as the data bases needed to use them. Among the project cited were the following:

(1) A small reactor analysis group of 4-to-5 staff members was formed in NRR's Division of Systems Technology. The group of analysts will provide computational support for the review of the ALWR designs in the areas of thermal-hydraulics, containment behavior, and associated disciplines.

(2) To improve staff understanding of the computer codes that they are developing, assessing, and maintaining, and to better manage its code development contractors, RES has initiated an in-house analysis capability using the RELAP5/SCDAP and MELCOR codes. Particular attention will be devoted to code assessment issues for the passive ALWRs. Supporting analyses of operating and advanced passive plants will also be 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 the NRC simulators and develop and use the Nuclear Engineering workstation and the RELAP5 desktop analyzer. In coordination with NRR and RES, AEOD is also using and developing the Reactor Safety Assessment System, which is used by the reactor safety teams in the NRC operations center and by regionally based teams to monitor the status of critical safety functions during reactor transients and the availability of success paths needed to maintain or restore such safety functions.

(4) A pilot program in the Division of High-Level Waste Management in the Office of Nuclear Material Safety and Safeguards will use high-performance computer workstations 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.

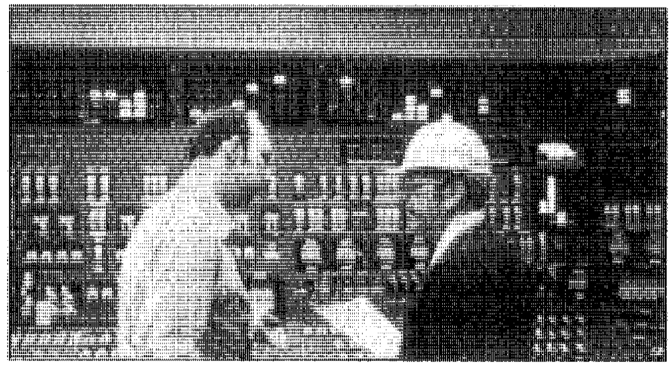
INSPECTION PROGRAMS

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 112 plants for which operating licenses have been issued (as of September 30, 1991), 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.

The operating reactor program, which went through major changes in fiscal year 1989, was modified again in fiscal year 1990 to incorporate feedback from the first year of implementation experience. Improvements continued to be made to the program throughout fiscal year 1991 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 "regional initiatives" phase of the inspection program, Regional Offices redirected certain of their inspection resources away from those 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, 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 undertakes to deal aggressively with unsafe or potentially unsafe events or conditions occurring at individual plant sites or other facilities involving licensed operations (so-called "reactive" inspections). In conducting reactive inspections, the NRC seeks to determine the root cause of the event or condition, evaluates the



NRC inspections within nuclear power plants include regular scrutiny of piping, such as the inspector at left is carrying out, and of control room procedures, being discussed at the right by an operator and an NRC inspector. The inspection programs of the NRC are mainly conducted through the five NRC Regional Offices (see Appendix 1).

licensee management's response to it, including action to prevent recurrence; and decides whether the problem is one that could occur at other facilities.

In the fall of 1989, the NRC staff initiated a regulatory impact survey. One conclusion coming out of that effort, with respect to the reactor inspection program, identifies the "scheduling and control of inspections, especially team inspections," as an area for regulatory improvement (SECY-91-172). Following up on this determination, an Inspection Manual Chapter, "Coordination of NRC Visits to Commercial Reactor Sites," was issued on October 19, 1991, limiting major activities at a site to no more than four, during a licensee's SALP cycle (see below) without approval of NRR and Region management. The Master Inspection Planning System (also discussed below), along with other proposed initiatives, address this mandate.

Reactor Inspection Program

The operating reactor inspection program is conducted by headquarters and regional 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 specialist, 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 1991, 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

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 prescribed responsibility. The NRC inspection program is audit-oriented, i.e., it does not undertake to examine every activity or item, but to verify, through carefully selected samples, that the activities under scrutiny are being properly conducted and carried out in a manner that ensures and/or enhances safety. What to sample, the sizes of the samples, and the frequencies of the inspection efforts are judgments based on the importance of the activity or system to overall safety and 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, from a systems approach, monitors the licensee's activity and provides feedback to the licensee's plant management, so as to allow it to take appropriate corrective actions. However, implementation of the NRC inspection program does not supplant either the licensee's programs or its responsibilities. Rather, the inspection program provides a feedback mechanism and an independent verification of the effectiveness of the licensee's implementation of its programs, to ensure that operations are being conducted 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 from the five Regional Offices, located in or near Philadelphia, Atlanta, Chicago, Dallas and San Francisco. These programs are supplemented by inspections conducted by special teams comprised of personnel from both Headquarters and Regional Offices.

Inspections are part of NRC's review of applications for licenses, as well as the NRC's 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 plans for fabrication and construction.

During construction, samplings taken across the spectrum of licensee activity are inspected to confirm that the requirements of the construction permit are followed and that the plant is being built according to the approved design and applicable codes and standards. Construction inspectors look for qualified personnel, quality material, conformance to approved design, and a well-formulated and satisfactorily implemented quality assurance program, to confirm the quality of construction.

As construction nears completion, pre-operational testing begins, in order to demonstrate the operational readiness of the plant and its staff. Inspections during this phase seek to determine whether the licensee has developed adequate test plans—both to 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 overall test management procedures; (2) examining selected test procedures for technical adequacy; and (3) witnessing and reviewing selected tests to examine the results and to verify the consistency of planned and actual tests. Inspectors also review the qualifications of operating personnel and certify that operating procedures and quality assurance plans are properly developed and implemented.

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 the actual startup test program begins. As in pre-operational testing, NRC inspection emphasis is given to testing management 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 completed satisfactorily, routine operations begin. Thereafter, NRC continues its inspection program throughout the operating life of the plant.

As stated, the responsibility for safe operation of nuclear plant lies with the licensee, and the NRC's role is to make sure that the licensee is meeting its responsibility. The NRC does this through a program of 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 individual in the Regional Office's determination as to when and what additional inspections are indicated for a specific plant. The activity of the resident inspector is supplemented by the efforts 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 reactors 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 done 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 finite resources available for inspection are used efficiently and effectively, with intensified attention devoted to those plants where, based on licensee performance, improvements in the levels of protection and of safety-consciousness may be in order.

The inspection program is an essential element in the NRC's regulatory operation. Its results are factored into NRC's overall evaluation of licensee performance under the Systematic Assessment of Licensee Performance (SALP) program, designed to ensure that nuclear power reactors 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.

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 will continue to be electrical distribution systems.

(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 1991, four TIs were issued, affecting such issues as maintenance, motor-operated valves, and site environment.

(3) *Discretionary 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 direction of the Regional Administrator in response to various plant events or issues.

Use of the team inspection methodology was continued in fiscal year 1991 to provide for an in-depth appraisal of the operability of safety systems at operating plants. The safety systems functional inspection (SSFI) continues to prove a useful approach for regional inspection, because it identifies significant safety issues that require the licensee to take corrective action. Another team inspection approach, the safety systems outage modification inspection (SSOMI), helps identify a need for licensees to maintain more effective controls over activity associated with the evaluation, design, procurement, installation, and testing of plant modifications. The Operational Safety Team Inspection (OSTI) element is employed to verify that the licensee organizations that control and support plant operations—such as Operations, Maintenance, Surveillance, Management Oversight Technical Support, Safety Review, Quality Assurance, and corrective actions—are functioning effectively to ensure operational safety. Because of their demonstrated success, these types of team inspections have been continued as a part of the regional initiatives portion of the inspection program. Team inspections, such as those carried out under the SSFI and SSOMI programs, have proved to be effective tools in assessing the operational readiness of key plant safety systems and licensee activity supporting plant operations. Headquarters and regional staffs will continue to employ them in fiscal year 1992.

The Master Inspection Planning System (MIPS), initiated in 1988, was developed and implemented to facilitate management of the inspection program. MIPS is a centralized, computer-based system providing the Regions with the ability to develop and maintain a current and unique inspection plan for each operating site. Besides providing accurate and up-to-date planning and tracking of inspection activity, MIPS provides the vehicle by which the Regions manage and coordinate reactor site activity. The Regional Offices, with input from Headquarters Offices, develop and schedule inspection plans for each plant. The plans are based on the NRC inspection program, SALP, inspection findings, operational events, senior management meetings, quarterly plant performance reviews, and other assessments of licensee performance. Inspection plans for each site include all NRC non-reactive inspections, third party activity, and any major activity planned by licensees, such as shutdowns or other activity which could affect inspection planning. Each plant's inspection plan in MIPS is consulted on a regular basis by Regional and Headquarters Offices to eliminate unnecessary duplication between the offices and to minimize instances where activity may be scheduled which conflicts with ongoing NRC, licensee or third party planning. NRR and the Regions also make use of MIPS in assessing the effectiveness of the inspection programs.

On September 2, 1991, the NRC began implementation of an Inspection Follow-up System (IFS). IFS is a tool useful agency-wide to track and manage the resolution of NRC-identified concerns at licensee facilities (commercial reactors, non-power reactors, and materials and fuel cycle sites) and at vendors. IFS permits the NRC to categorize and follow-up issues (including inspection findings) by SALP functional area, cause, and area of interest. The system provides a historical record of inspection findings, selected open items, and escalated enforcement information. Besides tracking docket-related questions requiring follow-up, IFS has the capability to track non-docket related items, including administrative items identified by the regional or headquarters staff. IFS is essentially an extension of MIPS; while MIPS facilitates the development of inspection plans and captures the actual hours spent on-site for the completion of inspections, IFS captures the results of inspections and the completion of items. Therefore, users of the fully-implemented IFS system will be able to follow an issue from the time it is first planned through each record of actual inspection accomplishment by inspection report, hours and status, to the final record of inspection closeout. (IFS was designed to incorporate most of the functions of the 766 System and to replace that older system.)

Special Team Inspections

During fiscal year 1991, the NRC headquarters and regional staffs performed 49 special team inspections. A special team inspection involves a team of 8-to-10 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. Depending on the nature of the inspection, the team inspects, as appropriate, the design, installation, testing, maintenance, and operation of the selected systems. The overall objective of these inspections is to determine whether, when called upon to do so in an emergency, the systems examined and plant personnel would perform their safety functions as described in the Safety Analysis Report.

The special team inspection program was described in detail in the *1988 and 1989 NRC Annual Reports,* on p. 21 and p. 22 respectively. Headquarters develops the concept for each new type of team inspection, tests it through a limited number of pilot inspections, and, when developed, incorporates the inspection methodology into the NRC Inspection Manual. The responsibility to complete the program is assigned to the Regional Offices.

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 include emergency operations, maintenance, ability of safety systems to function 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.

Electrical Distribution System Functional Inspections. A new type of special team inspection, called an Electrical Distribution System Functional Inspection (EDSFI), was developed in 1990 (see the *1990 NRC Annual Report,* p. 22). After testing the program at five plants in 1990 and evaluating the results of those initial inspections, the NRC decided to conduct an EDSFI at every plant in the country. As of the end of fiscal year 1991, an EDSFI inspection had been completed for plants on 24 sites. NRC plans are to complete the program at all sites by early 1993.

The principal types of deficiencies identified thus far by the EDSFI inspections—reported in two NRC Information Notices (Nos. 91-29 and 91-51, entitled respectively "Deficiencies identified During Electrical Distribution System Functional Inspections" and "Inadequate Fuse Control Programs")— represent errors in engineering implementation that were introduced during plant modifications or were present in the original design.

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 attention to electrical distribution systems, licensees are conducting their own electrical inspections, are devoting more effort to evaluating the design basis for their electrical distribution systems, and are improving the functional capability of these systems.

New Initiatives. Development work began in 1991 on two new types of team inspections in areas of concern to the NRC. The areas for which early development was under way at the end of 1991 were service water systems and shutdown risk. The first pilot inspection had been completed for service water systems while initial planning and inspection procedure development was under way regarding shutdown risk.

Inspection of Emergency Operating Procedures

During the report period, the NRC staff completed its long term program of improving Emergency Operating Procedures (EOPs). The objectives of the program were to improve the technical accuracy of EOPs and to ensure the incorporation of human factors principles in the procedures. Owners' groups representing the four nuclear power plant vendors re-analyzed relevant transients and accidents and developed generic technical guidelines for improving their EOPs. The industry revised the EOPs to reflect both the engineering guidance contained in the generic technical guidelines and the human factors principles contained in "Guidelines for the Preparation of Emergency Operating Procedures" (NUREG–0899, August 1982).

In order to gain a better understanding of the types and severity of problems that licensees may be having with the EOPs, the NRC staff began an accelerated inspection of the EOPs in fiscal year 1988, with the objectives of determining whether the EOPs were technically correct; whether they could be performed by plant operators during an emergency, taking into account locale, accessibility, and other physical factors; and whether the plant staff possessed the requisite knowledge and ability to correctly perform the EOPs in an emergency. Among other resources, the plant reactor simulator was employed, when available, in conducting this assessment.

The great majority of EOP problems identified during inspections conducted from March to October 1988 resulted from incomplete implementation of EOP programs. The most significant programmatic problems were the lack of a multi-disciplinary team approach in the development of EOPs, lack of independent review of the EOPs, and a lack of a systematic process for ensuring that the quality of EOPs does not deteriorate over time. These findings were discussed with NUMARC and the owners' groups and were published as "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures" (NUREG-1358, April 1989).

The EOP inspection program has been completed, with all operating plants inspected. Results of the fiscal years 1989 through 1991 inspections show some improvement in the implementation of EOP programs, but some problems identified in NUREG-1358 continue to exist. Significant findings from the recent EOP inspections will be addressed in a supplement to NUREG-1358. The staff continues to monitor plant performance in this area, and EOP follow-up inspections will continue to be conducted as necessary.

Vendor Inspection Program

The Vendor Inspection Program is centered in NRC Headquarters and is principally a reactive program 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 seeks to devise tasks and set priorities by which to identify and deal with issues according to their safety significance and generic applicability.

Inspections during fiscal year 1991, primarily reactive in nature, 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 received a number of allegations from current and former vendor and licensee employees, from the news media, from labor unions and other sources. The NRC determines the validity, extent, and safety significance of each reported and alleged deficiency and assures that licensees are apprised of potential problems so that appropriate action can be taken to prevent the use of defective components in nuclear plant safety systems.

In fiscal year 1991, the NRC conducted 33 vendor and licensee inspections, and several other vendor inspections were carried out by NRR involving technical support to the NRC Office of Investigations. The inspections covered vendors and distributors who manufacture/supply solenoid valves, steam generators, fuel assemblies and parts, motor-operated valves and related test equipment, pipe supports, hydraulic snubbers, filters, electric generators, diesel engines, insulation systems, molded case circuit breakers, and environmental and seismic qualification testing. Eight inspections of licensees were conducted to review their procedures and their implementation for the procurement of commercial-grade parts, components and materials for use in safety-related applications. Several inspections involved allegations of falsified records, defective materials, and suspect piping. The vendor inspection staff assisted the NRC Office of Investigations and various U.S. Attorneys in the 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 steam generators manufactured by Babcock & Wilcox in Canada for the Millstone (Conn.) plant; pipe supports and hydraulic snubbers fabricated in Germany by LISEGA GmbH for domestic nuclear plants; and an electric generator being manufactured for the Diablo Canyon (Cal.) plant by NEI Peebles in Edinburgh, Scotland.

As a result of inspection findings and other information in the vendor area, the NRC issued nine Information Notices and supplements to previously issued notices informing the nuclear industry of problems. The Information Notices dealt with falsified certificates of conformance for refurbished circuit breakers, nonconforming magnaflux magnetic particle prepared bath, inadequately qualified structural framing components supplied as safety-related equipment, an inadequate quality assurance program of a vendor supplying safety-related equipment, counterfeit valves, misrepresented resistors, improper assembly of certain molded-case circuit breakers, questionable certification of material supplied to nuclear power plants, and recognition of the equivalence of the ASME Accredition Program to Appendix B requirements.

The NRC staff continued to supply information to and participate in the Federal interagency working group on problem parts and suppliers, an activity that the NRC helped to sponsor and get under way in 1988 and 1989. An interagency data base for the interchange of information on counterfeit/ misrepresented parts is in development.

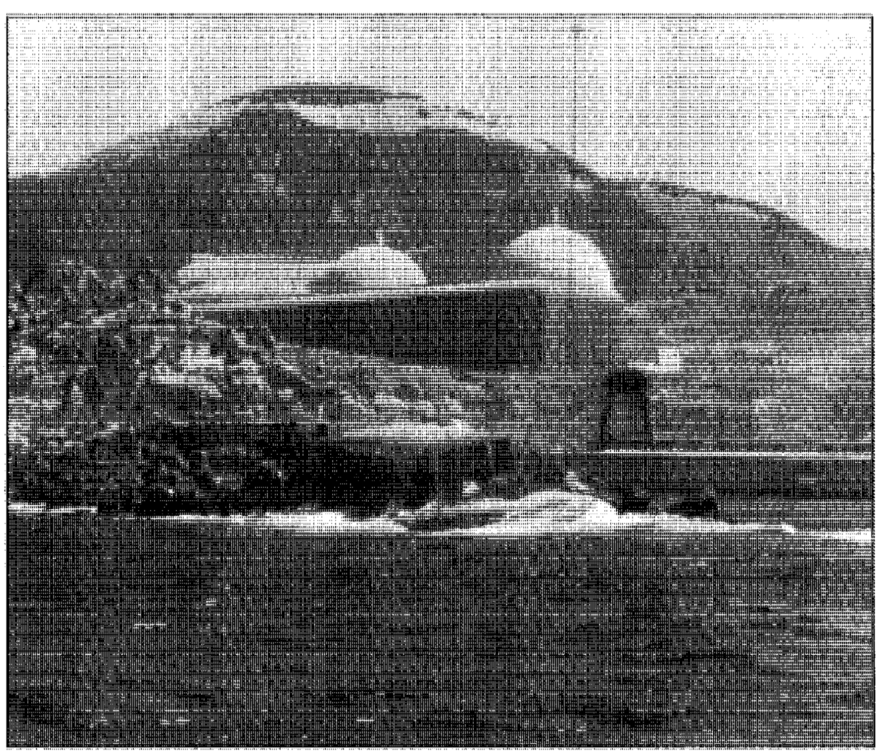
In an Information Notice issued in 1990, the NRC staff alerted licensees that 22 rotary, non-latching MDR-type Potter & Brumfield relays had been modified and/or refurbished and supplied to the Harris (N.C.) nuclear power plant by a firm called Stokley Enterprises, located in Vancouver, Wash. Similar relays were also provided to the Department of Defense (DOD) for use on nuclear powered submarines. Based on NRC and DOD investigations, the case was referred to a U.S. Attorney and resulted in the conviction of the company and its president. The president was sentenced to a jail term and both he and the company were assessed a fine.

In April 1991, the NRC staff issued a generic letter to licensees announcing a pause in the conduct of licensee procurement inspection and enforcement action and identifying a number of failures in licensees' commercial-grade dedication programs, as found during the inspections. The generic letter also expressed staff positions regarding certain aspects of licensee commercial-grade procurement and dedication programs which would provide acceptable methods to meet 10 CFR Part 50, Appendix B requirements. During the pause, as noted above, the NRC has conducted eight assessments of licensees' programs for procurement of commercial-grade parts and components for use in safety-related applications including the supply of fraudulent and misrepresented vendor products. The NRC is continuing to work with the Nuclear Utility Management and Resources Council (NUMARC) to address the nature, extent and safety significance of licensee procurement problems and to resolve differences in interpretation of the requirements of Appendix B for these procurement activities. Based on these concerns, the NRC plans to resume full inspections of licensees' procurement programs and their implementation in accordance with NRC regulations and the Electric Power Research Institute (EPRI) procurement guidelines as accepted by the staff. The resumption of these licensee inspections is planned for the first quarter of fiscal year 1992.

PERFORMANCE EVALUATION

The performance evaluation process is intended to improve the NRC's ability to evaluate the effectiveness of licensee performance at nuclear power plants. The effort involves the integration of information from various of the NRC's continuing 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 culminates in a semiannual meeting of NRC senior management to discuss and appraise operating plant performance. On that occasion, the plants of greatest concern to the agency are identified and a coordinated course of action is drawn up, including recommendations for special inspections and intensified management attention. The results of each meeting are presented to the Commission, and each licensee is informed of NRC's senior management's characterization of their overall performance.

As noted, 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 NRC's Vendor Inspection Program encompasses inspections of nuclear suppliers, both domestic and foreign. Among the latter efforts during the report period was the inspection of an electric generator manufactured by NEI Peebles of Edinburgh, Scotland, for use in the Diablo Canyon (Cal.) facility. The plant is located on the Pacific coast, not far from San Luis Obispo, Cal. Chapter 2 (continued)

The SALP program assesses, in an integrated manner, how well a given licensee's management is directing, guiding and providing the resources needed for the requisite assurance of safety. The purpose of the SALP review is to direct both NRC and licensee attention and resources precisely toward those areas that can most closely affect nuclear safety and that need improvement.

A part 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, to some degree, 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 certain specific functional areas—plant operations, maintenance and surveillance, emergency preparedness, and so forth.

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. Results of the assessment make up part of a data base for periodic reporting in the historical data summary, published semi-annually, and most recently in "Historical Data Summary of the Systematic Assessment of Licensee Performance" (NUREG-1214 (most recent issuance: Revision 8, August 1991)).

Human-Systems Interface

During fiscal year 1991, the NRC staff has given particular emphasis to the "human-systems" interface which is designed into the proposed advanced standardized nuclear power plants. (See "Advanced Reactors," under "Improving the Licensing Process," earlier in this chapter.) Considerable staff resources have been devoted to the analysis of human factors aspects of the EPRI ALWR Requirements Documents, of General Electric's ABWR and SBWR, of Combustion Engineering's System 80+, and of Westinghouse's AP-600 advanced reactor designs. The NRC staff has met with the applicants to discuss their approaches to human factors and particularly to the human-systems interface in the design of the control rooms.

Human factors is one of the areas greatly affected in proposed advanced reactor designs, because of the significantly different control rooms being proposed. These new control room designs incorporate small, compact work stations with computerized display and control functions, as well as some conventional hardwired controls. Current staff guidance for the review of control room designs of these kinds is not complete, and the NRC staff is working with the NRC Office of Research in the development of new guidelines that will exploit information gathered from other industries that use these kinds of control and display features. The staff is also working with advanced reactor vendors and with foreign utilities and research organizations to explore the studies and research they have performed to facilitate the design of advanced control rooms.

The NRC staff has initiated a revision of Section 18.0, "Human Factors Engineering" of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP; NUREG-0800). The revision to the SRP will incorporate new and revised regulatory guidelines regarding human factors and will cover human factors concerns related to future reactors.

The staff has also continued to conduct follow-up investigations of selected human-performance-related events. The investigations focus on identifying and evaluating the consequences to plant safety of human error and on analyzing conditions found conducive to human error.

Training

During fiscal year 1991, the NRC 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 producing effective training programs for nuclear power plant staffs. As part of the evaluation, NRC staff personnel are in attendance as observers when utilities' training programs are under review by INPO accreditation teams. NRC management personnel are also present as observers during utility presentations to the National Nuclear Accrediting Board. The NRC staff also continues to conduct training inspections in accordance with NRC Inspection Procedure 41500 when conditions at particular licensee sites warrant staff evaluation. During the report period, the NRC staff carried out training inspections at nine 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, requiring corrective action. The Commission continues to endorse the industry's accreditation program as an acceptable and effective means of ensuring proper nuclear power plant personnel training.

During fiscal year 1991, the staff began development of 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 June 10, 1991, the staff presented the proposed rule to the Commission. Based on comments from the Commission, the staff is revising the proposed rule and expects to issue it for public comment in early 1992. It is expected that the rule will have a minimal impact on the effectiveness of current industry training initiatives.

QUALITY ASSURANCE

The NRC continued to emphasize performance-based quality assurance (QA) during fiscal year 1991. The "Inspecting for Performance" course—training designed to broaden the scope and increase the technical depth of NRC inspections by imparting techniques based on observing and evaluating activity affecting plant safety and reliability—continues to be taught to NRC inspectors. A similar course is provided by private industry to licensee and other industry personnel.

A "Quality Assurance Program Description" (Section 17.3 of NUREG-0800) was issued late in the last fiscal year. It furnishes the criteria for development of a performance-based QA program. Although current plant licensees are not required to bring their QA programs into conformance with Section 17.3, most utilities are considering doing that as they update their QA program descriptions. This initiative is leading the industry to a more effective, performance-based approach to QA.

Software QA continues to be an area of substantial NRC interest and involvement. As digital systems replace analog systems in operating nuclear power plants, and in the design of advanced plants, the acceptability of these systems must be assessed with respect to the public health and safety. During fiscal year 1991, the NRC staff performed safety reviews of the microprocessors being retrofitted into operating plants. The QA provisions for software for advanced reactor plants are also under review by the NRC staff. And instructions are being prepared for NRC inspectors to inspect software control at operating nuclear power plants. Finally, the NRC staff is working with a standards development group to update the current safety-related consensus software QA standard to make it more broad-based in its application.

Maintenance

Proper maintenance is essential to nuclear power plant safety, and the results of plant maintenance activity must be monitored and evaluated to assure that they remain effective, particularly as plants continue to age. During fiscal year 1991, the NRC staff continued intensive evaluation of maintenance effectiveness in the nuclear power industry.

These efforts included:

(1) Completion of the 14 remaining maintenance team inspections (MTIs).

(2) Completion of the re-inspection of eight sites with poor MTI results.

(3) Evaluation of trends, including systematic assessment of licensee performance rating for current and past periods, performance indicators, licensee event report codes, and the Maintenance Effectiveness Indicator.

(4) Evaluation of maintenance at nuclear power plants through senior management perspective, based on inspections and plant and equipment performance.

(5) Evaluation of industry programs, including an Institute of Nuclear Power Operations standard for maintenance programs.

The NRC staff provided the results of these reviews to the Commission in SECY-91-110, "Staff Evaluation and Recommendation on Maintenance Rulemaking," on April 26, 1991. The staff also presented their recommendations to the Commission at a briefing on May 6, 1991.

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." This rule requires commercial nuclear power plant licensees to monitor the effectiveness of maintenance activity for safety-significant plant equipment, in order to minimize the likelihood of failures and events caused by the lack of effective maintenance. To implement the requirements of the new rule, licensees will have to monitor the performance or condition of certain structures, systems, and components (SSCs) against self-established goals to assure that those SSCs will be capable of performing their intended functions.

To facilitate the implementation of the rule on July 10, 1996, the NRC has begun the preparation of regulatory guidance which will describe methods acceptable to the NRC for implementing the new maintenance rule. The staff is also preparing inspection procedures to be used by NRC inspectors to verify that licensees have implemented the maintenance rule satisfactorily.

The staff will also revise and issue inspection procedures to be used for inspection of licensee maintenance activity in the interim period from the present until July 1996. These procedures will be revised to incorporate a "results-oriented" inspection approach, while remaining consistent with the scope of existing regulations. Procedures will also emphasize inspection of weaknesses noted during the MTIs, in order to assure appropriate and effective corrective actions have been implemented at the plants.

OPERATOR LICENSING

In the absence of new plant operating licenses, only initial and requalification examinations for power and nonpower reactor operators are currently being administered. The responsibility for administering written and operating examinations to license candidates at power reactors—and the issuance or denial of the license based on the results of the examination—continues to rest with the five NRC Regional Offices.

During fiscal year 1991, the NRC issued initial licenses for 329 reactor operators (ROs) and 368 senior reactor operators (SROs). Besides the initial license examinations, the NRC administered 1,206 requalification examinations at 65 reactor sites. The NRC requalification examination process is proving effective in confirming the competency of individual licensed operators and the quality of the facility licensees' requalification programs. The NRC also administered the Generic Fundamentals Examination (GFE) to 672 prospective ROs and SROs during fiscal year 1991. The GFE tests prospective licensed operators on their understanding of theoretical knowledge required for operating a nuclear power plant. In an effort to reduce the regulatory impact on facilities participating in the GFEs, the NRC has initiated action to administer future examinations at the individual sites, rather than in the NRC's Regional Offices.

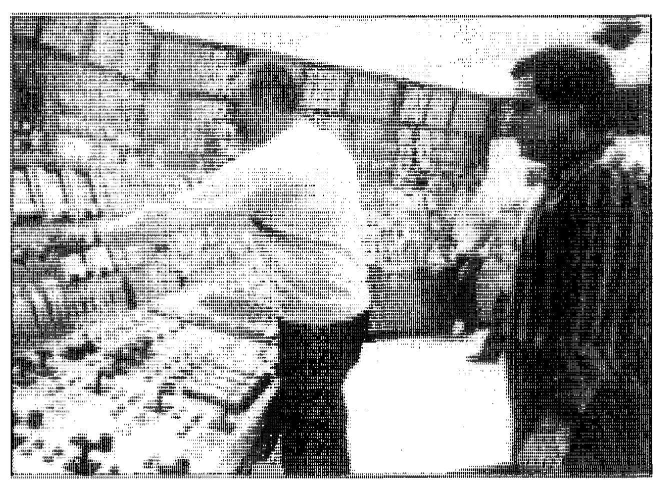
As a result of ongoing measures to improve the examination and licensing program, the NRC has made or is considering a number of changes designed to strengthen the operator licensing process. Among changes made or under consideration are the following:

(1) Three major studies were undertaken in an attempt to improve the NRC's requalification examination process. First, in an effort to enhance inter-regional consistency, the NRC evaluated the scope, depth, and complexity of a sizable sample of dynamic simulator scenarios administered across the NRC's five Regions. Second, the NRC's human factors specialists conducted a study to evaluate the level of stress created by the NRC's requalification examination process, and ways to mitigate that stress. And third, to encourage operator teamwork, the NRC is pilot-testing a revised simulator grading methodology based upon crew, rather than individual, performance. The results of all three studies are being evaluated.

(2) New examiner training and certification requirements designed specifically to enhance examination administration techniques are being implemented. A new examiner refresher training course was developed to meet the continuing training needs of certified examiners in the area of examination techniques.

(3) 10 CFR Part 55 was amended to make the cutoff levels for illegal drugs and alcohol in 10 CFR Part 26 applicable to licensed operators as a condition of their licenses. 10 CFR Part 2 was also amended to incorporate sanctions for licensed operators who violate fitness-for-duty requirements.

(4) The NRC staff completed its review of plant-referenced simulator certifications and exemption requests. Certified plant-referenced simulators are now being used for the conduct of operating tests at 102 of the 112 operating nuclear power plants. Nine facilities that were granted schedule exemptions are expected to submit their plant-referenced simulator certifications during fiscal year 1992. One facility has requested NRC approval to use a simulation facility other than a plant-referenced simulator. NRC approval of that application was pending at the close of the report period.



The NRC is responsible for the licensing of nuclear reactor operators as well as for the reactor plants themselves. During fiscal year 1991, the NRC issued initial licenses for 329 reactor operators and 368 senior reactor operators. Shown above is an NRC examiner observing a candidate for operator's license performing tasks in a control-room simulator.

(5) The responsibility for administering the operator licensing program at non-power reactors was transferred to a newly-created section within the Operator Licensing Branch at the NRC's Headquarters Office.

EMERGENCY PREPAREDNESS

The NRC staff assesses emergency preparedness 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 for 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. The staff also worked closely with the Federal Emergency Management Agency (FEMA) in addressing issues related to off-site emergency preparedness.

During fiscal year 1991, the NRC formed a special task force to review issues raised in a public meeting, held September 6, 1990, regarding off-site emergency preparedness for the Pilgrim nuclear power plant in Massachusetts. The task force was also asked to recommend whether the NRC should reconsider its "reasonable assurance" finding for Pilgrim emergency preparedness. The task force included individuals from NRC Headquarters and Regional Offices, and from FEMA Headquarters and its Region I office in Boston, Mass. Following a second public meeting in Plymouth, Mass., on June 12, 1991, to discuss its preliminary findings, the task force issued a final report (NUREG–1438) on June 18, 1991. That day, FEMA also informed the NRC of its finding that adequate protective measures can be taken off-site to protect the public in the event of a radiological emergency at the Pilgrim facility. The task force forwarded its recommendation to the Commission on June 24, 1991. Having appraised the information it had compiled, as well as FEMA's June 18 statement, and considered actions taken by the State, local communities and the utility, Boston Edison, the NRC task force concluded the it was not necessary for the NRC to reconsider its reasonable assurance finding for Pilgrim. On July 30, 1991, the Commission approved that recommendation, recognizing that responsible parties would still be examining several open issues.

The NRC 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. (The licensing of Seabrook was upheld in the U.S. Court of Appeals for the District of Columbia and subsequently challenged in the U.S. Supreme Court. The U.S. Supreme Court also upheld the decision to licensee Seabrook. (See Chapter 9.)) During fiscal year 1991, the State of Massachusetts once more became a participant in emergency preparedness planning for Seabrook, occasioning a transition from an existing utility-based off-site response organization, to a State-based off-site response. The NRC staff, with the assistance of FEMA, is reviewing the transition plans, including a 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 NRC staff also reviewed a methodology proposed by the industry's Nuclear Utilities Management and Resources Council (NUMARC) for categorizing events based on plant conditions (emergency action levels) at power reactors. The NRC, NUMARC and industry personnel participated in discussions and workshops which included "walk-through" scenarios to test the methodology against provisions of the guidance document (NUREG–0654) and also a pilot test at a licensed boiling water reactor (BWR). The staff will continue to work with NUMARC on a pressurized water reactor (PWR) pilot test, to be conducted in fiscal year 1992. When remaining issues have been resolved, the NRC intends to revise Regulatory Guide 1.101 to reflect endorsement of the NUMARC methodology as an acceptable means of meeting NRC requirements. Additional emergency preparedness licensing actions in fiscal year 1991 included review of advanced reactor submittals and of emergency preparedness issues related to the decommissioning of the Rancho Seco (Cal.) and Fort St. Vrain (Colo.) nuclear power plants. The staff also evaluated emergency planning aspects of actual events which occurred at operating plants during the year, including events at Maine Yankee, Yankee-Rowe (Mass.) and Nine Mile Point (N.Y.).

The NRC staff worked closely with FEMA in examining these elements of emergency preparedness: (1) portal monitors for use at reception centers, (2) exercise scenario requirements, (3) plume monitoring requirements, and (4) updating of the NRC/FEMA Memorandum of Understanding. The NRC staff also worked with the Environmental Protection Agency (EPA) to resolve issues leading to revisions of the Protective Actions Guide Manual, scheduled to be issued by the EPA early in fiscal year 1992.

SAFETY REVIEWS

Applications of Probabilistic Risk Assessment

In fiscal year 1991, the application of Probabilistic Risk Assessment (PRA) methods and insights to regulatory activity continued to expand. As in recent years, PRA applications were made in both traditional PRA-relevant activities and in new 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 plants performance.

The NRC staff has completed its preliminary review of safety improvements and PRA studies for the General Electric ABWR and the Westinghouse SP/90, both advanced plant designs. The estimated core damage frequency for the ABWR is significantly lower than previously estimates for more conventional boiling water reactor designs. Significant progress has been made in the review of Combustion Engineering's CESSAR 80+ design (also an advanced plant design) and related PRA.

The staff is continuing to review Individual Plant Examination (IPE) submittals. It has completed its review of the Yankee-Rowe (Mass.) and Seabrook (N.H.) IPEs. Most of the utility IPE submittals are expected in fiscal year 1992. They deal with accident sequences initiated by internal events or internal flooding. The NRC staff has completed its guidance for the external events portion of the IPE, and utility submittals on that aspect are expected within three years. These submittals will cover seismic events, fires, external floods, high winds and nearby industrial accidents.

The application of PRA results and insights to licensing and inspection efforts was considerable. The Commission's findings on the Yankee-Rowe Pressure Vessel Embrittlement Issues were influenced significantly by PRA, insights in conjunction with thermal hydraulic and fracture mechanics analyses. PRA insights were also influential in helping decide various utility requests for emergency Technical Specification relief.

The application of PRA results and insights to inspection and licensing activities continues to prove its worth. PRA-based information contributed to the planning of nine Maintenance Team Inspections (MTI). Risk-based insights were also provided for one Operational Safety Team Inspection (OSTI), one resident inspector "walkdown" inspection, two setpoint inspections, three Risk-based Operational Safety and Performance Assessment (ROSPA) team inspections and ten Electrical Distribution System Safety Inspections (EDSFIs). Additional risk-based inspection guidance were also completed for 14 plants and provided to the respective resident inspector and Regional Offices. These documents address single systems—12 Auxilary Feedwater Systems and two High Pressure Coolant Injection Systems. A similar risk-based inspection guide to support the Temporary Instruction for Service Water System Inspection is nearing completion. As a part of the ongoing new initiative, risk-based inspection guidance was developed to assess the risk during shutdown operations and outage management. PRA insights were also given with respect to two accident simulation scenarios, used in the control room operator examinations. The manual and syllabus of the PRA training course for inspectors were revised to reflect current practices and applications of PRA information in regulatory activity. Generic risk insights for General Electric boiling water reactors (BWRs) were also developed, and the study results were released to the utilities as NUREG/CR-5692, May 1991.

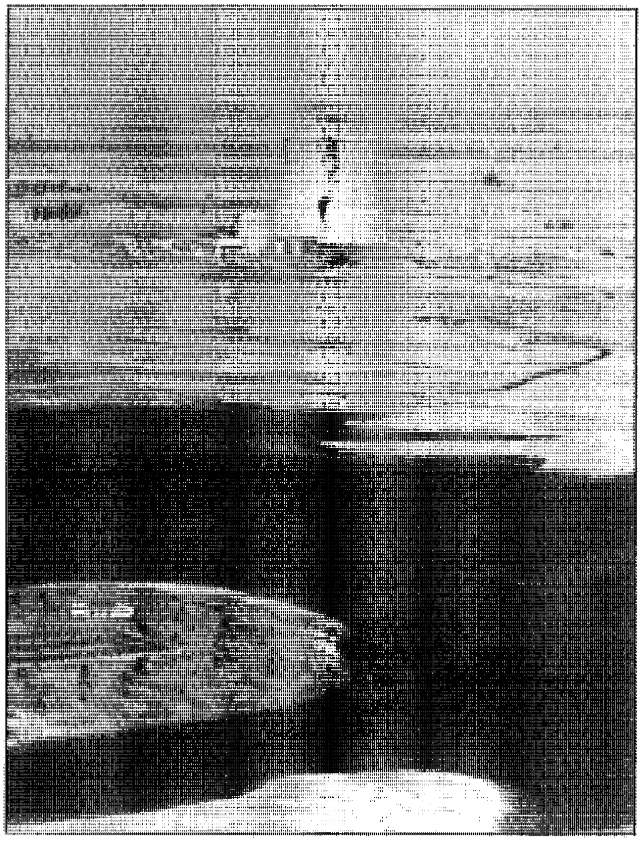
The Functional Incident Response System Tree (FIRST) for BWR-4 plants is being developed by NRR, in conjuction with NRC's Office for Analysis and Evaluation of Operational Data (AEOD), the NRC Regions, and Brookhaven National Laboratory, as an event assessment and screening tool for the Reactor Safety Team.

A new initiative is also under way to develop PRA insights into an Electrical Distribution System (EDS). PRA information of the EDS will be used to develop risk profiles of the EDSFI findings and the other regulatory or licensing activities.

In the area of licensing actions, PRA insights continue to constitute one of the bases for the review and evaluation of licensee submittals, including those for changes in existing Technical Specifications.

Interfacing Systems Loss-of-Coolant Accident Program

In fiscal year 1989, the NRC instituted new studies looking to the resolution of certain questions regarding the potential consequences of an event involving the interface between high- and low-pressure systems at nuclear plants. The postulated event centers on 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, and that condition contains the potential for the release of significant radioactivity directly to the environment.



NRC emergency preparedness licensing actions during fiscal year 1991 included reviews of issues related to the decommissioning of the Rancho Seco (Cal.) nuclear power plant, above, shut down two years earlier following the negative result of a referendum vote on its continued operation. Licensee for the pressurized water reactor plant, in operation since 1974, is the Sacramento Municipal Utility District.

Previous analyses had indicated that the risks associated with an ISLOCA were negligibly small. The basis for that judgment was a conviction that the problem could be appraised primarily in terms of the reliability of the redundant in-series check valves that formed the high-to-low pressure interface. Thus, by imposition of limited measures, such as surveillance requirements on the valves, it was deemed reasonable to expect that the probability of check valve failure could be kept sufficiently low. The new effort was initiated in response to certain operating experience that began to cast some doubt on the premise that ISLOCA is mainly a check valve-related issue.

Since fiscal year 1989, the NRC ISLOCA program has made substantial gains toward a better understanding of the factors that contribute to the likelihood and severity of an ISLOCA. At the close of the report period, NRR had completed a series of extensive plant inspections aimed at acquiring data for thorough evaluation of ISLOCA risks.

Specifically, ISLOCA team inspections were completed at four pressurized water reactor (PWR) plants, two of Westinghouse, one of Babcock & Wilcox, and one of Combustion Engineering design. Preliminary inspection findings for each were documented in inspection reports. In general, direct observation did not identify any inadequacies that could be considered compelling evidence of an immediate safety threat. But the inspections did uncover several concerns that may play an important role with respect to ISLOCA risks. At the same time, these are the areas that offer the best possibility of reducing ISLOCA risks. The matters in question encompass various inadequacies in maintenance, surveillance and testing, and also in human factor areas, including man-machine interfaces, procedures and training.

Detailed evaluation of the inspection data has been made by the NRC Office of Research (RES), in terms of engineering and human reliability analyses, as well as by PRA modeling. Draft reports for each of the vendor plants were issued for comment, and final reports will be issued in the near future.

It is expected that appropriate recommendations or requirements regarding ISLOCA will be forthcoming relatively soon. The objective is to identify those plant operational improvements that can actually reduce ISLOCA risks, based on a thorough technical evaluation of those risks and on cost-benefit considerations.

Performance of Motor-Operated Valves

Operating experience and research programs have raised concerns regarding the performance of motor-operated valves (MOVs) in nuclear power plants. Particular MOV problems have included inadequate MOV design and incorrect torque, torque bypass, and limit switch settings, which have led to failures of MOVs to perform their intended functions. As a result, the NRC staff developed and issued "Action Plans for Motor-Operated Valves and Check Valves" NUREG-1352 (June 1990), which discussed the staff and industry activities related to MOVs and specified staff plans for future action to improve MOV performance. The NRC staff is currently implementing those plans.

A significant task 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 NRC staff requested that licensees help ensure the capability of MOVs in safety-related systems by reviewing MOV design bases, certifying MOV switch settings initially and periodically testing MOVs under design basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The NRC staff requested that licensees complete the Generic Letter 89-10 program within three refueling outages or five years from the issuance of the generic letter.

The NRC staff issued Supplement 1 to Generic Letter 89-10 on June 13, 1990, to provide detailed information on the results of public workshops held to discuss the generic letter. On August 3, 1990, the NRC staff issued Supplement 2 to Generic Letter 89-10 to allow 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 NRC staff issued Supplement 3 to Generic Letter 89-10 on October 25, 1990, requesting that licensees of BWR nuclear plants take action in advance of the Generic Letter 89-10 schedule to resolve concerns about the capability of MOVs used for containment isolation in the steam supply line of the High Pressure Coolant Injection and Reactor Core Isolation Cooling systems, in the supply line of the Reactor Water Cleanup system, and also in other systems directly connected to the reactor vessel. Responding to Supplement 3 to Generic Letter 89-10, several BWR licensees have reported a need to replace or modify MOVs, within the scope of this generic letter supplement. The NRC staff is completing its review of the responses from the BWR licensees to Supplement 3 to Generic Letter 89-10 and is providing that information to NRC Regional Offices for the establishment of inspection priorities.

The NRC staff issued Temporary Instruction 2515/109 (January 14, 1991) to provide guidance for regional inspection of the programs being developed by licensees in response to Generic Letter 89-10. The staff also conducted workshops for regional inspectors regarding Generic Letter 89-10 and the temporary instruction. In January 1991, the NRC staff initiated inspections of the Generic Letter 89-10 programs, and, as of the close of the fiscal year, had conducted approximately 20 inspections of those programs. The staff has found many licensees to be making progress toward resolving concerns about MOV performance at their facilities. Where it was determined that licensees had not made sufficient progress, the staff has taken action to emphasize the importance of resolving the MOV issue at those facilities on a timely basis.

The staff is continuing to evaluate the industry's efforts in resolving concerns about the performance of MOVs at nuclear power plants. Regulatory action will continue whenever necessary to improve the performance of MOVs, so as to provide assurance that the health and safety of the public are adequately protected.

Diablo Canyon Long Term Seismic Program

The NRC issued Operating Licenses (OLs) for Diablo Canyon Units 1 and 2 (Cal.) in 1984 and 1985, respectively. The Unit 1 OL was conditioned to require that the licensee, Pacific Gas and Electric Company (PG&E), update the geological, seismological and ground-motion information, re-evaluate the magnitude of the earthquake used to determine the plant's seismic design basis, re-evaluate ground motion expected at the site, re-assess engineering and equipment response, and perform a seismic PRA and deterministic studies, as necessary. The license condition was imposed because of an Advisory Committee on Reactor Safeguards (ACRS) recommendation, in a 1978 letter, and because of significant advances in the geo-sciences that had occurred since the beginning of the site review.

As required by the license condition, PG&E submitted its final report on July 31, 1988. During its review, the NRC staff met a number of times with PG&E management and received several submittals in response to requests for additional information. PG&E conducted an extensive study and concluded that the Hosgri fault is the seismic source that could cause the maximum vibratory ground motion at the site. The utility evaluated the maximum credible earthquake associated with the Hosgri fault zone, the style of faulting expected on the fault, and the models for estimating ground motions expected at the site. Plant management also performed a deterministic analysis, as well as a PRA, and concluded that plant seismic margins are adequate.

The NRC staff review was conducted with the assistance of expert consultants, including the U.S. Geological Survey, the University of Nevada-Reno, Lawrence Livermore National Laboratory, Sandia National Laboratory and Brookhaven National Laboratory. As a result of the review, the staff concluded that PG&E had met all aspects of the license condition (NUREG-0675, Supplement No. 34), pending confirmation that plant structures can withstand the staff-assessed ground motion spectra which exceeds the PG&E evaluation spectra at certain frequencies, anywhere from 0-to-15 percent. The final Safety Evaluation Report was issued by the staff during fiscal year 1991, and completion of the program was reviewed and approved by the Advisory Committee on Reactor Safeguards.

Evaluation of Shutdown and Low-Power Risk Issues

Following the loss, on March 20, 1990, of all vital a.c. power, during a reactor shutdown at the Vogtle (Ga.) plant, the Commission requested that the staff address a number of issues pertaining to safety during shutdown operations. In response, the NRC staff is performing a broad evaluation of risks during shutdown and low-power operation, encompassing both the Vogtle follow-up issues and other matters. In October 1990, an action plan was forwarded to the Commission. On June 19, 1991, the staff discussed the status of its evaluation at a public meeting with the Commission. On September 9, 1991, the NRC staff issued SECY-91-283, describing its progress, and a revised action plan for completing the evaluation in fiscal year 1992.

During fiscal year 1991, the staff completed the following major tasks, in connection with this evaluation:

(1) Systematically reviewed operating experience, including reviews of foreign and domestic operating reactor event reports.

(2) Visited 10 plant sites to broaden staff understanding of shutdown operations.

(3) Reviewed, evaluated and documented the few existing domestic and foreign PRAs that address shutdown conditions.

(4) Completed a coarse Level 1 PRA of shutdown and low-power operating modes for a PWR and a BWR. (5) Completed several thermal-hydraulic studies that address the consequences of an extended loss of decay heat removal.

(6) Completed a compilation of existing regulatory requirements for shutdown operations and important safety-related equipment.

(7) Coordinated an international meeting of Regulatory Agencies to exchange information on current regulatory approaches to the shutdown issues in member countries.

In the spring of 1991, the staff conducted a three-day meeting of key NRC personnel who have been working on the shutdown and low-power evaluation, or have special expertise. Forty-five individuals—representing NRR, RES, AEOD, the Regional Offices, and several national laboratories—attended the meeting, held from April 30 to May 2, 1991. From these discussions, the staff was able to identify five issues that are especially important to shutdown operations and conditions, as well as a number of additional topics warranting further investigation. The five key issues are (1) outage planning and control, (2) stress on personnel and on programs, (3) the need for improvements in training and procedures, (4) technical specifications, and (5) PWR safety during mid-loop operation The more significant topics identified for furthere evaluation include a.c. power availability, containment capability, boron dilution events, fire protection, the potential for draining the vessel, the reporting requirements for shutdown conditions, the effectiveness of NRC Generic Letter 88-17, and NRC inspection programs to address shutdown issues.

The NRC staff has had extensive interaction during this period with the nuclear industry, represented by the Nuclear Management and Resource Council (NUMARC), especially in the area of outage planning and control.

The staff expects to complete its technical evaluation of key issues and topics and make recommendations regarding action in the near future, with recommendations for improvements to be tested through the NRC backfit process during 1992.

Pressurized Thermal Shock Issue

The Yankee-Rowe nuclear power plant is a 185-megawatt (electric) PWR located in Rowe, Mass., which is operated by the Yankee Atomic Electric Company. It is a four-loop Westinghouse design and began operation in 1960. In May of 1990, as part of the NRC staff's review of a license renewal document, questions arose regarding the ability of the Yankee-Rowe reactor pressure vessel to withstand a Pressurized Thermal Shock (PTS) event.

Thermal shock results when material undergoes a rapid temperature change on one surface, in such a manner that a large temperature gradient is developed across the material. The large temperature gradient can induce high thermal stresses resulting in crack initiation which poses the possibility of possible structural failure of the material. When this phenomenon occurs in a pressure vessel while it is still at a relatively high pressure, the phenomenon is called pressurized thermal shock.

The ability of a vessel to withstand the stresses resulting from a PTS event depends upon several factors, including the presence of flaws, the chemical composition of the material, and the amount of neutron irradiation that the vessel has experienced.

At the Yankee-Rowe facility, the chemical composition of important reactor vessel materials is uncertain. In addition, the results of inspections performed in recent years are not sufficient to give assurance that important flaws are not present in the vessel. The licensee, therefore, proposed a test, inspection, and material sampling program by which to narrow the range of potentially important uncertainties. The program is to be implemented in the spring of 1992.

To support its interim operation, the licensee provided a PRA, in July 1990, showing that the likelihood of vessel failure from a PTS event was low. In August 1990, the staff indicated its agreement with the results of these analyses and authorized continued operation of the plant until the spring of 1992.

In June of 1991, the Union of Concerned Scientists, in a petition to the Commission, proposed that the Yankee-Rowe reactor should be immediately shut down because of PTS concerns. The NRC reviewed the petition and, based upon calculations thought to be conservative, reaffirmed, in July 1991, the staff's earlier decision that continued operation was acceptable until April 1992. In issuing its decision, the Commission also ordered that the licensee seek measures to provide still greater protection against thermal shock events at Yankee-Rowe.

In response to the Commission's Order, the utility proposed specific modifications to further reduce the risk from a PTS event. Thermal hydraulic and fracture mechanics analyses were provided to support the proposed modification. The analyses were performed using more realistic models than in previous analyses, but they showed less conservative results. The unexpected results substantially reduced the staff's confidence that calculations of the likelihood of vessel failure probability are conservative. The staff, therefore, recommended that the plant be shut down until actual data from the test and surveillance program show an adequate margin of protection. In view of the staff's recommendation, the licensee voluntarily shut down the plant. After further meetings with the NRC, the utility agreed not to seek permission to restart Yankee-Rowe prior to April 1992.

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 conducted its initial reviews of licensee responses, including several site audits of the documentation supporting the responses. Based on these reviews, the staff determined that additional guidance and clarification was needed to correct the deficiencies found in the licensee's submittals and supporting documentation. The staff worked with the Nuclear Management and Resources Council (NUMARC) to develop augmented guidance. This guidance, reviewed and approved by the staff, was issued in January 1990, requesting that licensees provide supplemental responses on the station blackout rule to the NRC. Revised responses were received by the NRC in March 1990.

Although licensees generally followed the response guidelines proposed by NUMARC, the staff found it necessary to undertake further discussions with licensees, and in many cases, to obtain additional written information from the licensees, in order to complete the safety evaluation of the licensees' proposed method of coping with a station blackout. These added tasks have affected the overall review schedule.

As of September 30, 1991, the staff had completed reviews of 42 plant sites (65 units) and was continuing reviews of the remaining responses, giving priority to the sites for which changes are most beneficial. Nearly all sites require procedure changes and associated training in order to meet the station blackout rule. A few sites are adding or replacing safety-related batteries. Eight sites are adding non-safety grade diesel or gas turbine generators to serve as alternate a.c. (AAC) power sources. These sites are installing more safety-grade emergency diesel generators (EDGs) to provide an AAC source for station blackout and also to improve overall safety and operability. Some multi-unit sites are adding electrical crossties between units to permit an EDG of the non-blacked out unit to supply power to the blacked out unit. Other plants are making minor changes, such as removing ceiling tiles in the control room or adding emergency lighting.

At many of the sites for which procedural changes and minor modifications were proposed, these have already been implemented. At sites for which major additions were proposed, licensees have nearly completed the modifications or, at least, are moving toward completion.

Steam Generator Replacement at Millstone

In May 1990, Northeast Utilities informed the NRC staff of their decision to replace both steam generators at the Millstone (Conn.) nuclear power plant during its next refueling outage, which will occur beginning in April 1992. (Steam generators are very large heat exchangers that produce the steam that drives the plant's turbines.) Steam generator replacements have been successfully carried out at nine reactors on previous occasions in the United States. The Millstone Steam Generator Replacement Project (SGRP) is different from previous replacement projects. Only that portion of the lower section (the section containing the steam generator tubes), below the larger diameter steam drum upper section, will be replaced. The old steam generator will be cut at approximately the center of the cone area and removed. The new lower portion will then be installed in place and welded to the upper drum section. The upper drum section internals will be completely replaced before welding to the lower section. The plan was devised so that the new steam generators could pass through the containment access opening.

Millstone Unit 2 has been plagued with steam generator tube degradation since beginning operation in 1975. Caustic stress corrosion is the main contributor to the tube-cracking. Of the total of 17,038 steam generator tubes in both steam generators, 3,851 have been sleeved and more than 3,354 have been plugged, thus necessitating the steam generator replacement.

NRC staff oversight and inspection of the SGRP will cover all aspects of the program. Numerous meetings have been held at NRC Headquarters and Region I 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. Staff oversight and inspections will continue through to completion of the replacement. The licensee plans for an approximate five-month outage, beginning in April 1992, to carry out the project.

Progress continues to be made in reducing the number of USI items yet to be fulfilled. Approximately 85 percent of these items have been implemented at the 112 licensed plants; specifically, of the 1,819 relevant items, 1,545 have been completed and 274 remain open.

The staff will continue to closely monitor progress in carrying through on the TMI Action Plan items, USIs, and GSIs, to be updated in the next combined annual report.

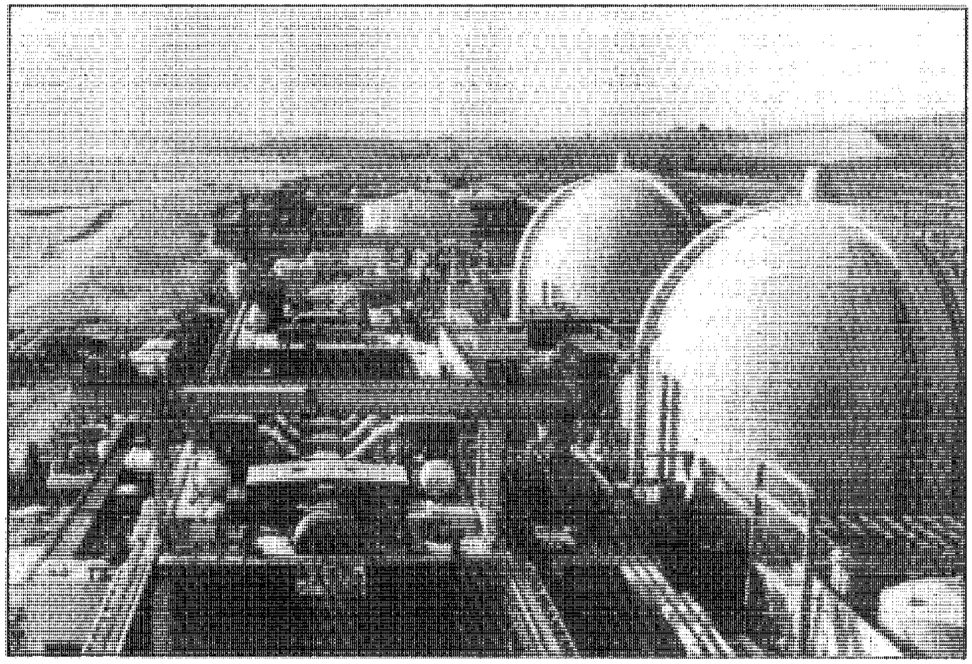
Radiation Protection at Nuclear Reactors

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 the cleanup of small leaks of liquid and gaseous materials. These initial alerts are followed up 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.

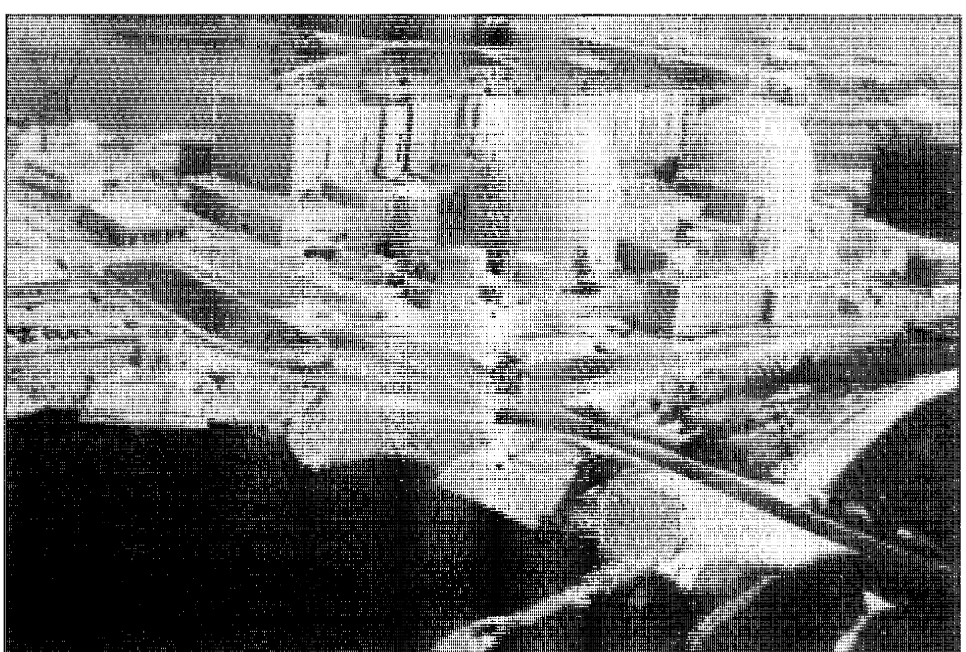
During fiscal year 1991, the NRC staff has provided radiation protection support in licensing activity at most of the operating nuclear power reactors, as well as reviews of design criteria and conceptual designs for advanced reactors. This work was initiated for the CESSAR SYSTEMS 80+ and continued for the Advanced Boiling Water Reactor (ABWR) and for the evolutionary and passive designs of the Electric Power Research Institute (EPRI). Such support included detailed evaluations of occupational radiation protection design features, systems and equipment; in this review, the staff discovered a serious shielding deficiency in the ABWR drywell. Evaluation continued for the off-site consequences of design basis accidents for the ABWR and EPRI projects. Also included were extensive discussions regarding control room habitability problems for such plants as San Onofre (Cal.), Zion (Ill.), and Cook (Mich.). Licensing action support during the period also included reviews of the radiation-protection histories at Millstone (Conn.), Indian Point (N.Y.), and Salem (N.J.) facilities, in connection with requests for operating license extensions. An important NRC staff function has been providing radiation-protection evaluations on the shutdown and decommissioning activity at the Fort St. Vrain (Colo.), and Rancho Seco (Cal.) power reactors, as well as the CINTICHEM (N.Y.) and Universities of Virginia and Washington production/research reactors. In addition, the staff has evaluated proposals from the Palisades (Mich.) and Maine Yankee (Me.) licensees 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 matters; during the report period, Information Notices were prepared and issued on such subjects as potential worker hazards from high radiation fields caused by irradiated components and by a radioactive liquid release incident at the Fitzpatrick (N.Y.) plant.

Inspection support was provided during the year for radiation-protection inspections at the Seabrook (N.H.), Browns Ferry 2 (Ala.), and North Anna (Va.) plants, and a special team inspection covering the ALARA ("as low as reasonably achievable") radiation exposure reduction program at the Brunswick (N.C.)plant. In cooperation with all five Regional Offices, the staff developed generic operating procedures for the regional mobile laboratories used at licensees' sites by the NRC to perform confirmatory measurements of effluent waste streams and other radioactivity measurements.

The Office of Nuclear Reactor Regulation (NRR) staff provided significant support to RES in issuing for public comment the Generic Environmental Impact Statement (GEIS) for a rule change, in 10 CFR Part 51, limiting 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 technical support to the NRC Office of Nuclear Regulatory



Radiation protection for workers at nuclear reactor plants is a continuing concern and focus of NRC study and assessment. One subject of such scrutiny during the report period was control-room habitability at such plants as the San Onofre facility, shown above, located on the coast of California near San Clemente, between Los Angeles and San Diego. Radiation protection inspections were also carried out during the year at the North Anna (Va.) facility, shown below, which is situated on the North Anna River in north central Virginia, between Richmond and Fredericksburg.



Research focused on the development and preparation for public comment of 10 regulatory guides associated with the revised 10 CFR Part 20.

Accident Management

Accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff (1) to prevent core damage, (2) to terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) to maintain containment integrity as long as possible, and (4) to minimize off-site releases. The objective of the accident management program is for utilities to perform strategic planning for severe accidents, in order to enhance their capabilities in each of these areas. Improvements would be realized through the development and implementation by each utility of an "accident management plan" which addresses important elements of accident response, such as procedures and training.

The NRC first described its approach to accident management in early 1989. Key aspects of the approach are that (1) NRC will work with the U.S. nuclear industry to define the scope and attributes of a utility accident management plan, and to develop guidelines for the plant-specific implementation of such a plan; (2) NUMARC will provide industry's perspective and coordinate the necessary industry-supported initiatives; and (3) NRC will conduct a supporting program of research to provide a technical basis for evaluating industry's activities and products. The NRC continues to follow these tenets in its overall approach to accident management.

The U.S. nuclear industry has instituted a major program in accident management. Industry efforts are being coordinated by NUMARC and involve participation of EPRI and the owners group for each U.S. reactor vendor. The industry program involves three principal activities, as described in the *1990 NRC Annual Report.* Substantive progress has been made on each of these activities, as summarized below.

Evaluating Accident Management Capabilities. An essential part of accident management is the process of planning for events beyond the design basis of the plant. To provide guidance to utilities as to how this process might be carriedout, industry has developed a method entitled "A Process for Evaluating Accident Management Capabilities." The method was submitted for NRC review in August 1991. It is a self-assessment approach consisting of four major steps, and a series of questions in nine different areas (e.g., procedures, training, instrumentation), intended to provide the structure for evaluating and enhancing accident management capabilities. The NRC has also developed a systematic process for developing and assessing accident management plans, as part of the NRC accident management research program. Both the industry and NRC methods for evaluating accident management are preliminary and will be employed on a trial basis at a limited number of plants. These demonstration applications were scheduled to be completed by late 1991.

Technical Basis Report. In September 1991, industry provided the NRC the first volume of a report entitled "Severe Accident Management Guidance Technical Basis Report." The purpose of the report is to summarize severe accident technology relevant to accident management concerns and to serve as the basis for vendor-specific accident management guidance. The staff is reviewing this document; a second volume of the report, containing more detailed information, will be issued and reviewed at a later date.

Vendor-Specific Accident Management Guidance. Each of the individual owners groups intend to develop vendor-specific guidance for accident management, based on the technical basis report. Licensees would then develop plant-specific guidance based on the vendor's guidance. In July 1991, the NRC staff initiated discussions with the industry's owners groups regarding the development of vendor-specific accident management guidance. At that time, each owners group representative described initial plans for developing such guidance. Each owners group intends to follow a systematic process in doing so and will survey and assess individual utility IPE results as an explicit element in their guidance development process. Vendors are just beginning to undertake these measures, and greater interaction between NRC and the owners groups is anticipated in the coming year.

The NRC continues its research into accident management, in order to develop a technical basis for evaluating the industry's activities and products. As part of the program, several strategies were deemed to merit more detailed investigation and are being further evaluated. PWR depressurization is one such strategy. Included in each of the detailed strategy evaluations will be an assessment of the risk reduction potential and cost/benefit of the strategy. Research is also continuing to better define other specific elements of the accident management framework and to assess options for NRC audits of industry capabilities in the longer term. This research should result in augmented insight and guidance for inclusion in either the industry products or the NRC Generic Letter on Accident Management.

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 within 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 the measurement of plant effluents and the analytical modeling of the environmental exposure pathways. In turn, the studies verify that the plant is in compliance with regulations and 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 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 year.

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 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.

The health impact of environmental radiation levels associated with nuclear power plants was assessed by two studies, and those studies in turn were evaluated by the NRC staff. A study conducted by agents of Columbia University and of the TMI Public Health Fund covered the vicinity of the Three Mile Island (Pa.) nuclear power plant and concluded that no ill effects from radiation could be found. The staff found no problems with this study and noted that its conclusions were consistent with other available evidence. A Massachusetts Department of Public Health (MDPH) study focused on the area around the Pilgrim (Mass.) nuclear power plant. One of the observations made in this survey was that, over a five-year period, the leukemia rate in their intermediate dose range was higher than in the low dose range. From this evidence, the MDPH concluded that a leukemia relationship with Pilgrim effluents was supported but not proven. The NRC staff rejected this conclusion because (1) the expression time was too short for the elevated leukemia rate to be radiogenic, (2) the dose distribution assumed by the MDPH was seriously wrong, (3) the data obtained from telephone interviews with surviving friends or relatives were too inaccurate to constitute a reliable basis for judgement, and (4) no correction was made for the anomalously low leukemia rate in a portion of the "low dose" area. The staff finds no reason to believe that radioactive effluents have any discernible deleterious effect on the environs of any nuclear power plant.

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. This average dose seems to have leveled off in 1990.

In 1990, the average dose-per-unit for LWRs was 339 person-rems. This is approximately the same as the 1989 average of 338 person-rems. In 1990, the average dose-per-unit for PWRs was 291 person-rems-per-reactor, up slightly from the 1989 average of 289 person-rems. In 1990, the average dose-per-unit for BWRs was 433 person-rems-per-reactor, down slightly from the 1989 average of 435 person-rems. The activities which most frequently contributed to PWR doses in 1990 were steam generator-related work, refueling operations, installation and removal of scaffolding and shielding, valve maintenance, and hanger modifications. Major contributors to BWR doses in 1990 included recirculation pipe replacement/crack repair (weld overlays), intergranular stress corrosion cracking related in-service inspection, valve maintenance, and installation and removal of scaffolding and shielding.

The 1990 dose compilation includes data from 72 PWRs and 37 BWRs. This total reflects the addition of one new PWR, South Texas Unit 1, and one new BWR, Limerick Unit 2 (Pa.). Plants which have not been in commercial operation for a full year are not included in this compilation. LaCrosse (Wis.), Dresden Unit 1 (Ill.), Humboldt Bay (Cal.), Three Mile Island Unit 2 (Pa.), Fort St. Vrain (Colo.), and Indian Point Unit 1 (N.Y.) are no longer included, because there are no plans to operate these plants in the future.

The NRC has ongoing contracts with Brookhaven National Laboratory in the area of occupational dose reduction at LWRs. The objective of one of the NRC-sponsored studies is to compare foreign and domestic processes which contribute to occupational dose. Other studies involve the compilation of a research data base on dose reduction projects at nuclear power plants, the impact of reduced dose limits, hot particle production, mitigation and dosimetry, and the licensing of irradiated gemstones. The NRC also has an ongoing contract with the Idaho National Engineering Laboratory to study contamination and recontamination in BWR primary coolant recirculation piping.

Operational Safety Assessment

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

In fiscal year 1991, the NRC assigned augmented inspection teams—as part of a formal program for the assessment of major incidents—to determine the facts regarding the following operating reactor events:

- Loss of reactor coolant system inventory at Braidwood Unit 1 (Ill.) in October 1990.
- Inadvertent lifting of two fuel assemblies from the reactor core along with the upper core internals during preparations for defueling at Indian Point Unit 3 (N.Y.) in October 1990.
- Discovery of multiple failures of main steam check valves at Sequoyah Units 1 and 2 (Tenn.) in October 1990.
- Rupture of two moisture separator drain lines and discharge of secondary plant steam/water into the turbine building at Millstone Unit 3 (Conn.) in December 1990.
- Loss of off-site power during refueling at Diablo Canyon Unit 1 (Cal.) in March 1991.
- Loss of decay heat removal capability at Oconee Unit 3 (S.C.) in March 1991.
- Loss of electrical power supply redundancy while shut down at Oyster Creek Unit 1 (N.J.) in March 1991.
- Unmonitored release of radioactive materials at Fitzpatrick Unit 1 (N.Y.) in March 1991.
- Partial loss of off-site power and reactor trip at Zion Unit 2 (Ill.) in March 1991.
- Loss of off-site power at Vermont Yankee (Ver.) in April 1991.
- Main transformer fault, main generator hydrogen fire and reactor trip at Maine Yankee (Me.) in April 1991.
- Degradation of decay heat removal capability during shutdown at Oconee Unit 1 (S.C.) in September 1991.
- Loss of reactor coolant system inventory at Oconee Unit 1 (S.C.) in September 1991.
- Spent fuel pool gate seal failure at Wolf Creek Unit 1 (Kans.) in September 1991.

Also as part of its formal program for the assessment of major incidents, the NRC, in fiscal year 1991, assigned incident investigation teams to investigate the following events:

- Fuel abnormalities at General Electric fuel fabrication plant (N.C.) in May 1991.
- Loss of control room annunciation and indication at Nine Mile Point Unit 2 (N.Y.) in August 1991.

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 request specific actions to be taken by utilities and require written confirmation when actions have been completed. In fiscal year 1991, the staff issued 98 Information Notices, including 16 supplements, and two Bulletin supplements. Generic Letters may also be issued to address operational safety matters having broad applicability. In fiscal year 1991, the staff issued 19 Generic Letters, including three supplements.

Implementation Status of Safety Issues

The NRC publishes a document annually giving the status of the implementation of planned 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 constitute the basis for a combined, updated annual report to the Commission, the first to be issued in November 1991.

As reported in volume 1 of the document, approximately 99 percent of the TMI Action Plan items have been implemented at the 112 licensed plants. Of the 13,527 applicable items, 13,404 have been completed or closed out, and only 123 remain open. About 50 percent of the remaining 123 open items are projected to be implemented by the end of calendar year 1992.

CLEANUP AT THREE MILE ISLAND

During fiscal year 1991, 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 is a passive monitored state similar to the SAFSTOR option of decommissioning.) The NRC has offered the opportunity for a public hearing prior to issuance of the license change which would authorize implementation of the monitored storage. Final residual fuel measurements and calculations for special nuclear material accountability at the facility are nearing completion. The evaporator system, used to dispose of the 2.1 million gallons of accident-generated water, began operation and has decontaminated and vaporized approximately one-third of the water.

In August of 1988, General Public Utilities Nuclear (GPUN) Corporation, the licensee, submitted a Safety Analysis Report to document and support their proposal to amend the TMI-2 license to allow the facility to enter PDMS. Through the end of fiscal year 1991, the licensee had issued 13 amendments to the report. The NRC staff and contractor consultants from Battelle Memorial Institute's Pacific Northwest Laboratory (PNL) have evaluated the licensee's proposals and are preparing a Safety Evaluation regarding the license conditions and technical specifications necessary to implement PDMS. The evaluation is expected to be issued early in fiscal year 1992. On April 25, 1991, the staff published a notice of opportunity for a prior public hearing regarding the license change to implement PDMS.

During July and August of 1991, the reactor vessel was drained to make final measurements of the residual fuel remaining in the vessel. The reactor vessel fuel measurement program is the final step in the special nuclear materials accountability program at TMI-2. The measurement technique made use of an array of helium-filled detectors to measure fast neutrons produced by the residual fuel. Calibrations were made using americium-berylium and californium sources. Because of the very complex geometries involved, data reduction and calculations are not expected to be completed until early in calendar year 1992. 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 continue to monitor and evaluate the licensee's reactor vessel fuel measurement program.

The evaporator system began vaporizing accident-generated water on January 24, 1991, after a prolonged period of system testing, modification, and repair. At the end of fiscal year 1991, a total of 738,800 gallons had been decontaminated and vaporized.

The 11-member Advisory Panel for the Decontamination of Three Mile Island Unit 2 is composed of citizens, scientists, and State and local officials. (See Appendix 2 for a listing of members.) The panel was formed by the NRC in 1980 to provide input to the Commission on major cleanup issues. During fiscal year 1991, the panel held two meetings in Harrisburg, Pa. Principal topics discussed at these meetings included decommissioning funding status and plans, results of cancer studies in the TMI area, status and progress of the cleanup at the TMI-2 facility, and the future of the Advisory Panel.

ANTITRUST ACTIVITY

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–0970, May 1985.) In addition, applications to amend construction permits permits or operating licenses resulting from a proposed transfer of ownership interest or operating responsibility in a nuclear facility are subject to antitrust review.

In previous years, the Commission's antitrust review responsibility has been primarily confined to reviews of construction permit and operating license applications; however, during the past two fiscal years, 1990 and 1991, the staff's antitrust activity has 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 1991, the staff conducted the following reviews pursuant to the NRC's antitrust review responsibility: (1) three operating license amendment requests resulting from proposed mergers involving owners of the Wolf Creek (Kans.), Millstone Unit 3 (Conn.) and Seabrook (N.H.) nuclear facilities; (2) an operating license amendment request by the licensee for the Farley (Ala.) nuclear plant to change the plant operator; (3) an outstanding request by two of the co-owners of the Perry and Davis-Besse (both in Ohio) plants to suspend antitrust license conditions; (4) a Section 2.206 request to enforce antitrust license conditions as part of the Diablo Canyon (Cal.) nuclear plant; and (5) a "significant-change" operating license review of the Watts Bar Unit 1 (Tenn.) nuclear power plant.

In the latter part of fiscal year 1991, the staff completed reviews associated with the proposed Kansas Gas and Electric Company (KG&E)/Kansas Power and Light Company (KP&L) merger and the proposed takeover of Public Service Company of New Hampshire (PSNH) by Northeast Utilities—involving the Wolf Creek (Kans.) nuclear power plant and the Seabrook (N.H.) and Millstone Unit 3 (Conn.) nuclear plants, respectively. The staff completed draft reviews associated with the change in ownership and operators of the Seabrook and Millstone facilities and was awaiting the advisory input of the Department of Justice on these two reviews, at the close of the report period. The review of the change in ownership resulting from the proposed KG&E/KP&L merger concluded that there were no post-operating-license significant changes resulting from the proposed change in owners. This finding was published in the *Federal Register*, and requests for re-evaluation were provided for. No such requests had been received as of the close of the report period.

The request by Alabama Power Company (APCO) to transfer operating responsibility for the Farley plant from APCO to a newly formed nuclear operating company, Southern Nuclear Operating Company, was reviewed by the NRC staff for significant antitrust changes. Pursuant to this review, the Commission issued a policy statement requiring new plant operators to be bound by a license condition limiting their role in the marketing or brokering of power or energy from the plant they intend to operate. This same policy statement indicated that no public comment would be sought on these license conditions.

The staff denied the request by Ohio Edison Company and Cleveland Electric Illuminating Company to suspend the antitrust license conditions which are a part of the Perry and Davis-Besse nuclear plants. A hearing on appeal of the staff decision was scheduled for the fall of 1991.

The staff issued a Notice of Violation (NOV) against the Pacific Gas and and Electric Company (PG&E) in fiscal year 1990 because of noncompliance with license conditions imposed on operation of the Diablo Canyon (Cal.) nulcear plant. In the fall of 1990, the Northern California Power Agency (NCPA) filed a Section 10 CFR 2.206 petition with the staff seeking modification and suspension of the Diablo Canyon licenses because of PG&E's alleged inadequate response to the NOV. Throughout fiscal year 1991, PG&E and NCPA engaged in discussions aimed at resolving their differences. NCPA and PG&E indicated to the NRC that a settlement would be concluded in the fall of 1991. The staff has developed a response to NCPA's most recent 10 CFR 2.206 request to be issued in late 1991 or early 1992, should the parties fail to resolve their differences within this time period.

The staff completed its "significant-change" operating license review of the Tennessee Valley Authority's Watts Bar (Tenn.) nuclear plant in fiscal year 1991. The staff found that there had been no significant antitrust changes in the licensee's activity since the previous antitrust review.

The "Solar, Wind, Waste and Geothermal Power Production Act of 1990" amended the Atomic Energy Act by deleting uranium enrichment from the activities taking place in a "production facility," as previously defined in the Atomic Energy Act. As a result of this amendment, uranium enrichment facilities are no longer reviewed under 10 CFR Part 50 of the Federal Code, but rather are reviewed under 10 CFR Parts 40 and 70. There is no requirement to conduct a pre-licensing antitrust review of applicants seeking 10 CFR Part 40 or 70 licenses, and, consequently, the staff's ongoing (fiscal year 1991) antitrust review of one uranium enrichment facility application was terminated.

In an effort to expedite and streamline the Commission's antitrust review process, the staff undertook a revision of NUREG-0970 in fiscal year 1990. Work on this revision was continuing at the close of the report period.

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 three layers, which involve a sharing of liability by the individual reactor licensee, the nuclear industry, and the Federal Government, are set forth in the 1990 NRC Annual Report, p. 39. Government indemnity for large power reactors was phased out in 1982.

Indemnity Operations

As of September 30, 1991, 132 indemnity agreements with the NRC were in effect. Indemnity fees collected by the NRC from October 1, 1990, through September 30, 1991, total $96,900. Fees collected since the inception of the program total $23,731,694. Future collections of indemnity fees will continue to be lower, since the indemnity program has been phased out for commercial reactor licensees. No payments have been made under the NRC's indemnity agreements with licensees during the 35 years of the program's existence.

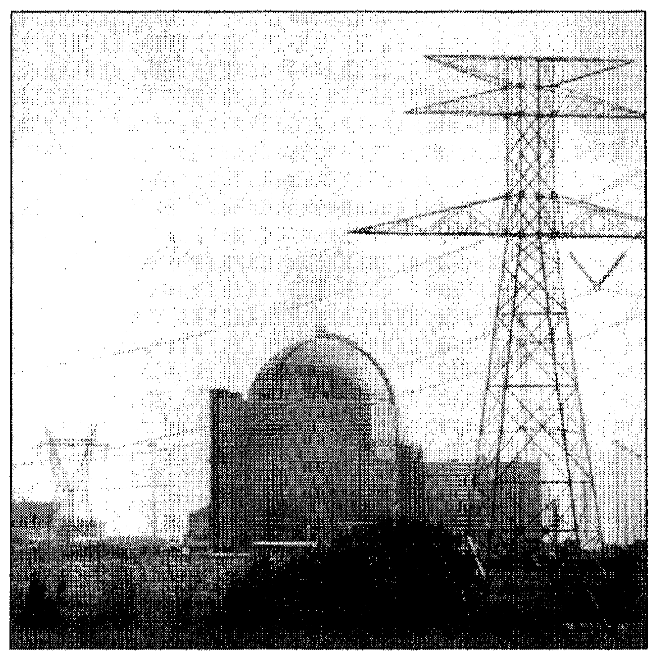
Insurance Premium Refunds

The two private nuclear energy liability insurance pools—American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid policyholders the 25th 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 loss experience of all policyholders over the preceding 10-year period.

Refunds paid in 1991 totaled $13,636,982, which is approximately 74 percent of all premiums paid on the nuclear liability insurance policies issued in 1981 and covers the period 1990-1991. The refunds represent 66.7 percent of the premiums placed in reserve in 1981.

Property Insurance

The ninth annual property insurance reports submitted by power reactor licensees indicated that, of the 76 sites insured, 71 are covered for at least the $1.06 billion required in the revised property/accident recovery insurance rule published on April 2, 1990. The remaining five sites have sought or have been granted exemptions from the full amount of required coverage, because of their small size or operating status. Thirty-five sites carry the maximum $2.035 billion currently available.



Not only does the NRC staff perform antitrust reviews of all applications for the construction and operation of nuclear power plants, it is increasingly involved in antitrust appraisals occasioned by proposed license amendments associated with changes in plant ownership. Among these was a review of proposed mergers involving owners of three nuclear plants, one of them the Wolf Creek (Kans.) plant, shown above. The plant is on the Neosho River, about midway between Wichita and Kansas City.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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

The ACRS reviews requests for pre-application site and standard plant approvals, each application for a construction permit or an operating license for power reactors, 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 necessary to conduct a balanced review is drawn from scientific and engineering disciplines and includes individuals experienced in conducting safety-related reviews of nuclear plant design, construction and operation.

During fiscal year 1991, 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 subjects:

- General Electric Advanced Boiling Water Reactor Design.
- Licensing Review Basis Document for the Combustion Engineering, Inc. System 80+ evolutionary light water reactor.
- Preliminary design approval for the Westinghouse RESAR SP/90 design. x Restart of the Browns Ferry Unit 2 (Ala.) nuclear power plant.
- Full-term operating license for the Oyster Creek (N.J.) nuclear power plant.
- Full-term operating license for the Palisades (Mich.) nuclear power plant.
- Full-term operating license for the Dresden Unit 2 (Ill.) nuclear power plant.
- Full-term operating license for the San Onofre Unit 1 (Cal.) nuclear power plant.

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

- Containment design criteria to accommodate severe accidents.
- Evolutionary light water reactor certification issues.
- NRC Safety Research Program.
- NRC staff's Regulatory Impact Survey Report.
- NRC computer codes and associated documentation requirements.
- "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (NUREG–1150).
- Consistent use of probabilistic risk assessment.
- Essential service water system failures at multi-unit nuclear power plant sites.
- Individual plant examination of external events for severe accident vulnerabilities.
- Loss of essential service water in nuclear power plants.
- Bolting degradation or failure in nuclear power plants.

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

- Nuclear power plant license renewal.
- Requirements for design certification under 10 CFR Part 52.
- Inspections, tests, analyses, and acceptance criteria for design certification.

- Emergency response data system.
- Fitness-for-duty requirements for reactor operators.
- Selection, training, and qualification of nuclear power plant personnel.
- Containment testing requirements.
- Nuclear power plant maintenance.
- Dry metallic spent fuel storage casks.

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

Operational Information/Investigations And Enforcement Actions

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. These 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 reactor plant safety performances. It is also responsible for the NRC's Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and the Incident Investigation Program. And the AEOD office provides support for the work of the Committee to Review Generic Requirements. There are two divisions in the AEOD office: the Division of Operational Assessment and the Division of Safety Programs. AEOD reports directly to the Executive Director for Operations (EDO).

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 regional and headquarters management.
- Development of the NRC program for diagnostic evaluations of licensee performance and direction of "diagnostic evaluation" teams.
- Development of policy, program requirements, and procedures for the NRC's investigations of significant operational events.
- Identification of needed operational data to support safety analyses, and development of agency-wide operational data reporting and retrieval methods and systems.
- 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.
- Development of an agency-wide technical qualifications program covering a broad range of technical positions within the NRC staff, and provision for technical training needed by NRC personnel, through operation 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, including backfit requirements, 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) about them, in the manner described below.

The members of the CRGR, as of the end of fiscal year 1991, 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.

L. Joe Callan, Director, Division of Radiation Safety and Safeguards, Region IV 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.

The Committee seeks to eliminate unnecessary demands on licensee and NRC resources. Through its review, 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, (3) is likely to result in significant safety improvement, or (4) 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.

Following its review, the CRGR recommends to the EDO either that the proposed requirements be approved, disapproved, modified, or conditioned in some way. The EDO considers CRGR recommendations, as well as those of the cognizant NRC office, in deciding whether a requirement has been adequately justified. From its inception in November 1981 through September 1991, the CRGR has held 209 meetings and considered a total of 356 separate issues. In fiscal year 1991, the CRGR held 18 meetings and considered 22 issues, including eight generic backfits in the form of three Rules, four Generic Letters, and one Bulletin. A listing of the 22 issues considered follows.

- Generic letter supplement on testing of motor-operated valves to address specific valve deficiencies found in NRC-sponsored tests.
- Final rule amendment on pressurized thermal shock.
- Generic letter on proposed resolution of a generic issue concerning reactor coolant pump seal failure.
- Final rule amendment on containment structure leak rate testing.
- Generic letter on extending surveillance intervals required by technical specifications to accommodate a 24-month refueling cycle.
- Generic letter on procurement and dedication of commercial grade products.
- Generic letter supplement on individual plant examinations for severe accident vulnerabilities related to external events.
- Revised policy statement and rulemaking alternatives on maintenance.
- Bulletin supplement on mechanical steam generator tube plugs manufactured by Westinghouse.
- Final rule on nuclear power plant license renewal.
- Proposed rule on training and qualification of nuclear power plant personnel.
- Proposed rule amendment on emergency preparedness to update and clarify requirements.
- Generic letter on partial resolution of generic issue concerning service water systems at multi-unit sites.
- Generic letter on resolution on generic issue concerning bolting degradation.
- Final rule on emergency response data systems.
- Final rule amendment on fitness for duty for licensed operators.
- Proposed rule amendment on decommissioning funding for prematurely shutdown plants.
- Proposed rule and guidance documents on environmental conditions for nuclear power plant license renewal.
- Generic letter transmitting new inspection guidance on determination of equipment operability.
- Generic letter transmitting a case study on problems with solenoid operated valves.
- Generic letter on upgrading emergency telecommunications system.
- Generic letter supplement on intergranular stress corrosion cracking in boiling water reactor piping.

The Committee periodically visits operating power reactors for discussions with the licensee's management, and operations personnel as another means of assessing the impact of NRC generic communications and new generic requirements on the operation and safety of power reactor facilities. During fiscal year 1991, the Committee visited Arkansas Nuclear One, Units 1 and 2, operated by Energy Operations, Inc.

Analyses of Operational Data

Domestic. AEOD analyzes and evaluates the operational experience of nuclear power plants by means of the following major data sources: reports submitted by plants to the NRC in compliance with the "Immediate Notification Requirements for Operating Nuclear Power Reactors" (10 CFR 50.72), and "Licensee Event Report System" (10 CFR 50.73), and the voluntary reports on component failure submitted to the Nuclear Plant Reliability Data System (NPRDS), which is managed by the Institute of Nuclear Power Operations (INPO). AEOD also uses plant operating profiles and shutdown data found in the licensees' Monthly Operating Reports to generate a context for event analysis and also as a source of data for normalization of event data (e.g., the calculation of reactor trips-per-1,000 critical hours).

As noted above, one of the primary sources of operational event data is the Licensee Event Report (LER). 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 the more frequent events of lesser significance that are more conducive to statistical analysis and trend detection. Since the implementation of the rule in 1984, the 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.

AEOD uses a 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 pieces of 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).

As a result of a regulatory impacts survey (NUREG-1395), AEOD and NRR staff conducted four regional workshops in the fall of 1990 on licensee event reporting. The workshops were designed to inform the industry on the reporting of data and to obtain industry feedback regarding implementation of the reporting rules (10 CFR 50.72, 50.73, and 73.71). Subsequently, in September 1991, AEOD issued "Event Reporting Systems 10 CFR 50.72 and 50.73" (draft NUREG-1022, Revision 1), for public comment. The final NUREG will contain clarifications to ensure complete event reporting and will consolidate guidelines for the two rules into one document.

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 Organization of Economic Cooperation and Development, the Nuclear Energy Agency (NEA), and the International Atomic Energy Agency (IAEA)—as well as through 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. There were about 100 foreign event reports reviewed during this period. The NRC also continues to exchange operational data with other countries, submitting 55 reports of U.S. operating experience to the NEA's international incident reporting system (IRS) during fiscal year 1991. (See "International Programs," in Chapter 7.)

Engineering Analyses of Operational Experience

In 1991, AEOD published one case study and one special study, and issued a number of engineering evaluations and technical reviews, all listed in Table 1. Substantial attention was given to identification of accident sequence precursors, to analyses of human factors in operating events, and to the ongoing Performance Indicator Program, as discussed below.

Operating Experience Feedback Report Solenoid-Operated Valve Problems at U.S. Light Water Reactors (C90-01—Published as NUREG-1275, Volume 6). The AEOD staff analyzed recent U.S. light-water reactor experience with solenoid-operated valves (SOVs), focusing on the vulnerability of safety-related equipment to common-mode failures or degradations of SOVs. It presented information on many representative events in which common-mode failures or degradations affected, or had the potential to affect, multiple safety systems or multiple trains of individual safety systems. While plant safety analyses may not have addressed such common-mode failures or degradations, operating experience indicates they are continuing to occur.

Table 1. AEOD Reports Issued During FY 1991

CASE AND SPECIAL STUDIES

Designation	*Subject*	*Issued*
C90-01	Operating Experience Feedback Report Solenoid-Operated Valve Problems at U.S. Light Water Reactors	1/91
S91-01	Performance of Emergency Diesel Generators (EDG) in Restoring Power to Their Associated Safety Buses_A Review of Events Occurring at Power	9/91

ENGINEERING EVALUATIONS

Designation	*Subject*	*Issued*
E90-09	Additional Factors Affecting the Lift Setpoint of Pressurizer Safety Valves	10/90
E90-10	Evaluation of Boiling Water Reactor Mode Switch Events	12/90
E91-01	A Review of Water Hammer Events After 1985	2/91

TECHNICAL REVIEWS

Designation	*Subject*	*Issued*
T90-13	Corrosion and Failure of Service Water Pump	10/90
T90-14	Seal Problems in Boric Acid Transfer Pumps	10/90
T90-16	Impact of Pipe Liner Failure on Pump Operation	11/90
T91-01	Causes of Incorrect System Flows	2/91
T91-02	Incorrect Rotation of PDP	2/91
T91-03	Overloaded Emergency Buses 2/91	
T91-04	Technical Review Report Turbine Overspeed Trip Due to Steam Valve Leakage and Condensate	4/91
T91-05	Setpoint Testing of Pressurizer Safety Valves With Water-Filled Loop Seals	5/91
T91-06	Technical Review Report Deficiencies in External Flood Protection	6/91
T91-07	Evaluation of Partial Loss of Station Power Events at Prairie Island Unit No. 2 On December 21 and December 26, 1989	7/91

Common-mode SOV failures and degradations have cut across multiple trains of safety systems, as well as multiple safety systems. Common-mode SOV failures have compromised front-line safety systems and important support systems, such as emergency alternating current (a.c.) power, auxiliary feedwater, high-pressure coolant injection, and scram systems, resulting in reductions in safety margins. Many of the common-mode SOV failures and degradations observed went beyond conditions analyzed in plant Final Safety Analysis Reports (FSAR) and are not modeled in present-day probabilistic risk assessments (PRAs).

The AEOD staff sought the root causes of the observed failures and degradations of SOVs and examined the widespread deficiencies found in design and application, manufacture, maintenance, surveillance testing, and feedback of failure data.

Some of the more significant common-mode SOV events discussed in the report are as follows:

- Simultaneous common-mode SOV failures that resulted in the failure of both emergency diesel generators to start at the Perry (Ohio) plant.
- Simultaneous common-mode failures within the scram system at the Susquehanna (Pa.) plant.
- common-mode scram pilot solenoid valve failures that resulted in primary system leakage outside primary containment at the Dresden (Ill.) plant.
- Simultaneous common-mode failures of two SOVs and the potential failures of 58 additional SOVs in multiple systems at the Kewaunee (Wis.) plant.
- Simultaneous common-mode failures of main steam isolation valves (MSIVs) to close upon demand at the Perry (Ohio) and Brunswick (N.C.) plants.
- Simultaneous common-mode failures of safety relief valves in the automatic depressurization system at the Brunswick (N.C.) plant.

The AEOD staff, concluding that correction of the root causes of the SOV problems would reduce the likelihood for common-mode SOV failures, recommended that, for safety-related applications, licensees (1) verify the compatibility of SOV design and plant operating conditions, (2) verify the adequacy of plant maintenance programs, (3) ensure that SOVs are not subjected to fluid contamination, (4) review SOV surveillance testing practices, and (5) verify that SOVs used in safety-related applications have been manufactured, procured, installed and maintained in a manner commensurate with their safety functions. The staff also recommended that an industry group take action to improve the mechanism for communicating SOV failure data to the manufacturers, for early detection and resolution of potential generic problems.

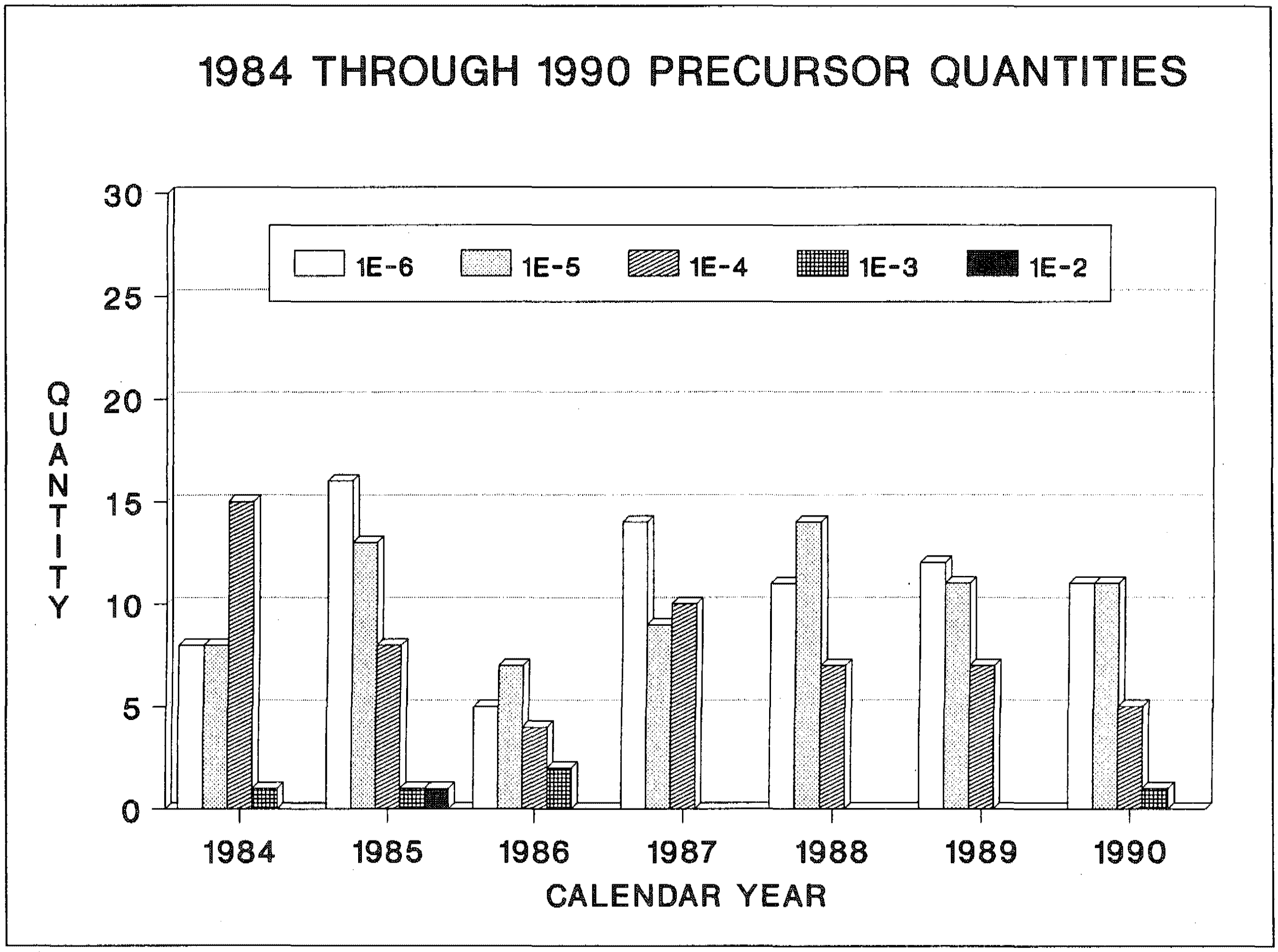
The case study was forwarded to the NRC Office of Nuclear Reactor Regulation (NRR) for implementation, in January 1991. NRR is considering actions to be taken on the recommendations in the case study. Major industry organizations (e.g., Electric Power Research Institute (EPRI), Institute of Nuclear Power Operations (INPO), Nuclear Management and Resources Council (NUMARC) have been given copies of the case study for their consideration, especially in light of the last recommendation noted above for industry action. The case study was issued as NUREG–1275, Volume 6, in February 1991, and was disseminated to communicate spread the lessons of operating experience presented in the report.

Accident Sequence Precursor Program. The Accident Sequence Precursor (ASP) Program, established at the Nuclear Operations Analysis Center at Oak Ridge National Laboratory in 1979, provides a structured means of evaluating the safety significance of operational experience. The program is administered and directed by AEOD and emphasizes evaluations of licensee event reports (LERs) of U.S. commercial light-water reactors to identify and categorize precursors to potential severe core damage accidents.

The ASP method models and evaluates plant equipment and human response that could affect the progression of an accident, employing actual failures that have occurred along with the probabilities for postulated additional failures. Precursors are important because they are combinations of events, actually experienced, which, if accompanied by additional events or failures, could result in reactor core damage. Precursors also can provide insights into the capability of a plant to respond to accidents or other incidents. Precursors identified in the ASP Program have an estimated probability of at least 1E–6 of resulting in core damage. (Probabilities designated E–3, E–2, E– 1 and 1 are considered "highly significant.")

A nuclear plant Accident Sequence Precursor is an actual, observed situation, event, or combination of events which, had it or they occurred along with other, postulated, events or failures, could have resulted in a plant condition leading to severe core damage.

As indicated by the above definition, a precursor can be more than a single event. It can be a combination of events in a given plant situation that form a part of postulated sequences of events leading to severe core damage. Nuclear plant damage cannot occur unless an accident or initiating event occurs and plant safety equipment fails to respond adequately. Thus, an initiating event, such as a loss of off-site power (LOOP) or a loss-of-coolant accident (LOCA), is required along with actual or postulated failure of plant equipment whose function is to mitigate the effects of the initiating event. In a precursor, not all parts of an accident scenario take place; subsequent



The graph above covers the period of 1984 through 1990 and shows the incidence each year of Accident Sequence Precursors—events at nuclear power plants judged to have had the potential, given certain other concurrent events or failures, to cause severe core damage accidents. The designations 1E–6 through 1E–2 identify categories of probability that precursors could result in core damage, with 1E–6 being those less likely to do so than 1E–5 events, and so forth (E–3, E–2, E–1 and 1 are categories considered "highly significant"). As the chart indicates, the frequency of precursor occurrence has remained fairly constant, especially in the latter portion of the period represented.

phases of a scenario, either initiating event or plant equipment failure, have to be postulated. The likelihood of these postulated occurrences are estimated using reliability data. The two major classes of precursor events are these:

(1) The first class of precursor event involves the unavailability of systems. These are events in which one or more safety system was found failed or degraded, during surveillance of the equipment. These events can be important because, while safety systems are degraded or unavailable, the plant can be at increased risk to initiating events (e.g., a LOOP) which call for response by the system or systems that are functionally unavailable. In these unavailability events, an initiating event and often additional system failures have to be postulated for core damage to occur.

(2) The second class of precursor event involves initiators or plant challenges. These events may be accidents or other initiators requiring plant response to mitigate the effects of the event. The initiator can be a LOOP, a LOCA, or other plant transients requiring response. The significance of this type of event is determined by assessing the degree to which plant equipment worked properly to mitigate the effects of the initiating event.

The precursor method uses event tree models to evaluate the likelihood of various possible outcomes (scenarios) for the events being modeled, resulting in a quantitative estimate of the significance of the event in terms of conditional core damage probability. The ASP methods and models are probabilistic, and they have corresponding limitations and uncertainties of results. One limitation of the results is that the evaluations require an estimate by the analyst of the likelihood of not recovering failed or unavailable equipment when needed.

The types of precursor found during calendar years 1989 and 1990 are generally similar to the types of precursors found in previous years; however, the reports tend to track the findings of other studies of reactor safety vulnerabilities, as new knowledge comes to light. For example, the instrument air system and components associated with this system, such as solenoid-operated valves, have been shown by AEOD studies to be important.

Steam generator tube rupture (SGTR) events were more frequent during 1989 (two events) than in previous years. The SGTR was known to be important because (1) if the SG with the leaking tube could not be isolated, continuous flow from the primary to the secondary system would occur, depleting water available for core cooling; and (2) risks of adverse health effects, though not evaluated by ASP, can be associated with these types of events. The concern is that, until the SG with the leaking tube is isolated, primary side coolant is discharged in a nearly direct path to the atmosphere, with potential consequences from the radioactivity in the primary coolant water. (The SGTR event that occurred at McGuire Unit 1 (N.C.), on March 7, 1989, had an estimated conditional core damage probability of 7.7 E–4. It was the most significant precursor event for 1989.)

Precursors are still being found that reflect hidden inherent equipment deficiencies, such as design errors, which are only uncovered during design reviews or from exceptional demands on the equipment. For example, during the Palo Verde Unit 3 (Ariz.) event of March 3, 1989, manual operation of atmospheric dump valves (ADVs) was hindered by equipment designs or conditions that increased the likelihood of human error. In this event, the ADV equipment room lighting was poor, the opening direction to manually operate the ADVs was not consistent between valves, and the available procedure to operate the equipment was not adequate or was poorly written.

Operational experiences with equipment failures or initiating events while the plants are shut down for maintenance or refueling have been subject to increased scrutiny over the last few years. For example, on March 20, 1990, while Vogtle Unit 1 (Ga.) was shut down for refueling, critical a.c. power from the off-site power grid was inadvertently lost, and the emergency diesel generators, which are intended to back up the off-site power for redundancy, initially failed; one was subsequently recovered. The event had an estimated conditional core damage probability of slightly less than 1E–3.

Table 2 lists precursor events that occurred in 1990. Precursors are also displayed in a chart for the years 1984–1990, according to degrees of importance, i.e., to the probabilities of conditional core damage associated with them. As the chart indicates, there were no highly significant events—events of a probability of 1E–3 or

The most significant precursor event in 1989 was a steam generator tube rupture at the McGuire (N.C.) facility, accorded a probability for potential core damage in the E–4 range (less probable than the "highly significant" categories of precursor events). The two-reactor facility is located on the Catawba River, 17 miles north of Charlotte, N.C.

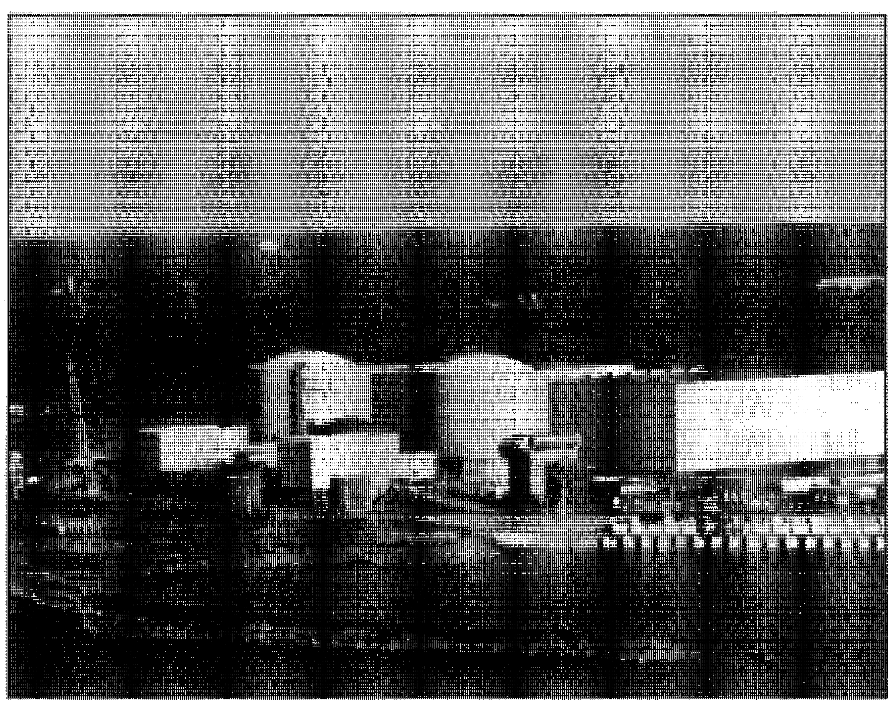


Table 2. Summary of 1990 Accident Sequence Precursor (ASP) Results

Conditional Core Damage Probability	*Plant Name*	*LER*	*Precursor Description*
1E–1 to 1		None	
1E–2 to 1E–1		None	
1E–3 to 1E–2		None	
1E–4 to E–3	Vogtle	190–006	*LOOP and both emergency diesel generators (EDGs) inoperable
	Fort Calhoun	90–020	EDG failure and similar problem on 2nd EDG
	Haddam Neck	90–008	Incorrect solenoid operated valve installation could gas-bind charging pumps
	Sequoyah 2	90–012	Gas accumulation in suction side of charging/HPI pump
	McGuire 1	90–017	Both emergency diesel generators inoperable
	Dresden	290–006	Stuck open safety relief valve followed by scram
1E–5 to 1E–4			11 events (includes one shutdown event)
1E–6 to 1E–5			11 events

*The evaluations this year include two precursor events while the reactors were "shutdown." In ASP analyses of previous years, shutdown events were not quantitatively evaluated.

higher—in the period from 1987 through 1989 (with one 1E–3 event in 1990). The overall volume of precursor occurrence has stayed relatively constant throughout the period, especially in the last four years shown.

Analyses of Human Performance in Operating Events. AEOD continued its program to expand the staff's understanding of human performance during reactor events. Under this program, teams of NRC staff and contractor specialists perform studies of selected events at plant sites shortly after the events occur. During fiscal 1991, studies were completed for eight events, as follows:

(1) Braidwood Unit 1 (Ill.) Loss of Reactor Coolant (10/04/90)

(2) Quad Cities Unit 2 (Ill.) Reactor Scram Caused by Erroneous Control Rod Withdrawal (10/27/90)

(3) Millstone Unit 3 (Conn.) Turbine Building Pipe Break (12/31/90)

(4) Oconee Unit 3 (S.C.) Loss of Decay Heat Removal (3/8/91)

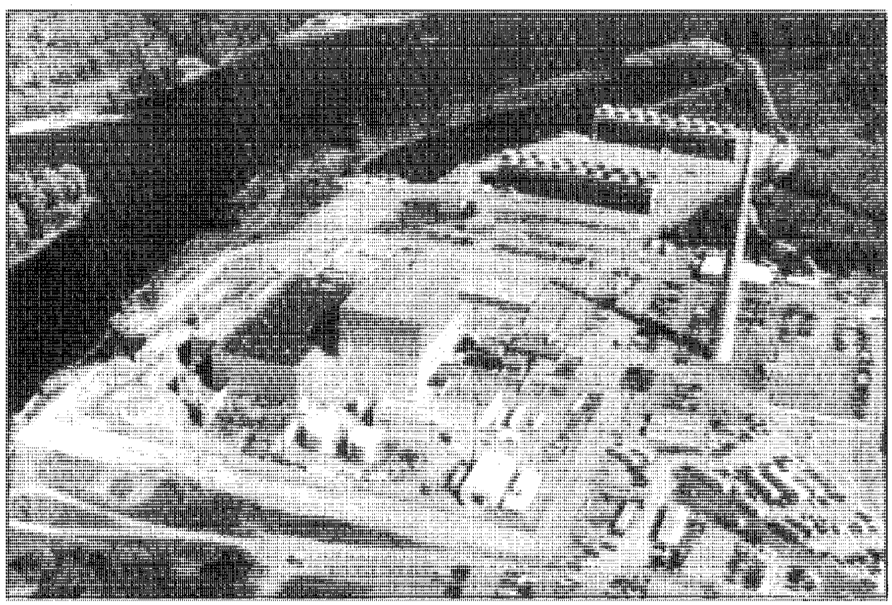
(5) Diablo Canyon Unit 1 (Cal.) Reactor Trip and Safety Injection (5/17/91)

(6) Monticello (Minn.) Reactor Trip Due to "Hi-Hi" Intermediate Range Monitor Channels (6/6/91)

(7) Waterford Unit 3 (La.) Manual Reactor Trip with Excess Steam Demand (6/24/91)

(8) Quad Cities Unit 2 (Ill.) Failure of Main Steam Isolation Valve (9/18/91).

The first AEOD human factors team study during this period was of an event at Braidwood Unit 1, which occurred on October 4, 1990, at 1:24 a.m., while the plant was in cold shutdown. In that event, approximately 600 gallons of reactor coolant was inadvertently discharged through a relief valve, resulting in contamination of licensee personnel. The study was conducted as part of a Region III Augmented Inspection Team (AIT) inspection. Region III issued the AIT report on October 23, 1990. Findings in the study concerned task awareness, coordination and teamwork. At the time of the event, two different tests were being performed simultaneously on the



Among the events involving human performance which were selected during the report period for special scrutiny was a reactor "trip," i.e., an unscheduled automatic shutdown of the reactor, at the Monticello (Minn.) plant, shown here. The facility houses a single boiling-water reactor; it is located on the Mississippi River, 45 miles above Minneapolis-St. Paul.

same system, under the direction of test engineers from the control room. The licensed senior control room operators, however, were unaware that the testing was in progress. It was noted that the test engineers had been on the job for more than 17 hours.

The second human factors team study involved a Quad Cities Unit 2 event that occurred on October 27, 1990, at 3:59 p.m., while the plant was in hot standby. The reactor scrammed on "hi-hi" intermediate range flux because, when the operator inserted the rods to decrease pressure, the reactor went subcritical, and, when the operator withdrew the rods to increase reactor pressure, he failed to recognize the need to follow procedures for re-establishing reactor criticality. The study found that the operations crew was not sufficiently aware that careful reactivity management is required while installing and removing test equipment, in order to avoid either super-criticality or short startup periods. The senior reactor operators did not adequately monitor control rod manipulations by the unit nuclear station operator. And the procedure governing operations when going from power operation to hot standby did not caution personnel about problems that might be encountered with high rod worths. Nor had the crew had been prepared and trained to handle the plant conditions. Finally, while an earlier shift had experienced high-notch worth, information regarding their experience was neither recorded nor passed along to later shifts. NRC Information Notice 91-04, "Reactor Scram Following Control Rod Withdrawal Associated With Low-Power Turbine Testing," was issued as a result of this event.

The third study concerned a Millstone Unit 3 event that occurred on December 31, 1990, with the reactor operating at 86 percent power. The study was performed as part of a Region I AIT inspection. During that event two six-inch-diameter moisture separator drain lines ruptured and discharged hot condensate system steam and water to the turbine building. The ruptures took place shortly after a licensed senior control operator (SCO) had manually closed a valve in one of the lines, as part of the process to isolate and repair a leak in the line. The SCO was able to return to the control room and report the failure. The control room operators manually initiated a turbine trip and a main steam line isolation, and began recovery measures that proved successful. The team study found that it may have been less than prudent for plant personnel to try to evaluate the significance of the through-wall leak without obtaining assistance from engineering personnel. There was apparently a lack of awareness that a through-wall pipe leak could be a precursor to a catastrophic failure. When the senior control operator (SCO) elected to personally isolate the leaking pipe section, control room command and control was temporarily degraded. The SCO escaped injury following the pipe rupture and returned to the control room, where he played an important role in recovery activity.

The fourth study was of an event at Oconee Unit 3, on March 11, 1991, at 9:00 a.m., in which the decay heat removal capability was lost for about 18 minutes during a refueling outage. The study was performed as part of a Region II AIT inspection. Several hours prior to the event, some technicians had requested authorization to perform testing on a Train A emergency sump suction valve. When the valve was opened by technicians, a gravity drain path was created from the hot leg. A blank flange, which was supposed to be installed between the valve and the sump, had been installed on the B Train line. The water level in the Reactor Vessel fell to the bottom of the hot leg causing a loss of shutdown cooling until the valve could be reclosed and water level restored. The event study found that procedures used for installation of the flange and the subsequent leakage testing did not provide sufficient information for properly identifying the line. Labels showing the correct penetration number for each line were available at the sump location, but were not given in the procedures for guidance. During the installation sequence, maintenance personnel acted in parallel, rather than independently, when verifying the flange location. Better communication would not have prevented this event, but it might well have facilitated a quicker termination of the loss of inventory. Operator actions were aided by the availability of diverse reactor level instrumentation during shutdown. For example, the operators initially doubled the level drop on the wide range instrument, but when the ultrasonic level instrument alarm was received, the crew took action to recover. It was noted that, since reactor coolant temperature is not measured directly, but at the RHR pump discharge, the operators were not aware of the extent of the temperature increase in the vessel. Although the consequences of this particular event were minor, the potential existed for the occurrence of more serious events, such as (1) exposure of irradiated fuel, if rapid draindown occurs while fuel is being moved; (2) more extensive flooding of the reactor building, if the borated water storage tank water source is not isolated; and (3) boiling in the core within a relatively short time, if reactor vessel inventory is not restored and decay heat removal is not returned to service.

The fifth study concerned an event at Diablo Canyon Unit 1, on May 7, 1991, at 6:28 a.m. In that event, the reactor automatically shut itself down from 100 percent power, because of an error by an instrumentation and controls technician. The technician took a nuclear instrumentation channel out of service, with another channel already out of service, and that act fulfilled the 2-out-of-4 trip logic that shuts down the reactor. Following the reactor trip, multiple steam dump valves "failed open," causing an excessive cooldown and depressurization of the primary system, which initiated a low pressurizer pressure safety injection. The study found that the control room crew worked quickly and effectively in responding to the trip and the safety injection. Several factors contributed to the surveillance test error. A calibration procedure did not comport with guidelines that would have made the error less likely; the technician had not completed training in individual verification; and the goal of completing the surveillance before shift change may have created a time-based stress. In addition, the technician, though still in training, was working without direct supervision. The control room annunciator "system acknowledge" circuit in the plant causes all blinking annunciator tiles to go to solid illumination and silences the alarm. Since the single acknowledge circuit affects all the alarms, there is an increased possibility that an incoming alarm may not be detected. Equipment problems can complicate decision-making and the conduct of an effective emergency response. But equipment problems should be anticipated and provided for in emergency operating procedures and in training.

The sixth study involved an event which took place at the Monticello facility on June 6, 1991, at 4:40 p.m., while operators were attempting to shut down the plant. The reactor automatically tripped because of "hi-hi" trips on both intermediate range monitor channels. The operators terminated a reactor startup and commenced a reactor shutdown because of a leaking safety relief valve. Because decay heat was low, the reactor cooled down and added positive reactivity to the core. The cooldown was not compensated for by the reactor operators, who allowed reactor power to increase, resulting in the reactor's automatic shutdown. The study found that the operators did not anticipate or understand the effect of the cooldown, nor did they question unusual plant responses. Operating procedures did not provide sufficient precautions and directions for the conditions that were possible. Finally, command and control by the operating crew did not ensure that plant evolutions were sufficiently planned and monitored.

The seventh study examined an event that occurred at Waterford Unit 3 on June 24, 1991, in which there was an excessive cooldown following a manual reactor trip. The event began when a lightning strike caused a turbine trip, which in turn caused an automatic power cutback to about 35 percent. Later, operators noticed that the steam generator (SG) #2 level was increasing and could not be controlled; they manually shut down the reactor. Following the trip, primary system temperature and SG pressure dropped rapidly, and the operators undertook a main steam isolation system actuation. The event study found that the overall response by the control room operators following the reactor trip was effective and timely. The control room operators felt that training was an important factor in their ability to respond to this excessive steam demand event. The steam generator high-level alarm was at 87.6 percent, and the high-level trip setpoint was at 87.7 percent. The shift supervisor felt he had no practical choice but to order the reactor scram when the high-level alarm setpoint was reached.

The eighth study concerned an event at the Quad Cities plant. On September 18, 1991, at 6:05 p.m., Quad Cities Unit 2 was in an end of cycle coastdown, when the "B" main steam line was isolated, and power spiked from 83 percent to 98 percent; the surge was not detected by the control room crew until over three hours later. The inboard "B" main steam isolation valve disc had separated from the stem and restricted flow in the "B" main steam line, causing reactor pressure to increase from 984 pounds-per-square-inch to 1,018, resulting in fluctuations in power, level, and core flow; but no alarms were activated because no setpoints were exceeded. The event study found that there was a low level of task awareness on the part of the unit operator. Command, control and communications were insufficient to detect the off-normal condition. Technical guidance did not exist for normal operations within alarm setpoints. Finally, there was insufficient control during an earlier reassembly.

Studies to date of this event have identified such human performance issues as task awareness, command and control, communications, stress, man-machine interface, simulator fidelity, shift technical advisor role, tagging, administrative controls, procedures, 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. An interim report will be issued in early 1992 summarizing the results and potential generic findings of the program.

Performance Indicator Program. The Performance Indicator (PI) Program is one element in the NRC's continuous monitoring of the performance of licensees operating commercial nuclear power plants in the United States. Under the direction of AEDO, the program currently calls for scrutiny of industry-wide data on eight indicators, in an effort to detect significant performance trends. The eight PIs are (1) the number of unplanned automatic reactor scrams while a reactor is critical, (2) the number of safety system actuations, (3) the number of significant events, (4) the number of safety system failures, (5) the forced outage rate, (6) the number of equipment-forced outages-per-1,000 commercial critical hours, (7) the collective radiation exposure, and (8) "cause codes." Each quarter, the AEOD staff provides a report containing plant-specific data for these eight indicators to the Commission and to NRC senior managers. These reports are placed in the NRC Public Document Room, and the staff transmits plant-specific information and industry average data extracted from each PI report to licensee managers.

Shown in the graphs are overall industry averages for the five recent calendar years for all of the indicators except the cause codes (for which industry-wide trends are not calculated). In computing these averages, data were excluded for the period when a plant (1) was in an extended shutdown that required Commission approval before either a startup or operation above low power could take place, or (2) was no longer in commercial operation. The trends indicate continued improvement in overall performance, although at a diminishing rate of improved performance.

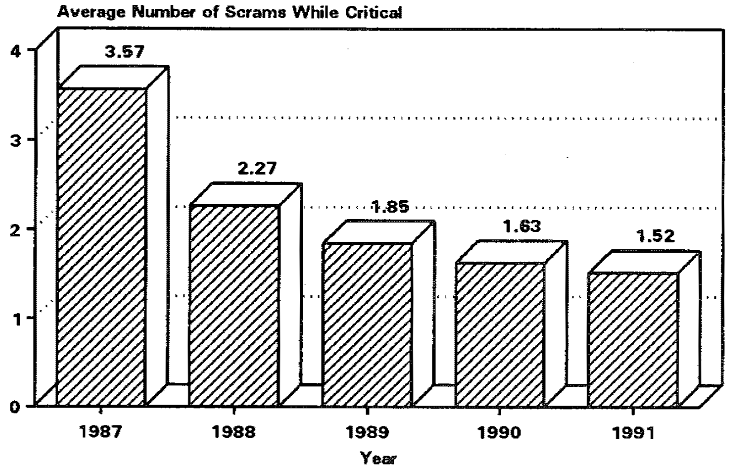
The PI Program is a single, coordinated, comprehensive NRC program that provides a useful perspective on operational performance that enhances the NRC's ability to recognize changing performance in operating plants with possible safety implications. The program is only a tool, to be used in conjunction with other tools—such as the results of routine and special inspections and the Systematic Assessment of Licensee Performance—to furnish the data to NRC managers by which they can decide whether any plant-specific regulatory requirements need adjusting. The PIs for a given plant, taken as a set, provide an additional source of data for appraising the meaning of changes in plant operational performance. The PIs often focus attention on the need to assess and understand underlying causes of identified changes by evaluating other available information.

During fiscal year 1991, the AEOD staff continued efforts to upgrade the PI Program through (1) creation of appropriate peer groups (of similar plants) among which to carry out comparisons of the performance of individual plants with the average performance of the group; (2) development of a methodology to account for the cyclic nature of cause code data during the operating cycle; (3) sponsorship of development by the NRC's Office of Nuclear Regulatory Research (RES) of a risk-based indicator of safety system unavailability; and (4) participation in the International Atomic Energy Agency (IAEA) program for the development of performance indicators.

AEOD staff proposed setting up peer groups (groups of comparable plants) among which to conduct comparisons of the reported event data. Nine peer groups, drawn up primarily according to design and regulatory issues, are to be evaluated, along with the operating cycle methodology described below, in a test to be carried out in fiscal year 1992.

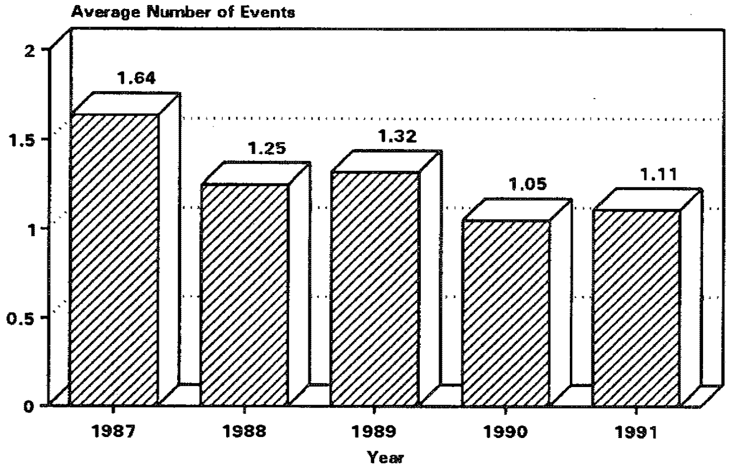
Early in its peer group development effort, AEOD staff recognized that cause code data were cyclic, with a period approximating the refueling interval. Further investigation of the phenomenon led to the conclusion that a plant's operating phase—startup, power operations, refueling, etc.—could have a considerable effect upon event reporting. AEOD therefore identified those phases of operation in which event reporting varies significantly, and then began development of a methodology to present plant trends and deviations as a function of operating phase. Upon completion of this study, the methodology will be evaluated, along with the peer group comparisons, in a test program during fiscal year 1992.

Average Number of Reactor Scrams While Critical



(1991 PROJECTED FROM 9 MONTHS OF DATA)

Average Number of Safety System Actuations



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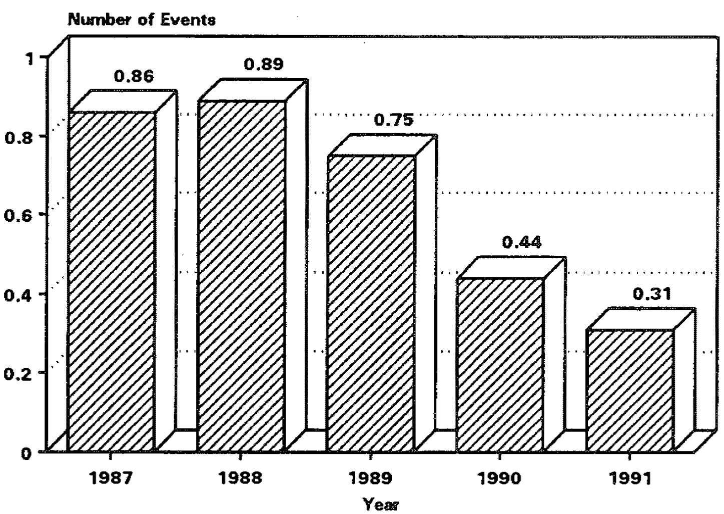
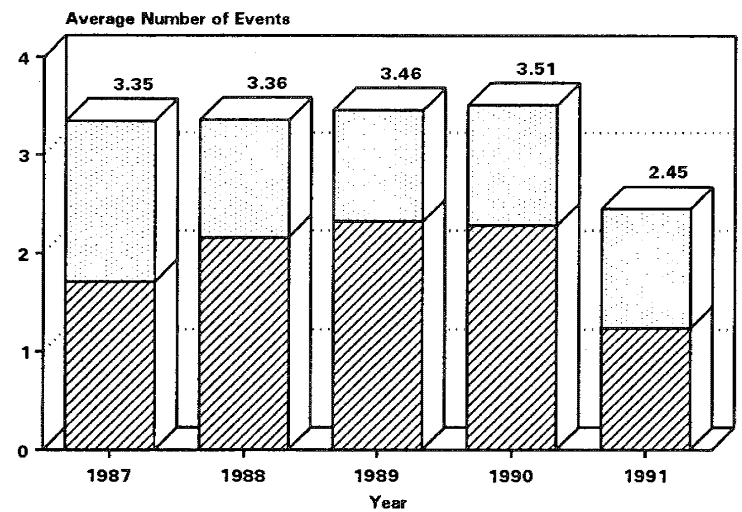
Average Number of Significant Events



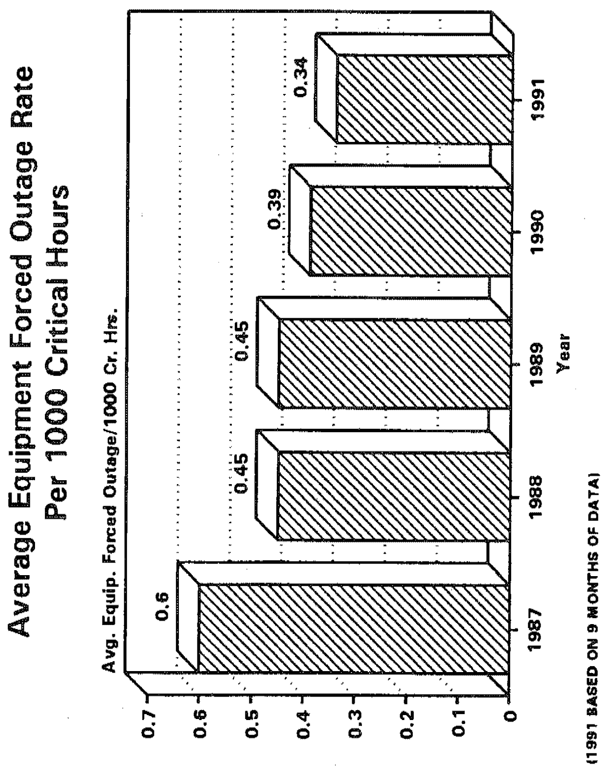
(1991 PROJECTED FROM 6 MONTHS OF DATA)

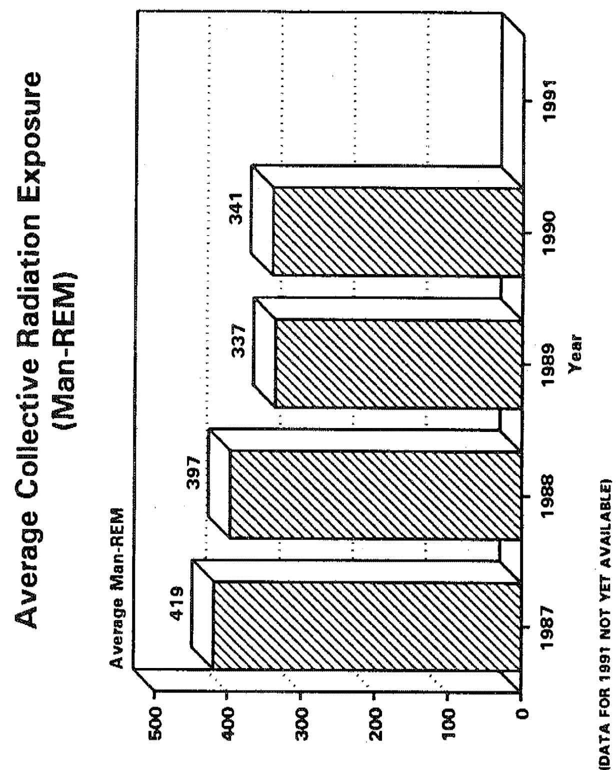
Average Number of Safety System Failures

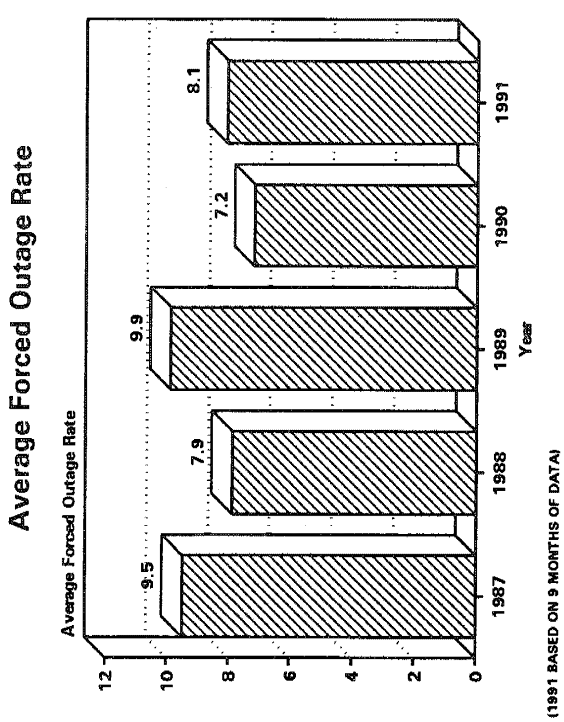


(1991 PROJECTED FROM 9 MONTHS OF DATA)

LIGHTER AREAS SHOW ADDED COUNTS FROM IMPROVED CLASSIFICATION PROCEDURES







The graphs on these two pages depict five-year trends for Performance Indicators (PI) 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 in that manner 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 evidence a continued improvement in overall performance, but at a diminishing rate of improvement.

In fiscal year 1991, RES continued to evaluate an indicator of the unavailability of selected risk-significant safety systems. The candidate indicator is the product of the fractions of time during plant operation in which each train of the system would not have functioned on demand. A persistently high value for this indicator could indicate that plant programs have not prevented the safety-system availability from degrading.

Since 1985, the IAEA has been involved in the development and use of performance indicators as part of its Operational Safety Indicator Program. Since 1986, AEOD has provided the IAEA with expert consultants to contribute to the development of indicators, and has also furnished evaluations by the Operational Safety Review Team and the Assessment of Safety Significant Events Team. In November of 1990, the IAEA convened a Technical Committee Meeting (TCM) on "The Exchange of Experience in Managing Nuclear Power Plant Safety Performance Using Numerical Indicators," to discuss nuclear power plant management practices seeking to gauge and to improve plant performance using numerical indicators. AEOD sent a representative to this TCM, which proved a useful exchange of information on current and future activity involving performance indicators. The meeting brought out the fact that the need for and uses of performance indicators on the part of regulatory bodies differ from those on the part of plant operators.

Analyses of Non-Reactor Operational Experience

Another AEOD responsibility 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.

A Quality Management rule for medical licensees was published by the NRC in 1991, containing new definitions for the kinds of misadministrations that must be reported to the NRC. The rule, to become effective in January 1992, is expected to reduce substantially the number of reports of diagnostic misadministrations received by the NRC.

In 1990, AEOD produced a videotape on the subject of desirable practices in preparing and administering radiopharmaceuticals. The videotape used data from reported medical misadministrations to identify those practices that result in the most frequent types of errors. The videotape illustrates practices designed to avoid errors in preparing and administering radiopharmaceuticals. The NRC staff developed the video with support from Oak Ridge Associated Universities and 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 have been distributed to all NRC medical licensees and regulatory agencies for the Agreement States.

During fiscal year 1991, the AEOD issued two surveys that included a review of 1990 non-reactor and medical misadministration reports. These reports were published in the *1990 AEOD Annual Report* (NUREG-1272, Vol. 5, No. 2). The non-reactor reports issued in fiscal year 1990 are listed in Table 3.

Report on 1990 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 1990, 24 non-reactor events were reported to the NRC, in which 30 individuals received exposures that were greater than those permitted by NRC regulations. All of the individuals were associated with NRC licensees. Most of these overexposures represent doses that exceed the quarterly regulatory limits by a small amount. There were four exposures to radiographers in which individuals received extremity or local exposures that ranged from 100-to-several-thousand rems.

Other types of incidents reported concerned lost, stolen or abandoned materials, or leaking sources. None of the events reported to the NRC in 1990 had a significant impact on public health and safety.

In May 1991, the NRC was notified that uranium in concentrations that exceeded the allowable limit had been added to a tank at the General Electric fuel fabrication facility at Wilmington, N.C. The uranium-bearing solution was removed without incident. An Incident Investigation Team (IIT) was set up to investigate the incident. One staff action resulting from the IIT was to expand the independent AEOD review of operating experience to include fuel fabrication facilities and to examine the independent review program for licensee groups not in the scope of AEOD activity. Staff action on this matter was under way at the close of the report period.

Medical Misadministration Report. The 467 misadministration reports received during 1990 involved 573 patients. Of these reports, 443 reports concerned diagnostic misadministrations and 24 concerned therapy misadministrations. Besides the 24 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 the dose for the diagnostic procedures for which they were scheduled.

Table 3. Non-Reactor Reports Issued During FY 1990

Subject	*Issued*
Report on 1990 Non-reactor Events, NUREG-1272, Vol. 5, No. 2, Appendix A	1991
Report on 1990 Medical Misadministration Events, NUREG-1272, Vol. 5, No. 2, Appendix B	1991

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

The error rate for therapy misadministrations ranged from 0.0002-per-procedure for brachytherapy and radiopharmaceutical therapy to 0.0003-per-patient for teletherapy; the error rate for diagnostic misadministrations reported was about 0.0001-per-procedure.

ABNORMAL OCCURRENCES

The NRC prepares a quarterly Report to Congress on Abnormal Occurrences (NUREG-0090 series), which also serves to communicate 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., or at Local Public Document Rooms (LPDRs) throughout the country (see Appendix 3 for list of LPDRs)).

There were five abnormal occurrence (AO) reports issued in fiscal year 1991: NUREG-0090, Vol. 13, No. 2 (April-June 1990); Vol. 13, No. 3 (July-September 1990); Vol. 13, No. 4 (October-December 1990); Vol. 14, No. 1 (January-March 1991); and Vol. 14, No. 2 (April-June 1991). There were no AOs reported for nuclear power plants. The five reports describe two AOs at fuel cycle facilities, 20 AOs at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc,), and six AOs reported by the Agreement States. The reports also update the status for certain AOs previously reported.

A list of the AOs reported in the reports cited above is given in Table 4, and each one is described below. Seven of the events (AOs 90-12, 90-14, 90-16, 90-18, 90-20, 90-22, and 91-1) resulted in civil penalties proposed by the NRC and four of the events (AOs 90-11, 90-12, 90-22, and 90-24) resulted in orders being issued by the NRC (see Appendix 6 for a list of all civil penalties proposed and orders issued by the Office of Enforcement during the report period, with capsule descriptions of the reasons therefor). One of the events (AO 91-6) was considered of potentially major significance and therefore was investigated by an NRC Incident Investigation Team (see "Incident Investigation Program" later in this chapter).

Abnormal Occurrences at Fuel Cycle Facilities

Significant Degradation of Plant Safety at Nuclear Fuel Services, Inc. Nuclear Fuel Services, Inc., in Erwin, Tenn., is a fuel production facility that produces nuclear fuel for the United States Navy. On November 30, 1990, licensee personnel discovered that, two days earlier, 395 grams of uranium-235, contained in liquid waste, had been processed through the waste water treatment system for collection and disposal of the uranium. This quantity was above the administrative criticality safety limit of 350 grams for the "unfavorable geometry" tanks used to hold the waste. (An "unfavorable geometry" tank refers to a process vessel that can hold enough uranium to produce criticality, or self-sustained fission, among the uranium atoms.)

While the amount of uranium–235 was well below the amount needed for criticality, the circumstances associated with the event were particularly safety significant. Highly concentrated uranium solutions in an adjoining part of the process were available in quantities that were more than sufficient to have caused a criticality accident in the unfavorable geometry tank. The hydrostatic head associated with those highly concentrated solutions would have been sufficient to force those solutions into the unfavorable geometry tank if the set of normally closed valves were faulty or were not fully closed.

Filling of storage tanks with liquid waste from the solvent extraction system in the high-enriched uranium recovery process began on November 27, 1990. When the tanks were full, the contents were recirculated, prior to sampling. An operator collected two samples of the liquid and submitted them for analysis. The analytical results were received on November 28, 1990, and revealed that the uranium concentration in the liquid was well below the authorized discard limit; hence, the quantity of uranium–235 was below the safety limit of 350 grams. The liquid waste was then pumped to another tank where it was mixed again, sampled for material accountability purposes, and then pumped to the Waste Water Treatment Facility (WWTF).

On November 30, 1990, the laboratory reported the results of the accountability sample to be above the authorized discard limit. This higher concentration was confirmed by analysis of another sample which had been obtained when the liquid was received at the WWTF. These analyses confirmed each other, and all discharges were halted as a special licensee investigation team undertook a detailed review to determine the cause and needed corrective actions. At about 4:15 p.m., the licensee reported the incident to the NRC.

The licensee identified the probable causes of the November 28 event to be (1) less than adequate piping layout that allowed uranium solutions to flow into the unfavorable geometry tank, and (2) personnel-related inadequacies, in that operators had no knowledge of the potential for crossover of highly concentrated uranium solutions into unfavorable tanks as the result of open valves or of other anomalies in the piping systems.

Following its review of the incident, the NRC concluded that there appeared to be other root causes besides those given by the licensee. These causes included:

(1) The safety basis for the plant was less than adequate because a documented safety analysis was not available.

(2) As a result of the lack of a detailed safety analysis, equipment important to safety, such as valves, were not properly identified, protected, emphasized in plant control documents and training sessions, or tested and maintained appropriate to their safety function, and did not possess positive closure indication.

(3) The design basis of the plant was less than adequate. The system drawings lacked adequate detail.

The licensee had missed an opportunity to eliminate the problems several years earlier, when modifications were made to the piping system. The licensee's reviews of the modifications had failed to identify the significant potential for uranium solutions to flow into unfavorable geometry vessels.

Corrective actions included modification of the piping system to prevent highly concentrated uranium solutions from flowing into the unfavorable geometry tanks. A review of the fuel recovery facility was initiated to identify the nuclear safety features and controls for each unfavorable geometry vessel. A Nuclear Criticality Safety Performance Improvement Program (PIP), that had been instituted prior to the incident, was accelerated and expanded to address the root causes of the problem. Training was also given to fuel recovery personnel to make them aware of it.

The NRC proposed a civil penalty of $10,000, which has been paid. In early 1991, the NRC prepared an action plan for the licensee's facility. The plan, which is updated quarterly, tracks the completion of the licensee's PIP items and calls for quarterly NRC and licensee management meetings on the PIP status, as well as NRC technical reviews of the PIP. Other items addressed in the plan include license renewal milestones and management meetings on decommissioning activity. A full-time NRC resident inspector began service at the facility on April 22, 1991.

Potential Criticality Accident at the General Electric Nuclear Fuel and Component Manufacturing Facility. On May 29, 1991, management of the General Electric Nuclear Fuel and Component Manufacturing facility in Wilmington, N.C., notified the NRC that it had identified higher than expected amounts of uranium in a process tank of the waste treatment system, posing a potential criticality safety problem. The amount was approximately 2,300 parts-per-million, or 150 kilograms total uranium (about 4 percent enriched in uranium–235). The administrative criticality safety limit for transferring uranium into the process tank vessel (an "unfavorable geometry" tank (see above)) was 150 parts-per-million.

Table 4. Abnormal Occurrences Reported During FY 1991

OCCURRENCES AT NUCLEAR POWER PLANTS

AO Number	*Subject*	*NUREG–0090 Issue*
		None reported during FY 1991

OCCURRENCES AT FUEL CYCLE FACILITIES

AO Number	*Subject*	*NUREG–0090 Issue*
91–1	Significant Degradation of Plant Safety at Nuclear Fuel Services, Inc.	Vol. 14, No. 1 June 1991
91–6	Potential Criticality Accident at the General Electric Nuclear Fuel and Component Manufacturing Facility	Vol. 14, No. 2 September 1991

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

AO Number	*Subject*	*NUREG–0090 Issue*
90–11	Deficiencies in Brachytherapy Program	Vol. 13, No. 2 October 1990
90–12	Radiation Overexposure of a Radiographer	Vol. 13, No. 2 October 1990
90–13	Medical Diagnostic Misadministration	Vol. 13, No. 2 October 1990
90–14	Administration of Iodine–131 to a Lactating Female with Uptake by October 1990 Her Infant	Vol. 13, No. 2
90–15	Medical Therapy Misadministration	Vol. 13, No. 2 October 1990
90–16	Medical Therapy Misadministration	Vol. 13, No. 3 January 1991
90–17	Medical Diagnostic Misadministration	Vol. 13, No. 3 January 1991
90–18	Significant Breakdown in Management and Procedural Controls at a January 1991 Medical Facility	Vol. 13, No. 3
90–19	Medical Diagnostic Misadministration	Vol. 13, No. 3 January 1991
90–20	Medical Diagnostic Misadministration	Vol. 13, No. 3 January 1991
90–21	Medical Therapy Misadministration	Vol. 13, No. 4 March 1991
90–22	Radiation Overexposure of a Radiographer	Vol. 13, No. 4 March 1991
90–23	Medical Therapy Misadministration	Vol. 13, No. 4 March 1991
90–24	Radiation Overexposure of a Radiographer	Vol. 13, No. 4 March 1991
90–25	Medical Diagnostic Misadministration	Vol. 13, No. 4 March 1991
91–2	Medical Diagnostic Misadministration	Vol. 14, No. 1 June 1991

Table 4. Abnormal Occurrences Reported During FY 1991
(continued)

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

AO Number	*Subject*	*NUREG–0090 Issue*
91–3	Medical Therapy Misadministration	Vol. 14, No. 1 June 1991
91–4	Medical Therapy Misadministration	Vol. 14, No. 1 June 1991
91–5	Medical Therapy Misadministration	Vol. 14, No. 1 June 1991
91–7	Multiple Medical Teletherapy Misadministrations	Vol. 14, No. 2 September 1991

OCCURRENCES AT AGREEMENT STATE LICENSEES

AO Number	*Subject*	*NUREG–0090 Issue*
AS90–1	Medical Diagnostic Misadministration	Vol. 13, No. 2 October 1990
AS90–2	Medical Therapy Misadministration	Vol. 13, No. 3 January 1991
AS91–1	Medical Therapy Misadministration	Vol. 14, No. 1 June 1991
AS91–2	Overexposure of a Non-radiation Worker	Vol. 14, No. 2 September 1991
AS91–3	Extremity Overexposure of a Radiation Worker	Vol. 14, No. 2 September 1991
AS91–4	Overexposure of a Radiographer	Vol. 14, No. 2 September 1991

During the morning of May 29, the licensee identified higher than expected amounts of uranium in a favorable geometry vessel in its solvent extraction system, the result of earlier problems with controls and equipment in that system. The licensee, having shut down the solvent extraction process, discovered that higher than expected amounts of uranium had also been improperly transferred into an unfavorable geometry waste tank. Licensee management was notified and a technical evaluation team was convened. In addition, sparging (i.e., mixing) was initiated in this tank to minimize the criticality potential by preventing an accumulation of material in the bottom of the tank. During the afternoon on May 29, the licensee notified NRC Region II of the incident. Later, the licensee began uranium recovery operations from this tank, by means of a centrifuge linked to the tank.

That same day, May 29, the NRC dispatched a Region II site team and activated the Headquarters and Region II Incident Response Centers. The site team arrived early on the morning of May 30. At 6:38 a.m., eastern daylight time, following discussions with the NRC response centers, the licensee declared an Alert in accordance with its Radiological Contingency and Emergency Plan.

On May 31, the NRC Executive Director for Operations (EDO) requested that the site team be upgraded to an eight-member NRC Incident Investigation Team (IIT). The licensee continued to remove uranium by centrifuge from the tank through June 1. By that day, the licensee had transferred sufficient amounts of solution containing uranium from the tank, via the centrifuge process, to other nearby tanks to reduce the uranium in the tank to an amount less than the criticality safety limit. The licensee then terminated the Alert status and the NRC went to a normal response mode in both its headquarters and regional response centers.

The IIT identified numerous problems at the plant, including inadequate management oversight, design deficiencies, procedural noncompliance, inadequate incident investigation, and a general deterioration of criticality controls. IIT conclusions regarding the interrelated root causes that contributed to the incident are described under Incident Investigation Program, later in this chapter.

Corrective actions taken by the licensee following this incident included: system "walkdowns" and verification that documentation matches current plant configuration; revision of procedures; retraining of operators; revamping of sampling procedures to ensure adequacy for the measurement of uranium; sensitivity training for all plant personnel to the importance of following procedures and reporting problems; documenting of a scheme for reporting events; additional management oversight of operators; establishment of an audit system; and development of a long term plan to improve performance in staffing, emergency response, equipment reliability, and in the engineered systems intended to replace administrative criticality controls. The license reported the status of short and long term corrective actions to NRC Region II on a biweekly basis.

The NRC IIT formal report—"Potential Criticality Accident at the General Electric Nuclear Fuel and Component Manufacturing Facility, May 29, 1991" (NUREG-1450)—was published in August 1991. Based on the IIT's findings, the EDO issued a memorandum, on August 13, 1991, defining and assigning NRC staff responsibility for generic and facility-specific actions. The resolution status or disposition of each IIT staff action will be covered in the AEOD annual reports (NUREG-1272 series).

Abnormal Occurrences Involving Other NRC Licensees

Physician Gives Brachytherapy Without Evaluation of Plans. On March 28, 1990, the NRC received allegations pertaining to brachytherapy treatments at the St. Mary Medical Center facilities in Gary and Hobart, Ind. The NRC also conducted a special inspection at Porter Memorial Hospital in Valparaiso, Ind. Although the original allegations did not include Porter Memorial Hospital, the NRC inspection was made because brachytherapy procedures at that facility were performed by the same physician as those at the St. Mary facilities.

The allegations made to the NRC affirmed that the physician, an authorized user of the licensed material, did not evaluate patients' treatment plans before giving treatment, and that the patients, therefore, did not receive the proper, prescribed dose of radiation from the brachytherapy. The NRC concluded that the two St. Mary facilities were not exercising adequate management control to assure that NRC requirements were being met.

The NRC determined that adequate records had not been maintained at the Porter Memorial Hospital to permit evaluation as to whether the brachytherapy procedures had been administered as prescribed and planned.

Following the NRC inspections, the NRC issued Orders to the three facilities suspending brachytherapy procedures until appropriate corrective actions are taken.

Radiographer's Assistant Wraps Source Around Neck. On the evening of April 6, 1990, Barnett Industrial X-Ray of Stillwater, Okla., notified the NRC that an incident had occurred that evening, while a radiographer and his assistant were working at a temporary job site in Ardmore, Okla. The radiographic operation involved the use of a radiography device containing an iridium-192 sealed source of approximately 80 curies of radioactivity. The licensee reported that the source had become disconnected from its drive cable and had remained in the source guide tube.

Failing to conduct a radiation survey of the exposure device, and thus being unaware that the source had remained in the tube, the assistant, having disconnected the source guide tube from the radiography device, wrapped the source guide tube around his neck while he moved equipment at the work site. The licensee's initial estimate was that the assistant received an exposure of 4,000 rems to the exposed area of his neck.

Both the assistant and radiographer were referred to a radiation oncologist for examination and blood samples were obtained. The cytogenetic studies revealed equivalent whole-body doses of 17 rems for the radiographer and 24 rems for the assistant. The assistant developed erythema, or reddening, on the left side of his neck, which later showed signs of more significant damage to skin tissue, in an area approximately 10 centimeters in diameter. The oncologist determined that the observed effect corresponded to a local skin dose of 5,000-to-7,000 rems. As of June 1990, the skin tissue in the individual's neck had regenerated, and the physician did not predict any long term effects as a result of this exposure. The assistant remains under the physician's care. There were no medical effects observed for the radiographer.

The NRC issued an Order prohibiting the radiographer and assistant from participating in licensed activity. The Order was later relaxed, after the licensee implemented corrective actions. The NRC staff proposed imposition of a civil penalty in the amount of $7,500.

Procedures Manual Gives Wrong Dosage. On June 8, 1990, Mercy Memorial Medical Center in St. Joseph, Mich., reported a diagnostic misadministration to the

NRC. A 70-year-old female patient was scheduled to undergo a diagnostic evaluation to determine whether she was suffering from an enlarged thyroid gland (substernal thyroid). No prescribed radiation dose was indicated. The technologist, attempting to order the proper amount of radioactive material, noted that her standard dose chart (created by authorized users) did not list a dosage for a substernal thyroid gland study. The technologist consulted the department's procedures manual, which indicated that the proper dose for a substernal thyroid gland study was 3-to-5 millicuries of iodine-131, or 100-to-200 microcuries of iodine-123. The technologist then asked an authorized user which isotope to use. The authorized user told the technologist to order enough iodine-131 to permit visualization of the thyroid gland. On June 5, 1990, the patient was given 4.3 millicuries of iodine-131, a dosage which conformed to the procedures manual. The dosage listed in the manual, however, was wrong. The standard dose for a substernal thyroid scan should have been 50-to-100 microcuries of iodine-131. (One millicurie is equal to 1,000 microcuries.) The mistake was discovered by the Chief of the Nuclear Medicine Department on June 6 and reported as a misadministration to the NRC on June 8, 1990.

The licensee estimated that the misadministration resulted in a mean dose to the thyroid gland of 5,752 rads. The NRC's medical consultant investigated the case and, on the basis of certain assumptions, the estimated the dose to be 3,400 rads to the thyroid gland; according to the consultant, this dose would yield a 10 percent chance of hypothyroidism in the patient, developing over the subsequent five years. The licensee is monitoring the patient's condition.

Treatment of Nursing Mother Irradiates Infant's Thyroid. On June 19, 1990, a nursing mother at the Tripler Army Medical Center in Honolulu, Haw., was given a 4.89 millicurie dose of iodine-131 that resulted in an unintentional radiation dose to her infant's thyroid gland, estimated at 30,000 rads, and a dose to the infant's whole-body of 17 rads. The error was detected on June 21, 1990, when the patient returned to the medical center for a whole-body scan, which indicated an unusually high breast uptake of iodine-131. In the opinion of the patient's physician and an NRC medical consultant, the infant's thyroid function will be completely lost. The infant will require artificial thyroid hormone medication for life to ensure normal growth and development.

The physician and nuclear medicine technologist failed to make sure that the patient was not breast-feeding. The patient had arrived at the medical center from a remote South Pacific island; communication between the island physician and the Army physicians was poor, and the Tripler hospital physicians were not aware that the patient had given birth on June 1, 1990.

The NRC proposed a civil penalty in the amount of $5,000, which was later reduced to $2,500, after consideration of the licensee's representations. The civil penalty has been paid.

Lung-cancer Patient Given Radiation to the Brain. On June 22, 1990, St. Luke's Hospital in Cleveland, Ohio, reported that a 57-year-old woman being treated for lung cancer had, that day, erroneously been given a 178-rem radiation dose to the left side of the head, instead of the intended 200-rem radiation dose to the chest area. A technologist had set the patient up for brain irradiation without looking at the treatment documents. Because the misadministration involved a single treatment and because of the dosage involved, no adverse medical effects were expected by the licensee.

Treatment Simulation in Prone Position Results in Wrong-side Irradiation. On September 19, 1990, the Muskogee Regional Medical Center in Muskogee, Okla., notified the NRC that a therapy misadministration had occurred involving treatment administered from February 20 to March 12, 1990. The radiation oncologist had identified the treatment error on September 6, 1990, but had not immediately recognized it to be reportable. The error involved administration of 2,160 rads to the right posterior neck of a patient, rather than to the left posterior neck, as prescribed.

The oncologist had initially participated in the treatment simulation and had approved simulation radiographs before the treatment. But the physician failed to notice that the wrong side of the patient's neck had been the subject of the simulation. The mistake was attributed to the fact that the treatment was simulated with the patient in the prone position, rather than in the routine supine position.

The oncologist had palpated an enlarged cervical lymph node on the patient's left side during a September 6, 1990 physical examination, which prompted his subsequent review of the treatment chart and identification of the error. All the records indicated that the right side of the patient's neck was treated, although the prescription clearly indicated that treatment was to be given to the left side.

The licensee's radiation oncologist has advised the NRC that no adverse effects were observed during routine follow-up examinations and that no significant effects are anticipated as a result of the misadministration.

The NRC proposed a civil penalty in the amount of $1,250, which has been paid.

Nuclear Medical Department Does Iodine–131 Scan Before Prescription Arrives. On June 1, 1990, NRC was notified by Overlook Hospital in Summit, N.J., that a diagnostic misadministration involving iodine–131 had occurred at the hospital. An outpatient had been scheduled for a nuclear medicine study in response to a phone call from the referring physician's office. The Nuclear Medicine Department personnel understood the doctor's request to be for an appointment for an iodine–131 scan. The patient brought the written prescription to the outpatient department and then proceeded to the Nuclear Medicine Department for the scheduled study. The written prescription was not received by the Nuclear Medicine Department until after the study was completed. When the Nuclear Medicine Department received the written prescription, the referring physician's written prescription requested a thyroid scan, not an iodine–131 scan. The patient had a normally functioning thyroid.

The intended dose to the patient's thyroid was approximately four rads from 300 microcuries of iodine–123. The administered dose to the thyroid was approximately 1,820 rads, from 1.4 millicuries of iodine–131. The licensee does not expect any significant consequences to the patient.

Significant Breakdown in Management and Procedural Controls. On August 14, 1990, North Detroit General Hospital in Detroit, Mich., reported to the NRC that films from diagnostic nuclear medicine studies were apparently fraudulent.

The films involved 30 studies performed on 27 patients, from July 19–to–27, 1990. During this period, a replacement technologist, supplied by a temporary services contractor, was engaged by the licensee. The licensee subsequently discovered that the films for 29 of the 30 procedures were fraudulent or indeterminate and were, therefore, unreliable for patient diagnosis. The films in question showed evidence of tampering. Their fraudulent character was revealed when a staff technologist made comparisons of them with later films, after the contract technologist had left. The licensee then reviewed the films from procedures performed by the contract technologist. The licensee's investigation determined "conclusively that the individual had doctored and provided fraudulent nuclear medicine studies for interpretation. The technologist had submitted nuclear medicine studies on patients who had previously been imaged within the department during the past two years and altered the names on those images and placed the names of the patients he was to have performed studies on in their place."

The licensee was unable to determine, in most cases, whether the diagnostic procedures had actually been performed or whether the patients had been administered the prescribed radiopharmaceutical for the procedures. The diagnostic procedures, with one exception, were not considered to be valid, were thus of no use in their intended diagnostic function. The licensee offered to do the procedures over, but some patients or their physicians elected not to have the studies performed again. (In those instances in which a second procedure was performed, the patient received additional radiation exposure.) The radiation doses associated with diagnostic procedures are small.

The NRC proposed a civil penalty in the amount of $2,500, which has been paid.

Wrong Iodine–131 Capsules Used in Thyroid Treatment. On August 14, 1990, NRC was notified by Copley Hospital in Morrisville, Ver., that a diagnostic misadministration involving iodine–131 (I–131) had occurred at the hospital on August 7, 1990. A 63-year-old woman undergoing I–131 treatment for primary hypothyroidism was administered 112 microcuries, instead of the routinely prescribed 10 microcuries. The hospital reported that a supply of I–131 capsules had been ordered with incorrect amounts of I–131. The dose to the thyroid, based upon the results of an uptake scan, was calculated at 3.9 percent uptake, resulting in an estimated actual dose to the thyroid of 29 rads. The licensee does not expect any adverse consequences to the patient.

Part-time Technician Administers Overdose of Technetium. On September 24, 1990, a consultant to West Shore Hospital in Manistee, Mich., informed NRC that an 84-year-old female cancer patient received a 175–millicurie dose of a technetium–99m (Tc–99m) labeled radiopharmaceutical for an imaging scan of her gall bladder, instead of the eight-millicurie dose prescribed in the Nuclear Medicine Department's procedures manual.

The radiopharmaceutical was prepared and administered by a part-time technician who had received only two weeks of training in the Nuclear Medicine Department procedures the previous February and had performed only two nuclear medicine procedures since.

An NRC consultant evaluated the medical consequences of the incident and concluded that no biological effects should be expected from the misadministration. It is estimated that the doses to the patient's bladder and upper large intestine were about 36 rads and 26 rads, respectively.

The NRC proposed a civil penalty in the amount of $4,375, which has been paid.

Iodine–125 Seeds Implanted Too Deeply in Prostate. On August 29, 1990, 86 "seeds" of iodine–125 (small sealed radiation sources) were permanently implanted in an 86-year-old patient at the University of Cincinnati, in

Cincinnati, Ohio. The seeds contained a total of 27.5 millicuries of iodine-125. A dose of 16,000 rads was prescribed for the prostate gland. The seeds were to be implanted in the prostate using an ultrasonic probe to view and position the implants.

Subsequent review by the licensee determined that most of the seeds had been implanted too deeply and had passed through the prostate into the surrounding tissue. Many of the seeds were 5-to-10 centimeters beyond the prostate gland. As a result, the radiation dose to the prostate was negligible, compared to the prescribed dose of 16,000 rads. while, the licensee estimated, the patient received a dose of 15,000 rads to the tissue beyond the prostate gland; this was a dose considerable greater than what would have been received if the seeds had been positioned as intended.

The primary cause of the misadministration appeared to be the difficulty of viewing the prostate area while using the ultrasonic probe. The licensee does not anticipate any significant effects to the patient as a result of the misadministration.

Radiographer Removes Dosimeters to Conceal Overexposure. During the evening of October 5, 1990, Western Stress, Inc., of Houston, Tex., notified the NRC that an incident had occurred earlier that evening while a radiographer and his assistant were working at a temporary job site in Bordentown, N.J. The radiographic operation involved the use of a radiography device containing an 80.5-curie, iridium-192 sealed source. The licensee reported that the source had become disconnected from the drive cable and remained in the guide tube.

Operations to perform 35 exposures of welds on a tank were planned. After cranking out the source for the sixth exposure, the radiographer heard a crash and saw that a magnetically mounted stand had fallen and was lying on the concrete pad. The source guide tube end-cap, with the collimator, had been approximately 10 feet above the concrete pad for this exposure.

The radiographer attempted to crank the source back into the camera but found that the drive cable could only be retracted a short distance because the guide tube was looped. The radiographer dragged the camera back by pulling on the drive cable housing to straighten out the guide tube. After straightening the guide tube, the radiographer retracted the cable fully, and, hence, thought that the source was in the camera. The radiographer removed his two self-reading pocket dosimeters and his thermoluminescent dosimeter badge. The radiographer later admitted that he did this to conceal the radiation exposure he would later receive.

The radiographer walked up to the end of the source guide tube with his survey meter in his hand but did not refer to the instrument. He grasped the end of the source guide tube with this left hand and removed the tape which held the collimator in place with his right hand. He then began to unscrew the source guide tube end-cap from the source guide tube to exchange the end-cap for a lighter one. As he removed the cap, the source chain containing the sealed source fell out of the end-cap assembly onto the concrete pad. The radiographer then dropped the source guide tube and end-cap, and left.

Two NRC inspectors investigated the event at the job site. Based on interviews conducted with the radiographer and the Corporate Radiation Safety Officer, the NRC inspectors estimated that the radiographer received a whole- body exposure of about 8.9 rems and an extremity exposure of about 1,070 rems.

The NRC issued an Order prohibiting the radiographer from engaging in NRC-licensed activity on behalf of the licensee for a period of one year. A proposed civil penalty of $15,000 was issued to the licensee, which has been paid.

False Assumptions Lead to Iodine-131 Overdose. On October 10, 1990, a 60-year-old female patient was referred to the Nuclear Medicine Department of the William Beaumont Hospital in Royal Oak, Mich., for iodine-131 thyroid ablation therapy after undergoing a thyroidectomy for cancer. After reviewing the clinical data on the patient, the authorized physician-user prescribed 175 millicuries of iodine-131 to be administered orally on October 15.

On October 15, the licensee received the patient's oral iodine-131 solution from a distributor, together with a second vial containing 140 millicuries of iodine-131. The latter vial was supplied to meet a weekly standing-order from the hospital, to be used as needed during the week.

The two vials were assayed by a technologist. After the assay, the technologist placed both vials side-by-side in the "fume hood" located in the nuclear pharmacy. Both vials were in their original leaded shields and labeled as to their contents.

The authorized physician-user was ready to administer the iodine-131 to the patient, and called for the material. Since the technologist who had prepared the dosage was not available, another technologist went to the pharmacy to obtain the radiopharmaceutical. The administering technologist picked up both vials, assuming they were to be administered to the patient. The technologist did not review the labels on the containers, assuming they were the proper doses. The technologist also was not alerted by the administration of more than one vial, since that was a common occurrence at this facility.

After receiving the dosage record, the authorized physician instructed the technologist to administer the dose to the patient. The authorized-user did not review the labeling on the containers, believing that because the patient's unit dose record was complete and indicated a dosage of 180 millicuries, the two vials were the proper ones for administration.

On October 16, the nuclear pharmacist received a request for 25 millicuries of iodine-131, but could not find the "standing-order vial." The resulting investigation disclosed that the vial had been erroneously administered the previous day.

An NRC consultant's evaluation indicated that the misadministration should not have any significant medical effects on the patient.

Radiographer's Assistant Irradiated When "Chirper" Stops. On November 26, 1990, Tumbleweed X-Ray Company of Greenwood, Okla., notified the NRC that on November 12, 1990, a radiographer's assistant may have sustained a possible radiation overexposure to his right hand at a temporary job site in Burns Flat, Okla. The licensee stated that it was not informed of the incident by the radiographer until the morning of November 25, 1990, because the radiographer did not think an overexposure had occurred until the assistant radiographer's right hand became red and his fingers began to swell.

On the day of the incident, the radiographer and his assistant were working with a radiography device that contained a 49-curie, iridium-192 sealed source. The radiographer and his assistant were performing radiographic exposures of welds on a 48-inch diameter tank at a fabrication shop. While the radiographer was away, the assistant set up an exposure and cranked out the source.

The assistant had turned the crank about two or three turns when he saw that the magnetically mounted stand that held the guide tube near the tank exterior had fallen. The assistant radiographer's alarming personnel dosimeter (chirper) had alarmed loudly when the guide tube had fallen. The assistant stated that he froze for about five seconds, and then cranked the source back to the shielded position. The assistant's chirper stopped alarming, so he thought the source was in the shielded position in the radiography device.

The assistant radiographer walked over to the tank and repositioned the magnetic stand and source guide tube. The assistant radiographer stated that he failed to pick up and use his survey instrument to survey the radiography device and the source guide tube because his chirper was not alarming. The chirper had been dropped a couple of times that night and upon subsequent testing was found to be malfunctioning because of a shorted ground wire. After the assistant radiographer correctly positioned the guide tube with his right hand, he returned to the crank handle to proceed with the exposure.

As he performed this exposure, he noted that his chirper did not alarm when the source was cranked out. Because of that, he looked at his pocket dosimeter and noticed that it was off-scale (greater than 200 millirem). At about the same time, the radiographer returned and the assistant told him what had happened and that his pocket dosimeter had gone off scale. The assistant told the radiographer that he did not think that he had received an overexposure, but that he thought his pocket dosimeter was off scale because he had bumped it earlier. The radiographer and his assistant continued to work and did not inform the Radiation Safety Officer of the incident until the assistant's hand showed clinical signs of a radiation injury.

The radiation injuries that the assistant radiographer sustained to his hand indicated that he had grasped the guide tube with his thumb, index, and middle fingers, and that the source must have been directly beneath the point grasped. This information may indicate that the assistant radiographer mistakenly cranked the source out, instead of in, when the incident first occurred.

From re-enactments, clinical observations, and calculations, the dose to the assistant radiographer's hand was estimated by the NRC to be from 1,500-to-3,000 rems. The whole-body dose to the assistant, as measured by his thermoluminescent dosimeter, was 365 millirem. Blood samples were taken from the assistant for cytogenetic tests, the results indicating an equivalent whole-body exposure of less than 10 rems.

On November 29, 1990, the NRC inspector noted that the assistant's thumb, index and middle fingers were severely blistered and swollen. The assistant was admitted to a burn center in Oklahoma City, Okla., for medical care. The assistant remained in the hospital for approximately two weeks, during which time he had a skin graft performed on his index finger. On January 22, 1991, the physician contacted NRC and stated that the assistant's middle finger and thumb appeared to be healing and that the index finger was grafted as a result of lesions that were not healing. The physician also stated that the assistant would remain under his care.

The NRC issued an Order prohibiting the radiographer and assistant from participating in licensed activity. Later, the NRC issued an Order suspending the licensee's General License, and its NRC materials license was terminated at the licensee's request.

Technetium Administered Instead of Indium. On November 26, 1990, a patient at the Veterans Administration Medical Center in San Diego, Cal., who was scheduled for the administration of five millicuries of indium–111 for diagnostic imaging of colorectal cancer, was mistakenly administered 168 millicuries of technetium–99m pertechnetate.

A technical assistant erroneously had picked up a syringe containing the technetium–99m pertechnetate and had given it to the physician. The physician failed to positively identify the label on the syringe before injecting the contents of the syringe into the patient.

The error was discovered by the licensee within minutes after the misadministration, and the patient was administered 10 drops of iodide and one gram of perchlorate to block and flush the thyroid gland respectively.

The patient was placed in a isolated room normally used for therapy for two days. The patient was scanned approximately 30 hours after the misadministration, and the thyroid gland showed no elevated radioactivity. A small residual amount of technetium–99m was detected in the bladder. Following the scan, the patient was noted to be clinically unchanged and was discharged from the licensee's medical center.

whole-body Scan Using Iodine–131 Mistakenly Administered. On January 24, 1991, Hutzel Hospital in Detroit, Mich., notified the NRC that a medical diagnostic misadministration had occurred at its facility on January 17, 1991. On January 16, 1991, a 37-year-old female patient (who had given birth two days earlier) was scheduled to have a thyroid scan. The licensee's normal procedure for such a thyroid scan usually involves administration of a 50–microcurie dosage of iodine–131. This would typically result in a thyroid dose in the range of 50–70 rads. The prescription for the procedure was prepared by a physician's assistant at the direction of the referring physician. The nuclear medicine technologist subsequently discussed the procedure with the physician's assistant and questioned whether or not the thyroid scan was the appropriate procedure. The technologist indicated a whole-body scan to identify thyroid tissue throughout the body would be the appropriate test. The physician's assistant agreed and submitted a new order for the whole-body scan. The iodine–131 was administered to the patient on January 17, 1991, with the whole-body scan performed on January 18, 1991. The procedure constitutes a misadministration because the referring physician had not intended to perform a whole-body scan using iodine–131.

The whole-body scan involved a dosage of five millicuries of iodine–131, instead of the 50 microcuries which would have been used for the diagnostic procedure actually prescribed by the referring physician.

Prior to administering the iodine–131, the technologist determined that the patient was not breast-feeding her baby and did not intend to do so. Some direct radiation exposure was received by the baby as a result of the presence of the iodine–131 in the mother's body. This exposure, however, was minimal (estimated to be approximately 0.5 millirads), because, as a result of the mother's medical condition, the baby was with her for only a 30–minute period.

An NRC consultant estimated that the patient received a dose of approximately 6,500 rads to the thyroid. This exposure would carry a slightly increased risk of developing hypothyroidism or thyroid cancer. Because the patient was lactating, thus concentrating the radioactive iodine in the breasts, there would also be an increase in the patient's risk of breast cancer.

Larynx Irradiated Instead of Brain. On February 1, 1991, the NRC was notified by Washington Hospital Center in Washington, D.C., that a therapy misadministration involving a teletherapy unit had occurred. A 74-year-old patient was to have received 250 rads to the brain for cancer treatment. The technologist identified the patient, but the technologist consulted the chart of another patient without confirming the name on the chart or examining the picture of the patient on the chart. No patient treatment area markers, such as tattoos, were used. Depending on the wrong chart, the technologist initiated treatment of the patient's larynx. The thyroid of the patient was not blocked from exposure to the teletherapy beam. While the patient was undergoing treatment to the larynx, the technologist realized that the wrong organ was being treated. The technologist immediately terminated the patient treatment.

It was estimated that 57 rads were delivered to the larynx, and about the same to the thyroid. After termination of the larynx treatment, the patient was given the proper treatment of 250 rads to the brain. An NRC medical consultant noted that there were no acute symptoms and that there should be no long term medical implications during the expected lifetime of the patient.

Radiation Therapy of the Eye Results in Overdose. On February 22, 1991, the NRC was notified by Hahnemann University Hospital in Philadelphia, Pa., that a therapy misadministration had occurred at its facility during the period from February 14–to–18, 1991, while a patient was undergoing radiation therapy for a tumor in the eye.

A radiotherapy physician prescribed a therapeutic dose of 30,000 rads to the base of the tumor and 14,300 rads to the apex of the tumor from a custom designed eye plaque, or patch, containing seeds of iodine–125. While the physicist was designing the eye plaque and calculating the anticipated dose, he decided to change to an eye plaque with a different radius of curvature. The physicist changed the coordinates for placement of each iodine-125 seed used in the plaque but failed to change the associated points for calculation of dose to various depths within the eye.

On February 18, 1991, the physicist suspected that an error had occurred, while planning treatment for another patient with a similar tumor. At that point, he retrieved patient data from the computer for the treatment started on February 14, 1991, reviewed the data, and confirmed that an error had been made. The patient's eye plaque was then removed. At that time, the total treatment dose was about 59,000 rads to the base of the tumor and 19,500 rads to the apex of the tumor. The licensee stated that the dose received by the tumor was within acceptable medical treatment protocols for the type of tumor involved, and that no acute effects were observed in the patient. An NRC medical consultant stated that there was an increased risk of long term adverse effects (e.g., cataract, tissue damage).

Two Hospital Patients with Identical Names: Wrong One Given Treatment. On March 28, 1991, officials at the Clara Maass Medical Center in Belleville, N.J., informed the NRC that a therapeutic misadministration, involving administration of iodine-131 to the wrong patient, had occurred earlier that day.

A radiotherapy physician prescribed a therapeutic dosage of 10 millicuries of iodine-131 to a patient for the treatment of hyperthyroidism. The physician who was familiar with the patient was not able to administer the therapeutic dosage and asked another physician to administer it. In the meantime, a transporter, reviewing the patient transport requests, noted that the patient was listed in a bed that she believed was assigned to another patient. The transporter advised the nuclear medicine secretary to check into the discrepancy. The secretary referred to a patient list for the patient's name, noted the area of the hospital where the patient's room was located, and changed the request form. The secretary did not know that there were two patients in the hospital with the exact same names. Also, the secretary did not know the computer program that generated the patient list did not print duplicate entries. The name of the patient who was to undergo treatment for hyperthyroidism was not printed on the list.

The physician who was to administer the dose picked up the request form and the iodine-131 dosage from the Nuclear Medicine Department and went to the nursing station on the floor of the patient. The physician did not inform the nursing staff that he was about to administer a therapeutic dose to one of their patients before going to the patient's room. There, he asked the patient his name and verified the name on the wrist band, but he did not cross check the patient number on the wrist band with the patient number on the request form. The physician completed the request form and returned the patient folder to the nurses' station. Within five minutes of the administration of the radiopharmaceutical, the nurses discovered the error and informed the physician and the Radiation Safety Officer. The licensee administered a thyroid blocking agent of 1,000 milligrams of potassium iodide immediately, with three subsequent doses of 1,000 milligrams each given at four-hour intervals.

The licensee ascertained that the thyroid of the patient had received an uptake of between 80 and 100 microcuries of iodine-131, which would imply a dose of between 112 and 140 rads. An NRC medical consultant concurred with these figures. The licensee informed the NRC that no adverse effects were anticipated during the lifetime of the patient as a result of the misadministration.

Multiple Teletherapy Misadministrations Discovered in Records Review. On April 12, 1991, NRC Region III was notified by St. John's Regional Medical Center in Joplin, Mo., that a number of cobalt-60 teletherapy misadministrations had occurred between September 1989 and March 1991. Misadministrations (defined as therapeutic doses varying by more than 10 percent from prescribed doses) were discovered during a review of past treatment data in March and April 1991. On April 25, the license formally reported that 12 misadministrations had occurred.

Of the 12, three patients received doses 10 percent to 18 percent higher than the prescribed doses, and nine patients received doses from 10 percent to 27 percent below the prescribed doses. All misadministrations resulted from erroneous information in the treatment planning computer program. All treatments, with one exception, involved the use of wedges which consist of a material, such as lead, placed in the radiation beam to more evenly distribute the prescribed dose of radiation to appropriate tissue. The one exception involved an arc treatment, which is a technique used to deliver a greater dose to a selected point while minimizing the dose to other areas by rotating the cobalt-60 source around the patient.

The treatment discrepancies were first discovered in March 1991, when a therapy technologist, preparing for an upcoming board certification test, pulled the files of previously treated patients to practice hand-calculated dosimetry. The technologist later informed licensee management that her results did not match the wedge-related treatment doses indicated in the patient files.

The Radiation Oncology staff began hand calculations of all patient treatments. Reruns of the original computer calculations also were initiated. By March 29, the recalculations supported the technologist's contention that actual administered doses had deviated significantly from prescribed doses. All of the patients' referring physicians were subsequently notified of the dose differentials, except for one physician who had left the area. In the latter case, the patient was notified directly. Subsequently, the patients have been examined by their physicians for follow-up care. The licensee stated that no adverse effects have been observed to date.

Abnormal Occurrences Involving Agreement State Licensees

One Hundred Millicuries Given Instead of Microcuries. On November 1, 1989, a patient scheduled for the administration of 100 microcurie capsules of iodine–123 for a diagnostic thyroid scan at Desert Samaritan Hospital in Phoenix, Ariz., was mistakenly administered a therapeutic dose of 100 millicuries of iodine–131 and sent home for 24 hours until the normal imaging was scheduled. (One millicurie is equal to 1,000 microcuries.)

When the patient returned for imaging, on November 2, the imaging camera flooded out, indicating a large overdose. The hospital immediately notified the Arizona Radiation Regulatory Agency (ARRA). The patient was immediately hospitalized and isolated (the standard practice for thyroid ablation patients). The patient was discharged on November 5, 1989.

The patient's family was contacted and a bioassay was performed to determine the thyroid body burden of each family member. The thyroid burdens were above the "action level" for radiation workers (0.4 microcurie), but the level was not considered a serious health threat to any family member.

A hospital employee and an ARRA representative surveyed and decontaminated the patient's house. Wipe tests were used to certify the efficacy of this action.

It was determined that the hospital staff had not assayed the dose in the dose-calibrator before administering it, had not compared the iodine–131 dose label with the physician's order, and had not maintained adequate records of incoming radiopharmaceuticals. In addition, Syncor International, Inc., the radiopharmacy that dispensed the dose, had not indicated the type of procedure to be followed in using it.

The ARRA issued an Order reducing the limit on the licensee's possession of iodine–131 until corrective actions could be taken. Later, a civil penalty of $12,000 was imposed.

Radioactive Seeds Spill from Receptacle into Patient. On April 19, 1990, at the Yuma Regional Medical Center in Yuma, Ariz., a patient's uterine tumor was implanted with 224 iridium–192 seeds by means of 32 trochars (a sharp, pointed surgical instrument fitted with a hollow tube), each one containing seven seeds on a ribbon. The prescribed dose was about 2,000 rads. A problem was noted with snagging of the ribbon in one trochar; five seeds were stripped from the trochar when an attempt was being made to remove both the trochar and the seeds. The trochar had inadvertently been placed in a necrotic cavity within the tumor, permitting the seeds to "pay out" into the cavity rather than being stopped by tissue.

An unsuccessful attempt was made to remove the five stripped seeds, during removal of the other seeds. When the trochar that had contained the snagged ribbon was removed, it was discovered that the tip of the trochar had been bent, presumably by the hardness of the tumor. The trochar had not been bent before it was inserted.

The five seeds were left in the necrotic tumor center. These seeds, from the time of emplacement until total decay, would deliver a dose considerably in excess of the prescribed dose. But a medical consultant affirmed that the poor prognosis for the patient, from the nature of her illness, outweighed any harm from additional radiation. (The patient subsequently died from her illness.)

Consultant Finds Multiple Deficiencies in Teletherapy Administration. On July 26, 1989, the Good Samaritan Medical Center in Phoenix, Ariz., reported to the Arizona Radiation Regulatory Agency that a series of three misadministrations had occurred involving the use of a cobalt–60 teletherapy unit in the licensee's Radiation Oncology Department.

The three patients received exposures of approximately 14 percent, 12 percent, and 12 percent greater than the prescribed doses of 6,200 rads, 6,480 rads, and 5,000 rads, respectively; the instrument involved was an AECL Theratron–80 unit containing 5,529 curies of cobalt–60, assayed on September 16, 1988. A beam-correcting wedge had been used along with a treatment planning computer. Although the computer already contained a wedge correction factor, the technologist and dosimetrist added a second wedge correction factor, after checking with the consulting physicist and being told that a wedge factor would be required.

While preparing to treat a fifth patient assigned the same treatment protocol, a hand calculation indicated a wide discrepancy when compared with the computer-generated treatment time. The discrepancy led to a comprehensive search of past cases, which revealed the three overexposures out of four possible cases.

All three patients showed signs of skin erythema, or reddening, and the first two patients (who had received radiation to the larynx region) reported hoarseness and pain on swallowing. The licensee stated that these symptoms are not unusual for patients undergoing radiotherapy, and, in fact, these same symptoms were mentioned to the patients as possible side-effects of the treatment.

A consulting physicist was retained to review patient records and the hospital's handling of this case. Among the consultant's findings were:

(1) The hospital staffing level was inadequate for the patient load.

(2) There was a loss of continuity in physics services with the departure of one physicist and the hiring of another physicist.

(3) There was poor communication (documentation) regarding the use of the computer-generated treatment plans.

The licensee has hired a full time qualified therapy physicist and a technical administrator, who will not have responsibilities outside of the therapy department. All computer-generated treatment plans will be subject to hand calculations to verify the computer readings. Procedures for use of this computer to generate patient treatment plans have been revised.

The State Agency proposed a civil penalty in the amount of $3,000.

Overexposure of a Non-radiation Worker. During radiography operations, an unmonitored, non-radiation worker employed by the Exxon Corporation received a whole-body exposure estimated to be between 1.8 and 3.9 rems from a radioactive source that was not properly shielded. The dose exceeds the abnormal occurrence reporting threshold of 0.5 rem in one calendar year for a member of the general public. In addition, a radiographer working at an Exxon site received a whole-body exposure of about 7.7 rems, on June 14, 1990, during operations by the H&G Inspection Company, Inc., of Houston, Tex.; the event occurred on a barge at Sabine Pass near Port Arthur, Tex.

On July 14, 1990, two licensee radiographers were performing routine radiography of welds at Exxon's Texas Well No. 1, located in Sabine Lake, using a Gulf Nuclear Model 20V camera containing 60 curies of iridium–192. At the completion of a radiograph, the lead radiographer (Radiographer A) cranked in the source, approached and surveyed the camera and guide tube, and locked the camera. He removed the exposed film and took it to the darkroom. Radiographer A returned to the weld to set up for the next exposure. During this procedure, an Exxon employee approached the radiography camera inside the restricted area to discuss the next shot with Radiographer A. Radiographer A had problems setting up the next shot and obtained Radiographer B's assistance. The Exxon employee left the area at this time.

The two radiographers completed the setup and were leaving to make the radiograph when Radiographer B noticed that the lead radiographer's survey meter was off-scale on the high side. This indicated that the source was not in the shielded position. They moved away from the camera and tried to return the source to the shielded position; they were not successful in this attempt.

They then unlocked the camera and retracted the crank-out handle one-half turn. The camera was relocked and pocket dosimeters were checked. The dosimeters were off-scale. The Radiation Safety Officer was notified of the incident, and the employees were ordered to return to the shop. Their thermoluminescent dosimeters (TLDs) were mailed in for immediate processing.

The TLDs indicated that Radiographers A and B received about 7.7 rems and 1.3 rems, respectively. Because the non-radiation worker was not wearing any radiation dosimetry, his exposure was estimated by a re-enactment of the event and subsequent calculation. This indicated that he had received a whole-body exposure between 1.8 and 3.9 rems.

There were three root causes identified for the event. The first was the locking of the camera with the source in the unshielded position. (The licensee stated that there is a design flaw in the lock box and that what had occurred is not unusual with the Gulf Nuclear Model 20V camera. The manufacturer of this camera is no longer in business.) The second cause was the failure of the radiographer to perform an adequate survey to determine whether the source was in the shielded position. Apparently, the radiographer went through the motions of performing the survey, was complacent about reading the meter, and failed to apprehend what his meter was indicating. The third cause was inadequate procedures regarding unmonitored personnel entering a restricted area.

The State agency was considering escalated enforcement action, at the close of the report period.

Extremity Overexposure of Radiation Worker. While extracting a 10–curie, cesium–137 source from its housing, a radiation worker at the Rosemount, Inc., Kay-Ray/Sensall Division, in Mt. Prospect, Ill., received an overexposure to his left hand. The actual exposure was not precisely known but was considered to most likely fall between 200–and–714 rems. Because the higher value, which was indicated by the worker's dosimetry, could not be disproved, 714 rems to the left hand was entered into the worker's radiation record. The event was investigated by the Illinois Department of Nuclear Safety.

On July 10, 1990, the worker was removing the source from a Model 7064P source housing, so that the source could be transferred to a Model 7067 housing, for resale to a customer. The worker had approximately three years' experience in source loading, although this was the first time that he had removed a 10–curie source. Operating on this particular source holder (constructed of stainless steel and holding a larger than usual activity of cesium–137) required precautions, including direct observation and timing of operations by the worker's supervisor. The removal of the source/source holder assembly from the source housing was routinely accomplished. Following that action, the source/source holder assembly was moved, by the use of tongs, to an area behind a lead-shielded work station and clamped into place.

Extraction of the source from the source holder then began. This procedure involves physically peeling back the crimp on top of the source holder, using a pair of sidecutter hand tools. This effort proved unusually difficult, because the material was stainless steel rather than aluminum. With about 25 percent of the crimp peeled back, the cylinder in which the source was contained separated from the base of the source holder. Using a pair of channel-locks in his right hand, the worker retrieved the cylinder containing the source and continued the extraction process, using the channel-locks to hold the source/source holder assembly in place. Following the uncrimping of the broken source holder, the worker twice tried to extract the source, being successful on his second attempt. The source was then placed in a lead pig for eventual loading into the new device. The total time reported by the worker's supervisor for the entire procedure was four minutes and 45 seconds.

Previous recorded extremity doses to employees involved with source changes on 10–curie, cesium–137 sources from stainless steel source holders were reported to be approximately 3–to–4 rems to the hand holding the sidecutters. However, because source manipulation was unusually difficult in this case, the supervisor suggested that the worker's ring thermoluminescent dosimeter (TLD) be processed. On July 12, 1990, the results indicated an exposure of 714 rems to the left hand.

The worker was examined by a physician on the evening of July 12. The examination included a physical inspection of his hand as well as a blood test. Aside from a slightly elevated white blood count, because of the presence of a virus, no unusual results were reported by the physician. The worker showed no visible signs of acute radiation overexposure to his left hand. He stated that there was no discomfort, reddening, swelling or other ill effects suffered as a result of this event. On July 20, after further blood tests and physical examination, an oncologist/hematologist informed the worker that all tests were normal and that he could find no sign of damage to the worker's hands or forearms. Based on these findings, the doctor believed that the worker had not been exposed to the high level of radiation reported.

The ring TLD had only been worn for two days. On July 9, the worker prepared source capsules for disposal, an activity which usually results in minimal exposure. On July 10, the worker only performed the 10–curie source extraction. When not in use, the ring TLD was stored in a drawer at his desk in the stock room. The worker stated that no sources are allowed in the stock room, and a survey of this storage area, performed by the State inspectors, revealed no evidence of any reading in excess of natural background.

The State agency witnessed a re-enactment of the event and concluded that an overexposure occurred but was likely considerably less than the 714 rems indicated by the TLD. The license was amended to include the licensee's corrective actions. The licensee was also cited for the overexposure.

Unshielded Source Causes Exposure; Rain Causes Survey Meter Malfunction. During radiography operations, a radiographer employed by Big State X-Ray of Eastland, Tex., received an estimated exposure of 35 rems to his right thigh from a radioactive source that was not locked in its shielded position. The event occurred at Pride Refinery in Abilene, Tex.

On November 7, 1990, two licensee radiographers were performing radiography outside the Pride Refinery when it started to rain. They moved their operations inside a building so they could continue working. At the completion of the first series of radiographs, Radiographer A proceeded to move the camera to the next weld for the next series of exposures. He stated that he surveyed the camera, got "no reading," locked the camera (but did not remove the key from the lock), and then moved the camera. He moved to the next weld by picking up and carrying the camera, survey meter, and other equipment, dragging the crank-out cables behind him. He stepped over some obstacles and believes the key turned in the lock and released the source, which was allowed to move outside the shield by the crank-out.

Upon arriving at the next weld, he resurveyed the camera and proceeded to set-up the next exposure. (It was later determined that the survey meter was not operating correctly because of the moisture from the rain.) After completing the set-up, he noticed that the camera was unlocked and checked his pocket dosimeter. It was off-scale. He went to the crank-out handle and retracted the source about one and one-half turns. He then notified Radiographer B of the incident and he stopped operations and had Radiographer A's film badge sent in for immediate processing. However, the film was damaged during shipment and could not be processed. Therefore, his exposure was estimated by a re-enactment of the event and calculations; these indicated he received a 35–rem exposure to the right thigh.

The primary cause of this incident was the failure of the radiographer to properly lock the source in the camera and remove the key prior to moving the camera. The radiographer also failed to determine whether his survey meter was operating correctly after it became wet in the rain.

The State Agency cited the licensee for the overexposure and the improper procedure.

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 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. Both are provided by the NRC team put together to identify the circumstances and ascertain the causes of an operational event. For an event of potentially major significance, an Incident Investigation Team (IIT) is established by the EDO, made up of a headquarters-directed team complemented by regional staff, 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 an Incident Investigation Team 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 following discussion with representatives of the National Transportation Safety Board, Federal Aviation Administration, and National Aeronautics and Space Administration and has been continually refined over the years.

The fourth Incident Investigation Training course was held from April 22 through May 5, 1991. The course emphasized training on IIT guidelines and accident investigation techniques, including laboratory case studies. It also included an update on changes to the Incident Investigation Program and covered lessons learned from the previous two IITs. Course members were pre-selected from a roster maintained by AEOD to provide for an adequate number of trained and qualified IIT team leaders and members with appropriate technical expertise to draw on in the event an IIT is initiated by the EDO.

Of reportable events occurring during fiscal year 1991, two were judged to have a significantly high level of safety significance to warrant an IIT investigation, while 15 events underwent an AIT evaluation.

IIT Investigation of the Potential Critically Accident at the Generic Electric Nuclear Fuel and Component Manufacturing Facility. On May 29, 1991, at the General Electric (GE) Company's Nuclear Fuel and Components Manufacturing (NFCM) facility approximately six miles north of Wilmington, N.C, an estimated 320 pounds of uranium was inadvertently transferred to an "unfavorable geometry" waste treatment tank. ("Unfavorable geometry" refers to a container or vessel that can hold enough uranium to produce criticality, or self-sustained fission, among the uranium atoms.) Because of the tank's configuration and the type and quantity of material available, the potential existed for a nuclear criticality accident. Such an accident would yield a burst of neutron and gamma radiation that would likely be fatal to anyone within 10 feet of the burst and would cause radiation exposures of approximately five rads at 45 feet. (There would be no expected off-site radiological impacts.)

On May 31, 1991, the NRC established an eight-member Incident Investigation Team (IIT) to identify probable causes and draw appropriate conclusions. The IIT arrived in Wilmington, N.C., on June 2, 1991. The team was assembled to represent a broad knowledge of facility event analysis, with individual members having specific knowledge of fuel fabrication operations, chemical operations, instrumentation and controls, maintenance, human factors, radiological emergency preparedness, and nuclear criticality safety.

The team concluded that there were a number of interrelated root causes which contributed to the incident. Among their findings were these:

- There was a pervasive licensee attitude that a nuclear criticality was not a credible accident scenario. While the licensee understood and recognized that a nuclear criticality with low-enriched uranium was technically possible, and that there were regulatory requirements establishing measures to guard against such an event, the licensee's perception was that the risk was so low that a criticality accident was intrinsically something that would not happen.

- Licensee management did not provide effective guidance and oversight of licensed activity to assure that operations were conducted in a safe manner.
- There was a deep-seated production-minded orientation within the licensee organization that was not sufficiently tempered by a "safety first" attitude, particularly regarding nuclear criticality safety.

The team also concluded that the NRC regulatory oversight of the fuel facility was deficient in some respects. The team noted shortcomings with respect to the NRC's regulations and regulatory guidance, license and licensing process, and inspection program. This lack of sufficient oversight had the effect of contributing to a situation where safety margins eroded to the extent that the licensee had little or no latitude to accommodate operator errors or system difficulties.

IIT Investigation of the transformer failure and common-mode loss of instrument power at the Nine Mile Point Unit 2 Nuclear Facility. 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 (i.e. automatic reactor shutdown). During the fraction of a second before automatic protective features isolated the transformer, the fault caused 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 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 the power supplies susceptible to failure initiated by degraded voltage.

Automatic reactor protection systems, including the scram, functioned properly. All necessary engineered safety features were available and used as needed. However, control rod position indication was lost, and the operators took conservative action, in accordance with their 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 augmented inspection team (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 NRC Executive Director for Operations (EDO) upgraded the NRC response to an Incident Investigation Team, on August 15, 1991.

The IIT's description of the incident, the methodology used in its investigation, and documentation of its findings and conclusions are to be issued in NUREG–1455. These findings and conclusions will be presented in the agency's annual report for 1992.

DIAGNOSTIC EVALUATION PROGRAM

The Diagnostic Evaluation Program (DEP) provides an 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 the 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 reactor facility, a Diagnostic Evaluation Team (DET) is authorized and established by the Executive Director for Operations (EDO). The DET consists of technical staff members from Headquarters Offices, regional and resident inspectors and contractors, if appropriate. Team members are selected to provide an unbiased and independent assessment of plant performance. Emphasis and focus is placed 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, technical support, conduct of operations, safeguards and security, plant modifications and design changes, radiation protection, quality assurance, and corrective action.

Diagnostic Evaluation of Oyster Creek Nuclear Power Plant. In June 1990, the Executive Director for Operations (EDO) directed that a diagnostic evaluation be conducted at the Oyster Creek (N.J.) nuclear power plant. The recommendation was based on inconsistent performance reflected primarily in the frequent problems there, often resulting in forced plant shutdowns, and the long periods required to implement corrective actions. Despite substantial plant improvements and a demonstrated safety-conscious attitude, equipment problems continued to challenge the plant operators. A 16-member team spent a total of three-and-one-half weeks on-site, and at the GPU-Nuclear offices in Parsippany, N.J., during November and December 1990. The subjects covered during the evaluation included operations and training, maintenance and testing, engineering and technical support, and management effectiveness and corrective actions.

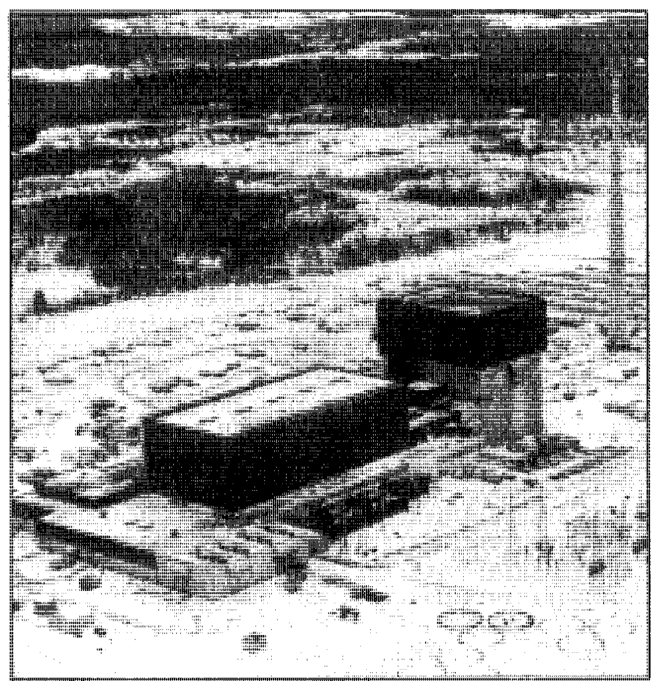
The DET concluded that, at the time of the evaluation, Oyster Creek was in a state of transition. The utility was still in the process of recovering the material condition of the plant, after deterioration resulting from minimal maintenance during the plant's early history, and was implementing a variety of improvements for future performance. Many of the strengths observed in the utility's performance were attributable to the experience and skills of individual personnel and managers. As a result of the these improving performance trends and currently effective conduct of plant operations, continued overall improvement was anticipated, although at a pace that was hampered by weaknesses in certain work practices.

The DET concluded that the root causes of Oyster Creek's inconsistent performance problems were weak administrative control of work, reflected in poor accountability; complex and inconsistent methods for setting priorities in, and tracking of, the work to be done; complex corrective action systems; reactive management practices; a lack of systematic means to determine when corrective action was necessary; weak supervision and lack of independent verification, which caused some inadequate evaluations and poorly reasoned decisions; and a general lack of rigor and inquisitiveness into work practices, which affects management's ability to achieve consistently effective solutions to problems.

Diagnostic Evaluation of 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 1991. The recommendation was based on an apparent decline in plant performance in three functional areas—plant operations, radiological controls, and safety assessment/quality verification. A 17-member team spent a total of three weeks at the FitzPatrick site, and at the corporate and engineering offices in White Plains, N.Y. Preparation for the report was under way at the close of the report period. The team's findings and conclusions will be presented in the agency's 1992 annual report.

TECHNICAL TRAINING PROGRAM

The NRC Technical Training Center (TTC) works with NRC Headquarters Offices and with the Regions in the development and implementation of NRC staff technical qualification training programs. Technical training is provided initially to impart and thereafter to sustain a level of knowledge among NRC personnel—inspectors, operator licensing examiners, reviewers, project managers, operations officers, technical managers, and others—with respect to reactor technology and other specialized technical subjects, as may be needed by them to perform their assigned tasks. Principles of the systems approach to training are routinely used throughout the life cycle of courses managed by the TTC. The Center, which is located in Chattanooga, Tenn., is part of the NRC headquarters organization, within the Office for Analysis and Evaluation of Operational Data (AEOD).



Under the NRC's Diagnostic Evaluation Program, a team made up of NRC headquarters and regional staff and, where appropriate, contractor personnel visits a facility for an extended and in-depth examination of operations, equipment and other aspects of plant performance. This kind of assessment is supplemental to the SALP program, the Performance Indicator program, and other routine or special inspections. A 17-member team spent three weeks at the FitzPatrick (N.Y.) plant toward the close of the report period. The plant is located on the shores of Lake Ontario near Oswego, N.Y.

The reactor technology curriculum at the Center comprises 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 upgrading of three full scope reactor training simulators that model the General Electric, Westinghouse, and Babcock & Wilcox reactor vendor designs, and associated computer equipment, in support of continuing training needs.

The core of the reactor technology training provided in support of the initial qualification programs for NRC staff has been the reactor technology full course series, which now consists 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 simulator course. Full course series training is provided in each of the U.S. light water reactor vendor designs. A variety of other stand-alone reactor technology courses have been made available to support other parts of NRC staff qualification programs.

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 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 on their own.

During fiscal year 1991, the TTC conducted or coordinated a total of 115 courses in the reactor technology areas and 82 courses in the specialized technical training areas. A total of 2,266 students attended TTC courses during the fiscal year, although a number of students in qualification programs attended multiple courses. These courses represent a total of 244 course-weeks, 146 of which involved reactor technology training and 98 of which involved specialized technical training. All courses falling under the TTC program element and listed in the annual syllabus of courses are included in these totals. This volume of training represents about 84,000 instructional hours of technical training received by students. Of this total, about 51,100 instructional hours involved reactor technology training and about 33,300 involved specialized technical training. (An instructional hour is a one-hour period of training in which a course instructor is present or readily available for instructing or assisting students. One hour devoted to any of the following activities is considered an instructional hour under this definition: lectures, seminars, discussions, problem-solving sessions, examinations, on-the-job training, laboratory exercises, programmed learning, and simulation exercises.)

Besides its technical training courses in support of qualification programs for NRC technical staff, the TTC also provided Reactor Concepts Courses in support of the orientation program managed by the Office of Personnel, reactor technology courses in support of the PRA Technology Transfer Program (also managed by the Office of Personnel), and National News Media Seminars in support of the public affairs function of the Office of Governmental and Public Affairs.

The TTC staff also accommodated a number of requests for special or non- scheduled courses during the year to meet a variety of needs. Special reactor technology courses are presented in General Electric and Westinghouse technology for State of Illinois personnel. Accident/Incident Investigation Workshops were given in Regions I, II, III, and V. Other special courses included: Mixed Waste Seminars for personnel from NRC's Office of Nuclear Material Safety and Safeguards (NMSS), a Beta Dosimetry Seminar for Region I, Site Access Training for Canadian employees under contract to NRC's Office of Nuclear Reactor Regulation (NRR), and special training sessions for Department of Defense personnel supporting the NRR safeguards function.

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 management with feedback and advice on training programs and help resolve issues involving curricula and training requirements associated with NRC staff qualification programs. Issues dominating these meetings were reactor inspector training requirements and technical training in support of intern programs.

A major effort throughout the year was that associated with identifying necessary changes to NRC Inspection Manual Chapter 1245 (IMC 1245), Inspector Qualifications. This task was accomplished through an IMC 1245 Work Group, which met at approximately two-month intervals throughout the year to develop the most significant changes to the directive ever issued.

Considerable effort was devoted during the year to the revision of the inspector initial qualification journals (Appendix B to IMC 1245). The number of journals was expanded from eight to 17, including 11 journals for region-based inspectors, two for NRR inspectors and four for NMSS inspectors. These generic qualification journals were last revised in 1983.

Major adjustments were made in the technical training program to keep pace with agency recruiting of technical interns. Programs were developed and implemented to address the surge in intern hiring during fiscal year 1991 and early fiscal year 1992. These programs included presentation of the Power Plant Engineering Course and development and implementation of special reactor technology courses for interns. The goal of these front-end technical training courses is to minimize the differences in experience, practical engineering knowledge, and basic reactor technology knowledge between technical interns and more experienced personnel who have usually attended the reactor technology full series courses.

A major initiative completed during the year brought expanded risk-based perspectives into the technical training curriculum. The intention behind the effort was to bring a risk-based culture to TTC courses, to complement the existing operationally oriented culture and to increase staff awareness of major risk contributors and risk-dominant sequences. The risk-based training development plan has now been completed for the courses of the General Electric, Westinghouse, Combustion Engineering, and Babcock& Wilcox reactor designs. All reactor technology full series courses beginning after January 1, 1991, have taken advantage of this new development.

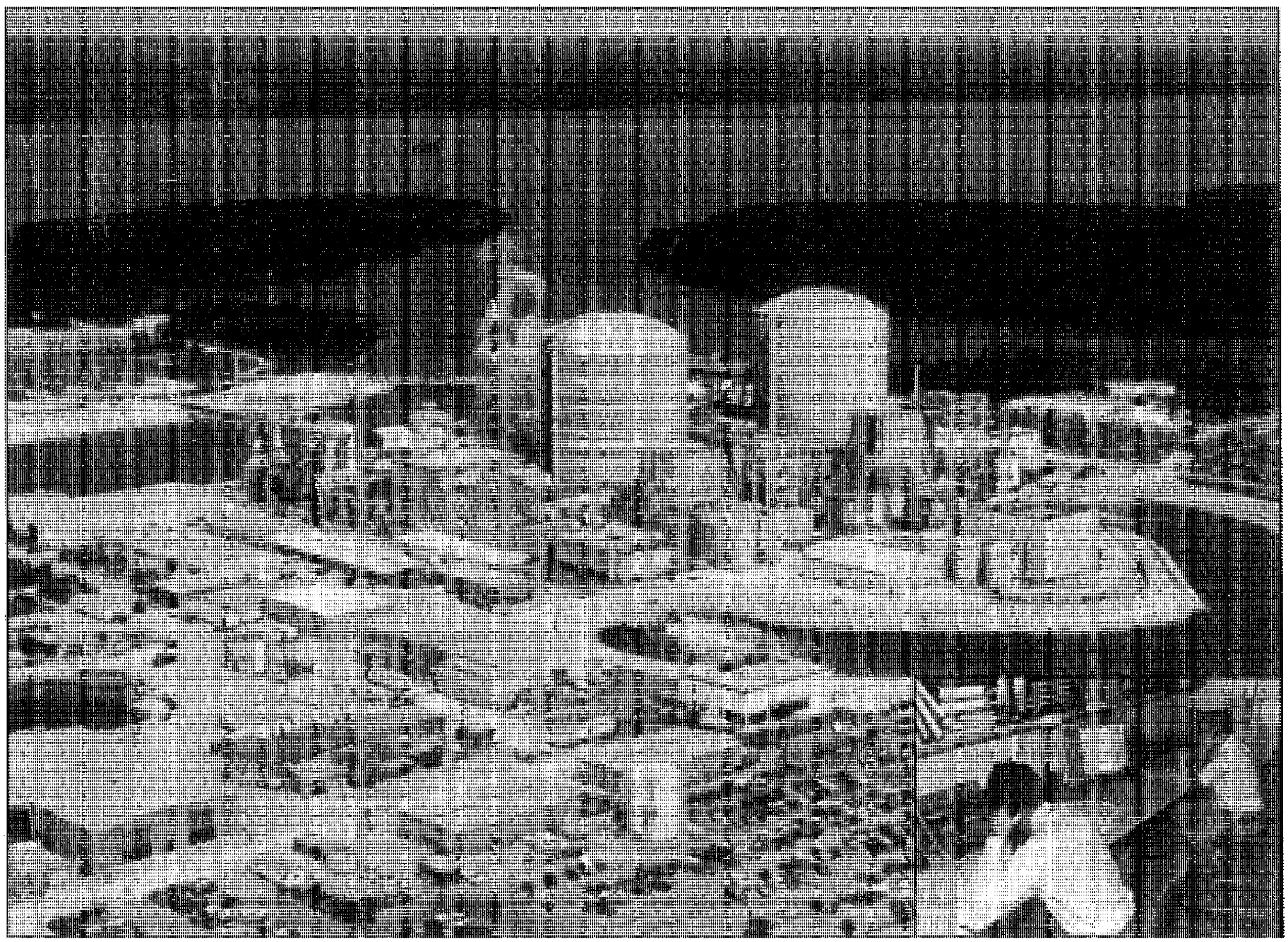
INCIDENT RESPONSE

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 licensed nuclear power plants and certain fuel cycle facilities, providing the capacity for the NRC 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 its licensees, primarily nuclear power plant operators. During fiscal year 1991, there were 184 incidents reported to the Operations Center under the NRC emergency classification system; of these, one was a "site-area emergency," seven were "alerts," and 176 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 potentially affected licensees and construction permit holders.

Operations Center. A prompt incident response capability entails continuous staffing by well trained individuals with the appropriate resources to receive information, assess that information, and communicate swiftly and reliably with other involved parties. During fiscal year 1991, the NRC entered the "Standby" response mode one time; this decision required the activation of the Headquarters Operations Center and the Region II Incident Response Center. It occurred when the General Electric Fuel Fabrication Facility, Wilmington, N.C., notified the NRC of a potential criticality event involving the loss of solvent extraction control for the nitrate waste treatment system. (See discussion earlier in this chapter, under "Abnormal Occurrences.") The event was subsequently classified by the licensee as an Alert. The Operations Center was involved in several other events. The Center was staffed to monitor a loss of off-site power at the McGuire Unit 1 (N.C.), Zion Unit 2 (Ill.), and Yankee-Rowe (Mass.) nuclear power plants; a partial loss of control room annunciators at the Millstone Unit 2 (Conn.) nuclear power plant; and a partial loss of control room annunciators and indications, coupled with a reactor trip from full power, at the Nine Mile Point Unit 2 (N.Y.) nuclear power plant (where a Site Area Emergency was declared). The Operations Center was also staffed to monitor the progress of Hurricane Bob as it threatened the east coast nuclear power plants. The telecommunications capability of the Operations Center was used by NRC management for teleconference discussions of a number of events of potential significance which, as they transpired, proved not sufficiently serious to warrant staffing of the Operations Center.



Various accident scenarios are devised and carried out in onsite exercises involving actual nuclear power plants and the NRC Operations Center. One of the exercises conducted during fiscal year 1991 took place at the St. Lucie facility, located on Hutchinson Island, off the east coast of Florida. The view shown above is toward the mainland. In the inset are two NRC Operations Center personnel taking part in the exercise, on March 20, 1991.

During fiscal year 1991, a number of exercises dealing with various accident scenarios involving the Operations Center were conducted, in order to confirm and maintain the capabilities of the agency response personnel. Most of the scenarios were concerned with reactor plant incidents. The exercises took place at the San Onofre (Cal.), St. Lucie (Fla.), Limerick (Pa.), Braidwood (Ill.) and Cooper (Neb.) nuclear power plants, and at the Nuclear Fuel Services Uranium Fabrication Facility (Tenn.). In addition, computer generated Nuclear Plant Analyzer reactor accident simulations were conducted in Region V. All of these exercises were supported through the Operations Center. The NRC participated in a three-day "plume phase and ingestion phase" post-emergency "tabletop" exercise at Madison, Wis., a simulated accident at the Kewanee nuclear power plant involving participation by the State of Wisconsin, Wisconsin counties, and Federal agencies. Planning is under way for the next Federal Field Exercise, scheduled for 1993.

Also during fiscal year 1991, the NRC established and implemented a 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. In addition, the NRC is working with the Office of State Programs to participate in meetings, workshops, and other vehicles that help describe the NRC assessment tools, response capabilities and federally sponsored accident assessment courses. As part of this program, the NRC participated in exercises at the Palo Verde (Ariz.), Three Mile Island (Pa.), and Harris (N.C.) nuclear power plants and made numerous presentations to State representatives during meetings.

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. About 120 State and local personnel received this training.

Throughout the year representatives of other Federal agencies, industry, State and local government, and foreign countries toured the Operations Center and were given detailed descriptions of the NRC response role and of typical activity within the Operations Center during an exercise or an event.

Region Response Capability. Each Regional Office also maintains its own incident response capability and an incident response center designed to support the agency response during a licensee Alert or in the NRC standby mode. The extent of Regional Office response to an incident is based on a pre-defined classification of the event. A regional base team and a regional site team are assembled for a significant event. Headquarters and the Region monitor licensee performance until a decision is made to dispatch a team to the site. An initial site team of 12–to–18 specialists, led by the Regional Administrator, will usually be at the site some 2–to–8 hours after being dispatched. When 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 would consider transferring appropriate responsibility and authority to the Regional Administrator, who would then be designated the NRC 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.

Each Region has prepared its own supplement, with specific implementation details, for the NRC Incident Response Plan. Regional response capabilities are assessed annually, and the Regions participate in several exercises each year, at least one of which includes participation by headquarters personnel.

Coordination with Other Federal Agencies. The NRC has participated actively in the development of the Federal Response Plan (FRP). The FRP is being developed by the Federal Emergency Management Agency (FEMA) as an umbrella plan for coordinating the Federal response to all major disasters. The NRC participates in the FEMA-chaired Federal Radiological Preparedness Coordinating Committee and its subcommittees on Federal Response, Training, Transportation, Public Information, Emergency Instrumentation and Potassium Iodine Use. The NRC was also very active in the initial planning and preparations for the Federal Field Exercise (FFE) scheduled for February 1993. The FFE will be a very large effort designed to demonstrate the integrated response of State, local and Federal agencies to a severe reactor accident.

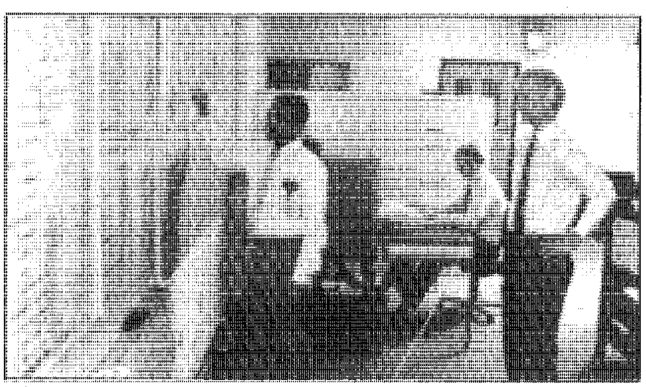
During 1991, substantial improvements were made in the level of coordination and understanding among the Federal agencies with roles during a reactor accident. These were mainly the result of:

- The NRC's conducting workshops in each Regional Office on Federal interface during a response for regional staff and representatives from the local offices of the other Federal agencies with response roles.
- The NRC's developing procedures for the prompt deployment of the Department of Energy (DOE) monitoring resources, when requested by NRC.
- The NRC's fully integrating representatives from Environmental Protection Agency (EPA), Department of Agriculture (USDA), Department of Health and Human Services (HHS), DOE, National Oceanic and Atmospheric Administration (NOAA), and FEMA into the NRC response.
- The NRC's participating in Response 91A, the first exercise of the FRP that simulated an earthquake involving seven States.
- The NRC's assisting EPA in revision of the Protective Action Guides for radiological emergencies.
- The NRC's assisting FEMA in revising evaluation documents for review of State and local emergency preparedness.
- The incorporation of Emergency Response Data Human Services (HHS), DOE, National Oceanic and Atmospheric Administration (NOAA), and FEMA into the NRC response.
- The NRC's participating in Response 91A, the first exercise of the FRP that simulated an earthquake involving seven states.
- The NRC's assisting EPA in revision of the Protective Action Guides for radiological emergencies.
- The NRC's assisting FEMA in revising evaluation documents for review of State and local emergency preparedness.

Emergency Response Data System. The Emergency Response Data System (ERDS) provides for licensee-activated transmission of pre-selected plant data from onsite computers to a computer at the NRC Operations Center, during emergencies at commercial nuclear power plants. Implementation of ERDS was initiated in 1988. During 1990 and 1991, the NRC portion of the system was installed and tested in the Operations Center. Initial licensee implementation was accomplished under a voluntary program. Under that program, 22 reactor units established operational ERDS links. Regulatory rulemaking to require implementation of ERDS at all commercial nuclear power plants was completed in 1991, and the final rule was published in the *Federal Register* on August 13, 1991. It is expected that the remaining plant connections will be completed by the end of 1993. In addition to licensee implementation of the system, State governments have expressed an interest in obtaining data during plant emergencies through the ERDS system. A memorandum of understanding is being developed between the NRC and the States of Michigan, Pennsylvania, Oregon, Ohio, Georgia, Washington, Tennessee, New York and New Jersey to formalize the establishment of an ERDS work station for use by State emergency response personnel.

Emergency Response Training. During fiscal year 1991, extensive staff response training was conducted for the NRC Headquarters, each Regional Office and other Federal and State response organizations. In all, about 700 people have attended one or more courses on some aspect of the NRC response program. This is about a 100 percent increase in Emergency Response training as compared with fiscal year 1990. 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.
- Emergency response workshops presented in each Region whose participants included representatives from NRC Headquarters, Regional Offices, EPA, DOE, and HHS. Topics discussed were the NRC response procedures and interfaces with other response organizations.
- Two-day courses in each Region on the standard technical procedures contained in the Response Technical Manual (RTM–91, NUREG/BR–0150). In addition, a one-week advanced course was held in Headquarters on these tools.
- A prototype course on Federal Radiological Monitoring and Assessment Center (FRMAC) operations.



Emergency staff response training continued during the report period for NRC Headquarters, each Regional Office, and other Federal and State response organizations. Region I (Philadelphia) personnel are shown above during an emergency exercise.

- Discussions of Emergency Response involving Headquarters, Regional Offices, EPA, DOE, and HHS. Topics discussed were the NRC response procedures and interfaces with other response organizations.
- Two-day courses in each Region on the standard technical procedures contained in the Response Technical Manual (RTM–91, NUREG/BR–0150). In addition, a one-week advanced course was held in Headquarters on these tools.
- A prototype course on Federal Radiological Monitoring and Assessment Center (FRMAC) operations.

Emergency Response Technical Tool Development. During fiscal year 1991, a Response Technical Manual (RTM–91, NUREG/BR–0150), was published. RTM–91 contains easy to use procedures and training materials for use during an accident for:

- Classification assessment
- Core condition assessment
- Projection of reactor accident consequences
- Assessment of UF_6 accidents
- Determination of protective actions for the public
- Implementation of EPA and FDA guidance on reentry and ingestion issues
- Radiation exposure control for NRC workers
- Obtaining assistance from DOE for monitoring and medical consultation
- Putting an accident in perspective for the public.

Work continued on the RASCAL model which is a computer code used to project consequences during accidents. This code is fast becoming the standard in the United States. Many utilities and states use RASCAL, and DOE is considering using it for their facilities. In addition, development of Graphic Image Systems and of improved electronic mail capabilities has continued.

A program has been initiated by the NRC to augment the assessment capabilities of the Reactor Safety Team (RST) during a response to a nuclear power plant emergency. The program involves the development of an expert system known as the Reactor Safety Assessment System (RSAS). The RSAS will be used as an independent tool by the RST to monitor and display the status of a plant's Critical Safety Functions, i.e., 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 or identify potential inconsistencies between RSAS and their assessment.

The first test of the Reactor Safety Assessment System (RSAS) for the Braidwood (Ill.) plant was conducted with on-line ERDS data feed, during an NRC and licensee exercise. Other tests were performed using a real-time ERDS feed from a NRC full scope training simulator at the Technical Training Center. Simulator testing at the Center will be the means of evaluating the RSAS for all four reactor models. Other work completed in fiscal year 1991 includes the development of building tools used in developing plant-specific models and the support system dependency matrix; the transfer of RSAS software to a UNIX based work station platform; and the collection of all PWR plant-specific data. During 1991, RSAS was demonstrated at the American Nuclear Society topical meeting, entitled AI91, Frontiers In Innovative Computing for the Nuclear Industry, held in Jackson, Wyo., and the International Atomic Energy Agency (IAEA) workshop entitled Demonstrate and Review Expert Systems Prototypes, held in Springfields, United Kingdom.

Work planned for fiscal year 1992 includes the completion and verification of plant-specific information; the creation of plant-specific files for most plants, the development and integration of the BWR generic knowledge structure; and continued testing of the software code and knowledge base. RSAS will be involved as a test case for a formal validation and verification project for expert systems, jointly sponsored by the NRC and Electric Power Research Institute (EPRI).

Office of Investigations

The Office of Investigations (OI) carries out 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 1991, OI opened 60 cases and closed 65 cases. Eighteen cases were referred to the Department of Justice for consideration of possible prosecution. Five cases were closed for administrative reasons.

During fiscal year 1991, OI continued to investigate the sale of counterfeit and substandard parts—such as fasteners, electrical relays, valves, and circuit breakers—to utilities operating nuclear power plants. OI also continues to participate in the interagency working group (IWG) on problem parts and suppliers, and the office plays a leading role in the IWG's subgroup of Federal investigative personnel. The interagency cooperation fostered by these groups has proved to be mutually beneficial. Cooperative efforts have resulted in the successful prosecutions of Stokley Enterprises and CMA International, based upon joint investigations by OI, the Naval Investigative Service, and the Seattle-based NORDECON Task Force.

Three Reports of Investigation concerning product substitution fraud were referred to Department of Justice for prosecutive consideration. One referral, involving Stokley Enterprises, Inc./Spectronics, Inc., resulted in the conviction of William Stokley and his company on charges of conspiracy to traffic in counterfeit goods and trafficking in counterfeit goods (electrical relays). William Stokley was sentenced to two years in Federal prison, fined \$7,500, and ordered to pay \$350,000 in restitution. Stokley Enterprises, Inc., was fined \$30,000 and ordered to pay \$2.5 million in restitution, less the amount of restitution personally paid by Stokley.

A case involving CMA International, which had been referred to the Department of Justice in September 1990, resulted in CMA and its owner, Clifford Ashley, each pleading guilty to conspiracy to traffic in counterfeit goods (valves). The conspiracy involved counterfeit valves which were installed at the Diablo Canyon (Cal.) nuclear power plant, the Vogtle (Ga.) nuclear power plant, and the U.S. Marine Corps Base in Quantico, Va. Mr. Ashley faces a maximum penalty of five years imprisonment and a \$500,000 fine; CMA faces a possible \$500,000 fine. Sentencing was set for December 6, 1991.

Cooperation between OI and other agencies was further demonstrated in the case of a Maryland firm, Data Measurement Corporation, which sold radioactive sources to an unlicensed plywood manufacturer in Virginia. Information developed by OI during this investigation was provided to the State of Maryland's Department of Environment, which conducted an inspection of Data Measurement. The inspection revealed several violations, including the shipment of unlicensed radioactive sources to the Oriented Strand Board Plant, Skippers, Va. As a result of the efforts of OI and the subsequent State inspection, Data Measurement agreed to pay a \$2,000 fine.

Other Convictions/Guilty Pleas

As a result of an OI:Region I investigation, Stanford Mining, Inc. (SMI), waived indictment in U.S. District Court, Pittsburgh, Pa., on August 5, 1991, and entered a plea of guilty to conspiring to improperly transfer and dispose of three nuclear weigh scales, in violation of Title 42, U.S. Code, Section 2273. SMI was fined \$30,000. The president of the company was scheduled for trial for his complicity in the transfer of the weigh scales.

OI:Region IV conducted an investigation of Saturn Wireline Services, Inc. (SWS), Tulsa, Okla., regarding unlicensed well logging activity, between July 1986 and December 1986. During the course of the investigation, the owner of SWS made false verbal statements to the investigator; the matter was referred to the Department of Justice. On August 23, 1991, the owner pleaded guilty. Sentencing was pending at the close of the report period.

Enforcement Actions/Civil Penalties

In Honolulu, Haw., a Finlay Testing Laboratories (FTL) employee was observed improperly transporting and storing a radiographic camera, by OI:Region V personnel. The FTL employee was operating as a radiographer for C&R Laboratories. A $1,500 civil penalty was proposed by the NRC, on November 28, 1990; it was paid by C&R Laboratories. (Department of Justice action was pending against FTL in U.S. District Court for numerous violations of NRC regulations regarding radiographic operations.)

In October 1990 an OI:Region V investigator and an NRC inspector observed and videotaped a radiographer improperly conducting radiographic operations. The radiographer then made a false statement to the NRC regarding his activity. An order was issued prohibiting the licensee from utilizing the individual as a radiographer for three years. The licensee was also assessed a $15,000 civil penalty. The licensee ultimately terminated its license.

An investigation by OI:Region III, at the Lafayette Clinic in Detroit, Mich., resulted in the levying of an $11,000 fine against the licensee. The investigation disclosed that a researcher at the clinic was responsible for the deliberate unauthorized use of radioactive material, as well as for employment discrimination against two clinic employees who raised concern over the matter. In addition to the fine, the NRC issued an order to the clinic prohibiting the researcher and the former acting clinic director from involvement in any NRC-licensed activity for the next three years.

As a result of an OI:Region III investigation, an order modifying its license was issued against Midwest Inspection Services, Ltd., a radiography firm in Green Bay, Wis. The order was based in part on the investigative finding that the owner of the firm deliberately allowed an untrained individual to perform the duties of both a radiographer's assistant and a radiographer. The order requires the licensee to notify the NRC before performing any radiographic activity and to employ an outside consultant to independently audit the firm.

An OI:Region IV investigation of Western Stress, Inc., for the violation of NRC regulations regarding radiographic activity resulted in the assessment of a civil penalty of $15,000, which was paid on August 1, 1991. In addition, an order was issued to prohibit the radiographer from supervising radiographic activity for one year.

Tumbleweed X-Ray Company was the subject of an investigation by OI:Region IV for violation of NRC regulations regarding radiographic activity. As a result of the investigation, an order was issued on September 6, 1991, suspending for three years the general authority of Tumbleweed to conduct radiographic operations in Oklahoma or in any jurisdiction where such work is regulated by NRC. Further, at the request of the company, NRC terminated its specific license.

An investigation of Patrick Chun, M.D., for false statements made to the NRC in his application for a materials license which led NRC to improperly issue the license was conducted by OI:Region IV during this reporting period. Based on the investigation, an order was issued to prohibit Dr. Chun from obtaining an NRC license or being named on an NRC license as a radiation safety officer or an authorized user for one year. The order also states that, before obtaining a license in the future, Dr. Chun will be required to provide assurances to the NRC that he can be relied upon to provide complete and accurate information and abide by other requirements incumbent on a license holder.

The following civil penalties were levied, based on OI:Region I investigative reports:

(1) Indian Point Unit 2: False Statements to the NRC Regarding Meggar Testing—$62,500 civil penalty.

(2) Professional Services, Inc.: Violation of a License Condition—$14,000 civil penalty.

(3) PX Engineering: Using Uncertified Radiographers—$7,500 civil penalty.

(4) Roche Professional Associates: Violation of a License Condition—$7,500 civil penalty.

Office of Enforcement

The Office of Enforcement is responsible for managing the Commission's enforcement program—subject to oversight by the Deputy Executive Director for Nuclear Reactor Regulation, Regional Operations and Research—for enforcement actions involving reactor licensees and by the Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support for enforcement actions involving all other licensees.

The NRC enforcement program has the objective of protecting the public health and safety by ensuring that NRC licensees comply with regulatory requirements. The program is currently carried out under the Commission's Enforcement Policy (10 CFR Part 2, Appendix C (1991)) which calls for strong enforcement measures to encourage full compliance and which will not permit operations by any licensees who fail to achieve adequate levels of protection. The NRC's Enforcement Manual provides further guidance for the enforcement program.

Appendix 6 provides a listing and brief summary of the civil penalties proposed, imposed, and/or paid during fiscal year 1991, and a listing and brief summary of the eight orders issued during fiscal year 1991. Recognizing that enforcement actions can sometimes span fiscal years, of the 92 civil penalties acted upon in fiscal year 1991, 76 cases were proposed, for a total of $2,762,175; 16 were imposed, for a total of $343,571; and 83 were paid (or payments were received for those civil penalties being paid over time), for a total of $3,215,614. In addition, 32 cases were issued as escalated enforcement actions, although no civil penalty was issued, for reasons unique to each case.

Enforcement conferences are normally conducted with licensees prior to the issuance of the enforcement action, in order to discuss the violations or nonconformance, as to their significance and causes; to discuss the licensee's or vendor's corrective actions; to determine whether there are aggravating or mitigating circumstances; and to obtain other information which may help define the appropriate enforcement action. In support of the enforcement actions discussed in the preceding paragraph, 157 enforcement conferences were held in fiscal year 1991.

The Enforcement Policy was modified on several occasions to reflect changes to the regulations issued during fiscal year 1991. These policy modifications related to new or modified regulations regarding maintenance, fitness for duty of licensed reactor operators, actions against non-licensed individuals, quality management of medical radioactive materials, and standards for protection against radiation.

LEVELS OF NRC ENFORCEMENT ACTION

The severity of NRC enforcement actions varies with the seriousness of the violations and factors such as whether the licensee identified the violation, the adequacy of the licensee's corrective actions, and prior performance of the licensee. Several levels of NRC actions are available:

- Written Notices of Violation are used in instances of noncompliance with NRC requirements.
- Civil penalties are considered for licensees who evidence significant or repetitive instances of noncompliance, particularly when a Notice of Violation has not been effective in achieving the expected level of corrective action.
- Orders for modification, suspension, or revocation of licenses are used to deal with licensees who do not respond to civil penalties or to deal with violations that constitute a significant threat to public health and safety or to the common defense and security. In the latter case, the order may be made immediately effective.

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 1991 comprised:

- Approximately 70 licensing actions dealing with fuel cycle plants and facilities.
- Approximately 3,000 fuel facility and materials licensee inspections.
- Team assessments and expanded inspections at eight major licensee facilities.
- Approximately 5,600 licensing actions on applications for new byproduct materials licenses and amendments and renewals of existing licenses.

FUEL CYCLE LICENSING AND INSPECTION

Fuel Cycle Licensing Activities

By the end of fiscal year 1991, the NRC had completed 73 fuel cycle licensing actions. Table 1 shows the number of licensing actions by category.

Uranium Enrichment

In November 1990, the Congress passed and the President signed the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990 (Public Law 101–575), amending the Atomic Energy Act to provide new requirements for regulation of uranium enrichment facilities. The principal effect of the new law is that uranium enrichment facilities will be licensed in accordance with those provisions of the Atomic Energy Act which pertain to source material and special nuclear material, rather than the provisions pertaining to a production facility. Uranium enrichment facilities remain "production facilities" for other purposes of the Act, such as controlling the export of specially designed or prepared uranium enrichment equipment and preserving Federal authority in Agreement States. Licensing is a single-step process, with one license issued pursuant to 10 CFR Parts 40 and 70, rather than the two-part licensing process under 10 CFR Part 50. Public Law 101–575 contains five new licensing requirements:

- A single adjudicatory hearing on the record before issuance of a license for construction and operation.
- A prohibition against issuance of a license until the hearing is completed and a decision issued.
- The preparation of an environmental impact statement in accordance with the National Environmental Policy Act (NEPA) before the hearing is completed.
- Verification by the Commission, prior to commencement of operation, that the facility has been constructed in accordance with the license, and publication of pre-operational inspection results in the *Federal Register*.
- Maintenance of public liability insurance against bodily injury, sickness, disease, death, and loss of or damage to property arising out of or resulting from the radioactive, toxic, explosive, or other hazardous properties of chemical compounds containing source material or special nuclear material.

The NRC published proposed rule changes implementing the amendment to the Act in the *Federal Register* on September 16, 1991.

Table 1. Fuel Cycle Licensing Actions Completed in FY 1991

Category	*No. of Actions*
Uranium Fuel Fabrication	32
Uranium Hexafluoride Production	5
Fresh Fuel Storage at Reactor Sites	2
Critical Mass Materials	14
Interim Spent Fuel Storage	9
Advanced Fuel Research & Development	2
Other Source Material	6
Enrichment	2
Decommissioning Plans	1
Total	73

Legislation was again proposed that would create a U.S. Enrichment Corporation which would own and operate the existing gaseous diffusion plants of the Department of Energy (DOE) and any new Atomic Vapor Laser Isotope Separation (AVLIS) plant. The legislation would require that the plants be licensed by the NRC. At the close of the report period, the legislation had not been acted upon, and the DOE's uranium enrichment plants remained exempt from NRC licensing.

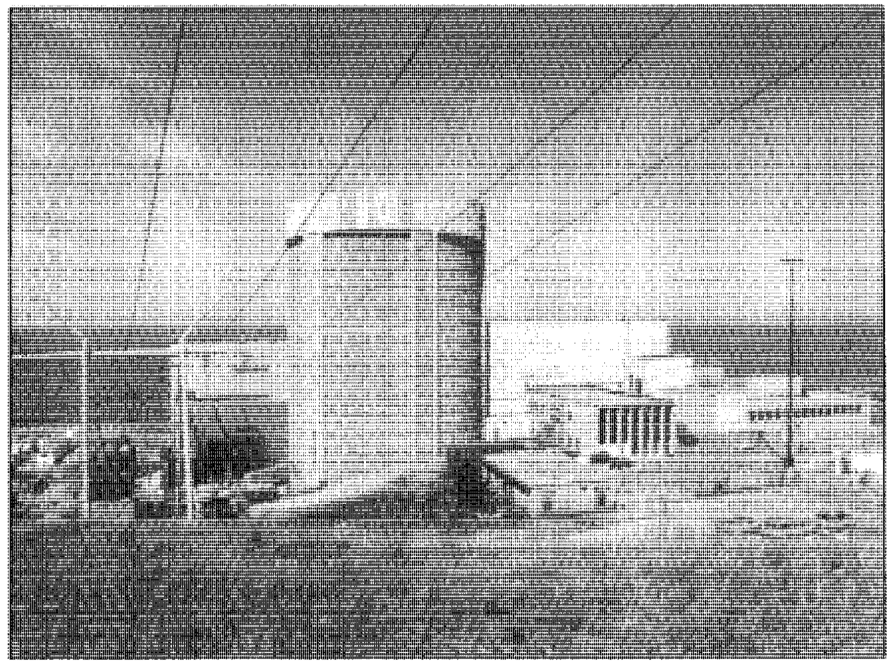
In January 1991, Louisiana Energy Services submitted an application for a license to construct and operate a gas centrifuge uranium enrichment plant, to be known as the Claiborne Enrichment Center. It would be located in Claiborne Parish, near Homer, La., and would have a capacity of 1.5 million kilograms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

Staff review of the license application continued throughout the remainder of the fiscal year. A public meeting was held in Homer, La., as part of the process leading to preparation of the required environmental impact statement. The draft environmental impact statement and safety evaluation report are scheduled for issuance in fiscal year 1992.

In January 1990, DOE submitted a plan to Congress for the demonstration, transition and deployment of the uranium AVLIS technology. The plan calls 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 may not proceed as scheduled. In anticipation of the possible passage of enabling legislation, the NRC staff has begun, on a low-priority basis, to familiarize itself with the AVLIS technology and some of the unique issues related to it.

West Valley Demonstration Project Oversight

Throughout fiscal year 1991, the Commission staff continued its safety oversight activity at DOE's West Valley Demonstration Project (WVDP), near Buffalo, N.Y. The purpose of the WVDP is to demonstrate the solidification and preparation for disposal in a Federal repository of high-level radioactive waste from reprocessing. Removal of dissolved cesium from the supernatant (liquid) portion of the waste, begun in early 1988, was declared completed in November 1990. The cesium will be combined with the solid portion of the high-level waste, which contains most of the other radionuclides. Beginning in 1996, the combined wastes will be solidified in borosilicate glass.



As the space available at reactor sites for the storage of spent fuel under water continues to diminish, utilities are turning to dry storage, approved by the NRC, in an Independent Spent Fuel Storage Installation (ISFSI). This may include the use of concrete storage casks, such as that to be used at the Palisades (Mich.) nuclear power plant, shown at right, which will have the capacity to store 24 fuel assemblies. The Palisades facility is located on the eastern shore of Lake Michigan; the pressurized water reactor plant was licensed for full-power operation in 1972.

The NRC staff monitors public health and safety aspects of the WVDP by inspections at the West Valley site and by reviewing Safety Analysis Reports submitted by DOE. DOE normally submits a separate Safety Analysis Report for each segment of the waste process, including solidification in glass-making. The staff reviews each submittal and issues a corresponding Safety Evaluation Report, presenting its conclusions regarding public safety implications of that process segment.

In 1991, the staff began its assessment of the safety of the West Valley sludge mobilization and washing system. DOE planned to begin this phase of operations in October 1991 and continue through 1993. The NRC agreed to become a cooperating agency in the preparation of an environmental impact statement (EIS) for site decommissioning. The NRC staff will develop decommissioning criteria for various aspects of the WVDP under NRC's oversight, which DOE will address in the EIS. A draft EIS is expected to be published by DOE and the State of New York in 1994.

Interim Spent Fuel Storage

Utilities are continuing to develop plans to increase storage capacity for reactor spent fuel, as the limit on available space in the on-site storage pools draws closer. Pursuant to 10 CFR Part 72, on-site dry storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI) is being adopted by a growing number of utilities to meet these needs. The ISFSIs utilize concrete or metal casks, modular vault dry storage, and concrete horizontal storage module designs.

The NRC staff is continuing to review the application submitted by Baltimore Gas and Electric Company for an ISFSI license for storage at its Calvert Cliffs, Md., site. The staff completed the review of the environmental report and issued the environmental assessment in March 1991. The safety evaluation review is ongoing.

The staff is also reviewing the application submitted by Northern States Power Company for an ISFSI license at its Prairie Island (Minn.) reactor plant. The ISFSI will use Transnuclear, Inc., TN-40 casks. A Petition to Intervene was filed by the State of Minnesota with respect to the Prairie Island ISFSI license application. The NRC, the State, Northern States Power Company, and the Prairie Island Indian Community signed an agreement for an open exchange of information concerning the company's plans to site, license and operate an ISFSI at its nuclear power plant. The agreement provides a forum for considering and resolving issues raised by the State and the Indian Community, and it establishes a cooperative working relationship with the State in conducting the license reviews. Following adoption of the agreement, the State withdrew its Petition for Leave to Intervene, obviating the need for any Atomic and Safety Licensing Board hearings. The staff's review of the environmental report and safety evaluation was under way at the close of the report period.

In October 1991, Sacramento Municipal Utility District submitted an application for a license to construct and operate an ISFSI on-site at its Rancho Seco (Cal.) nuclear power plant. The utility plans to use dual-purpose, storage/transportable casks, for ease of decommissioning at the end of useful life. The dual-purpose casks will be licensed under Part 72 for storage, and certified under Part 71 for transportation. This ISFSI will provide storage capacity for 493 spent fuel assemblies.

Throughout the year, the staff has continued the safety and environmental reviews in connection with this application. The staff is also reviewing safety analysis reports, under 10 CFR Part 72, Subpart K, for Certificate of Compliance applications to be used in general licenses.

In January 1991, Pacific Nuclear Fuel Services, Inc. (PNFS) submitted an application for a Certificate of Compliance for the standardized NUHOMS–24P cask. The NUHOMS–24P is a horizontal concrete cask with a storage capacity sufficient to accommodate 24 pressurized water reactor fuel assemblies or 52 boiling water reactor assemblies.

The staff completed the review of the Pacific Sierra Nuclear Associates (PSNA) topical safety analysis report (TSAR) in March 1991. The cask (VSC–24) is a concrete ventilated storage cask with a metal liner and with the capacity to store 24 fuel assemblies. PSNA submitted an application for a Certificate of Compliance in April 1991. In August 1991, the staff granted PSNA an exemption to fabricate three VSC–24 casks, referencing the approved TSAR in conjunction with the Certificate of Compliance review. The cask will be used at the Palisades (Mich.) ISFSI under the general license provision of Subpart K to 10 CFR Part 72.

In October 1991, Babcock & Wilcox (B&W) Fuel Company submitted an application for a Certificate of Compliance for the B&W CONSTAR cask. The B&W CONSTAR is a concrete cask with a storage capacity of 32 PWR fuel assemblies.

Independent Spent Fuel Storage at Fort St. Vrain

The High-Temperature, Gas-Cooled Reactor at Fort St. Vrain (Colo.) was permanently shut down in August 1989. The reactor is owned by Public Service of Colorado (PSC) which plans to proceed with the first stage of decommissioning by removing the fuel and other core components from the reactor vessel. Storage is required for up to 1,482 fuel elements, 37 keyed top-reflector-control rod elements, and six neutron-source elements. Because of ongoing litigation, the availability of facilities at the Idaho National Engineering Laboratory to store the fuel is uncertain. Since the Federal repository for commercial spent fuel is not available, interim storage has become necessary.

To meet this need, PSC has chosen an on-site ISFSI, licensed under 10 CFR Part 72. This ISFSI employs the Modular Vault Dry Storage (MVDS) System, designed by GEC Alstom of the United Kingdom and Foster Wheeler Energy Corporation. The system is designed to safely store the spent fuel and other core components in a contained, shielded system. The MVDS is composed of six vault modules containing a matrix of 45 storage positions each. A fuel storage container that may hold up to six spent fuel elements may be stored in each of the 270 positions. The steel-reinforced concrete vault module provides shielding around the array of stored fuel, and a cooling air inlet/outlet path. Because the ISFSI will be a stand-alone facility, the MVDS design provides the capability for direct off-site shipment of the fuel, following decommissioning of the facility.

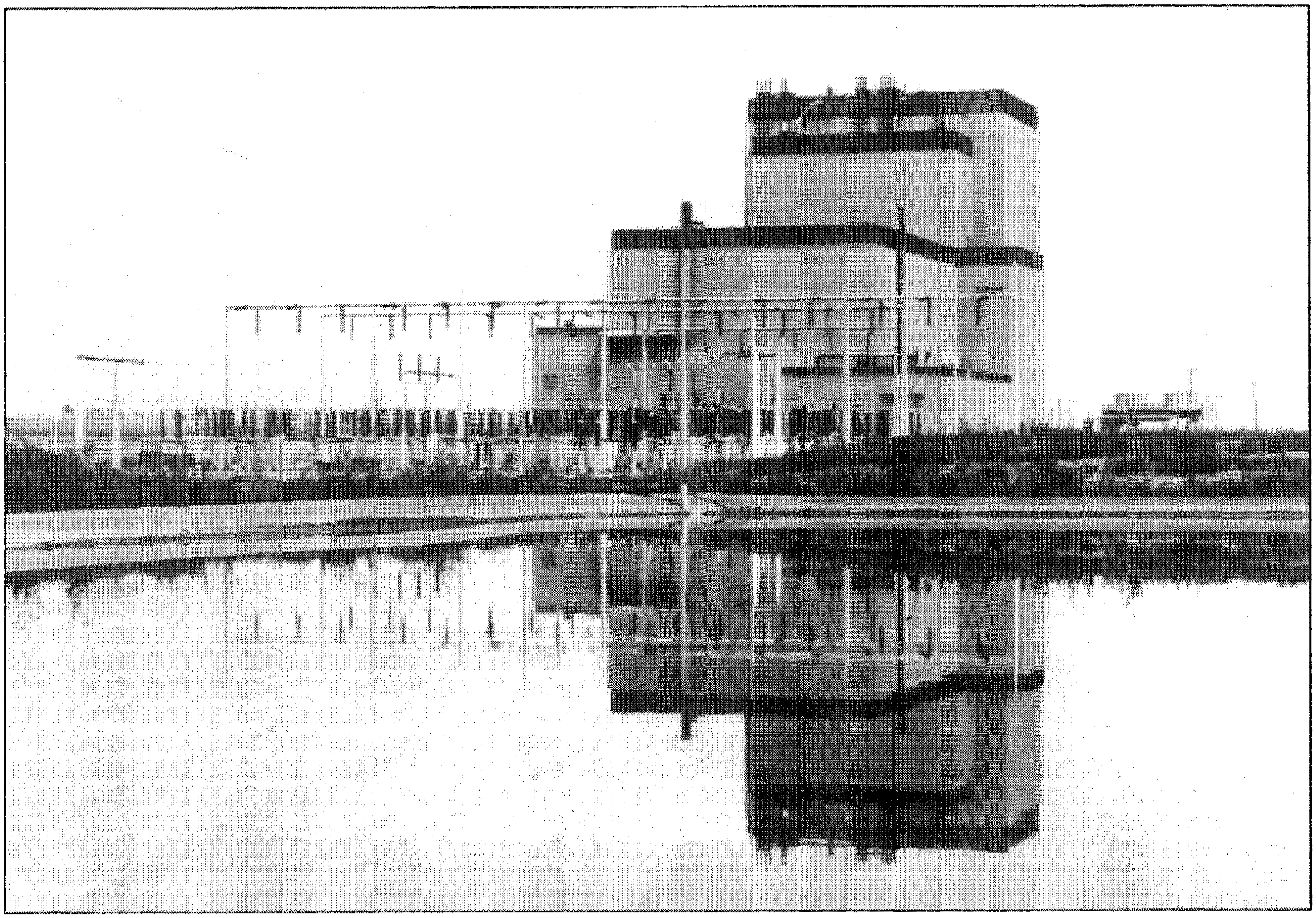
The ISFSI is designed to have a minimum life of 40 years, although it will be initially licensed, under 10 CFR Part 72, for only 20 years. The radiological consequences to the public from routine ISFSI operations are minimal. Even the postulated credible worst-case accident would result in off-site exposures well within all Federal guidelines.

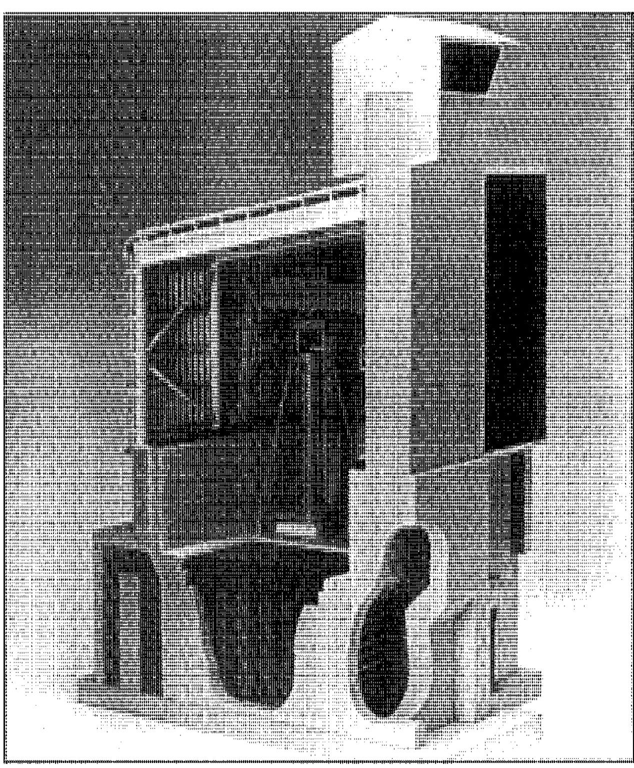
When the fuel and other core components are finally transferred to a permanent repository, the ISFSI system will retain only minimal residual contamination, and decommissioning should result in the release of the ISFSI site for unrestricted use.

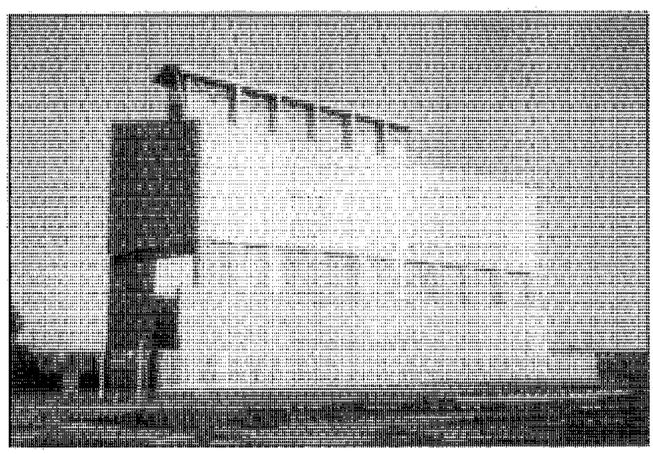
The environmental assessment and finding of "no significant impact" was published in February 1991. MVDS ISFSI construction, pre-operational tests, inspection, and the NRC staff safety review are complete. The 10 CFR Part 72 license issuance and the planned fuel loading were nearing completion at the end of the report period.

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 at the facility. The assessments evaluate 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 assessments. In fiscal year 1991, the NRC







The Fort St. Vrain (Colo.) plant was permanently shut down in 1989 and is in the process of begin decommissioned. The licensee needed storage capacity to accommodate nearly 1,500 fuel elements and other radioactive components from what was the only high-temperature, gas-cooled commercial reactor facility in the country, shown at top. The Independent Spent Fuel Storage Installation (ISFSI) adopted by the licensee employs a Modular Vault Dry Storage System, designed to provide safe interim storage of all elements and components, pending availability of a Federal repository for commercial spent fuel. At left is a cross-section model of the Fort St. Vrain ISFSI, and above is a photo of the front exterior of the completed installation.

conducted operational team assessments at two fuel facilities, and expanded inspections at the West Valley facility and several large universities holding Part 30 licenses.

MATERIALS LICENSING AND INSPECTION

The NRC currently administers approximately 7,800 licenses for the possession and use of nuclear materials in medical and industrial applications. Table 2 shows the distribution of these licenses by Region. Table 3 shows the distribution of licenses by type of use. The 28 Agreement States administer about 16,000 additional licenses. The program is designed to ensure that activities involving such uses of radionuclides do not endanger the public health and safety. NRC regional staff completed approximately 3,000 inspections of materials facilities in fiscal year 1991. The NRC Regional Offices administer all materials licenses, with the exception of exempt distribution licenses and sealed source and device design reviews, which are handled at NRC Headquarters.

The NRC completed nearly 5,600 licensing actions during the fiscal year. Of this total, about 500 were new licenses, 3,900 were amendments, 900 were license renewals, and 300 were sealed source and device reviews.

Human Factors. Human error associated with the production and non-reactor uses of byproduct material—i.e., medical and industrial uses—is a significant contributor to incidents resulting in unnecessary or excessive public and occupational exposures, to potential nuclear criticality safety incidents, and to the unintentional loss or release of material. Reduction of human error requires an in-depth knowledge of its causes. Human factors evaluations designed to acquire such knowledge with respect to applications involving teletherapy, brachytherapy using remote after-loaders, and industrial radiography continued during 1991. Contractors for the project have completed functional and task analyses of all three systems. Their analyses include an examination of factors (such as "human-machine" interfaces) that can influence to what degree persons working with the systems can reliably satisfy task requirements. Human factors evaluations of teletherapy, brachytherapy using remote after-loaders, and industrial radiography will ultimately identify and set priorities among human factors problems (i.e., human errors that affect system performance, along with the factors that contribute to those human errors). They will also seek to identify and assess various ways to resolve the problems.

In July 1991, an overview of the NRC program for the study of human factors in the medical uses of nuclear byproduct material was presented at the first International Symposium on Hospital Ergonomics. In September 1991, the NRC and its contractors made presentations at the annual meeting of the Human Factors Society on the goals and methods for the human factors evaluation of teletherapy, brachytherapy using remote afterloaders, and industrial radiography.

An NRC pilot project to better evaluate information in reports of nuclear medicine misadministrations continued during 1991. A key element in the project is a computerized data base. A preliminary summary of information in the data base indicated that continued development and use of the resource can help uncover underlying factors that lead to human errors in the field of nuclear medical applications.

Human error in the use of medical devices, including devices using nuclear byproduct material, may be reduced by means of 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 human factors analyst continued to participate in revision of the document, "Human Factors Engineering Guidelines and Preferred Practices for the Design of Medical Devices."

The NRC has actively encouraged producers and users of nuclear byproduct material to consider human factors as they attempt to improve their facilities and operations. During 1991, Syncor International Corporation, the operator of a major chain of nuclear pharmacies, began to implement a plan for human factors evaluation of its facilities. Extensive questionnaire, interview, checklist, photographic and video data were collected at a sampling of facilities. A preliminary summary and evaluation of those results was in progress at the end of the report period. Results of Syncor's human factors analysis are expected to reduce the likelihood of multiple misadministrations attributable to error at its pharmacies.

During fiscal year 1991, Nuclear Fuel Services, Inc. (NFS), a fuel cycle facility, added a human factors element to its performance improvement program through the selection of a human factors consultant to assist in development and implementation of the program. An initial assessment was made and a program plan, to be sent to NRC for comment, was under development at the end of the report period.

Table 2. Regional Distribution of NRC Nuclear Materials Licenses

(as of October 1991)

Region I	2,819
Region II	960
Region III	2,714
Region IV	841
Region V	268
Headquarters	233
Total	7,835

Industrial Uses

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 then issues a certificate of registration to the vendor. The certificate is used by the NRC or the Agreement State in its issuance of specific licenses to users of the products. This system avoids multiple filings of the source information by customers and thus expedites the licensing process.

The NRC maintains a nation-wide registry of sealed source and device designs. Agreement States also provide their certificates to the NRC registry and have access to all of the information in the registry. During the fiscal year, the staff completed nearly 300 safety evaluations for radioactive sources and devices. The computerized registry for source and device designs produced about 140 reports for NRC Regional Offices, the Agreement States, and foreign countries. The staff incorporated the Radioactive Materials Reference Manual of the Food and Drug Administration's Center for Device and Radiological Health into the nation-wide registry, as a service to the States. This manual includes a listing of products that contain naturally-occurring, accelerator-produced, radioactive material.

NRC and Agreement State representatives conducted a workshop on how to perform evaluations of source and device designs. The workshop provided, among other services, information on the development of audit procedures for vendors to ensure that the products that are sold continue to meet the terms of the certificate. The NRC has also initiated a contract to test products to determine if conditions of use and prototype design testing procedures are adequate.

Sealed Sources Exceeding Part 61, Class C. Licensees with certain sealed sources are experiencing problems disposing of the sources when they are no longer needed. Certain well-logging sources, gauges, irradiators, and teletherapy sources are not accepted for disposal at commercial burial sites because, when packaged for disposal, concentrations of radioactivity exceed the limits for Class C low-level waste, as set out 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 DOE have discussed the need for the DOE to accept and store these 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.

DOE has retrieved and is storing several gauges that were abandoned in the public domain. The NRC staff continues to inform the DOE of its concerns and has requested that the DOE identify an interim storage facility or establish a fee schedule for interim disposal, and also expand its emergency retrieval provisions to prevent sources from being abandoned because of high disposal costs or lack of a disposal site.

General License Effectiveness. For several years, the NRC has been studying the regulatory framework for licensing the possession and use of certain measuring and

Table 3. Distribution of Nuclear Materials Licenses by Type of Use

(as of October 1991)

Byproduct Material		7,384
ACADEMIC USERS	(86)	
MEDICAL USERS	(2,442)	
*Medical Institutions	1,513	
*Private Practices	547	
*Teletherapy	188	
*Veterinary	123	
*Other Medical	71	
MEDICAL USERS	(4,856)	
*Well Logging	94	
*Measuring Systems	2,993	
*Manufacturing/Distribution	329	
*Waste Disposal	14	
*General/Exempt Distribution	210	
*Radiography	239	
*Irradiators	241	
*Research & Development	736	
Source Material		196
Nuclear Material		255
Total		7,835

gauging devices containing nuclear materials, under the general license requirements of 10 CFR Part 31. In particular, the NRC has evaluated the performance of users possessing devices that are held under 10 CFR 31.5. An NRC mail survey sought data from about 3,000 general licensees in non-Agreement States for three categories of devices: gauges, analytic instruments, and self-powered exit lights. The questionnaire was designed to obtain information about the respondents' knowledge of the regulatory requirements for general licensees and their practices and procedures concerning maintenance, testing and disposition of the generally-licensed devices. The response rate for the survey was between 84 and 94 percent, depending on the type of device in question. Although a high proportion of the general licensees evinced accurate knowledge of the regulatory requirements and compliance with them, a significant number of mistaken and/or uncertain notions were exposed.

Particularly noteworthy results of the survey included the following:

- Survey results indicate that uncertainty about general license requirements is greatest among firms with tritium exit signs. These firms frequently reported that they did not possess a copy of the general license requirements and had not appointed a contact to monitor compliance with those requirements. In contrast, licensees with analytic devices and gauges were much more likely to report knowledge of general license requirements and assignment of a company representative to monitor their general license operations and to ensure compliance with all requirements.
- No category of survey respondents reported numerous or frequent relocations of devices. In most cases, respondents reported that specific licensees carried out those relocations that had occurred.
- Based on survey results, loss, theft, or damage to devices is not a major problem for survey respondents. The most numerous reports of inability to locate devices came from respondents for tritium exit signs,

and the only reports of theft of devices also came from them.

The staff is working on a proposed rule to make general licensees more aware of NRC requirements and to have them respond to an annual questionnaire about devices in their possession.

Quality Assurance and Control for Manufacturers. The staff has developed a draft Quality Assurance and Control Manual for manufacturers and vendors of sealed sources and devices containing byproduct material. The manual includes an overall description of a quality assurance and control program, a checklist for auditing programs, and examples of program procedures and documentation. A pilot program was instituted to evaluate the draft manual and its impact on both NRC and the vendors. The project included staff visits to manufacturers representing a variety of products in distribution, both as to size and type. Information and insights gained from these visits will be reflected in the revision of the manual, which will then be used to complete the pilot evaluation program.

Irradiator Rule. On December 4, 1990, the NRC staff published for public comment, a proposed new section of the regulations, 10 CFR Part 36. The proposed regulation specifies 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 affect a product's condition (e.g., to sterilize disposable medical supplies, such as syringes or gloves, or to polymerize compounds in wood finishes. For more information on irradiators, see *1990 NRC Annual Report,* pp. 82–83.)

On February 12–13, 1991, the staff held a public meeting to discuss the proposed rule. The staff answered questions from the participants, clarified various matters and urged the participants to submit comments and suggestions in writing. The staff is evaluating the public comments and preparing a final rule which should be published in fiscal year 1992.

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. The NRC has a total of 239 radiography licenses in effect. Portable radiography devices may contain radioactive sources with as much as 200 curies of iridium-192 or 100 curies of cobalt-60. Devices employed at fixed facilities may contain sources of several hundred curies.

Workers in the radiography industry who do not follow required procedures exactly incur a high potential for overexposure and have, in fact, sometimes received significant overexposures. The NRC staff has several initiatives under way aimed at reducing these incidents. One of them is a rule change, "Safety Requirements for Industrial Radiography Equipment," published in final form in January 1990. One provision of the rule, a requirement for radiographers to use alarm rate-meters as an additional form of personnel monitoring, became effective on January 10, 1991. Other provisions of the rule concern design, manufacture and testing of radiographic equipment. Some of these measures become effective in January 1992, and the remainder in January 1996. The NRC's enforcement policy was also revised to specify that failure to use NRC-required radiographic equipment, radiation survey instruments, or personnel monitoring during radiography operations is a violation which causes NRC significant concern and for which civil penalties will be considered.

Another initiative in this area is the development of a certification program for industrial radiographers. As described in the *1989 NRC Annual Report,* p. 81, and the *1990 NRC Annual Report,* pp. 74 and 75, 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 fiscal year 1991, the NRC staff worked closely with the ASNT, the Conference of Radiation Control Program Directors, the State of Texas, and other States to foster cooperation and understanding about implementation of the ASNT certification program. As of August 1991, ASNT had administered its IRRSP examination (prepared by the State of Texas) to 159 applicants and had issued certification documents to 41 individuals.

The first of two planned rulemakings on radiographer certification became effective on April 18, 1991. This recent amendment gives existing NRC licensees and applicants the option to affirm that all their active radiographers will be certified before beginning their duties as radiographers, in lieu of meeting the current regulatory requirements to describe an initial radiation safety training and qualification program. NRC's existing radiography licensees are also allowed to substitute ASNT certification for certain training and experience verification procedures described in their license applications. This rule change is intended to encourage voluntary participation in ASNT's IRRSP certification program.

In a second rulemaking action, the staff plans to develop a rule for Commission consideration that would mandate third-party certification. The staff has accelerated its work on this rulemaking and anticipates publishing a proposed rule in fiscal year 1992.

Medical Uses

Advisory Committee on the Medical Uses of Isotopes. The Advisory Committee on the Medical Uses of Isotopes (ACMUI) met in January and May of 1991; the topics discussed included the Quality Management (QM) Rule, low-level waste concerns, the Interim Final Rule on the radiopharmacy petition, issues related to the use of the term "Supervision" in Part 35 and the Practice of Radiopharmacy. An ACMUI sub-committee met in March to discuss the QM Rule.

The Commission directed the staff to expand the ACMUI, in order to achieve a more balanced representation of the medical community. In addition, members' terms will be limited to two years with the option of one reappointment. To fulfill this directive, four members have been added to the committee: an individual qualified to address patients' rights and care; a person with broad experience in medical regulation, as conducted by individual States; a radiation oncologist, with experience in brachytherapy; and a representative from the Food and Drug Administration (FDA). Three members of the committee, each of whom had served for many years, were retired. The staff plans to seek three additional members in 1992, including an individual qualified to address medical research, an individual experienced in hospital administration/management, and an oncology physician with experience in teletherapy. (Current membership of the ACMUI 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. Eleven persons applied, and the NRC evaluation panel reviewed each application, conducted interviews and selected a physician and a radiopharmacist for the first one-year fellowships.

The physician is an expert in the diagnostic and therapeutic application of radiopharmaceuticals. He is a retired professor, Division Chairman, and Director of a nuclear medicine department at a large teaching hospital and medical school. He is a past president of both the American College of Nuclear Physicians, and a chapter of the Society of Nuclear Medicine.

The radiopharmacist is a board-certified nuclear pharmacist, an expert in the field of radiopharmacy, and expert in the diagnostic and therapeutic application of radio-labeled monoclonal antibodies.

Quality Management Rule. On January 27, 1992, regulations will become effective requiring licensees to establish a quality management program, in compliance with 10 CFR 35.2 and 35.32, if they administer radiation from sealed sources containing byproduct material for therapy, or if they administer radiopharmaceuticals containing quantities greater than 30 microcuries of either sodium iodide I-125 or I-131 and radiopharmaceuticals for therapy. Their programs must be submitted to NRC and in effect by January 27, 1992. Implementation of this rule should provide high confidence that the byproduct material or radiation from byproduct material will be administered as directed by an authorized user-physician.

Petition for Rulemaking: Nuclear and Pharmacological Issues. 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 that are normally regulated by the FDA Center for Drug Evaluation and Research and Center for Biologics Evaluation and Research.

On August 23, 1990, the NRC published an interim final rule (55 FR 34513) addressing two issues raised in the petition. This rule permits physician-directed departures from the manufacturer's instruction for diagnostic reagent kit preparation and generator elution, and from the package's administrative indications for use and route of administration for therapeutic radio-pharmaceuticals, provided that certain conditions are met and records are kept. The interim final rule will be effective for three years. Data on the use and the frequency of physician-directed departures made in accordance with the interim final rule are being collected during NRC's inspections of medical facilities and commercial nuclear pharmacies. The NRC will continue to work closely with the FDA, the nuclear medicine community, and the radiopharmacy community, to resolve the remaining issues raised by the petition.

EVENT EVALUATION AND RESPONSE

The NRC continued 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. The NRC conducted an exercise in June involving the fuel facility in Erwin, Tenn. Exercises of this nature allow the NRC to evaluate new emergency response procedures related to events at materials licensees, and to give the licensee a better perspective of the response it might expect from the NRC in such an event.

Contaminated Steel Fence Parts. On August 9, 1991, the State of Washington notified NRC that radioactive material had been detected when a Tri-Cities Fencing truck was surveyed as it attempted to leave the Hanford Reservation in Washington. The Department of Energy

(DOE) staff found two chainlink fence tension bars on the truck to be contaminated with cobalt–60. The States of Washington and Oregon radiation control programs were alerted and they traced the material to a Portland, Ore. wholesale fence distributor. The Oregon inspectors found two pallets of tension bars in the distributor's yard and further analysis gave the concentration as 837 picocuries of cobalt-60-per-gram of tension bar steel. The surface readings ranged from 11–to–220 microrad-per-hour for individual bars, to 2,500 microrad-per-hour for a pallet of 500 bars. Within days, the NRC and several States had assessed the extent of the contamination at other wholesale fence distributors throughout the United States.

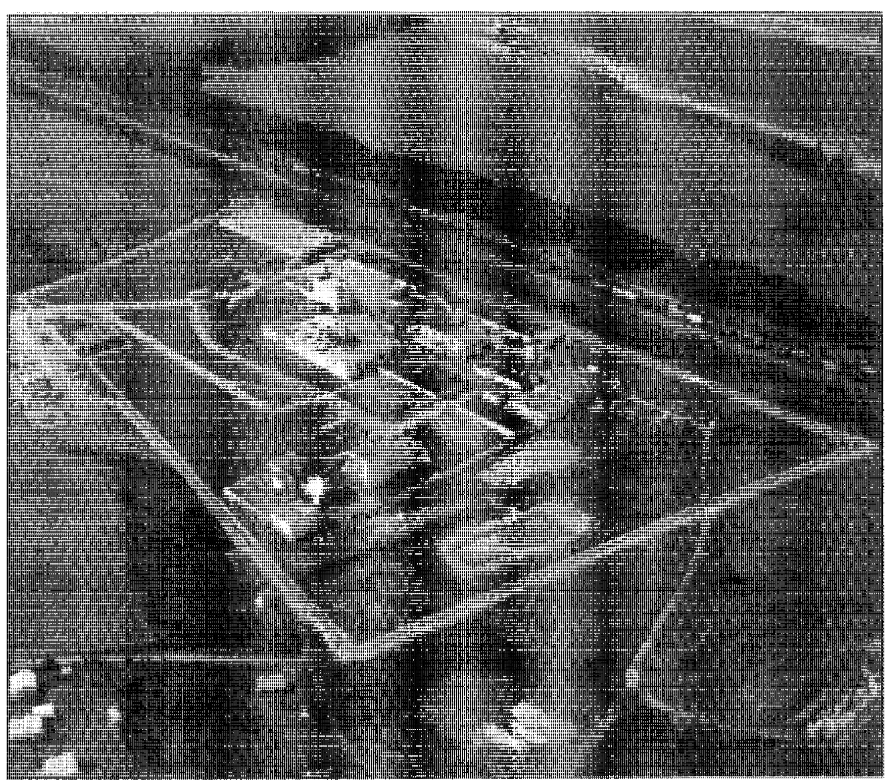
A confirmatory action letter was sent to the two importers and their known distributors of steel fencing products from India. The letter mandated that addressees would survey their inventories, segregate any contaminated products, and provide the NRC with survey results and any information about incoming shipments. Addressees were also requested to contact their suppliers in India and to ask them to coordinate with Indian authorities.

After a thorough analysis, the staff concluded that no action was necessary for bars already installed in fencing or in the possession of retail companies, because of the estimated low risk and wide distribution of the fence products. It was decided as well that contaminated bars possessed by wholesale distributors should be returned to India or transferred to a low-level waste site for disposal.

The Government of India also was informed and has initiated its own investigation into the cause of the incident.

Potential Criticality Accident. On May 28–29, 1991, at the General Electric (GE) Nuclear Fuel and Component Manufacturing facility, located near Wilmington, N.C., approximately 150 kilograms of uranium were inadvertently transferred from "safe geometry" process tanks to an unsafe geometry tank located at the waste treatment facility, thus creating the potential for a localized criticality safety problem. The excess uranium was ultimately safely recovered when the tank contents were centrifuged to remove the uranium-bearing material. Subsequently, NRC dispatched an Incident Investigation Team (IIT) to determine what had happened, to identify probable causes, and to make other findings and conclusions. The IIT report (NUREG–1450) was published in August 1991, and identified problems at GE which the team determined to be the incident's root causes, as well as a number of weaknesses in NRC's regulatory guidance, licensing and inspection programs.

Following the incident, a number of corrective actions were defined both at GE and within NRC. The licensee proposed a number of steps to tighten its criticality safety program, its process control system, and its maintenance evaluation program. Within the NRC, short term and long range initiatives were framed regarding emergency preparedness reviews, facility operational safety and operating experience, and criticality safety.



In its continuing effort to maintain the capability to respond to unplanned events at nuclear fuel fabrication facilities, the NRC conducted an exercise in June 1991 at the plant in Erwin, Tenn., operated by Nuclear Fuel Services.

Materials Regulatory Review Task Force. Following the GE incident, the NRC assembled a four-person Task Force to re-examine the regulatory process for large material facilities and to identify any generic weaknesses that may have contributed to the GE incident. The Task Force also reviewed other significant events and shortcomings that the staff had uncovered from other events but had not had the opportunity to correct. The Task Force report was completed in late September, and circulated for staff comment. At the end of the report period, the staff was beginning evaluation of the draft report findings. A report is expected in early fiscal year 1992.

Sequoyah Fuels Corporation. In August 1990, while excavating two underground storage tanks, Sequoyah Fuels Corporation (SFC) discovered elevated uranium concentrations in the groundwater. The NRC's inspection efforts discovered many weaknesses in SFC's programs. Because of concerns over the licensee's safety and environmental programs, the NRC issued a Demand for Information in November 1990. Responding to the Demand, SFC set up oversight team, comprised of consultants, to oversee SFC's daily operational activity. SFC also contracted for a management appraisal of its organization. The May 1991 management assessment report contained 47 recommendations, in the areas of policy, planning, communications, organization, management controls, human resources management, training, and regulatory relations. In its July 1991 response, SFC agreed to implement most of these recommendations. The SFC also conducted a facility environmental investigation. The report of this investigation was submitted to NRC in July 1991.

In early October 1991, the NRC issued an Order Modifying License and a Demand for Information. The NRC ordered that the "Manager, Environmental" be removed from supervisory or managerial responsibilities over NRC-regulated activity for a period of one year. It also ordered SFC to provide information to demonstrate why the NRC should not remove the "Manager, Environmental" from serving in any capacity involving licensed material. And the NRC ordered SFC not to conduct production operations until SFC performed certain tasks. SFC was required to submit, 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 document also demanded information to demonstrate why SFC's license should not be modified to prohibit the Senior Vice President, the Vice President of Regulatory Affairs, and the Health Physics Supervisor from serving in any capacity involving NRC-regulated activity.

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 of nuclear facilities. In general, safeguards for licensed nuclear fuel facilities and non-power reactors emphasize protection against theft or diversion of SNM, while safeguards associated with power reactors stress protection against radiological sabotage. (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. In general, SSNM is high-enriched uranium–235 (HEU), uranium–233, or plutonium.)

During fiscal year 1991, NRC safeguards requirements were in effect with respect to 113 power reactors, 47 non-power reactors, and 15 active non-reactor facilities. Requirements also affected 11 shipments of spent fuel, 28 shipments of SNM involving more than one but less than five kilograms of HEU, and 3 shipments 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). These two agencies have delineated their respective regulatory responsibilities in this area through a Memorandum of Understanding (MOU). For international shipments, DOT is the designated U.S. Competent Authority and is responsible for implementing International Atomic Energy Agency (IAEA) standards. The NRC advises DOT on technical matters.

STATUS OF SAFEGUARDS AND TRANSPORTATION IN 1991

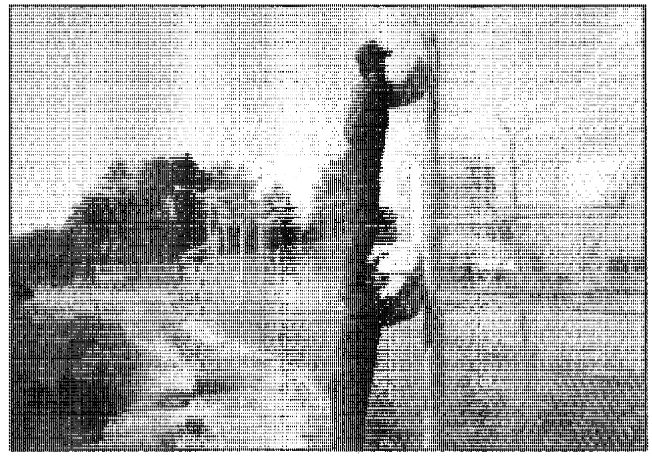
Reactor Safeguards

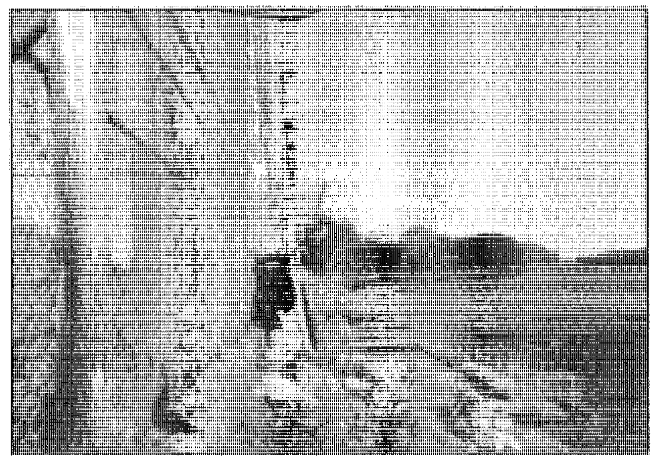
Power Reactor Safeguards Inspection and Licensing. Within the five NRC Regional Offices, a total of 170 safeguards inspections were conducted at the 74 sites which currently house the 113 licensed nuclear power reactors under NRC safeguards requirements. Approximately 228 revisions to licensee security, contingency, and guard training plans were reviewed by both regional and headquarters staff and found acceptable.

Safeguards Effectiveness Reviews at Power Reactors. The NRC staff, assisted by U.S. Army Special Forces personnel, completed the Regulatory Effectiveness Review (RER) Program in May 1991. The objectives of the program were to assess NRC's reactor safeguards regulations and to evaluate the practical effectiveness of licensees' safeguards programs for protecting vital equipment at power reactors. Since the start of the program in 1981, RERs have been conducted at each operating reactor site. The reviews have confirmed the basic soundness of safeguards regulation and contributed to the implementation of over 500 significant safeguards improvements at operating power reactors.

Effectiveness reviews by interdisciplinary teams will continue, but with increased specificity and concision. The revised program, Operational Safeguards Response Evaluations (OSRE), will evaluate licensees' contingency response capabilities by focusing on the interactions between operations and security personnel in establishing priorities for the protection of equipment, and also by scrutinizing the defensive strategies used. OSRE teams will also conduct safety/safeguards interface reviews to continue to assure that safeguards measures do not adversely affect the safe operation of the plant.

Fitness-for-Duty at Power Reactors. Power reactor licensees are required to implement fitness-for-duty (FFD) programs, under 10 CFR Part 26. The Commission





In May 1991, the NRC completed a program begun 10 years earlier under which NRC staff, with the assistance of U.S. Army Special Forces personnel, from Fort Bragg, N.C., tested the safeguards protections in place at every operating reactor site in the nation. These reviews have confirmed the effectiveness of safeguards regulations and improvements brought about in the protection of vital equipment at nuclear power plants. A typical assessment involved practical exercises, as shown here. Clockwise, from top left, NRC and Fort Bragg personnel test perimeter intrusion detection systems; the team checks out system ability to detect fence-cutting; and the team puts the test to the closed-circuit TV system's reliability in detecting intruders.

amended this rule during the report period (56 FR 41922, August 26, 1991) to clarify its intent and affirm the unacceptability of taking actions against an individual solely on the basis of *preliminary* results from a drug-screening test, as well as the acceptability, under certain conditions, of employer action—up to and including the temporary removal of an individual from unescorted access or from normal duties—on the basis of initial screening tests for marijuana or cocaine. Although the existing rule appears to be achieving the desired effects, the Commission is considering other changes that would take advantage of lessons learned during the first 18 months of the program.

An overview of the lessons learned during the first full year of implementation and a discussion of a variety of relevant issues is provided in a document entitled, "Fitness for Duty in the Nuclear Power Industry: A Review of the First Year of Program Performance and an Update of the Technical Issues" (NUREG/CR-5784). Program performance data provided by the licensees have been summarized in "Fitness for Duty in the Nuclear Power Industry: Annual Summary of Program Performance Reports, CY 1990" (NUREG/CR-5758). That report indicates that over 278,000 tests for the presence of illegal drugs and alcohol were conducted during calendar year 1990, of which 2,409 were positive. The majority of the positive test results (1,548) were obtained through pre-access testing (a 1.26 percent positive rate). There were 550 positive tests from random testing (a 0.37 percent positive rate). The positive rate varied by worker category: for example, 0.28 percent of the random tests of licensee employees were positive; for long term contractors, the rate was 0.49 percent; and for short term contractors, the rate was 0.58 percent.

Non-Power Reactors. The NRC conducted 29 safeguards inspections in this sector during fiscal year 1991. Efforts are continuing toward converting 25 non-power reactors from the use of high-enriched uranium (HEU) to low-enriched uranium (LEU) fuel. The NRC regulation governing this effort is predicated on (1) availability of Department of Energy (DOE) funding, (2) availability of a suitable replacement fuel, and (3) whether or not a reactor has a "unique purpose" requiring the use of HEU. At the end of the fiscal year, six licensees for non-power reactors had their HEU-LEU conversion orders issued and had converted, or were in the process of, converting; six had been funded by DOE; and four university licensees for non-power reactors were awaiting funding. Two commercial licensees with non-power reactors were not scheduled to receive DOE funding; one commercial licensee for a non-power reactor was in the process of decommissioning; and, in cases involving two university non-power reactors, there was no suitable fuel developed (a "unique purpose" application is being considered for one of these). One "unique purpose" application was under consideration by the Commission for a government-owned non-power reactor; two universities are planning to decommission their non-power reactors; and one non-power reactor license was terminated.

Advanced Reactors. Safeguards reviews of advanced light water reactor standard designs continued to emphasize 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 the operating procedures developed for new plants." In fiscal year 1991, the Electric Power Research Institute included in its "Advanced Light Water Reactor Utilities Requirements Document" a requirement for plant designers to analyze their designs for potential sabotage vulnerabilities that could be minimized through design modifications. In its "Advanced Boiling Water Reactor Standard Safety Analysis Report," General Electric included an insider sabotage analysis and made changes to some design details as a result of that analysis. Staff review of these design documents takes into account the need to assure that safeguards measures will not interfere with safe reactor operation.

Fuel Cycle Facility Safeguards

There were 15 active licensed non-reactor nuclear facilities subject to NRC safeguards requirements during the report period. Of these, nine were major fuel fabrication facilities. Three of the 15 facilities had "formula quantities" of SSNM, requiring extensive physical security and material control and accountability measures. ("Formula quantity" means SSNM, in any combination, in a quantity of 5,000 grams or more, with a specified amount of the uranium-235 isotope.) One of these three, UNC Naval Products, of Montville, Conn., was in the process of being shut down and decommissioned, because of reductions in orders for naval reactor cores. Work in progress was either to be completed or to be transferred to the Babcock & Wilcox facility in Lynchburg, Va., by the end of calendar year 1991.

To ensure that the physical protection provided for the SSNM at these facilities is essentially equivalent to that afforded weapons-usable materials in the government sector, the NRC upgraded its protection requirements in several areas. Licensees for the two remaining facilities possessing formula quantities have installed additional barriers at the site perimeter (including substantial vehicle barriers), conducted periodic tactical drills and exercises (including force-on-force techniques), and armed the Tactical Response Teams with more powerful firearms. Other improvements under development include physical fitness training programs for facility security personnel and upgraded firearms qualification requirements. Besides the physical protection regulations, the NRC requires licensees possessing SNM to have systems in place for control and accounting of nuclear materials in process and in storage. In fiscal year 1990, licensees possessing quantities of SSNM adopted control and accounting systems designed to ensure rapid detection of the loss of five or more kilograms of SSNM. Shutdown requirements associated with these systems were implemented in July 1991.

An application for a license to control and operate a commercial uranium enrichment plant was submitted by Louisiana Energy Services, Limited Partnership. Review of the application is being conducted under the provisions of Public Law 101-575; initial operation is planned in calendar year 1995.

Fuel Cycle Facility Inspections. Comprehensive physical security and material control and accounting inspections were conducted at the nine major fuel fabrication facilities. Special teams also completed inspections of the newly implemented physical security upgrades at two facilities possessing formula quantities. Although no items of noncompliance were identified, certain minor improvements of the installed system were deemed necessary.

Transportation

Spent Fuel Shipments. The NRC approved 31 transportation routes as acceptable for protection against radiological sabotage, and 11 spent fuel shipments were made over approved routes during fiscal year 1991.

Spent fuel shipping included two series of shipments by rail. One series began in fiscal year 1989. Four more of 35 programmed shipments—from the Brunswick (N.C.) nuclear power plant to the Harris (N.C.) nuclear power plant- -have now been completed. The other series—from the Robinson (S.C.) nuclear power plant to the Harris plant—involves a series of four shipments, two of which have been completed. The spent fuel pool at the Harris plant is configured to store a large number of spent fuel assemblies. The planned shipments 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. NRC regional personnel continued to work with local law enforcement agencies in conducting field surveys of routes proposed for shipments of spent fuel. A "Public Information Circular for Shipments of Irradiated Reactor Fuel" (NUREG-0725, Revision 7), published by the NRC, reports on all 1,101 highway and 82 rail shipments of spent fuel within the United States subject to NRC safeguards regulations, from 1979 through 1989.

SSNM Shipments. Twenty-eight shipments of less than five but more than one kilogram of HEU were completed during fiscal year 1991. These included 12 export shipments, five foreign shipments that entailed transient transport through the United States, and 11 domestic shipments of SSNM. Three export shipments of five or more kilograms of HEU were also made during the fiscal year; the domestic portions of these shipments were made by DOE.

Tracking International Shipments of SNM. The NRC regulations require licensees to notify the NRC of international shipments of SNM and natural uranium. During fiscal year 1991, the NRC received more than 300 such notifications, about two-thirds of which were forwarded to the Department of State or the Department of Transportation 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. This total effort involved more than 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 with quality assurance programs of the container suppliers continued in fiscal year 1991. Inspections were conducted at seven facilities, representing a broad spectrum of the industry. The container-supplier inspection program includes designers, fabricators and distributors that 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. In fiscal year 1991, the quality assurance inspection program was expanded to include inspection of spent fuel dry storage casks licensed under 10 CFR Part 72. Two inspections of dry storage casks were conducted, supplying information on the implementation of the quality assurance requirements in fabrication, loading and maintenance.

Plutonium Air Shipment Criteria Development. Section 5062 of Public Law 100-203 imposes requirements

A loaded spent fuel dry storage cask is transported from the spent fuel pool to the dry storage area at the Surry (Va.) nuclear power plant. Quality assurance inspections of spent fuel storage casks was initiated during fiscal year 1991, and this cask (Model I28, made by the Nuclear Assurance Corporation) was among those inspected during the report period.

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 1991, the NRC prepared an interim report on the conduct of feasibility studies related to the testing of such packages. Development of the feasibility studies was requested and funded by the Power Reactor and Nuclear Fuel Development Corporation, on behalf of the Japanese Government. Contract support for this effort is provided by the Lawrence Livermore National Laboratory.

Incident Response Planning And Threat Assessment

The NRC staff assesses threats to NRC–licensed facilities, materials and activities, and prepares the NRC's safeguards incident response plans for responding to actual thefts of nuclear material or radiological sabotage of nuclear facilities or activities. The safeguards staff maintains close and continuing contact with the intelligence community, including participation in regular interagency meetings of Federal agencies that are concerned with and prepared to deal with terrorism. Other liaison activity includes briefings and consultations with the representatives of other governments regarding NRC threat assessment and incident response activities. As part of these cooperative efforts, the NRC and the Federal Bureau of Investigation promulgated a revised Memorandum of Understanding regarding information exchange, incident response and related mutual support.

In response to events in the Persian Gulf, the staff expanded the scope of its safeguards concentration. Each Region was visited and provided an updated threat assessment, and the NRC Headquarters Duty Officers were provided training on responding to telephonic nuclear extortion threats. All reported threats related to Desert Shield and Desert Storm were closely monitored and analyzed, and the Commission and other NRC management were provided briefings about them, some on a daily basis. Increased liaison with other Federal agencies was conducted during Desert Shield and, in coordination with the DOE, the NRC issued four advisories to selected licensees. During this time, no significant change in the domestic threat environment emerged.

During fiscal year 1991, the staff discerned no significant change in the threat environment that would affect the NRC's current safeguards regulations. Two techniques are employed in assessing reported threats to the NRC's 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 the report period, the fuel cycle safeguards incident response plan was reviewed and updated. In June 1991, incident response training was completed and a safeguards exercise involving NRC headquarters and regional staff and Nuclear Fuel Services was conducted. Several improvements were identified. In October 1990, a less extensive "Table Top" exercise was completed by the Fuel Cycle Safeguards Incident Response Team.

The staff continued to analyze safeguards events related to threats and incidents in search of trends, patterns and anomalies. The "Safeguards Summary Event List" (NUREG–0525), a compilation of safeguards events, was revised in July 1991 (Rev. 17) to include events occurring through December 1990. This document was distributed to the licensed nuclear community, foreign governments, the Congress, and other Federal agencies.

During the fiscal year, the Safeguards Event Analysis Program continued to focus on establishing consistent event reporting throughout the industry and on providing more definitive and meaningful feedback to licensees and the NRC staff. Four reports, distributed to NRC staff and to all reporting licensees, presented statistical data on hardware system and human error events, and furnished root–cause analyses performed by some licensees that resulted in improved equipment operation or reduced human error. The staff also participated in four regional workshops on Backfit and Event Reporting, sponsored by the NRC Office for Analysis and Evaluation of Operational Data, and gave briefings to licensees and NRC staff at two regional nuclear security association meetings.

International Safeguards

NRC/IAEA Interaction. The principal interaction between the NRC and the International Atomic Energy Agency (IAEA) during 1991 involved the application of international safeguards (pursuant to the US/IAEA Safeguards Agreement) to two NRC–licensed facilities. The General Electric (GE) Low–Enriched Uranium Fuel Fabrication Plant in Wilmington, N.C., and the Babcock & Wilcox (B&W) Fuel Company in Lynchburg, Va., were visited by IAEA Inspectors on several occasions during the year for ledger audits and material verifications. Annual physical inventory verifications were performed at B&W and GE in July and October 1991, respectively. A second area of cooperation with the IAEA during fiscal year 1991 involved monitoring and coordinating the reporting of nuclear material inventory and transaction data to the IAEA by Westinghouse (Columbia, S.C.), Siemens Nuclear Power Corp. (Richland, Wash.), and

Combustion Engineering (Windsor, Conn.). This information was reported pursuant to the Protocol to the US/IAEA Safeguards Agreement.

In May 1991, representatives of the IAEA, the NRC and other U.S. agencies met in Washington, D.C., to discuss safeguards issues related to implementation of the Safeguards Agreement. During this meeting, the IAEA noted that all safeguards goals had been attained at the two facilities (GE and B&W) that were subject to safeguards inspections.

The NRC works to improve the technical effectiveness of IAEA safeguards both by means of direct interaction with the IAEA and by means of participation in U.S. Government interagency efforts. The NRC is contributing to the design of safeguards systems by participating in the consultants group on the development of methods for the evaluation of effectiveness of safeguards systems. The NRC supports improved safeguards at reprocessing plants by participation in the international forum called LASCAR (Large Scale Reprocessing plant safeguards). LASCAR is helping to assure that information on all technical considerations related to safeguards is available to those designing reprocessing plant safeguards systems and to the IAEA. The NRC is also contributing to the improvement of IAEA safeguards for reprocessing plants through development of the Adjusted Running Book Inventory Technique. The most important interagency effort is participation in oversight of the U.S. Government Program of Technical Support to Agency Safeguards (POTAS).

International Physical Protection. In connection with its export licensing program, the NRC conducts an exchange of information on the physical protection of nuclear facilities. Information exchange visits are made to all countries which have imported a significant amount of nuclear material from the United States or received retransfers of U.S.-origin material. During fiscal year 1991, visits were made to the United Kingdom, Indonesia, Australia and Mexico. Similarly, teams from France, Denmark, Japan, and Sweden visited the NRC in the United States.

REGULATORY ACTIVITIES AND ISSUES

Proposed Rules

The following rulemaking actions were initiated during fiscal year 1991:

- Work has begun on a proposed rule to amend 10 CFR Part 73 to clarify physical protection requirements. This rule would amend the language of Section 73.40(a) to clearly indicate that this section is intended as a general statement of the requirements for physical protection and that the detailed protection requirements for each class of licensed facility or material are provided in other sections of Part 73. The proposed rule would also require protection against radiological sabotage at non-power reactors when it is deemed necessary to protect the public health and safety.
- Work is under way on a proposed rule to amend 10 CFR Part 26 to extend the fitness-for-duty rule to licensees who possess, use, or transport Category I (unirradiated formula quantity) material. The final rule is expected to be published in November 1992.
- Work has began on a proposed rule to amend 10 CFR Part 11 to include acceptance of the DOE-L or DOE-Q Reinvestigation Program for NRC-R SNM access authorization renewal requirements. The final rule is scheduled to be published in late 1991.

The following rulemakings continued during fiscal year 1991:

- Work is continuing on a rulemaking to upgrade the requirements for the physical protection of SSNM in transit. Commercial shipments of SSNM are currently being made by DOE. The proposed rule would upgrade NRC regulations to make commercial transport protection comparable to that provided by DOE. The proposed rule is expected to be published in early 1992.
- Work continues on a rulemaking to amend 10 CFR Part 73 to establish upgraded Weapons Firing Qualification Requirements and Physical Fitness Training and Performance Testing Requirements, for all security personnel at Category I fuel facility licensees. The final rule is expected to be published in June 1992.
- Work continues on a final rulemaking to modify 10 CFR Parts 70, 72, 73, and 75. These changes will: (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. The final rule is expected to be published in December 1991.
- Work is continuing on a final rule to amend 10 CFR Part 74 to establish material control and accounting measures for uranium enrichment facilities that would produce low-enriched uranium for commercial light water reactors. The final rule is expected to be published in November 1991.

Experts in safeguards technology from around the world met in Williamsburg, Va., from July 15–to–19, 1991, to discuss international safeguards for large reprocessing plants. NRC staff represented the U.S. Government interagency team, and they met with counterparts from Germany, France, Japan, the United Kingdom, the International Atomic Energy Agency (IAEA), and the EURATOM Safeguards Inspectorate.

Final Rule

The following rulemaking was completed and published in fiscal year 1991:

- On April 25, 1991, the NRC published Section 73.56 of Title 10 of the Code of Federal Regulations, "Personnel Access Authorization Requirements for Nuclear Power Plants", to provide increased assurance that individuals granted unescorted access to protected and vital areas are trustworthy and reliable and do not pose a threat to commit radiological sabotage. The rule requires an access authorization program that consists of three elements: background investigation, psychological assessment, and behavioral observation. The required elements have long been practiced in varying degrees by most licensees as part of their Physical Security Plans. The NRC also published regulatory Guide 5.66, "Access Authorization Program for Nuclear Power Plants" in June 1991 to provide an approach acceptable to the NRC staff by which licensees can meet the requirements of the rule.

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 Systems (NMMSS). Basically, this is an accounting system for 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.

Waste Management

Chapter 6

The Office of Nuclear Material Safety and Safeguards (NMSS) of the Nuclear Regulatory Commission (NRC) manages and coordinates NRC regulation of all commercial high-level and low-level radioactive waste and of uranium recovery facilities. Specifically, NMSS responsibilities include:

- Developing the criteria and the framework for regulating high-level waste (HLW), including development of the technical bases for the licensing of HLW repositories.
- Providing program management for NRC responsibilities under the Nuclear Waste Policy Act of 1982 (NWPA), as amended.
- Leading the national effort to license, inspect and regulate commercial low-level waste (LLW) disposal facilities.
- Developing guidance and providing technical assistance to States and State Compacts so that the goals of the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA) are met.
- Providing national program management for the licensing, inspecting and regulating of uranium recovery facilities and of radioactive waste associated with the uranium milling process (mill tailings).
- Reviewing and concurring in significant Department of Energy (DOE) decisions regarding inactive mill tailings sites and the licensing of stabilized tailings piles for monitoring and maintenance programs.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development Activities

During this fiscal year, the NRC continued to take steps to ensure that the regulations governing high-level waste disposition (10 CFR Part 60) were clear and complete. The NRC staff completed analysis of those portions of the rule that were either unclear as to the rule's intent or unclear as to which organization was responsible for implementation. The staff also undertook an evaluation of 10 CFR Part 60 with a view to determining if all of the repository functions dealing with radiological safety and waste isolation were adequately covered. Based on these two evaluations, the NRC will determine what changes are needed in 10 CFR Part 60 to make the rule as comprehensive and unambiguous as possible.

With respect to specific regulatory activity, the NRC continued to conduct the technical work necessary to respond to the DOE's April 19, 1990 petition for rulemaking. In that petition, DOE requested that 10 CFR Part 60 be amended to include quantitative dose criteria for a design basis accident. The NRC expects to complete preparation of its response to the petition during fiscal year 1992.

In fiscal year 1991, the NRC also continued to work with the Environmental Protection Agency (EPA) as that agency revised its high-level waste standards. The NRC staff participated in EPA's efforts to issue its environmental standards for disposal of HLW. This participation included reviewing, preparing comments on, and consulting with EPA on important features of the draft standards. The NRC will work toward implementing the revised EPA standards in parallel with EPA's revision. The NRC also will conduct its own rulemaking to bring 10 CFR Part 60 into conformance with the final EPA Standards.

Regulatory Guidance Activities

Regulatory guidance issued during the report period included one final and two draft Staff Technical Positions (STPs). STPs are key mechanisms for providing guidance to DOE, and they contain criteria for the development of methods acceptable to the NRC staff in demonstrating compliance with 10 CFR Part 60. One STP, "Regulatory Considerations in the Design and Construction of the Exploratory Shaft Facility" (NUREG-1439) was published in final form during fiscal year 1991. This is a compilation of previous NRC staff positions on the subject transmitted to DOE and is based on the premise that the Exploratory Shaft Facility (ESF) will eventually become part of a future geologic repository. The second STP, "Investigations to Identify Fault Displacement and Seismic Hazards at a Geologic Repository," was issued as a draft for public comment in May 1991. When completed, this STP will provide guidance to DOE on appropriate geologic repository investigations that can be used to identify fault displacement and seismic hazards. The staff also issued the draft STP, "Geologic Repository Operations Area (GROA) Underground Facility Design—Thermal Loads," for public comment in July 1991. In this document, the NRC staff position is set forth that DOE should develop and use a defensible methodology by which to demonstrate the acceptability of the underground facility design of a geologic repository operations area.

In addition to the STPs, the NRC staff issued Draft Regulatory Guide DG-3003, "Format and Content for the License Application for the high-level Waste Repository," in November 1990 for public comment. This Draft Regulatory Guide reflects the staff's attempt to provide early guidance on the information to be contained in the license application and to establish a format acceptable to the NRC staff for presenting this information. DG-3003 recommends a repository systems-based approach for the license application, reflecting the organization of the requirements in 10 CFR Part 60. The final version of this Regulatory Guide is to be completed in fiscal year 1994.

Technical Assessment Capability For Repository Licensing Reviews

The staff continued development of its independent capability to conduct performance assessments, as part of its repository licensing reviews. Such assessments will be used by DOE in its license application to show compliance with the EPA high-level waste standard and with NRC regulations. The HLW staff briefed the Advisory Committee on Nuclear Waste, the Nuclear Waste Technical Review Board, and the Electric Power Research Institute on the results of an initial demonstration (1990) and is enhancing its performance assessment capability by a second iteration—using more refined predictive models and treating a more comprehensive set of phenomena. The staff expects to complete this task by June 1992.

Yucca Mountain Site Characterization Analysis

In December 1990, the DOE submitted its response to the July 1989 Site Characterization Analysis (SCA) by NRC staff of DOE's Yucca Mountain Site Characterization Plan (SCP). The staff had identified 198 concerns, classified as either objections, comments or questions. In July 1991, the NRC transmitted to DOE the results of its evaluations of the DOE response. That response appeared generally to focus on ways of improving the site characterization program, rather than dealing with any of the SCA concerns. The staff took note, however, of the progress DOE has made toward resolving two objections—one regarding quality assurance, and the other on the ESF design and design control process.

With respect to the ESF design and design control process, the staff believes that DOE is addressing the NRC's concerns. Before the NRC can make a final determination, however, it will have to receive and review DOE's formal submittal on the design and design control process. Details on the quality assurance objection are provided in the section of this report entitled, "Quality Assurance Activities." Besides making progress on resolving the objections, the staff was able to close out 59 of the 198 open items, i.e., staff concerns, on the basis of DOE responses. The staff will continue working with DOE towards resolution of all remaining open items.

One way that the NRC staff gains early cognizance of DOE's planned site characterization activity is by review of the DOE's study plans, which describe how the investigations presented in the SCP are to be implemented. By the end of fiscal year 1991, the NRC had received 29 DOE study plans and had completed, or was engaged in, review of those plans. In no instance did the NRC staff find it necessary to object to a startup of activity consistent with a study plan, but the staff did convey its concerns to DOE with respect to certain details in several study plans.

Questions of particular significance arose from staff reviews of two DOE study plans pertaining to new site characterization activity in Midway Valley, a potential location for the surface facilities of the proposed repository. The The staff's conclusion that it had no objection to those two study plans, along with the staff's acceptance of DOE's quality assurance programs for those activities, meant that DOE could conduct its Midway Valley activities, once the appropriate permits were issued by the State of Nevada. Upon receiving two environmental permits from the State of Nevada, DOE proceeded with work under those plans.

Interactions with Governmental Entities and Indian Tribes

State of Nevada and local representatives continue to participate in the technical exchanges and meetings between NRC and DOE. State, local and Tribal representatives also continue to receive notification of upcoming NRC/DOE HLW meetings, as well as NRC Advisory Committee on Nuclear Waste (ACNW) transcripts and letter reports related to the HLW program. Access to these activities and reports thereon were also provided to those counties designated by the Secretary of Energy as "affected units of local governments."

Quality Assurance Activities

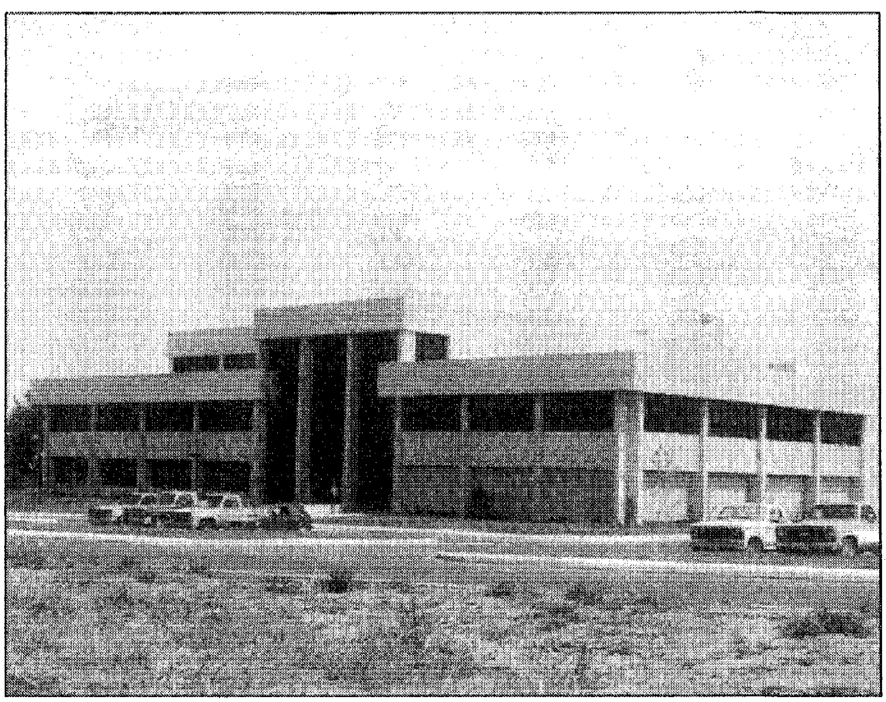
During the report period, the staff continued its reviews of DOE's and DOE contractors' QA plans and procedures (document reviews); evaluations of DOE's effectiveness in auditing its program, to identify and correct problems in program implementation; and evaluations of DOE contractor effectiveness in implementing QA programs. The NRC staff completed review of the majority of QA plans in earlier years. Work in this area for fiscal year 1991 included review of revisions to those accepted documents. In making its evaluation of DOE's effectiveness in auditing and DOE contractor effectiveness in QA program implementation, the NRC staff conducted observations of DOE audits. The DOE audits were performed at all major DOE contractor organizations participating in the site characterization program for the Yucca Mountain Project. Formal NRC staff reports were issued for all of the audits observed, and DOE will be required to respond to those reports where improvements in the audit process are needed.

All of these activities represent the steps DOE has been taking to resolve the quality assurance (QA) objection raised by the NRC staff in its SCA. The staff found, in January 1991, that the DOE Office of Civilian Radioactive Waste Management QA program was acceptable and that limited new site characterization activity in Midway Valley could begin. In August 1991, DOE requested that, because of its improvement in the QA area, the NRC remove its SCA objection concerning the lack of an acceptable QA program. The staff took the request into consideration, and a decision was expected early in 1992.

Center for Nuclear Waste Regulatory Analyses

The Center for Nuclear Waste Regulatory Analyses (CNWRA), an NRC contractor, completed its fourth year of operation in October 1991. The level of support that the Center provided to NRC has continued to increase, pursuant to its fundamental mission of providing sustained, high-quality technical assistance and research in support of NRC's high-level waste program, under the National Waste Policy Act of 1982 (NWPA), as amended. The Center provided a broad range of support to NMSS and to the Office of Nuclear Regulatory Research, as well as the Office of the Licensing Support System Administrator (see "Licensing Support System," below). The CNWRA continued to develop its technical and analytical capabilities, including the hiring of additional technical staff—fully integrating physics, geosciences and specific engineering disciplines. 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 has developed and, together with the NRC staff, is actively implementing a computer-assisted "systems approach" to identify and reduce uncertainties,



The NRC's Center for Nuclear Waste Regulatory Analysis (CNWRA) completed a fifth year of operation at the end of fiscal year 1991, providing a broad range of support to the agency offices dealing with nuclear waste regulation. CNWRA staff work out of the Southwest Research Institute, shown here, in San Antonio, Tex., and at the Washington Technical Support Office in Arlington, Va.

to select strategies and methods for confirming compliance with NRC regulatory requirements, and to define risks in licensing a HLW geologic repository. This approach is being taken by the NRC to assure that all of its HLW activities under the NWPA are planned, integrated, implemented, documented and managed as thoroughly and effectively as possible. Additional support from the Center is its assistance in the NRC review of study plans and design reports; participation in NRC/DOE pre-licensing technical exchange meetings; assistance with QA observation audits; technical support to NRC's rulemaking and regulatory guidance development programs; and assistance in the development of technical assistance capabilities and methods (e.g., computer codes). Specific accomplishments include a critical evaluation of natural resources assessment methodologies, a pilot study on the use of advanced three-dimensional interactive graphics information systems in license review, and development of the technical basis for evaluation of containment of nuclear waste. Significant progress was also made in the joint NRC/CNWRA development of a performance assessment methodology.

A broad-based integrated research program continued at the CNWRA during fiscal year 1991. Activities under the program included studies on the thermodynamic and ion exchange properties of sorbing minerals; selection of a geochemical natural analog site 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 hydrogeological responses to induced seismic events at an active mine; evaluation of state-of-the-art seismic rock mechanics computer codes; and laboratory investigation of the degradation of nickel- and copper-based alloy container materials.

Nuclear Waste Negotiator

On July 26, 1991, NRC Chairman Ivan Selin and the U.S. Nuclear Waste Negotiator signed a Memorandum of Understanding (MOU) between NRC and the Office of the U.S. Nuclear Waste Negotiator, similar to the existing MOU between DOE and the Negotiator. This document outlines the initial procedures governing interactions between NRC and the Negotiator in meeting their responsibilities, as set forth in the Nuclear Waste Policy Amendments Act of 1987. The MOU establishes a working relationship between both parties and assures a timely flow of information between the two agencies. It also provides the Negotiator with the use of such NRC services, facilities and personnel as the Chairman determines to be appropriate, while assuring the independence of all parties.

LOW-LEVEL WASTE MANAGEMENT

The main objective of the NRC's low-level waste program is to provide adequate protection of public health and safety and the environment in the management of low-level radioactive waste, in conformance with the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA).

Regulations and Guidance

Waste Form Technical Position Revision. On January 24, 1991, the NRC staff issued Revision One to the "Waste Form Technical Position." This document was initially developed in 1983 to give guidance to generators of low-level radioactive waste (LLW) regarding the test methods and acceptance criteria to be employed in meeting the waste form requirements in 10 CFR Part 61. Since the initial issuance of the technical position, field experience and laboratory testing of cement-solidified LLW have indicated that some unique chemical and physical interactions can occur between cement constituents and various chemicals and compounds which may exist in the waste materials. More comprehensive guidance was clearly needed on cement stabilization of LLW, and so the technical position was revised to include an appendix which specifically addresses cement stabilization. The guidance provided in Appendix A is the culmination of extended study, research and discussion between the NRC staff and representatives of various organizations.

Title Transfer Provision of Amendments Act. NRC staff completed analysis of the issues associated with the waste title transfer and possession provisions of the LLRWPAA and presented its recommendations to the Commission (SECY-90-318, issued September 12, 1990). As part of its review of SECY-90-318, the Commission sought public and State views on the issues presented in the analysis, and also on eight specific questions related to title transfer and possession of LLW. The issues focused on the role NRC should play in managing LLW, in light of LLRWPAA requirements, and on the health, safety, and environmental impacts of LLW storage. A solicitation of public comment was published in the *Federal Register* on December 4, 1990, and a total of 74 individuals or organizations responded by the end of the comment period (March 2, 1991). The staff reviewed the comments received and provided alternatives for the Commission for its consideration in selecting a course of action anticipating the 1993 and 1996 milestones written into the law. The NRC recognizes the need for licensees, States and Agreement States to plan ahead for these milestones and will continue efforts to clarify and resolve issues related to title transfer, possession and storage of LLW.

Standard Review Plan. By the end of September 1991, the Low-Level Waste Management and Decommissioning staff had completed draft revisions to 14 sections of the Standard Review Plan (NUREG-1200). The plan (SRP) provides guidance to regulatory personnel performing safety reviews of license applications to construct and operate a low-level radioactive waste disposal facility. The draft revised SRP covers topics on: surface water hydrology; the design of soil cover systems; waste disposal operations; performance assessment and analysis of radioactivity releases; and occupational radiation protection. In response to requests from State regulatory organizations for a fuller description of the data necessary to demonstrate compliance with regulatory requirements and to support the issuance of a license, a new section (SRP 1.0, Licensing Process) has been added to the SRP.

Technical Assistance to the States

The LLWM staff continued during fiscal year 1991 to support the Office of Government and Public Affairs (GPA) in providing technical assistance to the States as they implement their plans for low-level waste disposal facility development and licensing.

This technical assistance included:

- Support to GPA in holding a Regulators' Workshop and Performance Assessment Workshop for Agreement State regulators.
- Support to GPA in conducting program reviews of Agreement State regulatory programs.
- Participation in meetings of the LLW Forum, a group of State and compact officials which meets quarterly to discuss areas of common interest.
- Participation on a blue ribbon panel appointed by the State of California to review the environmental monitoring program for the proposed Ward Valley, Cal., site.
- Presentations and written reports 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 of these areas of technical assistance are discussed in more detail below.

Review of Site Characterization Plan for Vermont Yankee Site. In June 1991, the staff completed its review of the Site Characterization Plan for a potential low-level waste disposal site at the Vermont Yankee nuclear power plant. The Vermont Low-Level Waste Authority, in compliance with State legislation, had prepared the plan to characterize the features of this site for the purpose of determining whether it is suitable for development as a low-level waste disposal facility. (The Vermont Yankee plant generates most of the low-level waste in the State of Vermont.) The staff review focused on whether the plan would provide the Vermont Authority the data and information it would need to prepare a license application complete enough for NRC assessment. The issues identified by the NRC staff for consideration by the Vermont Authority included the high water table at the site, discharges of groundwater to the surface within the disposal site, and high groundwater velocities.

Performance Assessment Guidance. The staff has prepared and is implementing a program for low-level waste performance assessment (LLWPA). The planned program has two primary goals:

(1) To enhance the NRC staff's capability to review and evaluate a LLWPA from an applicant.

(2) To develop an in-house LLWPA modeling capability that will be the basis for development of regulatory guidance on LLWPA.

This guidance will provide license applicants with acceptable criteria and technical bases for evaluating the long term performance of a LLW disposal facility. The program will also improve NRC's ability to provide technical assistance to Agreement States on LLWPA issues. The program—which has been developed jointly by the NRC Offices of Nuclear Material Safety and Safeguards (NMSS) and Nuclear Regulatory Research (RES)—involves integrated staff and contractor work, supported by research projects. In order to provide inter-office coordination of LLWPA activity, staff from NMSS and RES have joined to form a Performance Assessment Working Group, to be responsible for developing and carrying out the LLWPA Program. The NRC staff's approach for improving in-house expertise in LLWPA modeling and in developing regulatory guidance is to do so through direct experience with LLWPA applications. The Performance Assessment Methodology developed by Sandia National Lab (NUREG/CR-5453 and NUREG/CR-5532), under a technical assistance contract to NRC, constitutes a comprehensive framework for conducting the LLWPA modeling program. The staff is also active in a variety of LLWPA areas involving interagency coordination of LLWPA projects, participation in international LLWPA and validation exercises, and consultation with Agreement State personnel on LLWPA issues.

Performance Assessment Workshop. On September 10–12, 1991, NMSS and the Office of Government and Public Affairs hosted a second LLW disposal facility performance assessment workshop for Agreement State regulatory staff. The workshop was conducted by contractor staff from Sandia National Laboratories who developed the performance assessment methodology (PAM) under a technical assistance contract with the NRC. The workshop consisted of an in-depth lecture on PAM development and implementation, as well as computer laboratory sessions that provided Agreement State staff with hands-on experience in exercising the computer codes of the methodology. The workshop was attended by 15 regulatory staff members from nine Agreement States that currently regulate the disposal of LLW or are developing licensing programs.

LLW Disposal Regulators' Workshop. On July 15–17, 1991, NRC staff hosted the annual Agreement State Regulatory Workshop in Bethesda, MD. The purpose of the meeting was to enable the States and the NRC to exchange information of common interest on the licensing of LLW disposal facilities. Topics discussed included the staff's recent changes to the Standard Review Plan for licensing LLW disposal facilities (NUREG–1200), recent State experiences in licensing of facilities, waste form criteria and issues, and the NRC's consideration of the title transfer provisions of the Amendments Act of 1985. The workshop included a visit to the Beltsville Agricultural Research Center to examine the Cover System Research Project, being funded by the NRC.

Agreement State Program Reviews and Visits. LLWM staff participated with the Office of State Programs in reviewing several State LLW disposal facility regulatory programs. NRC staff visited or reviewed programs in Illinois, South Carolina, North Carolina, Utah and California during the year.

Cooperation With Other Federal Agencies

During 1991, the NRC aggressively pursued cooperative efforts with other Federal agencies, seeking to resolve issues associated with low-level radioactive waste management and decommissioning, as well as other issues of common interest. Most of these efforts have involved the Environmental Protection Agency (EPA) and the Department of Energy (DOE). At the staff level, cooperation has continued toward the resolution of issues associated with the joint regulation of radioactive mixed waste management and its disposal, and with dual regulation of radionuclide emissions to air. During fiscal year 1991, the principal focus of the agencies' efforts in the mixed waste area have been on the development of a "National Profile" on the volume, characteristics and treatability of mixed waste in the United States. The profile was requested by the Host States' Technical Coordinating Committee and is being developed, as a joint NRC-EPA project, through a contract with Oak Ridge National Laboratory. Initial results from the project are expected by May 1992.

Regarding emissions of radionuclides to the air, NRC and EPA are cooperating in considering 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 under the Clean Air Act. (Section 112(d)(9) of that Act states that EPA does not need to regulate radionuclide air emissions if it determines that NRC's regulatory program already provides an ample margin of safety.) Specific efforts involved in this project include coordination of a survey of air emissions data from NRC and Agreement State licensees, proposed rescission of EPA standards that would control air emissions of radionuclides from nuclear power reactors, and development of a staff-level Memorandum of Understanding (MOU) that includes the States of Colorado, Texas and Washington and pertains to rescission of EPA standards for limiting radon emissions from uranium mill tailings disposal. The MOU was signed on October 24, 1991, and published in the Federal Register on October 25, 1991. The agencies also consulted on other issues across the broad spectrum of shared responsibility.

Cooperative efforts with DOE during the report period focused 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), 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, DOE plans to store small quantities of the GTCC waste as it is transferred to DOE by commercial licensees. The NRC and DOE are cooperating in developing procedures and criteria for managing the transfer of limited quantities of GTCC waste to DOE prior to operation of the disposal facility or dedicated storage facility. The NRC has also developed draft guidance on waste-disposal activity averaging, in order to confirm compliance with the Class C limits in 10 CFR Part 61. These cooperative activities are being coordinated with interested parties, such as the States and LLW Compacts.

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 plans for inactive uranium mill tailings sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA).

Regulatory Development and Guidance

In August 1991, NRC staff issued a final position on the disposal of *in-situ* wastes. In order to avoid the proliferation of small waste disposal sites, NRC regulations require wastes from *in-situ* solution mining facilities to be deposited at existing mill tailings disposal sites, in the absence of compelling reasons justifying disposal by other means. As these disposal sites closed, *in-situ* operators became concerned that there would be no off-site disposal options available to them and requested that they be allowed to dispose of their wastes on-site. In response to these concerns, the staff issued a draft position in April 1990 and an interim final position for public comment in August 1990. As a result of comments received, several clarifying changes were made to the position.

The commingling of low-level 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 meeting the definition must be disposed of in low-level waste facilities. In July 1988, the staff issued guidance on the disposal of such material in uranium mill tailings impoundments. A concern identified in that guidance is the question of DOE acceptance of title to the site for custody and long term care. Early in fiscal year 1991, DOE clarified its position on that issue, and, in August 1991, NRC staff prepared revised guidance for Commission review and public comment.

An area of interest to both the licensed mill program and the DOE inactive mill tailings remedial action program has been the use of alternate concentration limits (ACL) in meeting groundwater protection standards. In June 1988, NRC staff issued a draft technical position on ACLs for uranium mills and in October 1988 held a workshop for mill operators in Denver, Colo. A second workshop was held in Bethesda, Md., in December 1990, focusing on the DOE program—but also addressing the licensed mill program.

Licensing and Inspection Activities

The NRC Uranium Recovery Field Office (URFO) performed 35 inspections of uranium recovery facilities during the fiscal year. In other regulatory action, the URFO staff completed 56 major license amendments and 58 minor license amendments.

Of the 27 NRC-licensed uranium recovery facilities, 19 are uranium mills, three are either heap leach or other byproduct recovery operations, one is a research and development solution mining operation, and four are commercial *in-situ* solution mining facilities. At the close of the report period, only two commercial *in-situ* mining facilities were in operation, one was in standby, and one was authorized to begin construction at the end of the year. Only one conventional uranium mill was in operation, with two others in standby. Because of the low market price of uranium, few new facilities are expected to be licensed in the near future, except for *in-situ* solution mining facilities (one of which is under licensing review). As another result of market conditions, the two standby conventional mills are not likely to resume operation, except for short operating runs, and continued operation of the one operating facility is unlikely. Over the next few years, much of the casework confronting the uranium recovery program will be in the areas of remedial activity for the shutdown facilities, including decommissioning of mills, reclamation of mill sites and tailings disposal areas, and remediation of groundwater contamination. Continuing 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.

In 1991, NRC staff undertook a review of a proposed facility for the commercial disposal of uranium and thorium mill tailings and wastes. In the fall of 1989, the NRC received an application from Envirocare of Utah, Inc., to commercially dispose of 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 action. 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 commenced, and the staff completed two acceptance reviews and a first round of questions to the applicant. The scoping process for the environmental review was also completed in fiscal year 1991.

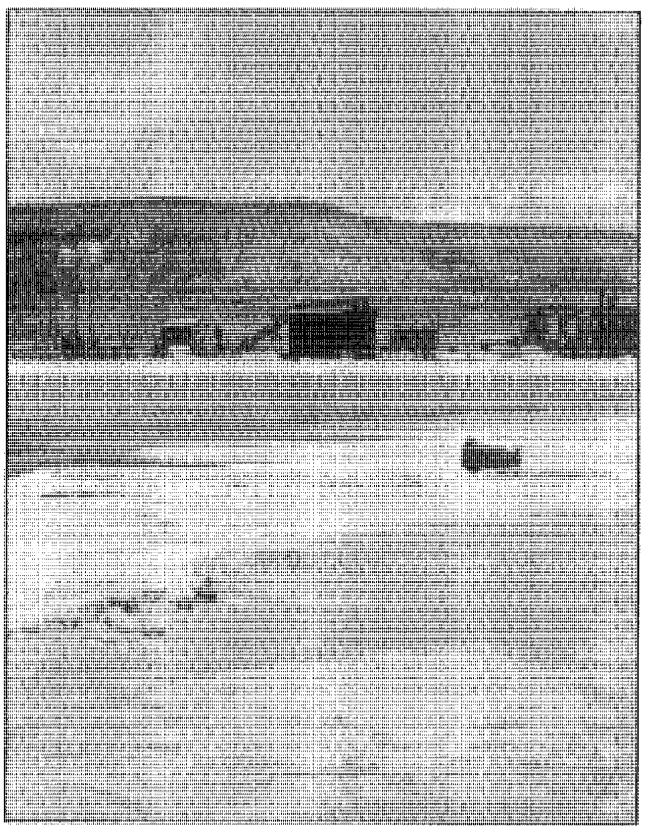
Remedial Action at Inactive Sites

During fiscal year 1991, NRC staff completed 70 review actions pursuant to its responsibilities at inactive uranium mill tailings sites. These included 10 Remedial Action Plan (RAP) reviews, 7 inspection plan reviews, 4 RAP modification reviews, 20 other site-specific reviews, 4 Completion/Certification Report reviews, and 14 reviews of generic items. The staff prepared six Technical Evaluation Reports, documenting its review of DOE's remedial action selection for the Grand Junction (Colo.), Lakeview (Ore.), Ambrosia Lake (N.M.), Lowman (Idaho), Falls City (Tex.), and Durango (Colo.) sites. Of particular note, the staff issued its first Final Completion

Review Report, documenting the review of and concurrence in DOE's remedial action performance at the Shiprock (N.M.), site.

Besides dealing with site-specific casework, the staff visited many of the sites for various purposes. Inspections of remedial action, in progress or completed, were conducted at the Durango, Salt Lake City (Clive), Lowman, and Grand Junction sites, and site visits associated with Remedial Action Plan reviews were made by NRC technical staff to the Falls City site and three sites in Colorado—Rifle, Maybell and Gunnison.

During the fiscal year, a final rule amending 10 CFR Part 40 for general licenses for the custody and long term care of uranium and thorium mill tailings disposal sites was published. Periodic NRC/DOE management meetings have been, and will continue to be, held to coordinate the UMTRCA Project activities through discussion of schedules and programmatic actions and issues.



The NRC is responsible for regulation of remedial action taken by the Department of Energy at inactive uranium mills, where the radioactive detritus of the milling process, called tailings, remains on-site. Among the six Technical Evaluation Reports prepared by NRC staff during the report period was one for the abandoned Ambrosia Lake (N.M.) site, shown above. The NRC must concur in all remedial action planned for such sites.

DECOMMISSIONING OF NUCLEAR FACILITIES

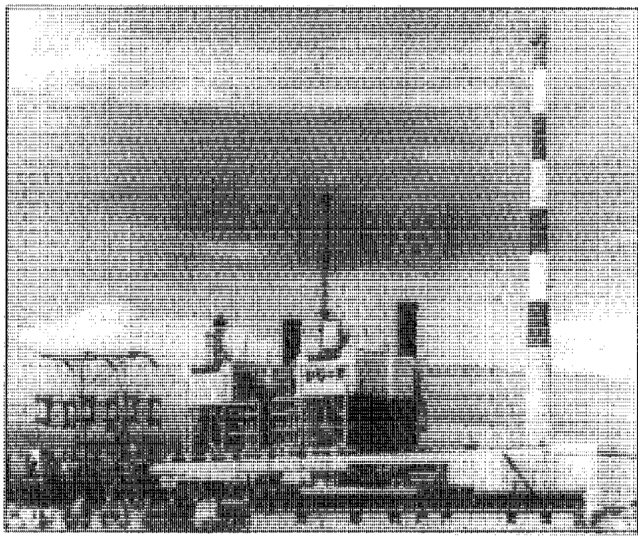
The staff has continued developing the guidance that licensing staff and licensees will need to implement amendments to Commission regulations with respect to the decommissioning of nuclear facilities. The staff is also performing decommissioning reviews for both nuclear reactors and materials facilities.

Guidance Documents

The staff is developing guidance documents for license reviewers and licensees giving needed information on acceptable methods for decommissioning, among them Standard Review Plans (SRP) 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 licensees, and a guidance document on how to perform final termination surveys is also in development. In 1989 and 1990, the staff prepared decommissioning financial assurance guidance for materials licensees and licensees with uranium recovery facilities and those operating low-level waste disposal facilities. These guidance documents will be updated to reflect staff experience in reviewing materials licensee financial assurance submittals. The staff is also preparing a rulemaking on timeliness of decommissioning that will set a time limit for decommissioning a facility where active operations have ceased. A rulemaking on record-keeping is also under way that would ensure that decommissioning records are maintained which include "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 Humboldt Bay Unit 3 (Cal.), Vallecitos (Cal.), Fermi Unit 1 (Mich.), Peach Bottom Unit 1 (Pa.), and LaCrosse (Wis.). In 1990, the staff approved a dismantlement plan for the Pathfinder (S.D.) power reactor, 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



The NRC has implemented a transfer of responsibility for reactor plants that have been approved for decommissioning from the Office of Nuclear Reactor Regulation to the Office of Nuclear Material Safety and Safeguards. The latter office has assumed oversight of decommissioning of the Humboldt Bay Unit 3 (Cal.) facility, among others. The plant, which began commercial operations in 1963, is located near Eureka, Cal., about 200 miles north of San Francisco.

shipped by rail to the commercial low-level waste disposal site in Hanford, Wash. The staff is also reviewing the decommissioning plans for the Fort St. Vrain (Colo.) high-temperature gas-cooled reactor, the Rancho Seco (Cal.) pressurized water reactor, and the Shoreham (N.Y.) boiling water reactor.

Site Decommissioning Management Program

In March 1990, NMSS prepared its Site Decommissioning Management Program (SDMP) for the Commission. The SDMP addresses the cleanup of about 40 contaminated materials sites. To meet its objectives, the SDMP defined the following measures:

- Structuring SDMP program management.
- Identifying sites requiring cleanup.
- Setting priorities among sites by which to apportion NRC review efforts.
- Estimating program schedules and resources needed for NRC action.
- Resolving generic policy and Congressional issues for SDMP implementation.

Since the formulation of the SDMP, NRC staff has reviewed site characterization plans, remediation plans, and final survey data. Confirmatory surveys have also been conducted. The SDMP was fully revised in April 1991.

LICENSING SUPPORT SYSTEM AND LSSA

The Licensing Support System (LSS) is an information management system established to contain and organize the documentary material generated by the Department of Energy (DOE), the NRC, the State of Nevada and other parties or potential parties to the licensing proceedings related to DOE's high-level radioactive waste repository. All potential parties to the hearing will have electronic access to the system both before and during the hearing.

The NRC Office of the 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 requirement of statutory law.

The LSS Advisory Review Panel was established to provide advice from future users of the LSS regarding its design, development, operation and maintenance. The panel includes representatives of the State of Nevada, local government entities, the National Congress of American Indians, the nuclear industry, and the Department of Energy. Representatives of other Federal agencies having significant experience with developing automated information management systems serve on the panel to provide their expertise on specific design, procurement and operational issues.

LSSA Activities

During the report period, the DOE's LSS design contractor—Science Applications International Corporation (SAIC)—completed work on conceptual design documents for the LSS. These documents describe each of the major components of the system. In the final phase of system development, the LSSA, DOE and SAIC carried out intensive reviews and evaluations of each document, including an updated cost-analysis. These documents provide an excellent foundation for the development of functional specifications for LSS procurement purposes. LSS design and development, including program responsibilities, continued to be explored jointly by the NRC and DOE at the close of the report period.

Most of the materials in the Licensing Support System will be textual. The LSS rule calls for this material to be stored in the LSS in both searchable text and digital-image formats. Descriptive information will also be stored in the LSS to help users locate material according to particular attributes, such as author, date, title, etc. During the report period, 29 descriptive fields were agreed upon by the LSS Advisory Review Panel. Because future users of the LSS will need timely and efficient access to millions of pages of non-textual data, produced during scientific investigations of the candidate repository site, special access procedures will be required. During the year, the LSSA examined numerous alternatives which would allow for effective identification and retrieval of these kinds of materials—e.g., handwritten field notes, maps, photographs, logs, computer tapes.

To assure that the LSS becomes the comprehensive and accurate data source intended—both for technical review and litigation support—LSS participants must identify, prepare and submit their documentary material to the LSS in a proper and timely manner, and in compliance with the LSS rule. During the report period, the LSSA undertook to create detailed document submission standards, set realistic document production schedules, explore the feasibility of setting priorities for document submission, and develop a cost-effective compliance evaluation program.

The LSS Advisory Review Panel, which is supported administratively by LSSA, held two public meetings during the report period. Four additional counties were invited to participate on the Advisory Panel, as part of the coalition representing local governments in areas adjacent to the Yucca Mountain site in Nevada. (See Appendix 2 for a listing of LSS Advisory Panel 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. (See Appendix 2 for a listing of the current membership of the ACNW.)

During fiscal year 1991, the ACNW reported to the Commission on a variety of issues, including:

- Stringency of the U.S. Environmental Protection Agency high-level radioactive waste repository standards.
- A recommendation that EPA consider incorporating into the high-level radioactive waste repository standards some guidelines on limits for doses and risks to individual members of the general population.
- Regulation of mixed wastes.
- Subsystem performance requirements of Part 60.
- Issues that deserve priority attention in the field of radioactive waste management. x An NRC staff technical position on regulatory consideration in the design and construction of the Exploratory Shaft Facility.
- Consideration of the possibility of human intrusion among concerns to be addressed in the licensing of a high-level waste repository.
- Individual and collective dose limits and radionuclide release limits.
- Uncertainties associated with implementing the EPA high-level waste standards.
- Response to questions accompanying Working Draft No. 3 of the EPA high-level waste standards.
- Classification of events that may affect repository performance.

- Review plan for Regulatory Guides implementing Revised 10 CFR Part 20.
- Resistance of low-level waste forms to leaking by groundwater.
- Comments on the Center for Nuclear Waste Regulatory Analyses.
- Role of formal elicitation of expert judgment in the performance assessment of a geologic high-level waste repository.

In performing its reviews and preparing reports of the kind cited above, the ACNW holds regular full committee meetings and working group sessions, as needed.

Communicating With Government And The Public

Chapter 7

The Nuclear Regulatory Commission is in 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 regarding NRC activities. The Commissioners and senior management frequently take part in Congressional Hearings (see Table 1), and appropriate Congressional Committees are kept regularly informed of NRC actions and decisions. Liaison with Federal and State agencies, Indian Tribes and local community organizations, the news media, Congress and the international community was formerly provided through the NRC Office of Governmental and Public Affairs (GPA). That office was reorganized in November 1991, after the close of the report period. NRC communications are currently conducted through the Office of Congressional Affairs, the Office of Public Affairs, the Office of International Programs, and the Office of State Programs.

PUBLIC COMMUNICATION

Commission Meetings

The NRC Commissioners meet in public session at the NRC Headquarters building in Rockville, Md., to discuss agency business. Members of the public are welcome to attend and observe Commission meetings, except on those unusual occasions when the Commission decides that a meeting should be closed. A meeting may be closed if its subject deals with one or more of the subjects specified in the Government in the Sunshine Act, which allows the closing of meetings involving certain kinds of subjects or documents—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 participate 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, disclosing the time, place and subject matter of the meeting, stating whether it is an open or closed meeting, and giving 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, by publication in two Washington newspapers, and by mailings to individuals who have requested copies of such notices. Announcements of Commission meetings are also regularly furnished by means of a recorded telephone message, on (301) 504–1292, providing the schedule for upcoming Commission meetings and/or voting sessions.

Advisory Committee Meetings

The Nuclear Regulatory Commission benefits from the knowledge and experience of numerous members 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 spectrum of the scientific and technical community, as well as from State and local governments, and from among private citizens.

NRC's advisory committees meet, in accordance with the requirements of the Federal Advisory Committee Act, in public session at Headquarters locations and in venues throughout the United States. Their members discuss and provide advice and recommendations to NRC on a broad range of topics and issues affecting NRC policies and programs. Appendix 2 gives a listing of the membership of the NRC's standing advisory committees.

Notice of advisory committee meetings is published in the Federal Register, in NRC press announcements, and by posting of meeting dates and topics in the NRC Public Document Room located at 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.

Meetings of the Nuclear Regulatory Commission are open to the public, except in rare instances, when the subject(s) to be discussed include classified documents, internal personnel matters, etc. Shown here is the Commission in session on October 29, 1991, dealing with the status of advanced reactor programs. From left-to-right are Commissioner Forrest J. Remick, Commissioner Kenneth C. Rogers, Chairman Ivan Selin, and Commissioner James R. Curtiss. (The fifth position on the Commission was filled with the appointment of E. Gail de Planque, sworn in on December 16, 1991.)

Public Information

Efforts to keep the public informed about NRC activity and programs were expanded in 1991. NRC Chairman Ivan Selin has charged the staff to be even more open than in the past in dealing with safety issues and to actively communicate to the public as to what the NRC is doing, why and how. As part of this expanded effort, the NRC has initiated a program of periodic news media briefings by the five Regional Administrators. The briefings are in addition to press conferences conducted on specific events or incidents in a particular Region and will include NRC activity of local interest, as well as agency-wide issues.

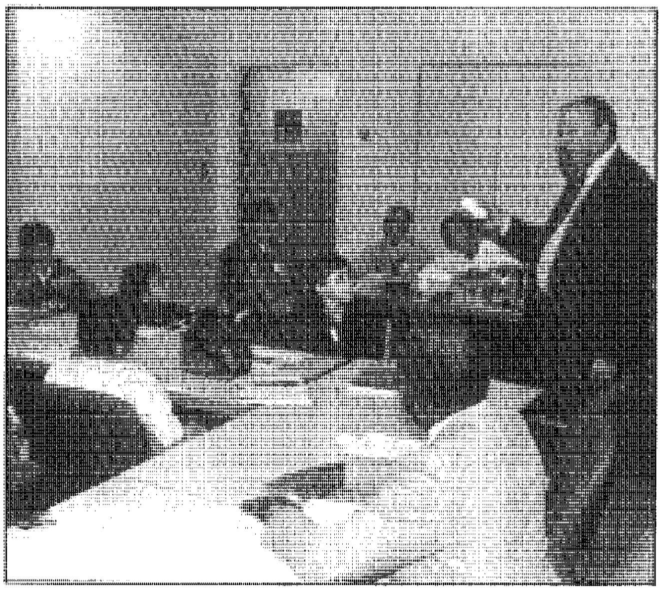
Throughout the year, the Office of Public Affairs conveys a wide variety of information to the news media and the general public by means of news releases, formal orders, fact sheets and pamphlets on the major decisions and actions taken by the Commission and the NRC staff. After important meetings of the Commission and/or of the staff, interviews and news conferences are arranged for reporters, in order to provide more detailed explanations of rulemakings, policies and programs.

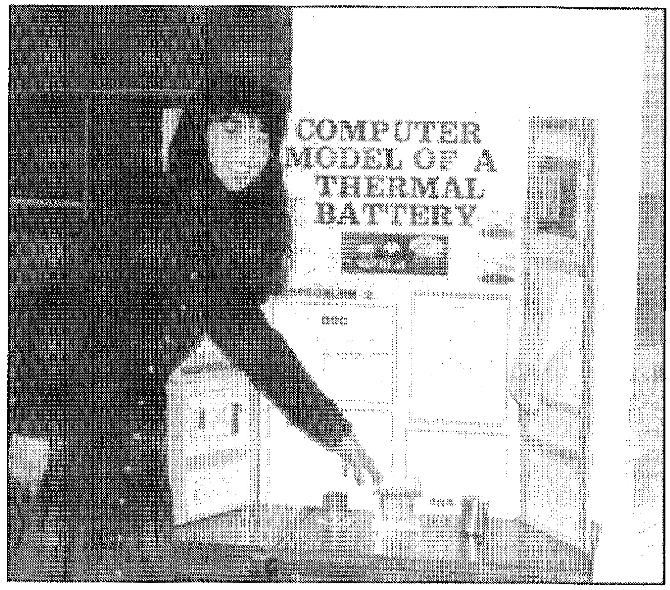
A videotape on the mission of the NRC to protect public health and safety and the environment has been broadly used by schools and television stations, reaching about 100,000 students the first year and an estimated 570,000 television viewers in the first six months of its availability for TV use. A slide show explaining NRC programs and operations is also available, for use in presentations by NRC personnel or others.

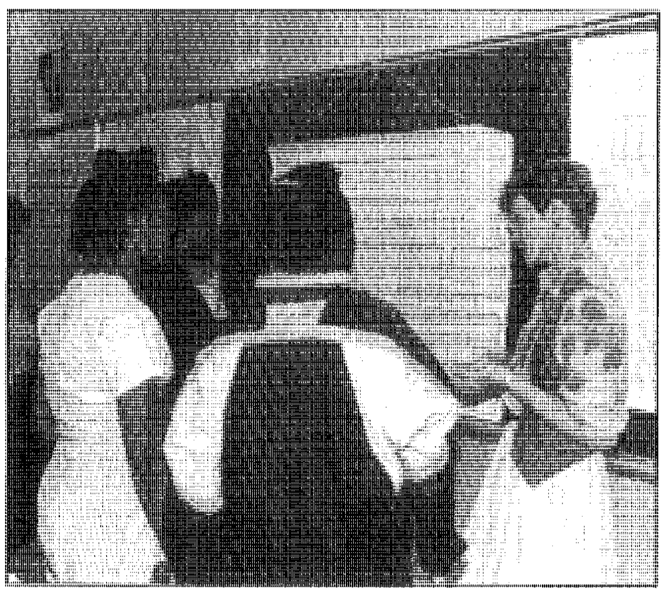
Media Seminar Workshop. Regional public affairs officers continue to schedule seminars for reporters and editors from newspapers, television and radio stations around the country. Reporters are given the opportunity to receive "hands on" instruction using actual nuclear power plant simulators at the NRC Training Center, at Chattanooga, Tenn., and to acquire the fundamental knowledge of how a nuclear power plant is operated.

NRC School Volunteers Program. This report year marks the seventh year that NRC volunteers have worked with schools throughout the Washington Metropolitan Area, as part of the national Partnerships in Education Program initiated by the President in 1983. In any given week from September through June, NRC volunteers typically visit 2-to-3 schools, and students are visiting NRC Headquarters about once a month. In fiscal year 1991, over 150 volunteers responded to 265 requests from 115 schools, reaching over 6,000 students and faculty, primarily in Montgomery County (Md.) Public Schools. An award was given to the NRC from Montgomery County for outstanding service to education during the 1990–1991 school year.

Volunteer presentations in the schools covered a broad range of activity, involving all grade levels from kindergarten through college (the latter including the United States Naval Academy, Howard University, the University of the District of Columbia, and Catholic University). NRC volunteers worked with students of advanced academic standing, those interested in science, and also those who are "at risk" of dropping out, or are otherwise not doing well in school. Specifically, volunteers provided hands-on science demonstrations, academic tutoring, mentoring, assistance on science projects, opportunities for students to "shadow them" on the job, judging for







NRC staff interact with students throughout the Washington, D.C., metropolitan area through a variety of programs and events. At top left, Dr. Harold Denton of the NRC describes the workings of the agency for 7th-and-8th grade students from Gaithersburg, Md., who are visiting NRC Headquarters. Top right is Ann Chi, the winner of a special award sponsored by the NRC at the Science Fair conducted by Montgomery County, Md., at the National Institute of Standards and Technology. Ms. Chi subsequently came to work for the summer at the NRC. Above is Donna Smith of the NRC (at right), explaining her work as a human factors engineer to students at the Seneca Valley (Md.) High School. At the middle of the right column are a group of "Youth Governors" attending the 30th annual Youth Governors' Conference; these high school students visited NRC Headquarters in June 1991. At right, Frank Ashe of the NRC (foreground) describes his work as an electrical engineer to students at the Seneca Valley High School, while (at back) the NRC's Louis Grosman demonstrates information access and retrieval from the agency's voluminous electronic files.

science and math fairs, assistance to faculty in developing curriculum for special study areas, responses to student interviews, lectures on the use of math and science on the job, and career awareness discussions. Commissioner Forrest Remick joined in the efforts to reach out to minority students, encouraging them to develop and pursue career goals.

This year, several noteworthy initiatives were effected through the School Volunteers Program. For the first time, NRC provided special awards at the Montgomery Area Science Fair, with Commissioner James Curtiss making the presentations. The award-winning students presented their projects to the full Commission, in a meeting open to all NRC employees. Also this year, for the first time, the NRC hosted over 40 teachers from Maryland, Virginia, and the District of Columbia in an all-day seminar dealing with the mission and programs of the agency and including a tour of the NRC Operations Center. Among the participants were Chairman Ivan Selin, who welcomed the teachers; Commissioner Kenneth Rogers, who spoke on the subject of risk perception; and numerous NRC managers and staff. On another occasion, more than 35 students—elected by their peers as Youth Governors from across the country—discussed a number of current nuclear-related issues with Commissioner Curtiss and other NRC staff.

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 significant decisions and actions. The computerized, on-line Bibliographic Retrieval System (BRS) features 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., in 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 (see below) about nuclear facilities and other licensees will find this specialized research center to be a major resource. PDR users may have documents from the collection, with some exceptions, reproduced for a nominal fee.

The PDR makes available to the public a variety of agency documents, such as 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 an average month, the PDR serves about 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 the terminals in the Reading Room, or off-site, via modem. off-site access (by both 1,200 and 2,400 baud) 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 reference number below.

The PDR/BRS users group consists of members of Congressional staffs, media representatives, other government agencies, foreign embassies, law firms, utilities, State agencies, consulting firms, public interest groups, individual members of the public and foreign governments. Foreign use of the PDR includes users from England, France, Italy, Japan, the Netherlands, and Spain. NRC staff may also access the data base in either Headquarters or Regional Offices.

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 the report period, the NRC was maintaining 87 Local Public Document Rooms (LPDRs) throughout the country. These collections of documents are related to nuclear power reactor, research reactors, fuel cycle facilities, and low-level and high-level waste disposal facilities, operational or prospective. Financial assistance, by means of cooperative agreements, was provided to 69 LPDRs during the period. (See Appendix 3 for a complete listing of LPDRs.)

A primary goal of the LPDR program in fiscal year 1991 was to replace paper records, dating from January 1981 forward, with microfiche, for the power reactor LPDRs. This effort involved the production of over 3,000,000 microfiche. Approximately 47,000 microfiche were sent to each LPDR library. NRC LPDR staff visited 36 of these LPDRs to set up the microfiche files, during the last quarter of the fiscal year, reducing shelf space required for paper records by approximately 50 linear feet at each library. The post-1981 paper records will be replaced by backfit microfiche at the remaining power reactor LPDRs in fiscal year 1992. The conversion from paper to microfiche gives the public in each locale access to all records made available publicly by the NRC, and not only to records pertaining to the nearby nuclear plant. The new arrangement also reduces and stabilizes NRC costs for support of the LPDR program. The conversion to microfiche has been favorably received by LPDR librarians and patrons.

Eighteen 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 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.

One facility was relocated during the report period: the LPDR for the Grand Gulf (Miss.) nuclear power plant was relocated from the Hinds Community College, Raymond, Miss., to the Judge George W. Armstrong Library, Natchez, Miss. A new LPDR was established for Louisiana Energy Services' proposed uranium enrichment facility; the LPDR is located at the Claiborne Parish Library, Homer, La.

Commission History Program

Under the Commission History Program, the origins and evolution of regulatory policies in their historical context are researched through review of records maintained in the archives of various government agencies, review of personal papers of former government officials, and personal interviews with such officials. Based on these sources, the History Office is currently completing a sequel to its book, *Controlling the Atom: The Beginnings of Nuclear Regulation, 1946–1962,* published in 1984 by the University of California Press. The new volume, *Containing the Atom: Nuclear Regulation in a Changing Environment, 1963–1971,* focuses on reactor safety and siting, radiation protection, and environmental issues; it will be published in 1992. Like the first volume, it is intended to serve as a reference for general readers, as well as for agency staff.

CONGRESSIONAL OVERSIGHT

The Office of Congressional Affairs is responsible for developing, managing, and ensuring coordination of relations with the Congress, and is the principal point of contact between the agency and Congress. The office coordinates appearances by NRC officials at Congressional hearings, monitors and tracks bills relevant to the NRC during each Congress, monitors and coordinates the preparation of all testimony given by NRC witnesses, and keeps authorizing committees and the Congressional leadership informed of all NRC activity.

During the first session of the 102nd Congress, NRC witnesses testified at 17 hearings before Congressional Committees and Subcommittees, as shown in Table 1. Congressional Affairs staff attended and prepared summaries and reports for more than 50 hearings and mark-ups.

In the first session of the 102nd Congress, the office was involved in the process leading to confirmation of Dr. Ivan Selin as Chairman of the NRC, and initiated the process to confirm E. Gail de Planque as the fifth member of the Commission. (See Chapter 1.)

COOPERATION WITH THE STATES

The NRC's contacts with regional, State and local agencies, and with Indian Tribes are administered through the Office of State Programs (SP), except for interactions related to inspection, enforcement, and emergency planning. Cooperative activity encompasses the State Agreements Program and various other liaison programs administered in accordance with policies and procedures established by Headquarters and implemented primarily by the Regional Offices.

State Agreements Program

A total of 28 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 1991, there were about 16,000 radioactive

Table 1. Congressional Hearings at Which NRC Witnesses Testified – FY 1991

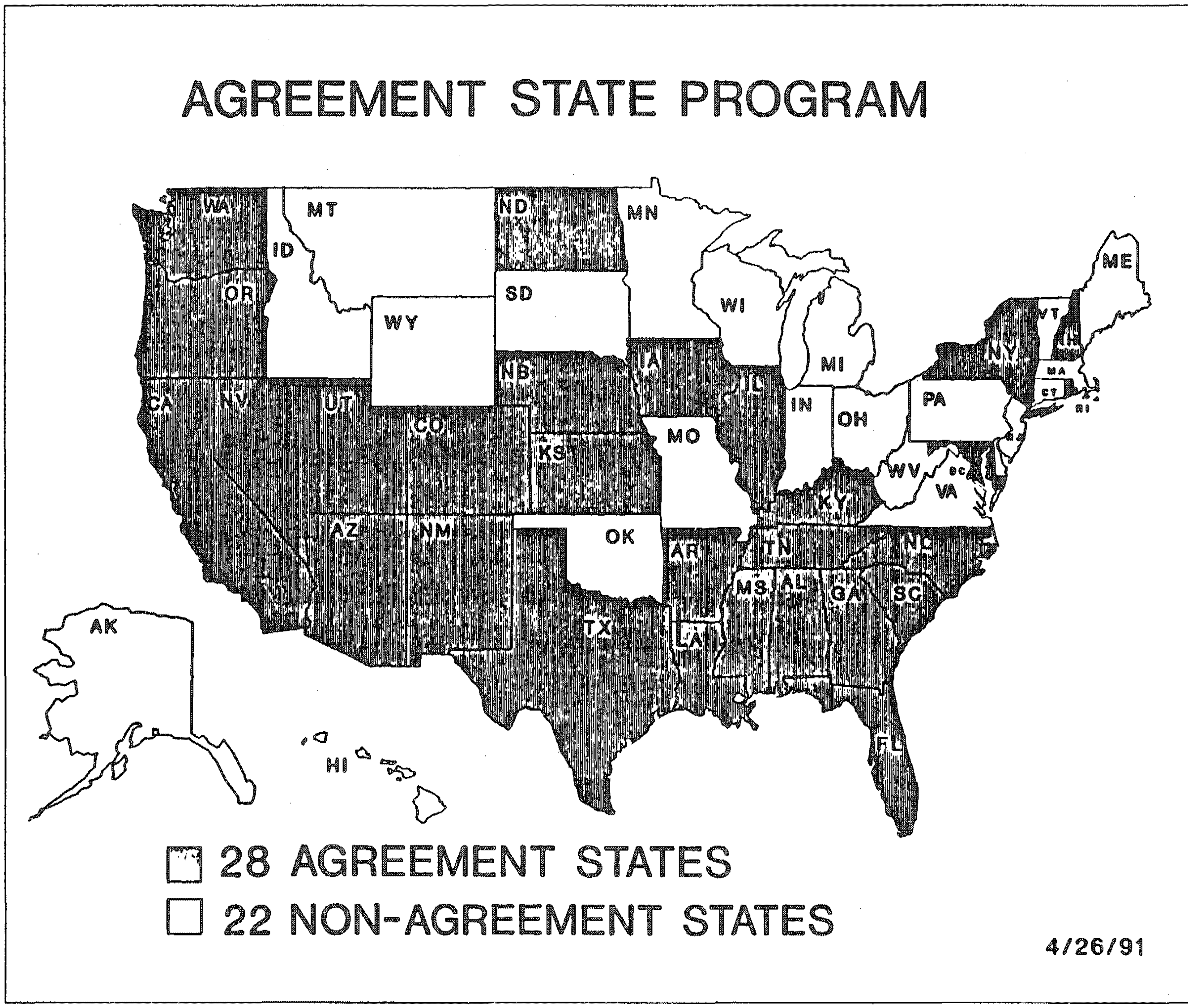
Date	*Committee*	*Subject*
10/02/90	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	High-Level Waste
10/30/90	Committee on Interior & Insular Affairs Subcommittee on General Oversight and Investigations (House)	Pilgrim Restart/ Emergency Planning
02/28/91	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	Budget Review
03/07/91	Committee on Governmental Affairs (Senate)	Substandard Parts
03/13/91	Committee on Appropriations Subcommittee on Energy & Water Development (House)	NRC's FY 92 Budget
03/19/91	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	NRC's Financial & Programmatic Management
03/21/91	Committee on Energy & Natural Resources (Senate)	Civilian Nuclear Waste Program
04/10/91	Committee on Science, Space and Technology Subcommittee on Energy (House)	AVLIS Enrichment
04/30/91	Committee on Interior & Insular Affairs Subcommittee on Energy and Environment (House)	Energy Facility Siting
05/08/91	Committee on Energy and Commerce Subcommittee on Energy and Power (House)	Nuclear Issues

(Table 1 continued)

Date	*Committee*	*Subject*
05/16/91	Committee on Environment & Public Works Subcommittee on Nuclear Regulations (Senate)	Nuclear Titles of National Energy Strategy
05/21/91	Committee on Environment & Public Works (Senate)	Chairman Selin's Nomination
05/23/91	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Nuclear Waste Title of National Energy Strategy
07/25/91	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	Licensing Reform Titles of NES
07/25/91	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	International Nuclear Reactor Safety
08/01/91	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	Yankee-Rowe
09/12/91	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	BRC Policy H.R. 645

material licenses administered by the Agreement States, representing about 65 percent of all the radioactive materials licenses issued in the United States. The States of Maine and Pennsylvania were negotiating agreements with the NRC at the close of the report period. The Pennsylvania proposal would give that State regulatory authority over the land disposal of byproduct, source and special nuclear material only. In May 1991, Idaho returned its Agreement State status to NRC because of fiscal concerns within the State.

Review of State Regulatory Programs. The NRC is required by the Atomic Energy Act of 1954 to review Agreement State radiation control programs periodically, and the programs are normally reviewed annually. The NRC conducts three distinct kinds of reviews—routine reviews, review visits, and follow-up reviews. Routine reviews are complete, in-depth examinations of State regulatory programs, normally conducted every other calendar year. Review visits are usually conducted between routine reviews and serve to maintain familiarity with



Agreement State radiation control programs, to provide an opportunity for discussion of areas of mutual concern on an informal basis, and to 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 activity in specific areas.

In fiscal year 1991, 15 routine program reviews, eight review visits and three follow-up reviews were conducted. Two additional reviews focused on low-level radioactive waste. 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, during these reviews. When appropriate, multi-discipline teams are used to conduct reviews of agreement programs. These teams include NRC Program and Regional Office staff, as well as other Agreement State representatives. In general, the States have been found to be doing an excellent job in maintaining adequate and compatible programs.

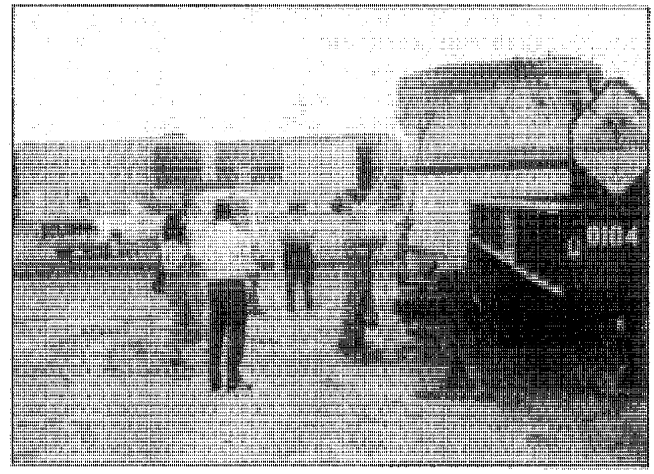
The NRC reviews help identify potential problems in State programs for the benefit of State management. In doing so, the NRC employs a "Category I" designation for the more serious concerns. If no significant Category I comments are provided, then the program is deemed adequate to protect the public health and safety and is 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 provided technical assistance to Agreement States during the report period in the areas of licensing, inspection, and enforcement, and also alerted the States to proposed statutes and regulations. Technical assistance ranged from responding to requests for information to assisting in State reviews of license applications and State inspections. Technical assistance was also given in interpreting information generated by specialized expertise.

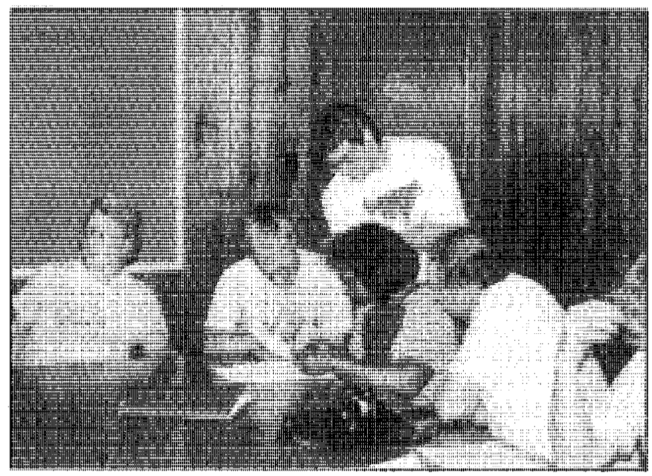
Training Offered by the NRC. State radiation control personnel regularly attend NRC-sponsored courses to improve their ability to maintain high quality regulatory programs. The NRC sponsored 24 training meetings and courses. The training sessions were attended by 481 persons (381 of them were State radiation control personnel). In addition to State personnel, the sessions were attended by NRC staff and by two military personnel, and two students from the Canadian Atomic Energy Control Board. Courses covered such subjects as health physics, industrial radiography safety, nuclear medicine procedures, inspection procedures, well logging, radiation protection engineering, transportation of radioactive materials, nuclear materials, and low-level waste. Two of the formal training courses were hosted by Agreement States. And there was other training activity, e.g., State employees visiting other Agreement States or NRC Offices to obtain on-the-job training in licensing and inspection of radioactive materials.

Annual Agreement States Meeting. The annual meeting of Agreement States radiation control program directors was held in October 1991, in Sacramento, California. The official welcome was extended by Carlton Kammerer, Director, Office of State Programs. The meeting included panel discussions on low-level waste, material licensing, and materials regulation. This year's meeting was attended by representatives from the U.S. Navy and the U.S. Air Force, as well as representatives from the Non-Agreement States of Alaska, Connecticut, Maine, Massachusetts and Pennsylvania. Also in attendance were representatives from Canada and Mexico.

Regulation of Low-Level Waste. The NRC provided technical assistance to the States of Washington, Illinois, Utah, California, and New York in the development of low-level waste regulatory programs that meet the requirements of the Low-Level Radioactive Waste Policy Amendments Act of 1985. Technical assistance was also accorded the States of Pennsylvania and New York in the promulgation of low-level waste regulations. 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.



Training sessions involving NRC and State personnel in all aspects of regulatory programs are a constant. Above the students are performing a radiological survey during a laboratory exercise, part of a Nuclear Transportation Course, in September 1991. Below the students are examining the shipping manifest for an incoming shipment of low-level nuclear waste.

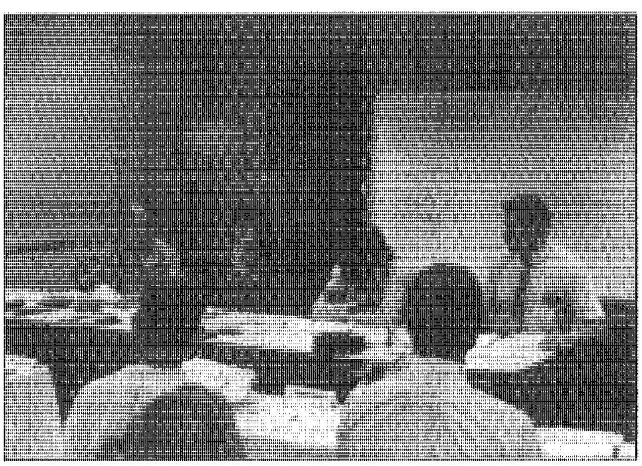


A Low-Level Waste Regulatory Workshop was held in July 1991, providing an opportunity for the NRC to discuss current regulatory issues related to low-level waste disposal with the State personnel who are expected to regulate a low-level waste facility. A second Low-Level Waste Performance Assessment Methodology Workshop was held in September 1991. That workshop provided State regulatory personnel the opportunity to familiarize themselves with the computer codes used in the evaluation of an application for a low-level waste facility.

Regulation of Uranium Milling. The NRC assisted Agreement States in their programs for regulating uranium milling. Assistance was given in the areas of groundwater monitoring requirements for milling facilities, reclamation design reviews, guidance document reviews, license termination determinations, and the conformity of uranium mill regulations with revised NRC regulations. This assistance was provided to the States of Colorado, Texas, and Washington.

State Regulations Workshop. State Programs sponsored a workshop entitled "The Process of Rules Development and Changes" on September 19–20, 1991, in Bethesda, Md. The purpose of the workshop was to explore the various issues surrounding State adoption of regulations designated as matters of compatibility by the NRC. Areas for discussion included States' political and fiscal obstacles in adopting regulations; advantages and disadvantages of coordinating Federal and State rulemakings on major issues; assistance from the NRC, from other Federal agencies, and from the Conference of Radiation Control Program Directors in a State's rulemaking process; use of Suggested State Regulations and adoption of regulations by reference. Participants in the workshop included representatives from various State radiation control programs, representatives from the Attorneys General offices of Arizona, Illinois, and Colorado, as well as from the National Association of Attorneys General, the NRC and other Federal agencies.

Operational Events In Agreement States. Information on events taking place in Agreement States is routinely shared through 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 all appropriate regulatory agencies and interested parties.



A Special Topics Workshop on the "Sealed Source and Device Registry" brought together 49 staff personnel from the Agreement States, NRC Headquarters and Regional Offices, the Food and Drug Administration, and the Canadian Atomic Energy Control Board. The registry is a system for setting standards and maintaining records for sealed radioactive sources and devices that the States and the NRC approve for use by licensees. The workshop is another example of the regular interaction between Agreement State and NRC staffs, and cooperation with other agencies and other nations.

Improving Cooperation With the States. In April 1991, the Commission directed the staff to develop a process that will ensure early and substantial involvement of the Agreement States in rulemakings and other regulatory efforts that affect facilities licensed under 10 CFR 30, 40, 61 and 70, or equivalent regulations. Some examples of steps taken in response to this directive are:

- Holding public meetings of NRC with the Agreement States in which NRC plans for future rulemakings are presented.
- Consultation with the States on compatibility determinations for proposed NRC rulemakings.
- Continued consultation with the States during development of rulemakings of importance to the States.
- Holding joint NRC-Agreement State Workshops on inspection priorities of materials licensees and on sealed source and device evaluations which included opportunities for State input for future changes.
- Expanded consultation with the Organization of Agreement States to assure that this process has the support of all concerned.

Compatibility Issue. In June 1991, the Commission was briefed by representatives of the Agreement States on compatibility issues in the Agreement State program. While the Atomic Energy Act, as amended, makes reference to "compatibility" of Agreement States programs with that of the Commission, the Act does not define the term. A reading of the legislative history of Section 274, "Cooperation with States," indicates a Congressional intent that Agreement State programs need not be identical to the Commission's, but the degree of flexibility to be permitted is not clear. The Commission, besides receiving Agreement State views on the issue, requested that the staff seek the comments of other interested parties (e.g., materials users and waste generators) on the general matter of compatibility and on the advantages and disadvantages of a uniform national approach to radiation safety matters. The staff will be developing recommendations to the Commission for a policy on compatibility issues.

State, Local and Indian Tribe Liaison Programs

The NRC Five Year Plan calls for the agency to assume a more active role in fostering better cooperation and communication between NRC and State and local governments and Indian Tribal representatives, in order to promote a wider and deeper understanding among all concerned of issues and activities related to nuclear safety, including those discussed below.

Low-Level Radioactive Waste Compacts. The Low-Level Radioactive Waste Policy Amendments Act of 1985 ensures that currently operating disposal facilities will remain available until the end of 1992, subject to specified limitations on volumes of waste and to certain milestones for specific action by the States. The Act set up a system of incentives and penalties to promote steady progress toward new facility development, and it granted Congressional consent to seven interstate low-level waste disposal compacts. There are now a total of 43 States participating in nine separate interstate compacts, although Michigan's status is being litigated, as explained below.

Three Agreement States are currently reviewing applications for licenses and license issuance is anticipated by early 1992 for California, by the fall of 1993 for Nebraska, and by late 1993 for Illinois. Texas, another Agreement State, was expected to have a license submitted before January 1, 1992, in order to meet the Congressional milestone of the Low-Level Radioactive Waste Policy Amendments Act of 1985. The Agreement State of North Carolina was in the site characterization phase, and the

States of Connecticut and Maine had identified a number of candidate sites, as of the close of the report period.

The remaining States are in earlier phases. Vermont had identified a site which was later disqualified and will be initiating a site screening of the State. New Jersey and Pennsylvania will be identifying candidate areas in 1992. New York and Massachusetts are in earlier phases, with progress set back by revised legislation and budget cuts, respectively. The remaining States of New Hampshire and Rhode Island, the District of Columbia, and Puerto Rico have taken little action in terms of establishing their own disposal capacity.

On July 29, 1991 the Midwest Compact filed a lawsuit in Minnesota Federal Court on the effective date of revocation of Michigan's status as host State. Earlier, on July 24, 1991 the Compact had revoked Michigan's host State status because of failure to serve as host State. The outcome of the suit may also lead to the suspension of all compact privileges for Michigan. All siting activity has ceased in Michigan because the Michigan Low-Level Radioactive Waste Authority has been disbanded. All low-level radioactive waste in Michigan is being stored temporarily, because the State has lost access to the operating facilities in South Carolina, Washington and Nevada. At the same time that Michigan's status was revoked, the Midwest Compact voted Ohio as the host State. Ohio officials are working on enabling legislation.

A tentative contract between the Northwest Compact Commission and the Rocky Mountain Compact Board was released for comment on October 11, 1990. The contract has not yet been signed, pending the resolution of Idaho Governor Andrus' concerns with the Department of Energy's accepting spent fuel from Colorado's Fort St. Vrain nuclear power plant.

Because many host States will not have disposal facilities operating until after the Congressional deadlines of 1993 or 1996, interim management options, such as storage, are under consideration. The NRC is developing a policy on management alternatives that is supportive of the objectives of the 1985 Act.

As reported in the *1990 NRC Annual Report* (p. 112), New York State, the State of Michigan, and the Concerned Citizens of Nebraska are seeking to have the 1985 Act declared unconstitutional. Defendants in the suits include the NRC, the Department of Energy and the Department of Transportation. The U.S. Court of Appeals for the Second Circuit affirmed the decision of the Federal District Court dismissing the lawsuit brought by the State of New York and the New York counties of Allegany and Cortland against the Federal Government.

Plantiffs appealed the case to the U.S. Supreme Court on September 29, 1991. Thirteen States have filed *amicus curiae* briefs in support of New York. (See "Significant Judicial Decisions," under Judicial Review, in Chapter 9.)

On August 28, 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. The State, in addition to challenging the constitutionality of the Act, included claims brought under the National Environmental Policy Act (NEPA), which among other things, directs Federal agencies to prepare an environmental impact statement for major Federal actions. Michigan filed, in mid-October 1991, a notice of appeal in the Sixth Circuit Court of Appeals.

In July 1991, the Concerned Citizens of Nebraska filed a notice of appeal with the U.S. Court of Appeals for the Eighth Circuit in their suit challenging the constitutionality of the 1985 Act. Briefs for the defendants, including the NRC were due by November 14, 1991. The U.S. District Court for Nebraska had earlier dismissed the suit, on October 19, 1990.

State Liaison Officers. The governor-appointed State Liaison Officers (SLOs) continue to be the NRC's primary contact with States concerning proposed rulemaking, policies and other matters. SLOs are the also the main point of contact for "observation agreements" with States, which allow States to accompany NRC inspectors during certain inspections at reactor facilities.

A Regional State Liaison Officers' Meeting was held August 28–29, 1991, in Arlington, Tex. Representing the NRC were Robert Martin, Region IV Administrator, Carlton Kammerer, Director, and Sheldon Schwartz, Deputy Director, Office of State Programs, State Programs staff, and Region IV staff; they participated in the meeting with SLOs from the Region IV States of Idaho, Montana, Wyoming, North Dakota, South Dakota, Nebraska, Kansas, Colorado, Utah, New Mexico, Oklahoma, Arkansas, Louisiana and Texas. Topics discussed included risk communication, low-level radioactive waste, NRC's enforcement policy, NRC's fee schedule, radioactive waste "below regulatory concern," uranium mill tailings, emergency preparedness, Agreement State compatibility, operator testing, and decommissioning.

Memoranda of Understanding With States. The NRC and the State of Illinois signed an agreement (Subagreement #3) allowing Illinois Resident Engineers to participate in NRC inspections at nuclear power plants in Illinois. The agreement is one of the first to be concluded under the NRC's policy entitled "Cooperation With States at Nuclear Power Plants and Other Nuclear Production or Utilization Facilities" (54 FR 7530; 2/22/89). The agreement will be offered as a model for other States wishing to enter into Resident Engineer agreements.

The NRC and Michigan Officials are currently negotiating a Memorandum of Understanding designed to provide Michigan's emergency response officials with electronically transmitted "real-time" reactor data, in the event of an accident at nuclear power plants in Michigan (excepting the 69-megawatt Big Rock Point nuclear facility). The NRC also began negotiations with the State of New Jersey for a Memorandum of Understanding which would provide the basis for cooperation on the interim storage, shipment and management of low-level radioactive waste in the State.

The NRC staff has also proposed to amend its policy statement on Cooperation With States, to allow 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. The proposed amendment has been published in the *Federal Register* for public comment. Once comments are received, they will be analyzed and, where indicated, incorporated into the proposed policy amendment, which will then be sent back to the Commission for final approval and publication.

The Office of State Programs, in cooperation with the Office for Analysis and Evaluation of Operational Data (AEOD), coordinated an "ERDS" Memorandum of Understanding (MOU) with the State of Michigan. "ERDS," the Emergency Response Data System, 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 acquire the capability to receive ERDS data during events at power plants, by executing an MOU with the NRC. Eight other States have requested a link-up with ERDS.

Other Training. The State Programs staff, in cooperation with the NRC Office for Analysis and Evaluation of Operational Data, completed a full series of Protective Measures Workshops for State emergency preparedness and radiological health officials in each of the five Regions. The course covered the NRC's severe accident classification philosophy, methodology and tools. The program focused on recommended courses of action that State personnel should concentrate on during certain postulated severe accidents. The NRC expects to update this information and present the workshops on a two-year cycle at various locations throughout the country, to increase its accessibility to State personnel.

Regional State Liaison Officers. On the staff of each NRC Regional Office is a Regional State Liaison Officer (RSLO) who acts as the Region's principal contact with SLOs and other State and local officials. The RSLOs are the coordinators for NRC activity involving State and local government or Indian Tribes. The RSLOs often attend and participate in State and local meetings when issues under NRC purview are to be discussed. 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 NRC's jurisdiction. The RSLOs attend meetings dealing with regional low-level radioactive waste issues and monitor State progress in developing needed capacity for the disposal of low-level waste. They also participate in emergency planning exercises involving State and local governments.

Liaison with American Indian Tribes. The NRC continues to maintain communications with those American Indian Tribes, including their national organizations, who are potentially affected by or otherwise interested in NRC regulatory activity. While no Tribes have been formally accorded "affected status" under the 1987 Nuclear Waste Policy Amendments Act, the 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 activity of the NRC's Advisory Committee on Nuclear Waste.

During the past year, the Commission has met with a number of Tribes to hear their concerns and provide information concerning nuclear activity on or near Tribal land. Informative meetings have been held with the Prairie Island Tribal Council in Minnesota regarding the Council's concerns with the Northern States Power Company's plan to construct an independent spent fuel storage installation at its Prairie Island nuclear plant site. The staff also met with the Naragansett Tribe in Rhode Island to discuss their concerns with the license termination for the UNC Recovery Systems facility, located at Wood River Junction. The Shoshone-Bannock Tribes in Idaho have also expressed an interest in spent fuel shipments from the Fort St. Vrain reactor in Colorado to the Idaho National Engineering Laboratory. The Tribe has requested advance notification of future spent fuel shipments crossing their reservation.

Interagency meetings are another means by which NRC keeps up-to-date on American Indian issues. Meetings sponsored by the Environmental Protection Agency afford the opportunity to exchange new information of potential relevance and importance to Federal and Tribal activities. NRC also maintains liaison with Department of Interior/Bureau of Indian Affairs in an effort to keep its constituency abreast of nuclear-related issues affecting Indian country.

INTERNATIONAL PROGRAMS

The NRC's international activities, serving the agency's world-wide objectives through the Office of International Programs, are intended to:

- Improve world-wide cooperation in nuclear safety and radiation protection.
- Assist U.S. efforts to restrict U.S. nuclear exports to peaceful use only.
- Support U.S. foreign policy and national security objectives.
- Contribute to the safe operation of licensed U.S. reactors and fuel cycle facilities and the safe use of nuclear materials.

The NRC's international program in nuclear safety includes bilateral and multilateral regulatory and research cooperation, including extensive interaction with the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development/ Nuclear Energy Agency (OECD/NEA).

Power reactor safety, the primary focus, and materials safety—including radioactive protection, waste management, source and by-product material fuel handling, and international transportation of radioactive waste—are an important part of the NRC international agenda.

Through its international programs, the Commission is continuing bilateral cooperation with a number of countries, is broadening its focus on reactor safety to include Soviet-designed reactors in Eastern Europe, and is working with the IAEA, the NEA, the European Community (EC) and other international groups on nuclear safety and regulatory matters.

Highlights of Fiscal Year 1991

- Arranged for Chairman Selin to participate in the NEA Senior Regulators meeting in Paris; to hold policy discussions with senior officials concerned with nuclear safety in Bulgaria, the Czech and Slovak Federal Republic, the former U.S.S.R., Finland, Sweden and the United Kingdom, while visiting nuclear power plants in these countries; to participate in the IAEA General Conference in Vienna; and to hold bilateral discussions during the General Conference.
- Continued substantial bilateral cooperation with the former U.S.S.R. 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 (JCCCNRS).
- Participated in the second U.S.-U.S.S.R. reactor inspector exchange, the U.S. team spending a month at the Soviet regional office in Kiev, and on other site visits, during September 1990, and the Soviet team visiting in Regions I and IV in January 1991, in conjunction with site visits.
- Expanded the NRC's focus on the safety of Soviet-designed reactors in Eastern Europe.
- Responded to concerns about threats to the Krsko reactor in Slovenia during the civil war in Yugoslavia.
- Met with the Japanese Agency for Natural Energy Resources (ANRE) of the Ministry of International Trade and Industry (MITI) for the Sixth Regular Meeting on Nuclear Regulatory Matters. The occasion included meetings of Commissioner Remick with senior Japanese officials.
- Signed a Memorandum of Understanding with Argentina and held discussions with senior Argentine and Brazilian nuclear officials, during then Chairman Kenneth Carr's visits in November 1990. Chairman Carr also visited the United Kingdom in June 1991, including a visit to the Sizewell B plant under construction there.
- Arranged regulatory safety information exchanges involving the the Commission and senior-staff level personnel and senior nuclear safety authorities from Spain, Italy, Mexico, Brazil, Argentina, Peru, Korea, and the United Kingdom.
- Arranged temporary assignments at the NRC, as regulatory staff members, of 14 individuals from nine countries, to work in the areas of inspection, technical assessment, radiation protection, emergency preparedness, analysis and evaluation of operational data, accident evaluation and advanced reactor research.
- Completed 208 export licensing actions. Of these, 101 involved routine exports of low-enriched uranium (LEU) fuel for various power reactors around the world.
- Issued licenses for export of more than 183 kilograms of high-enriched uranium (HEU), for use in research and test reactors in Canada, France and Japan.

- Participated in the NPT Nuclear Exporters Committee (Zangger Committee) meeting on nuclear-related issues and in the Nuclear Suppliers Group (NSG) discussions to develop an international export control list for "dual-use" nuclear-related items.

- Participated in the Coordinating Committee for Multilateral Export Controls (COCOM) negotiations in France to consider the development of a new list of controlled nuclear-related commodities for export to China, Eastern Europe and the Soviet Union.

- Coordinated visits by Commissioner Rogers to Korea, Norway, Finland, Sweden, the United Kingdom, Mexico and Canada to discuss nuclear safety issues.

- Worked closely with the Executive Branch and the IAEA in strengthening international safeguards and physical security, including sending experts to the United Kingdom, Indonesia, Australia and Mexico.

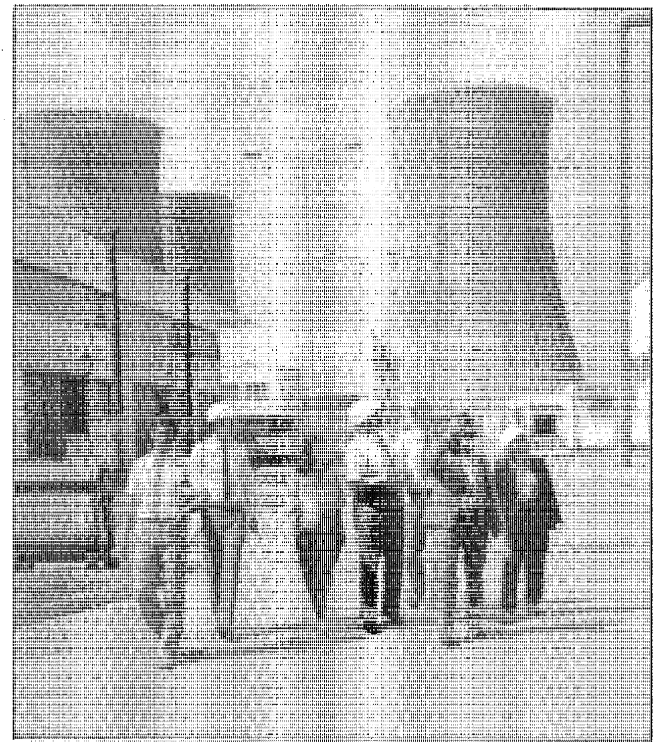
- Participated in interagency working groups to review international safeguards in light of lessons learned from the Iraqi breach of its nuclear non-proliferation obligations.

- Participated actively in the IAEA's September International Conference on the "Safety of Nuclear Power: Strategy for the Future," with Commissioner Remick chairing the advanced reactor session, and NRC senior staff working on the steering committee and chairing a working group. Commissioner Remick also visited the Dukovany nuclear power plant in the Czech and Slovak Federal Republic.

- Sent 47 participants to IAEA meetings on nuclear safety issues, such as emergency operations, accidents, radiation sources, aging, waste management, decommissioning, safety research, research reactors, nuclear liability, maintenance, probabilistic risk assessment, and transport of radioactive materials.

- Sent a total of 13 U.S. experts to participate in eight IAEA Operational Safety Review Team (OSART) missions to Finland, Sweden, the Czech and Slovak Federal Republic, Bulgaria, the former U.S.S.R. (two missions), and pre-OSARTs in China and Romania.

- Participated in the IAEA's annual Nuclear Safety Standards Advisory Group (NUSSAG) meeting in Vienna to review reactor safety standards-related activity.



Commissioner Forrest J. Remick, fourth from left, visits the Dukovany nuclear power plant together with aides and escorts, in September 1991. The plant is located inside the Czech and Slovak Federal Republic.

- Sent an NRC expert with an IAEA Assessment of Safety Significant Events Team (ASSET) mission to the Vandellos nuclear power plant in Spain and a U.S. expert to the Kozloduy nuclear power plant in Bulgaria.

International Cooperation

U.S.-Soviet Civilian Nuclear Safety Cooperation. The U.S.-U.S.S.R. Agreement for Peaceful Uses of Atomic Energy provided the legal basis for the Joint Coordinating Committee for Civilian Nuclear Reactor Safety (JCCCNRS). Under the JCCCNRS Protocol, the NRC participated in working groups in Moscow and in Washington during fiscal year 1991, to discuss specific nuclear safety issues and make on-site visits to exchange operational experience. In working group sessions, both sides indicated that potential future topics of mutual interest include fire protection in design of new plants and in backfitting operating plants, as well as issues involving changes in material properties as a result of irradiation. A two-part Reactor Vessel Annealing Team visited Moscow in February and again in March to observe the Soviet annealing process, followed by detailed discussions. Both contingents of the team reported very positive results from the experience. In September 1990, an NRC inspection team spent a month working at the Soviet regional office in Kiev and participating in other site visits. A reciprocal Soviet team visited the United States in January 1991 for five weeks of discussions and site observations in Regions I and IV, with side visits to Comanche Peak (Tex.), Three Mile Island (Pa.), Haddam Neck (Conn.), and Indian Point Unit 1 (N.Y.).

Chairman Selin led an NRC delegation to hold policy discussions with senior Soviet and republic officials in Moscow and Kiev on the safety of nuclear power and the restructuring of organizations dealing with nuclear power issues, in the wake of the attempted *coup d'etat* in August. The team also visited the VVER–1000 nuclear facility at Khmelnitskiy in Ukraine and the RBMK–1000 plant at Chernobyl. The visit occurred at a time when major changes were taking place in the Soviet Union and major reorganizations were being implemented, leading to greater authority in the republics, both for plant operations and for safety regulation. The Chairman sought to establish good links with emerging republic groups, while maintaining contacts with remaining central authorities.

Bilateral Information Exchange Arrangements

The NRC participates in a wide-ranging, mutually beneficial program of information exchanges and cooperative safety and research activity with its counterparts in the international community. Since 1974, when it formalized the information exchange arrangement program, the NRC has conducted most of its technical information exchanges through a series of 27 general safety cooperation arrangements with regulatory authorities in Argentina, Belgium, Brazil, Canada, China, The Czech and Slovak Federal Republic, Denmark, Egypt, Finland, France, Germany, Greece, Hungary, Indonesia, Israel, Italy, Japan, the Republic of Korea, Mexico, the Netherlands, the Philippines, Spain, Sweden, Switzerland, the former U.S.S.R., the United Kingdom, Yugoslavia 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 assist in the identification of possible precursor events that merit 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. The arrangements are usually effective for five years, but they include provision for renewal by mutual written agreement of the parties.

During fiscal year 1991, the NRC concluded its first information e exchange arrangement on civil power and research reactors with Argentina. Active negotiations continue on renewal of the NRC's arrangements with China, Germany, Indonesia, Japan, Korea, the United Kingdom and Yugoslavia.

Nordic Countries. In May, Commissioner Kenneth Rogers visited Finland, where he gave a presentation at an international conference and met with officials of the Finnish Center for Radiation and Nuclear Safety (STUK) and visited the Loviisa nuclear power plant. In Norway, he toured the OECD Halden Reactor project and met with officials from the Norwegian Nuclear Safety Authority. In Sweden, the Commissioner met with officials of the Swedish Nuclear Power Inspectorate (SKI), the National Institute of Radiation Protection (SSI), the National Board for Spent Nuclear Fuel, and the Swedish Nuclear Fuel and Waste Management Company (SKB). He also visited the Forsmark and Oskarshamn reactor sites, the Subseabed Final Repository for for low- and intermediate-level waste and the Central Storage for Spent Fuel facility.

As a part of his European visit in September, Chairman Selin visited the Loviisa nuclear power plant in Finland and held discussions with officials at STUK, the Finnish nuclear regulatory body. He also visited Sweden briefly to tour Unit 3 of the Forsmark reactor site and the Subseabed Final Repository for low- and intermediate-level waste. He held discussions with nuclear safety officials from SKI, SSI, SKB and the Forsmark reactor site.

Spain. In September, during the IAEA General Conference in Vienna, Chairman Selin signed a renewal of the general research agreement between the NRC and its Spanish counterpart, the CSN. Chairman Donato Fuejo Lado signed the renewal agreement for Spain. The renewal assures continuation of a very productive nuclear safety research relationship between the NRC and the CSN.

In June, Mr. Gaspar Arino, a member of the Spanish Congress of Deputies, met with NRC staff to discuss the regulation of the U.S. nuclear industry and the current status of the U.S. nuclear program. Deputy Arino was in the United States for a briefing on the regulation of U.S. utilities for use in drafting legislation to decentralize and privatize Spanish utilities. The purpose of the Spanish legislative initiative is to facilitate third party access to the Spanish market for the 1992 integration of the European Community.

United Kingdom. In April, after attending the World Association of Nuclear Operators (WANO) biennial meeting in Atlanta, Mr. James Hann, Chairman of Scottish Nuclear Ltd., met with former Chairman Carr, Commissioner Curtiss and members of the NRC staff. Discussions focused on British energy strategy, high-level radioactive waste activity, privatization, and the 1994 review of the United Kingdom's nuclear energy program.

In June, former Chairman Carr presented a paper in London, "A View from a Regulator's Perspective," at the British Nuclear Forum Conference on "Nuclear Power: Clean, Safe Energy with a Future," and visited the Sizewell B nuclear power plant under construction in the United Kingdom. In September, Commissioner Rogers presented a paper, "Issues Underlying Public Attitudes Toward Nuclear Power in the United States," at the British Nuclear Forum Ditchley Park Debate near London.

Canada. Commissioner Rogers visited Canada in October 1990 and visited the AECL's Sheridan Park Engineering Laboratory and Ontario Hydro's Darlington nuclear power plant, to discuss AECL's Integrated Design program and tour the Chalk River Laboratory. He returned to Canada in August of 1991 to discuss Canadian nuclear fuel waste management and toured related facilities.

During 1991, NRC staff continued intensive exchanges with Canada in such areas as CANDU technology, nuclear power plant inspections, industrial radiography, emergency response and procedures, reactor project management, licensing evaluations and safety analysis.

Mexico. In February, Commissioner Rogers visited the Mexican Ministry of Energy, Mines and Parastatal Industry to hold discussions with senior Mexican nuclear officials at Mexico's nuclear regulatory organization (CNSNS), the Federal Electricity Commission (CFE), the utility operating the Laguna Verde reactor, the main nuclear research organization, and several professors from Mexican universities. He also toured the Laguna Verde facility, which came on line in July 1990. During the discussions, it was agreed that a program of joint evaluation of five BWR operational events by a team of specialists from NRC, the CNSNS, and the CFE might be of mutual benefit to the NRC and the appropriate Mexican organizations.

In May, CNSNS Director General Miguel Medina visited NRC Headquarters for nuclear safety discussions with Commissioners and senior management. Mr. Medina brought with him copies of operational events at Laguna Verde for NRC analysis.

Germany. An NRC team visited Germany in May to exchange information with local personnel on plant operating experience; to learn how operating data are being used; to obtain information on feedback mechanisms; and to find out to what degree NEA Incident Reporting System (IRS) data are being integrated in safety analyses. While in Germany, the team held productive discussions with personnel of the Reactor Safety Company (GRS) in Cologne, the BMU (Ministry for Environment, Nature Conservation and Nuclear Safety) in Bonn, and the Philippsburg nuclear power plant. This visit was one stop on an itinerary which included France, Switzerland and Belgium.

Signing the renewal of the Research Agreement between the NRC and the Spanish Nuclear Safety Council are Council Chairman Donato Fuejo Lado, at left, and NRC Chairman Ivan Selin, at right. The renewal was ratified in Vienna, during the 1991 IAEA General Conference.

Dr. Walter Hohlefelder, Assistant Secretary for Nuclear Safety (BMU), and Dr. Adolf Birkhofer, General Manager (GRS), visited the NRC in July to exchange views with senior NRC officials on objectives and results of ongoing nuclear safety cooperation with the former U.S.S.R. and other Eastern European countries and on anticipated safety characteristics of new reactors.

Eastern European Cooperation. In September Chairman Selin visited Bulgaria to visit the Kozloduy nuclear power plant and to hold discussions with Bulgarian nuclear officials in Sofia on the safety of the plant. He also visited the Bohunice reactor in the Czech and Slovak Federal Republic.

A visit to the United States by a joint Czech and Slovak Federal Republic-Hungarian delegation was scheduled in December 1991. In the interim, the NRC made arrangements for the release to the above two countries of several computer codes (e.g., RELAP 5/Mod2 Thermal-hydraulic Code and MELCOR Severe Accident Code) for use in performing safety analyses of their plants.

A senior NRC staff member visited Budapest, Hungary, in April 1991 to participate in the second part of a joint American Nuclear Society-Hungarian Nuclear Society seminar on PWR safety issues. The seminar brought together representatives from U.S. industry, academia, and government agencies (the NRC and the Department of Energy (DOE)), and their West and East-European counterparts) to give Hungary with up-to-date information on nuclear safety developments in western countries, as well as to provide for consultation with an expert from the Sandia National Laboratory.

The NRC concluded an agreement with the U.S. Agency for International Development in September whereby the NRC is provided $575,000 in funds during fiscal year 1991 in support of nuclear safety assistance to the Czech and Slovak Federal Republic and Hungary. The funds will be used to pay actual expenses in the United States for nuclear specialists from these countries, who are participating in approved activities, and to pay for their memberships in two nuclear code-users groups. Funding for later years was also approved.

Argentina and Brazil. In November 1990, former Chairman Carr visited Argentina and Brazil to meet with government and utility representatives to discuss nuclear safety and non-proliferation matters and to visit the Atucha and Angra nuclear power plants, respectively. While in Argentina, he also signed the first NRC-Argentine National Atomic Energy Commission (CNEA) Memorandum of Understanding for the Exchange of Technical Information Directly Applicable to the Safety of Operating Civil Power and Research Reactors. Both countries subsequently sent senior delegations to the United States for follow-up nuclear safety and non-proliferation discussions.

Korea. Commissioner Rogers travelled to Korea in April to deliver a paper on Directions in U.S. Nuclear Regulatory Policy, during the Plenary Session of the 6th Annual Conference of the Korea Atomic Industrial Forum/Korean Nuclear Society, and to participate in detailed safety discussions with the Korean nuclear establishment. He also visited the Korea Electric Power Corporation's Kori-3 nuclear power plant and the manufacturing facilities of Korea Heavy Industries Construction, Inc. In May, the NRC participated on the U.S. interagency delegation to the 13th meeting of the U.S.-Korea Joint Standing Committee on Nuclear and Other Energy Technologies in Seoul and in a side trip to Daeduk Science Town for meetings with, among others, the Korea Institute of Nuclear Safety, the NRC's regulatory counterpart. In September Chairman Selin hosted a visit to NRC by Korean Minister of Science and Technology Jin-Hyun Kim, who was interested in discussing approaches to low-level waste management and public acceptance.

Japan. In May, a team of senior staff from NRR, AEOD, Region II, and the Office of International Programs met in Tokyo with representatives from the Agency for Natural Energy Resources, Ministry of International Trade and Industry, for the Sixth Regular Meeting on Nuclear Regulatory Matters. The two-day meeting, also attended by Commissioner Forrest Remick, covered topics on current nuclear power plant operating experiences, accident sequence precursors, loss-of-electrical-power events, plant life extension, research on severe accidents, shutdown risk and operating events during shutdown. Discussions were also held with the Science and Technology Agency's Nuclear Safety Bureau to discuss their recently completed review of TEPCO's advanced boiling water reactor construction application.

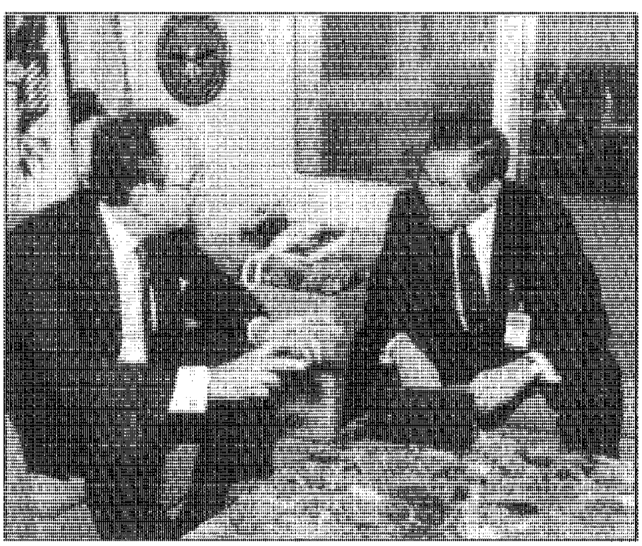
While in Japan, Commissioner Remick met with officials from the Ministry of International Trade and Industry and the Nuclear Safety Commission. He also visited the Toshiba Nuclear Engineering Laboratories in Yokohama, the Fukushima Daini nuclear power plant near Tomioka, and the research laboratories of the Japan Atomic Energy Research Institute and the Power Reactor and Nuclear Fuel Development Corporation in Tokai.

Taiwan. The NRC team members visiting Japan also visited Taiwan, where they participated in the American Institute in Taiwan (AIT)-Coordination Council for North American Affairs' Joint Committee Meeting on Civil Nuclear Cooperation. There were presentations on the current operation of U.S. and Taiwan nuclear power programs, advanced reactors, nuclear waste handling and storage, and long-range energy needs. In fiscal years 1991 and 1992, work will continue in thermal hydraulics, reactor aging, mechanical engineering, seismic research, operational safety, and emergency planning. Planned new items of cooperation include attendance by several engineers from the Taiwan Atomic Energy Council in courses at Chattanooga's Technical Training Center and several visits to Headquarters and the Regions for one-day technical discussions on topics including quality assurance programs, radiation safety issues, spent fuel management, decontamination and decommissioning, and low- level radwaste treatment.

India. The Office of International Programs, together with the Office of Nuclear Material Safety and Safeguards and the Office of State Programs, alerted the Government of India to the discovery of steel fencing material, imported from India into the United States, that was found to be slightly contaminated with cobalt-60. The Indian Government was also apprised of the potential contamination problem at the involved Indian steel scrap and rolling companies and was given the information gathered in the United States about the extent of the problem and the U.S. response to it. The Indian Government shared the results of its investigation with the NRC.

Participation in International Organizations and Conferences

IAEA General Conference and Board of Governors Meeting. Chairman Selin represented the NRC at the 35th Session of the General Conference of the International Atomic Energy Agency in Vienna in September.



Chairman Selin hosted a visit to NRC Headquarters, during the report period, by Korean Minister of Science and Technology Jin-Hyun Kim. Chairman Selin, at left, and the Minister discussed low-level radioactive waste management, among other issues of mutual interest.

The NRC was also represented at the June IAEA Board of Governors meeting.

International Nuclear Safety Conference. Commissioner Remick chaired a session on advanced reactors at the IAEA's International Conference on the "Safety of Nuclear Power: Strategy for the Future," in early September. Harold Denton, Director of the NRC's Office of International Programs, was a member of the Conference Steering Committee, and Edward Jordan, Director of the NRC's Office for Analysis and Evaluation of Operational Data, was a member of the working group on operating reactors. Though it had not been announced as part of the agenda for the conference, the issue of whether there should be an international convention on safety standards for nuclear power facilities and radioactive waste management was raised and explored, with the NRC representatives actively involved in developing the U.S. position on the matter.

Helsinki Symposium. Commissioner Rogers presented a luncheon address at the Senior Expert Symposium on Electricity and the Environment held in Helsinki in May. The meeting was sponsored by 11 international organizations, including the IAEA. Nuclear power was one of several energy options discussed at the meeting, which looked at environmental problems (and possible solutions) caused by electric power generation.

OSARTs. NRC staff members participated as experts on two IAEA Operational Safety Review Team (OSART) missions to Finland and Sweden this year. The NRC arranged to have U.S. utility experts take part in OSART missions to Bulgaria (three missions), the Czech and Slovak Federal Republic and the former U.S.S.R. (two missions), as well as in a pre-OSART trip to China.

Activities in the OECD/NEA

The NRC maintained its active involvement in OECD/NEA activities by serving on key standing committees and working groups and participating as members of the U.S. delegation to the Steering Committee meetings. The NRC also participated in several international NEA projects, including the TMI Pressure Vessel Examination Project and the establishment of the International Information System on Occupation Exposure (ISOE), supporting the operation of the ISOE North American Technical Center.

In April, NEA Director General Kunihiko Uematsu and NEA Deputy Director for Nuclear Safety and Regulation Klaus Stadie visited the NRC to discuss a wide range of current safety issues associated with multilateral cooperation between the Commission and the NEA, including improved coordination of activity between the NEA and the International Atomic Energy Agency

Commissioner Forrest Remick chaired the advanced reactor session at the International Conference of the Safety of Nuclear Power: Strategy for the Future, held in Vienna in September of 1991.

(IAEA), cooperation between the NEA and Eastern European countries, and a proposal for cooperation with Asian countries.

In September Chairman Selin attended a special NEA meeting for the heads of nuclear safety organizations of seven major OECD countries (France, Germany, Italy, Japan, Sweden, the United Kingdom and the United States) to discuss possible steps to be taken to improve the safety of Eastern European reactors and the prospects for a wider international safety regime. A large part of the meeting was devoted to the question of nuclear safety assistance to Eastern European countries and to mechanisms for the transfer of technology and regulatory safety information. On a more general basis, the participants also discussed the prospects, and limitations of, an enhanced international nuclear safety regime.

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 1991, the NRC completed 208 export licensing actions. Of these actions, 101 involved exports of low-enriched uranium fuel for various power reactors around the world using uranium of U.S origin or purchasing DOE uranium enrichment services. Countries using the low-enriched uranium fuel include France, Germany, Japan, South Korea, Spain, Sweden and Taiwan.

The NRC also issued four licenses authorizing the export of more than 183 kilograms of high-enriched uranium for use in research and test reactors (as target material), in Japan (JRR-2 and JMTR), France (HFR-Grenoble) and Canada (NRX-NRU.

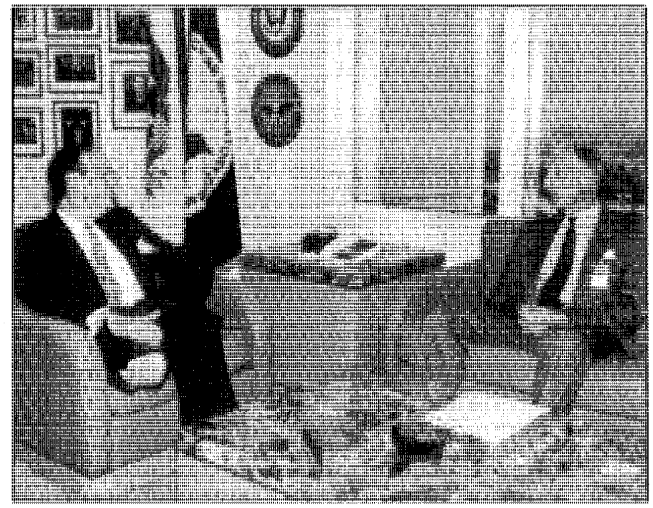
Consultations with the Executive Branch on Export Matters. The NRC, in addition to its usual licensing actions, consults with the Executive Branch on other nuclear-related exports. These involve nuclear-related commodities licensed by the Department of Commerce, Executive Branch requests for retransfers of nuclear material originating in the United States, and nuclear technology transfers. Cooperation with the former Soviet Union and East European countries continued during the report period in transfers of safety-related nuclear technology.

TOPAZ II Space Reactor. NRC Export regulations were amended by the Commission in June to permit the return of the TOPAZ II space reactor to the former Soviet Union. The reactor was imported in January 1991 by a contractor of the U.S. Department of Defense for the purpose of being exhibited at an international space nuclear power symposium in New Mexico. Prior to issuing the import license for the reactor, the NRC staff called attention to the difficult obstacles in the Atomic Energy Act, standing in the way of any planned return of the device to the U.S.S.R. The change to NRC's export regulations made in June was not a generic amendment, but rather a measure adopted to facilitate the return of the one already imported Topaz II device.

Enrichment Plant Safeguards. Negotiations continue between the United States (the NRC and the Executive Branch Agencies) and the URENCO governments (the U.K., the Netherlands and Germany) on an agreement that will lead to the licensing and construction in the United States of a privately owned centrifuge enrichment facility, equipped with URENCO-designed and built centrifuges. The negotiations were expected to lead, by the end of 1991, to a formal inter-governmental agreement that would contain the necessary proliferation and safeguards assurances. The agreement would also resolve the issue of export control requirements on the transfer to Europe by the proposed facility operator of operating and design information that could include "Restricted Data."

Non-Proliferation, International Safeguards and Physical Protection. The NRC staff reviews pending export cases with a view to confirming the IAEA safeguards and physical security arrangements to be applied to the exports in 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 also participates in the Subgroup on Nuclear Export Coordination (SNEC), the 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. SNEC provides an effective 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.



At the right, visiting with Chairman Selin at NRC Headquarters, is Ambassador Andreas van Agt, of the Commission of European Communities, to explore the many regulatory issues affecting U.S. and European nuclear programs.

In May, the NRC was represented at the Non-proliferation Treaty (NPT) Nuclear Exporters' Committee meeting (Zangger Committee) in Vienna, as part of the U.S. delegation, headed by the Department of State. The Committee consists of 23 NPT countries (signatories to the Treaty on Non-Proliferation of Nuclear Weapons Treaty) and meets regularly to discuss and consult on nuclear-related issues. The IAEA-sanctioned Zangger Committee has developed an internationally agreed upon list (the trigger list) for the control of proliferation-sensitive items, the export of which triggers IAEA safeguards. Most of the trigger list items are incorporated in NRC's regulations.

The NRC was represented on the U.S. delegation at the Coordinating Committee for Multilateral Export Controls (COCOM) meeting, held in Paris, to consider reductions in the COCOM list of controlled items on the International Atomic Energy List. COCOM is an international export control group of 17 western countries and Japan that was formed to maintain common controls on exports to Eastern Europe, China and the Soviet Union. In light of events in the Soviet Union and the democratization of Hungary, the Czech and Slovak Federal Republic and Poland, COCOM members in late 1990 began consideration of an entirely new core list of controlled commodities and goods to replace the existing COCOM list. (The new list would liberalize many of the controls on these items.) Technical negotiations were to continue in December 1991.

The NRC also was on the U.S. delegation to the Nuclear Suppliers Group (NSG) third Dual-Use Working Group meeting held in Annapolis, Md., in early October 1991. The 26 countries of the NSG are developing a multilateral control list for dual-use nuclear items, as a way to strengthen dual-use export controls world-wide on such items. Deliberations on the new list began early in 1991, after the Persian Gulf War revealed the weaknesses in international dual-use nuclear export controls, and, especially, in light of the substantial reductions that had occurred in COCOM export controls (a number of dual-use nuclear-related items had been decontrolled). Before the NSG initiative, there had been no mechanism to coordinate controls among the various countries that produce items of potential use in nuclear weapons applications. Some of the items on the new list are being considered for IAEA safeguards, which means these items would most likely be controlled under NRC export regulations.

The U.S. Program for Technical Assistance to IAEA Safeguards (POTAS) provides the largest share of voluntary technical support by IAEA member states. In 1991, the NRC provided one staff member to the IAEA Department of Safeguards for a POTAS-funded research project on the effectiveness of IAEA safeguards. Through its participation in the Technical Support Coordination Committee, an 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 and, during 1991, 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 was assembled in response to the discovery of the clandestine Iraqi nuclear program and its impact on the perception of the effectiveness of the international safeguards regime. The group is charged to produce proposals to strengthen the nuclear non-proliferation regime that will 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 1991, U.S. delegations visited the United Kingdom, Indonesia, Australia and Mexico. The NRC also participated in an international conference in Vienna to review the Convention on the Physical Protection of Nuclear Material.

Nuclear Regulatory Research

Chapter

8

Activities of the Office of Nuclear Regulatory Research (RES) contribute 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 and probabilistic risk assessments. It is also a RES function to conduct 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 fiscal year 1991 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which comprises a listing of those guides issued, revised or withdrawn during fiscal year 1991.

This chapter summarizes RES activities during fiscal year 1991 under the following major headings: Preventing Damage to Reactor Cores, Reactor Containment Performance, Integrity of Reactor Components, Confirming Safety of Nuclear Waste Disposal, and Resolving Reactor Safety Issues and Developing Regulations.

Preventing Damage To Reactor Cores

The research effort dealing with the prevention of damage to reactor cores and the mitigation of severe accident consequences encompasses the operations of the reactor as a system; the establishment and maintenance of accident management programs to minimize the risk to the public, in the event of severe accidents; and consideration of the operator as an integral part of the reactor system.

PLANT TRANSIENT ANALYSIS

Modeling

As a result of an event at the Vogtle (Ga.) plant on March 20, 1990, the staff was asked by the Commission to perform a comprehensive evaluation of safety levels during reactor shutdown and during low-power operation. In particular, RES was asked to evaluate the effectiveness of alternate decay heat removal methods after the loss of residual heat removal (RHR). A report entitled "Thermal-Hydraulic Processes Involved in Loss of RHR During Reduced Inventory Operation" (EGG–EAST–9337, Revision 1) was forwarded to the Commissioners on March 1, 1991. Of the two cooling methods identified in a staff document on the subject (NUREG–1410), i.e., gravity feed from outside sources, such as the refueling water storage tank, and reflux cooling, the report EGG–EAST 9337 concluded that gravity feed phenomena were well understood but that reflux cooling at reduced pressures in the presence of air was not as well understood.

In order for reflux cooling to work in a system partially filled with air, the pressure must increase enough to expose a condensing surface in the steam generator. The concern was that such a pressure rise may lead to blowing out the nozzle dams or the instrument thimble tube replacements, possibly causing a loss of coolant while the plant was in this particular shutdown condition. The design pressure of the thimble tube replacement is about 50 pounds-per-square-inch (psi), while the setpoint is well above that, at about 100–psi.

Bounding analyses were performed on U-tube steam generators showing that the pressure increase required to expose a condensing surface was only 15–psi, well below the 50–psi design pressure of the thimble tube replacements. In order to confirm these results with test data, initial commitments were obtained from the French and Japanese to perform relevant tests in their BETHSY and ROSA facilities, respectively.

The pressure rise needed to expose a similar condensing surface in Babcock & Wilcox reactors with once-through steam generators may be higher. Special tests were performed at a University of Maryland facility to obtain data on the magnitude of this pressure rise.

> ### 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 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 lay out the steps taken by the staff in evaluating 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 known as "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 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 its likely impact, in terms of resources and effort involved.

Another request made of the staff sought an evaluation of the software used in in calculating heat and smoke propagation in a multi-compartment structure, in order to assess the possibility of inadvertent actuation of fire protection systems. The results were to be used in a probabilistic risk assessment (PRA) study associated with the resolution of Generic Issue 57, "Inadvertent Actuation of Fire Suppression Systems." A survey disclosed that the National Institute of Science and Technology (NIST) produced candidate software. This code was evaluated, found appropriate, and used to provide the analyses needed for the PRA study. Additional modeling of vertical plumes and horizontal ceiling sets was identified, and a more comprehensive NIST code was used to provide results with these models to corroborate previous findings.

Operating Reactor Assessments

In order to extend the duration of fuel cycles, utilities operating pressurized water reactors (PWRs) have taken to increasing the enrichment of reload fuel. The fresh reload assemblies may be highly reactive when they do not contain control element assemblies or many burnable poison rods. In certain loading configurations, this reactivity could lead to a loss of the required shutdown margin, below 5 percent, or, in the extreme, to an inadvertent criticality.

The NRC alerted PWR owners to this potential problem and requested that licensees ensure, by specific action, that any intermediate fuel assembly configuration in the reactor is maintained within the required shutdown margin. NRC's Office of Nuclear Reactor Regulation also requested that RES use the most appropriate deterministic and probabilistic techniques to analyze different refueling configurations, the better to understand the potential for losing shutdown margin and for inadvertent criticality.

Research results showed that a cluster of at least four fresh fuel assemblies is needed before the required shutdown margin is lost, and the frequency of that event is expected to be very small, less than 1.0E–6/year. If the conservative assumption is made that it only takes five fresh assemblies to cause an inadvertent criticality, then the frequency is 4.8E– 9/year, which is an acceptably low probability. If such a criticality event were to occur, there would be radioactivity released from the fuel but no dose to the public, provided the containment isolation systems remain effective.

The LaSalle (Ill.) nuclear power plant underwent an "oscillation event" on March 9, 1988, which brought into question the ability of current analytic techniques to predict the instability boundary in boiling water reactors (BWRs), as well as the magnitude of the power and flow oscillations that might occur when the reactor does become unstable. This past year, the NRC was able to assure itself and the industry that its analytic tools can predict such oscillatory behavior in BWRs. A major finding was that it is important to model major systems in the balance of the plant (not only in the nuclear steam supply system), especially the feedwater temperature and flow, in order to correctly predict the oscillations in the reactor vessel. It should be noted that General Electric has now improved their modeling capability to calculate similar oscillatory magnitudes.

Research results during the fiscal year revealed that containment pool heatup after an "anticipated transient without scram," or ATWS, with unstable oscillations, is well below any safety margin. However, it was found that there could be a large increase in fuel rod temperature. Two potential improvements to operating guidelines were investigated. First, it was found that significant oscillations can start about two or three minutes after an ATWS, and thus the operator action of boron injection through the standby liquid control (SLC) system—which may take about 10 minutes to influence the power oscillations—will not be immediately effective. Second, it was found that early tripping or shutdown of the feedwater pumps is very effective in suppressing oscillations.

The RES team reviewing the Yankee-Rowe (Mass.) licensee's evaluation of their pressure vessel performing under pressurized thermal shock (PTS) conditions raised a question on the calculation of the fluid temperature in the downcomer during a postulated small-break loss-of-coolant event. A detailed study was performed, and it showed that the Yankee-Rowe calculation with the REMIX code was a best estimate of the downcomer temperature. The conclusion was based on the assessment of REMIX against data from six scaled test facilities, published in "A Unified Interpretation of 1/5 to Full Scale Thermal Mixing Experiments Related to PTS" (NUREG/CR-5677, April 1991).

An analytic study exposed difficulties in the use of thermal/hydraulic (T/H) codes to calculate the time to fuel pin failure after a loss-of-coolant accident (LOCA). The study explored the possible use of new source term information to increase the allowed technical specification time for containment isolation valve closure following a LOCA. After close scrutiny of the key code issues for the analyses being performed, the study was completed showing a time of about 30 seconds to fuel pin failure. These results could ultimately prove useful to the development of isolation valve testing strategies and improvement of valve design reliability.

With completion of testing and subsequent shutdown of large scale U.S. T/H test facilities in 1989, the NRC was concerned whether small scale facilities—such as the University of Maryland at College Park (UMCP) facility—could provide useful data for code assessment and issue resolution. The UMCP program was completed during the fiscal year and showed that useful data could be obtained, after a proper scaling methodology was used to specify the test boundary conditions and to interpret the resulting T/H data. UMCP results agreed with major results of the larger scale MIST facility, after they were scaled to vessel inventory and after appropriate adjustments of initial and boundary conditions were made.

The UMCP scaling report was reviewed by a group of experts; their conclusions on the usefulness of such a small scale facility were mixed. The review recommended that a small scale facility must be designed and operated with careful attention to scaling principles. There is much less experience with the scaling used for the UMCP than with the power-to-volume, full-height, full-pressure scaling used for MIST and most other large scale facilities. In addition, the large scale facilities are thought to be more useful for stimulating actual loop-to-loop interactions.

A small scale facility may be more useful for uncovering qualitative surprises in system interactions among components than in providing quantitative data for code assessment. Thus, such a facility can meet scaling goals that are modest, but achievable. The UMCP program showed that a small scale facility can be cost-effective in reproducing the key phenomena expected in the full-scale plant; they can be reproduced in the same time sequence, and their quantitative characteristics can be reasonably approximated.

ACCIDENT MANAGEMENT

During the report period, NRC research continued in its dual tasks of (1) defining the necessary components of a functioning utility severe accident management plan, and (2) providing the technical bases for evaluating industry-documented products on accident management.

In the first category, two reports were completed and transmitted to industry. The first, "A Systematic Process for Developing and Assessing A/M Plans" (NUREG/CR-5543), clarifies how the five framework elements—strategies, instrumentation, guidance, decision-making, and training—could be integrated into a working accident management plan. The second, "Instrument Availability for a PWR with a Large Dry Containment During Severe Accidents" (NUREG/CR- 5691), demonstrates how to assess the potential availability or unavailability of certain instrumentation that would be useful in following the course of a severe accident.

In the second category, significant progress was made in identifying and assessing accident management mitigative strategies. Two workshops held at the University of California at Los Angeles, one on PWRs and the other on BWRs, abetted the process by focusing on a limited set of specific strategies. The key mitigative strategies identified for detailed assessment include BWR boration, external vessel flooding, PWR primary depressurization, late primary bleed and feed, use of containment sprays, containment venting, hydrogen control, fission product control, and late secondary "bleed and feed."

A study of potential recriticality in a BWR following a core damage event (NUREG/CR-5653) showed that about 700 parts-per-million of B-10 (boron) is sufficient to ensure sub-criticality for conceivable core configurations, including standing fuel rods and melted control rods. A related strategy of initiating timely cooling of the suppression pool as quickly as possible was suggested as a way to extend the time available for boration.

The strategy of PWR primary depressurization (NUREG/CR-5447) was shown to be most effective if performed late rather than early. For a loss-of-heat-removal accident, "late" refers to time after the core begins to come uncovered. A survey was completed showing which plants would respond favorably to this strategy, which plants for which the strategy would not work at all, and which plants would need further analysis to confirm the effectiveness of the strategy.

There is some question as to whether natural circulation might lead to failure of the pressurizer surge line and consequent unintentional depressurization. To address this question in a systematic way, a workshop was held, out of which four processes were identified for inclusion in code modeling: (1) steam generator plenum mixing, (2) surge-line flow, (3) hot-leg/vessel flow behavior, and (4) hydrogen distribution. Models and/or bounding assumptions were developed for all four processes so that analyses can be performed next year to resolve the question by predicting the possibility of inadvertent depressurization.

HUMAN PERFORMANCE

In close coordination with three other NRC Offices—Analysis and Evaluation of Operational Data, Nuclear Material Safety and Safeguards, and Nuclear Reactor Regulation—RES is conducting research to confirm and to provide the technological bases that will ensure that regulation and regulatory guidance for nuclear operations appropriately address human performance. This research is a multi-disciplinary endeavor, relying upon both the behavioral sciences and a variety of engineering disciplines. The research is continuing to follow the program described in "Human Factors Regulatory Research Program Plan" (NUREG-1384).

The human factors research program is designed to conduct research regarding regulatory issues related to personnel performance, human-system interfaces, organizational factors, data acquisition and management systems, human reliability analysis and probabilistic risk assessment (PRA) methods and applications, and human factors generic issues. This activity—divided into (1) human factors research, (2) organizational factors research, and (3) reliability assessment research—is discussed below.

Human Factors Research

Through its personnel performance program, the NRC seeks to improve its understanding of the effect of human performance on the safety of nuclear operations and maintenance, whether at power plants or materials facilities. A continuing project is the development of a human factors investigation process by which to provide a standardized method for investigating events with a view to identifying the root cause of human errors. Work continued on three projects involving a human factors evaluation of processes employed by materials licensees (e.g., those engaged in industrial radiography, brachytherapy using remote after-loaders, and teletherapy). The projects were undertaken to identify the human factors contributing to error in these processes.

Personnel performance research also continued into the impact of overtime and shift scheduling effects on operator performance, using nuclear power plant data, and seeking specifically to provide a quantitative data base for the evaluation of the safety implications of 12-hour shift schedules. Work also continues on the development of a methodology to assess the effectiveness of training programs at nuclear power plants. To that end, a workshop engaging training evaluation experts was held, resulting in the development of a framework on which to base a training effectiveness evaluation methodology. Research on the factors that go into operations staffing decisions and on how they relate to safe startup, shutdown, and operation of nuclear power plants is an ongoing effort. Also continuing is a study of the impact of environmental influences on human performance. Following a comprehensive review of the literature in this area, it has been decided that the focus of the study should be on the effects of heat and noise.

A new research project concerns training for severe accidents, focusing on cognitive skills development and training in how to deal with stress. A second, related, project will bring together a group of experts in a workshop to hear their independent views on issues related to accident management training and decision-making. Among the issues considered in the workshop will be the interactions between the training issues and the decision-making issues and the potential impact of such interactions on human performance during severe accidents.

Human-systems interface research continued with NRC participation in the "Halden Project." As a follow-up to an assessment of the costs and benefits of expanded regulatory guidance on normal and abnormal operating procedures (NUREG/CR-5458), a new project has been initiated to develop guidance for the review of procedures upgrade programs. Activity continued toward the resolution of Generic Issue 5.1 ("Local Control Stations"; see Table 3) with an historical survey of plant incidents caused, at least in part, by inadequate consideration of human factors at local control stations (LCS), and with a series of nuclear power plant site visits to document the status of component-level LCS human factors upgrades. And work continued on developing a guideline for use in performing human factors reviews of advanced control and display technology. A study is continuing 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. Research also continued on computer classification—which involves review and appraisal of existing regulatory guidance documents and quality assurance methods—and on their adequacy when applied to computer-based safety systems.

Research has continued with experiments to evaluate a performance indicator of the effectiveness of human-machine interfaces.

Another survey was performed to develop the technical bases for regulatory guidance on the design, development, test and acceptance of computer systems performing safety functions.

Organizational Factors

In the area of organizational factors, the development of modeling and data-gathering techniques to support PRA studies and inspection and diagnostic evaluation activity has continued. Data-gathering techniques included surveys, interviews, direct observation, and job performance sampling. Field testing continued at several NRC-licensed facilities and additional research was started on alternative quantification methods for incorporating organizational factors into PRA. Initial qualitative and quantitative validation studies of leading indicators of safety performance in the areas of organizational learning, resource availability, and resource allocation have been completed. Research continued on the feasibility of transferring established leading indicators of safety performance from the chemical processing industry to the nuclear power industry and on methods for interpreting the relative safety significance of leading indicators, individually and in combination.

Reliability Assessment Research

This research is intended to furnish tools and data for applying reliability technology to help improve the regulatory program. In particular, the research has developed analytical tools for evaluating the risk impact of requirements in nuclear power plant technical specifications. In fiscal year 1991, the research produced methods for evaluating and optimizing surveillance test intervals from a risk standpoint. Future work is planned to develop criteria for use with these methods. Reports published include "Issues and Approaches for Using Equipment Reliability Alert Levels" (NUREG/CR-5611) and a "Study of Operational Risk-Based Configuration Control" (NUREG/CR-5641).

A computer simulation, based on artificial intelligence, models the cognitive tasks required of operators during accident scenarios. When the computer simulation of operators is linked to a computer simulation of a nuclear power plant, they can together simulate the chronological progress of an accident and operator responses. In fiscal year 1991, data were gathered from operators responding to accident sequences on a training simulator. These data will be used to calibrate the computer simulation. The ultimate aim of the effort is to expand the operators' ability to deal with a wide range of potential accident conditions.

Other RES activity involving human performance has included resolution of questions regarding procedure violations during the Chernobyl accident and direct support in such areas as the design of the new operations center, plant inspections, and materials licensee investigations. A new study was initiated during the report period to assess the feasibility of establishing NRC human factors regulatory research facilities.

Research continued on the resolution of Generic Issue B-17, "Criteria for Safety-Related Operator Action" (see Table 3). A final draft of a revised industry standard, "Time Response Criteria for Safety-Related Operator Actions" (ANS 58.8), was developed and will be evaluated in fiscal year 1992.

Reactor Containment Performance

In order to ensure that existing regulations adequately protect the public from the consequences of severe accidents, research is carried out to confirm the technical bases upon which the regulations are founded. The technical bases include such matters as the behavior of fission products released from melting fuel, the temperatures and pressures produced during a core-melt event, and the capabilities of containment buildings to retain radioactive materials during such events. The behavior of radioactive materials released to the containment and potentially to the environment is also an important consideration. With data derived from this research, the NRC is better able to confirm the adequacy of its requirements for the siting, design, construction and reliability of those safety systems installed to mitigate the effects of severe accidents and also to determine when and where improvements in the regulations are indicated.

SOURCE TERMS

Fission Product Behavior

"Source Term" refers to the magnitudes of the radioactive materials released from the core to the atmosphere, the timing involved, and other information needed to calculate off-site consequences following a postulated severe reactor accident. The NRC conducts research in this area to help define and focus accident-management concerns and containment performance improvements and to help seek out potential latent vulnerabilities in individual nuclear plants.

At present, research is under way to develop 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 is being 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.

A version of the VICTORIA code has been completed and a user manual has been published (NUREG/CR-5545). Model development related to those phenomena encountered during late phases of severe accidents (e.g., fission product release during late phases of core degradation, re-entrainment of deposited fission products in the reactor coolant system) has been completed. For the containment, the TRENDS models have been developed to calculate the partition of iodine between the aqueous phase and the gas phase in the containment, the production of organic iodide species, containment water pool chemistry, and the extent of iodine revaporization and resuspension from containment surfaces and sumps. The models were used to calculate the revolatilization of iodine from containment water pool and the production of organic iodine in containment. The calculations were completed and the results documented in a report entitled, "Iodine Chemical Forms in LWR Severe Accidents" (NUREG/CR-5732 (Draft for Comment)). The results will be employed in a revision of the source terms delineated in the report TID-14844 (1962), which outlines a procedural method to calculate the off-site radiation dose from iodine exposure. In fiscal year 1992, the TRENDS models will be incorporated into the CONTAIN code, used for analyzing containment response to severe accident conditions.

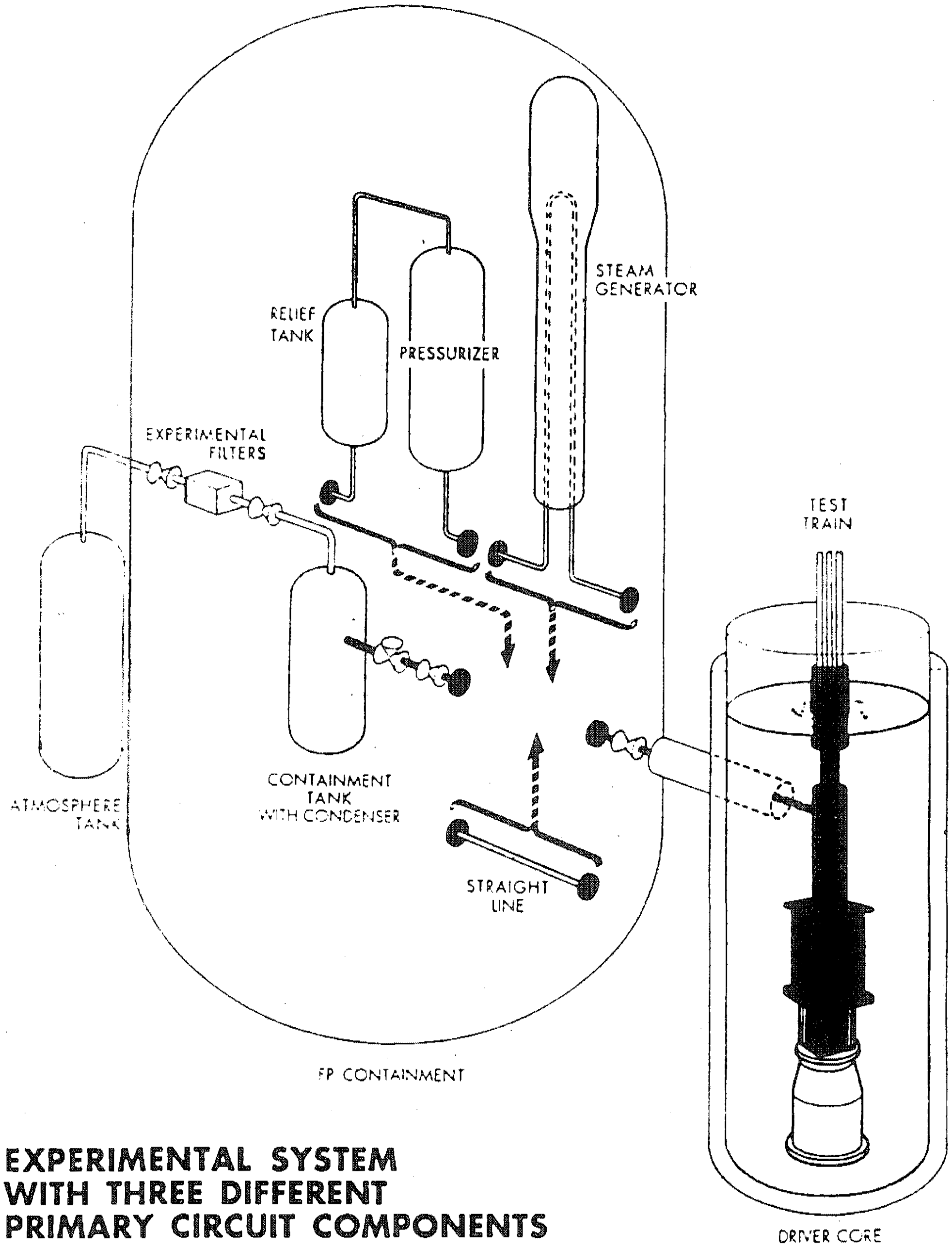
The NRC has also entered into an international agreement with the Commissariat a L'Energie Atomique of France (CEA) to participate in the PHEBUS-FP program. This program, sponsored by the CEA and the Commission of the European Communities, consists of "in-pile" severe fuel damage experiments and a study of the fission product behavior and transport in the reactor system and the containment system. The program consists of six integral tests for five different simulated severe accidents. The first test is scheduled for October 1992. 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. This information is confirmatory in nature with regard to current efforts to revise the source term assumptions now based on TID-14844 and for other aspects of the NRC's "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147).

Besides the fission product research cited above, the NRC is participating in an internationally sponsored project called Advanced Containment Experiments. The project comprises four phases: phase A deals with large scale filtration tests, using filter designs from different countries; phase B involves experiments on the physical and chemical behavior of iodine in a containment that includes the presence of hygroscopic aerosols, steam, and water pools; phase C deals with molten core-concrete interaction; and phase D deals with melt cooling issues, seeking a determination as to what debris configurations (power level and depth) can be cooled by an overlying pool. Phase C has now been completed; the seven integral core-concrete interaction tests that were conducted addressed the effects of various compositions on typical concrete substrates. Several small scoping experiments have been performed under phase D, and two integral tests are scheduled to be completed during fiscal year 1992.

REACTOR CONTAINMENT SAFETY

Core-Melt Progression

In-vessel, core-melt progression describes the state of a light-water reactor core from uncovery of the core up to reactor vessel melt-through, in "unrecovered" accidents,



EXPERIMENTAL SYSTEM WITH THREE DIFFERENT PRIMARY CIRCUIT COMPONENTS

The NRC is participating in the PHEBUS program, sponsored by the French nuclear regulatory authority and the Commission of European Communities. PHEBUS is a loop-type test reactor with a low-enriched drive core. A cluster of rods—one meter in length and in a PWR configuration—is inserted in a test train and located in the central hole of the driver core of the reactor. The test fuel is re-irradiated in the in-pile section of two weeks, using the existing pressurized water loop to generate a sufficient inventory of fission products. The loop is then slowly blown down with a simultaneous reduction of the reactor power and with the in-pile section isolated from the loop. The text phase follows, in which is the in-pile section is connected to a circuit and vessel that simulate the primary circuit and containment building of a PWR. The released fission products and aerosols are swept by a flow of steam and hydrogen into the circuit which simulates the primary cooling system up to the point of a pipe break. The flow then enters a vessel that simulates the containment building.

or through temperature stabilization, in accidents "recovered" by core reflooding. Melt progression study examines the initial conditions for assessing the loads that may threaten the integrity of the reactor containment. Significant results of melt progression are the melt mass; rate of release; composition, and temperature (superheat) of the melt released from the core, and later from the reactor vessel at melt-through. Melt progression research reveals the in-vessel hydrogen generation and the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system, and also provides the core conditions for assessing accident management strategies.

Much has been learned about the processes involved in the early phase of melt progression from integral tests in several test reactors, from tests in the German CORA ex-reactor fuel-damage test facility, and from "separate-effects" experiments on significant phenomena. Most of the available information on late-phase melt progression has come from the post-accident examination of the core of the pressurized-water reactor at the Three Mile Island Unit 2 (Pa.) plant (TMI–2). Despite the core reflooding that successfully terminated the TMI–2 accident, the general—but not necessarily detailed—late-phase melt progression phenomenology of the TMI–2 accident appears to be applicable to unrecovered as well as to recovered accidents and possibly to some boiling-water reactor (BWR) accidents as well.

Results of the integral tests and the TMI–2 core examination have provided a consistent picture of melt progression. The picture involves the development of a debris-supporting metallic blockage across the lower core during coolant boildown, from the relocation and freezing of metallic melt. The TMI–2 core examination has shown that a pool of mostly ceramic uranium-oxide fuel melt grows from decay heat in the particulate debris bed. This molten material is supported by the metallic core blockage. The growing pool melts through the blockage that surrounds the ceramic melt pool, either at the bottom or out the side of the core (as happened at TMI–2). The melt then drains into the vessel lower plenum.

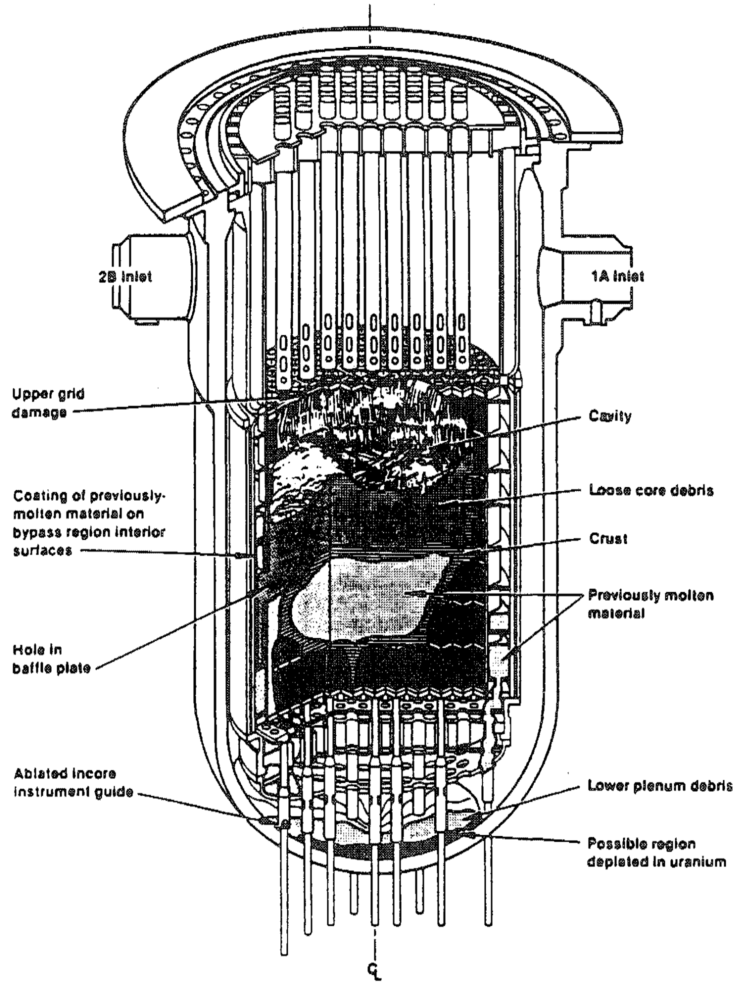
Current NRC research on melt progression is concentrated on two major issues or tasks. The first task is to determine if there are any accident conditions for BWRs in which a metallic core blockage similar to that at TMI–2 would not be formed. In the TMI–2 event, the metallic core blockage forms the lower crust that supports the resolidified pool of molten ceramic fuel in the core. 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.

A program of experiments and corollary analysis has been started on both these melt progression issues. On the issue of blockage of the core by metallic melt, TMI–2 and the results of the experiments cited above have indicated that, for "wet core" conditions (with water in the bottom of the core), the relocating molten metallic Zircaloy in the core freezes to block the lower core (as happened at TMI–2). All previous experiments, for both PWRs and BWRs were performed for these wet core conditions. The emergency operating procedures for BWRs in the United States, however, call for reactor depressurization that 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 1992.

Preparations for experiment and analyses were also begun on the process of melt-through of the pool supporting metallic and ceramic crusts by the growing ceramic (fuel) melt pool in the damaged core, for blocked-core accident sequences like TMI–2. These experiments will be performed in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories. The 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.

In 1988, the NRC—in cooperation with 10 foreign countries, under the auspices of the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD)—undertook a follow-on program to the Department of Energy's (DOE) TMI–2 evaluations. Under this program, called the TMI–2 Vessel Investigation Project (VIP), test specimens from the lower head (bottom) of the TMI–2 reactor vessel (which did not fail) were removed in 1990, and initial examinations were begun to obtain information on the molten attack on the lower head during the accident. The United States and seven foreign countries participating in the OECD/NEA project are performing metallurgical and mechanical examinations of the TMI–2 test specimens. Results of metallurgical studies

of the vessel steel samples completed in fiscal year 1991 have provided preliminary estimates of temperature histories of the lower head samples. The specimens indicated that some regions of the lower head reached temperatures during the accident that exceeded the critical transformation temperature of the steel. The VIP Management Board decided in fiscal year 1991 to extend the project until March 1993, in order to perform more



This schematic shows the final condition of the reactor core at Three Mile Island Unit 2 (Pa.), depicting the progress of the molten material toward the bottom of the reactor vessel and the molten attack on the lower head of the vessel, as occurred during the 1979 accident. The NRC has carried out extensive research on specimens obtained from the TMI-2 core.

detailed testing and examinations 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 these examinations are expected to provide 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 will then be used to perform analyses of potential reactor vessel failure modes, such as penetration tube failures and global or local failure of the reactor vessel lower head.

A major activity in fiscal year 1991 was the preparation of a comprehensive research plan for melt progression. This draft plan will be revised in fiscal year 1992 to incorporate comments from a peer review.

Natural Circulation in Severe Accidents

"Natural circulation" in severe accidents refers to the buoyancy-driven steam circulation between the reactor core and upper-plenum region of a vessel (in-vessel circulation), with or without countercurrent flows in the hot legs and steam generators (ex-vessel circulation). This kind of multi-dimensional flow may exist during the core-uncovery and core-melt periods of certain severe accidents in a PWR. If such flow should occur, it will provide a means of transferring the decay heat from the core to the upper-plenum structures, hot leg piping, and steam generator tubes. As a result, the reactor coolant system (RCS) pressure boundaries may be heated to high temperatures, which could challenge their structural integrity.

Experiments sponsored by the Electric Power Research Institute (EPRI) and the NRC at a 1/7-scale Westinghouse test facility indicated that multi-dimensional natural circulation does indeed exist under certain simulated accident conditions. Analyses using the COMMIX code (valid for intact-core geometry and single-phase flow) were compared with the Westinghouse data, and good agreement found. (For a description of calculation analyses, see the *1987 NRC Annual Report*, pp. 134 and 135.)

Recent SCDAP/RELAP5 and COMMIX analyses for the station blackout sequence in the Surry (Va.) nuclear power plant, reported in NUREG–1150, have concluded that primary system depressurization caused by creep failure of the surge line may be more likely than previously envisioned. Uncertainties related to this study have not yet been quantified but are reflected in the Revised Severe Accident Research Program Plan (NUREG–1365). Uncertainties associated with various modeling assumptions, code and user input errors, code application and limitations using the SCDAP/RELAP5 and COMMIX codes for natural circulation were assessed and reviewed. It was concluded that certain underlying modeling assumptions in the SCDAP/RELAP5 and COMMIX analyses have been shown to impose a great amount of uncertainty on the results. It appears that further evaluation and analyses are still needed. A step-by-step approach to resolve this natural circulation issue has been recommended by the reviewers.

Fuel-Coolant Interactions

An Integrated Fuel Coolant Interaction (IFCI) computer code is nearing completion at the Sandia National Laboratories. This code treats the major fuel-coolant interactions in an integrated manner. A draft documentation of the IFCI was completed in fiscal year 1990. Data review and validation of the interactive models in IFCI were continued in fiscal year 1991, with the intent of completing the code manual in fiscal year 1992.

The NRC and the Safety Technology Institute of the Joint Research Center (STI–JRC) of the Commission of the European Communities at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of fuel-coolant interaction experiments at the FARO facility in Ispra. At the STI–JRC FARO facility, large masses of real 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 are considered to be based on more prototypical conditions and will greatly enhance the existing data base in the United States. This technical exchange arrangement is of significant benefit because it enables the NRC, at a modest cost, to obtain prototypic integral experimental data to benchmark analytical models used for predictions of molten fuel-coolant interactions in containment. The information will supplement those data obtained under the NRC Cooperative Severe Accident Research Program and will provide confirmatory information to support the source term assumptions and other aspects of the "Integration Plan for Closure of Severe Accident Issues" (SECY–88–147). The first scoping test at FARO is planned for the last quarter of calendar year 1992.

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 interacts with structural concrete. The consequences of these thermal and chemical melt-concrete interactions can have a significant effect on containment loading, the modes of containment failure, and the radiological source terms. To define and gauge the threat to containment integrity and the nature of the ex-vessel releases, a number of experiments are under way, and mathematical models are being developed and assessed.

A scoping test was conducted in the MACE series of tests, a cooperative research program involving NRC, DOE, EPRI, and several foreign countries. Approximately 130 kilograms of UO2–ZrO2–Zr melt mixtures that interacted with limestone-sand concrete were flooded with water. The test resulted in a stable crust zone that inhibited continuous contact of the melt by the overlying water. A series of larger scale (50 x 50 cm) tests are planned to be performed in the MACE program. Sandia National Laboratories are also performing some oxidic material coolability tests in their WETCOR facility, within the 1991–1992 time period, under NRC sponsorship.

The CORCON code was developed as a best-estimate computational tool to calculate the physical and thermodynamic variables needed to characterize the progression of high-temperature core debris as it erodes concrete in the reactor cavity. A significant update of the code has recently been accomplished.

This work produced improved axial and radial heat transfer models, the inclusion of condensed phase chemistry (for oxide-metal reactions), improved coolant heat transfer models (including the effects of subcooling and gas injection on film boiling), the addition of models for interphase mixing and stratification, and improved models for bubble behavior (e.g., bubble size, bubble rise velocity, and void fraction). A topical report has been prepared to describe the phenomenological models and correlations incorporated in the code and to identify accepted limits of validity for the models and correlations. The code is used in research institutions throughout the world. The code was used to check analyses of core-concrete interactions involved in calculations of the failure of the BWR Mark I liner. The code was also successfully used to conduct calculations for the International Standard Problem (ISP 30) of the German Beta Test V5.1, which involved molten steel and zirconium interactions with concrete.

Large scale integral experiments with sustained induction heating were continued, in order to study the effect of overlying water pools on core debris mixtures of various compositions interacting with limestone and concrete.

The VANESA code models the physical and chemical processes that occur when gas bubbles generated by the decomposition of concrete pass through the molten debris pool and break at the surface. The WITCH tests of aerosol generation by mechanical processes and the GHOST tests of aerosol generation by vapor-condensation have been started, and those test data are used to assess the VANESA code. This code has been incorporated into the CORCON code to form a single code, CORCON MOD3, in order to improve the accuracy of code calculations and directly include the effects of vaporization on the energy balances solved in CORCON.

A number of transient phenomena that may occur in the reactor cavity during, or closely following, primary vessel failure are now under study. Experiments to study the hydrodynamic behavior of core debris are also being considered to determine the manner in which it may spread and relocate within the reactor cavity. These efforts will confirm the ability of the BWR Mark I steel drywell shell to survive a core-melt accident.

High-Pressure Melt Ejection—Direct Containment Heating

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. Left unmitigated, a molten core 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 as fine particles, thermal energy would be quickly transferred to the containment atmosphere. The metallic components of the ejected core debris could further oxidize in air or in steam, and that could generate a large quantity of chemical energy and further pressurize the containment. This process is called direct containment heating (DCH).

To help develop a data base to estimate the risk associated with high-pressure core-melt accidents, the following two activities were completed in fiscal year 1991: (1) the development of a system level scaling methodology to ensure the applicability of future integral effects tests at different scales to full size containment, and (2) the modification of the 1/10th scale facility at the Sandia National Laboratories (SNL) and the 1/40th scale facility at the Argonne National Laboratory (ANL) to conduct companion tests. In fiscal year 1992, most of the integral effects tests at SNL and ANL will be conducted. Appropriate analysis will also be performed to assess the extrapolation of computer code models crucial for DCH phenomena to full-size containment.

Hydrogen Combustion

Hydrogen combustion research seeks to assess the possible threat to containment and safety-related equipment. It is necessary to understand how hydrogen is transported and mixed within 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. During the reporting period, several hydrogen research programs were initiated.

The largest program 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). This program is to address high-temperature, hydrogen-combustion-related, high-speed combustion modes, i.e., detonations and DDT. Another joint agreement between the NRC and Germany involves a program to evaluate data from the German KfK/PHDR hydrogen behavior experiments.

A hydrogen research program has also been initiated at Rensselaer Polytechnic Institute to investigate diffusion flame behavior. And the HMS code—a three-dimensional finite difference analysis tool developed at the Los Alamos National Laboratory—is also used to provide more detailed hydrogen transport and mixing calculations. The assessment and documentation of the HMS code will continue.

CONTAINMENT STRUCTURAL INTEGRITY

Structural Tests. The major effort 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, and the other related to pre-stressed concrete containments. The current generation of Japanese PWR containments are pre-stressed concrete designs.

A reinforced concrete model was chosen for the NRC-sponsored test at SNL, since it would provide a greater challenge for analytical models. 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 thought to be 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. The test would complement the test-to-failure of a steel containment model performed by SNL in 1984, under NRC sponsorship. That model was cylindrical in cross section and was representative of PWR ice condenser and BWR Mark III containments. The proposed Japanese model would include the "knuckle regions" that are present in BWR designs in the United States. 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.

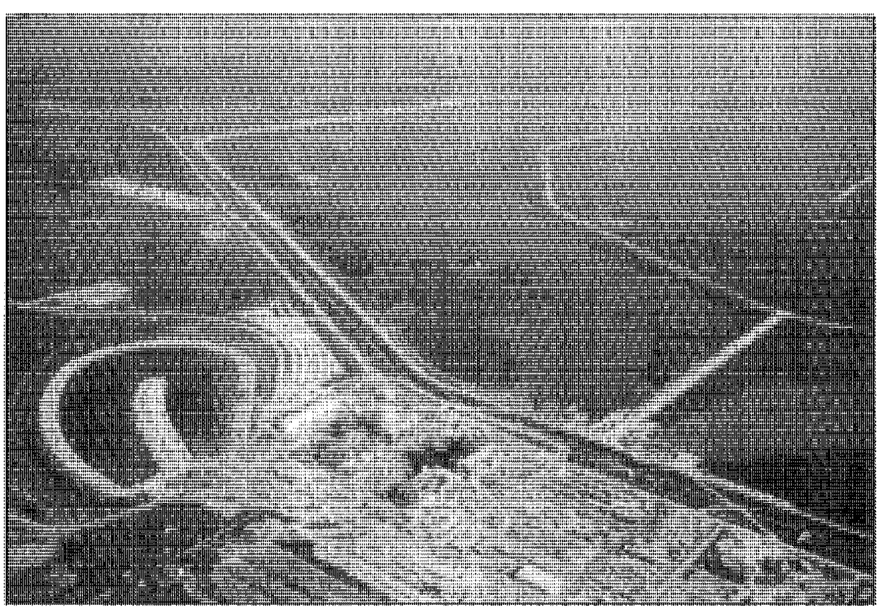
Equipment Hatch Tests. The final report for the pressure-unseating equipment hatch test program was completed in fiscal year 1991. This report, besides presenting the results of the test program, presents analytical procedures to estimate the pressure and temperature conditions at which leakage can be expected by unseating the hatch. The test program has therefore provided information useful in accident scenario analyses and in future hatch designs.

REACTOR ACCIDENT RISK ANALYSIS

Review of PRAs

Probabilistic risk analysis (PRA) is used by the NRC staff to support the resolution of a broad spectrum of regulatory issues. For licensed plants, PRAs are sometimes voluntarily submitted by licensees to support their specific proposed means for resolving such issues. For advanced plants of the future, applicants are required to perform and submit PRAs as part of their overall license applications. Reviews performed in fiscal year 1991 included the following:

South Texas. This PRA was a voluntary submittal by the licensee, who plans to use the document as a reference in future technical discussions on regulatory issues. The review was completed in fiscal year 1991.



The licensee for the South Texas nuclear facility voluntarily submitted its Probabilistic Risk Assessment (PRA) to the NRC. The Houston Lighting & Power Company's two PWRs, shown still under construction, came on line in 1988. The facility is located near Bay City, Tex., not far from the Gulf of Mexico.

Diablo Canyon (Cal.). In order to comply with a license condition, the licensee for Diablo Canyon has developed a long term seismic program. As part of this program, the licensee has performed a PRA for seismic as well as other potential accident initiators. The review was completed near the end of fiscal year 1991.

GE Advanced BWR. A PRA has been submitted as part of the licensing application for this advanced BWR. In May 1991, a draft safety evaluation report was transmitted to NRR. This was subsequently transmitted to General Electric.

EPRI Requirements Document

In support of the advanced reactor design certification process, the Electric Power Research Institute (EPRI) has developed a set of requirements to guide the design of such reactors. One part of this guidance relates to the performance and use of PRA methodologies. A review of this guidance was made with a draft evaluation transmitted to NRR in August 1991. This evaluation was reviewed by NRR and transmitted to EPRI in October 1991.

Completion and Review of Reactor Risk Reference Document

In February 1987, the NRC issued the draft version of the "Reactor Risk Reference Document" (NUREG–1150), as well as a series of supporting contractor reports, for public comment. The draft report assessed the risks from possible core damage accidents in five U.S. nuclear power plants—Surry (Va.), Zion (Ill.), Sequoyah (Tenn.), Peach Bottom (Pa.), and Grand Gulf (Miss.). The report discussed the implications of the five analyses on regulatory issues such as implementation of the Commission's Safety Goal and Severe Accident Policy Statements. Two NRC-funded reviews of the draft report were obtained and published, as NUREG/CR–5000 and NUREG/CR–5113. The American Nuclear Society sponsored and published a review of the draft report.

The NRC staff and supporting contractors updated the five risk analyses. The updates, which were quite extensive, were intended to reflect comments received, to reflect the present plant design and operating characteristics, to improve the methods used, and to incorporate new experimental data on severe accidents resulting from the research programs of NRC and others.

The newly completed version of NUREG–1150 was delivered to the Commission in April 1989 and published as a second draft for peer review in June 1989. A peer review panel, organized under the Federal Advisory Committee Act, completed its formal review of the document and provided generally positive findings. The final version of the report (NUREG–1150) was issued in December 1990.

Analysis of Low-Power And Shutdown Accident Risks

Since 1989, the staff has had under way a study of the risks associated with accidents initiated during low-power and shutdown plant operating conditions. The first phase of this analysis was completed, providing a rough categorization of potential accidents by their frequency and consequences. This information is being used by NRR in its analysis of the need for additional regulation under these operating conditions.

Computer Tools

Risk Model Development, Quality Assurance, and Maintenance. Probabilistic risk analysis has become an important tool in the NRC's assessments of safety issues in the design and operation of commercial nuclear power plants. To use this tool well, it is necessary to use state-of-the-art methods for performing and reviewing PRA and to develop, maintain, and provide quality assurance for such methods.

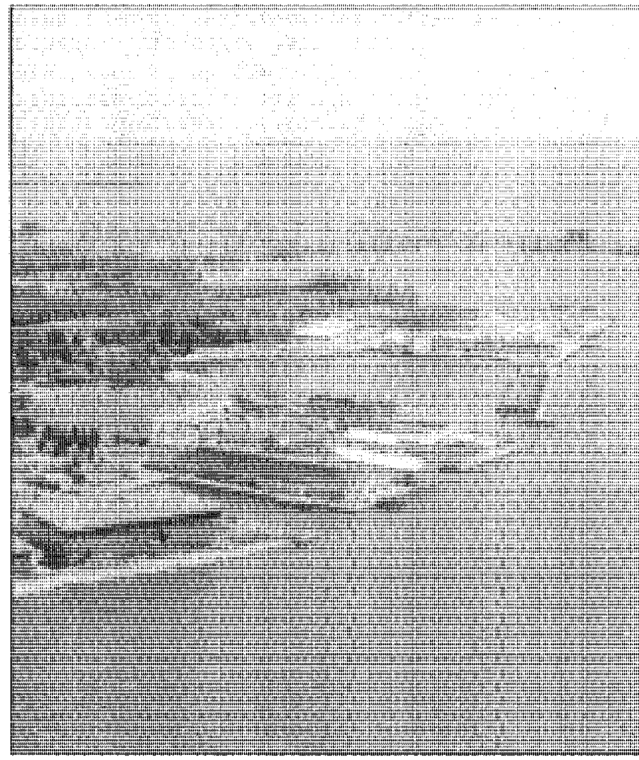
Version 1.5 of the MACCS code—a computer code that estimates the post-accident release of radioactive material to the environment and health and economic consequences to the public—was completed and made available to the public in fiscal year 1990. Final benchmarking of the code with international standard problems is under way and is expected to be completed in early fiscal year 1993.

Risk Model Applications. In regulatory decision-making, it is necessary to ask what impact a proposed modification to plant hardware or procedures will have in terms of risk. Generally, the most appropriate way to answer such a question is to examine existing PRAs, change the affected parameters, perform the analysis again, and observe the resulting change in core damage frequency and public risk. Such calculations are currently employed in setting priorities in the use of agency resources and for regulatory analyses of generic safety issues. Other uses, such as targeting inspection activity, are also emerging.

The System Analysis and Risk Assessment (SARA) system and the Integrated Reliability and Risk Assessment System (IRRAS) were conceived to address the need described above, as well as to provide the NRC with the kinds of reliability data that are currently available only on large mainframe computers. The development of high-performance microcomputers has provided greater capacities to interact with extensive data bases for a large number of users. During fiscal year 1991, versions of these codes were used by NRC contractors to perform risk studies of accidents initiated during low-power and shutdown operations (described above), and also by NRC staff, to assess such things as the sensitivity of NUREG-1150 results to variations in human error rates and motor-operated valve failure rates, and the benefit potentially achievable by the resolution of certain generic issues.

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 will 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 leak-tight. 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.



Research at the U.S. Navy's David Taylor Research Center, in Annapolis, Md., produced analyses of pressure vessel failure which contributed to development of changes in the ASME Boiler and Pressure Vessel Code. The Center is located at the mouth of the Severn River, near the U.S. Naval Academy.

REACTOR VESSEL AND PIPING INTEGRITY

Pressure Vessel Safety

The reactor pressure vessel is the crucial component of the primary pressure boundary. It houses and supports the reactor core and provides for channeling of 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 engineered safety systems cannot provide protection in case of rupture. 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. The effort addresses methods for judging the potential for vessel fracture under operating and postulated accident loads, the effects of the reactor operating environment on vessel integrity, and the mechanisms controlling vessel degradation.

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 research includes both analytical and experimental efforts. During fiscal year 1991, research continued on evaluating the validity and accuracy of reactor pressure vessel fracture analyses; evaluating the effects of parameters that affect the fracture analyses, in order to identify those warranting additional research; 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.

During fiscal year 1991, a significant effort was completed in the development of data and analyses that could be used to evaluate the potential for non-ductile failure of reactor pressure vessels. Research at the U.S. Navy's David Taylor Research Center in Annapolis, Md., and at the Oak Ridge National Laboratory (ORNL) generated independent analyses for pressure vessel fracture and for evaluating the fracture resistance of the material, based on results from small laboratory specimens. The results contributed to the development of proposed changes to Section XI of the ASME Boiler and Pressure Vessel Code—changes that are being endorsed by the staff as acceptable criteria for evaluating low upper-shelf welds.

Also taking place during the report period was the development of equations for predicting the material properties needed in pressure vessel fracture analyses. Using pattern recognition techniques, researchers at Modelling and Computing Services developed mathematical models capable of predicting the fracture toughness of pressure vessel steels and weldments, with particular emphasis on the low upper-shelf welds. This work, funded under a Small Business Innovation Research contract, provides the NRC and the industry a statistically based methodology for estimating the material properties needed in these analyses.

As the technology for predicting the fracture behavior of reactor pressure vessels has matured, the emphasis in NRC's research has moved from broad spectrum scoping research to research aimed at developing analyses and the supporting data that can eliminate some of the very conservative assumptions incorporated in the early regulatory analyses. A significant initiative in the pressure vessel research program is aimed at evaluating the apparent increase in fracture resistance for shallow flaws.

During fiscal year 1991, tests were performed to confirm pressurized thermal shock (PTS) analyses showing that shallow cracks were initiated at higher fracture toughness values than were deep cracks. The analytic findings led to an expanded program that seeks to validate and quantify fracture behavior, in an effort expected to continue for 2–to–3 years. Once completed, its results could have a major impact on pressure vessel safety assessments, significantly reducing the currently perceived risk associated with accidents such as PTS.

Work to evaluate the spatial distribution of fabrication defects was initiated during fiscal year 1991 and is expected to continue for several years. The issue has been largely overlooked in PTS analyses but has a significant impact on the results of such analyses. The success of this long term effort will be heavily dependent on the availability of the proper materials for detailed examination.

Besides the research efforts, the fiscal year 1991 program included an unusually extensive effort in support of the Office of Nuclear Reactor Regulation (NRR). Significant effort was given to performing independent analyses of the vessel failure frequency attributable to PTS transients for a particular plant. These efforts drew on expertise in probabilistic fracture mechanics, embrittlement trends, flaw size distributions, and inservice inspection techniques. While the regulatory decisions about this plant were made in NRR, the research efforts contributed substantially to the decision process. Besides the PTS analyses, other analyses were performed to evaluate alternative methods for determining pressure-temperature limits and low-temperature, over-pressure protection setpoints. The results of that work contributed directly to staff appraisals of industry proposals. The overall effort during fiscal year 1991 demonstrated that the results and expertise developed by the research program provide a valuable resource in the search for answers relevant in unusual plant-specific applications.

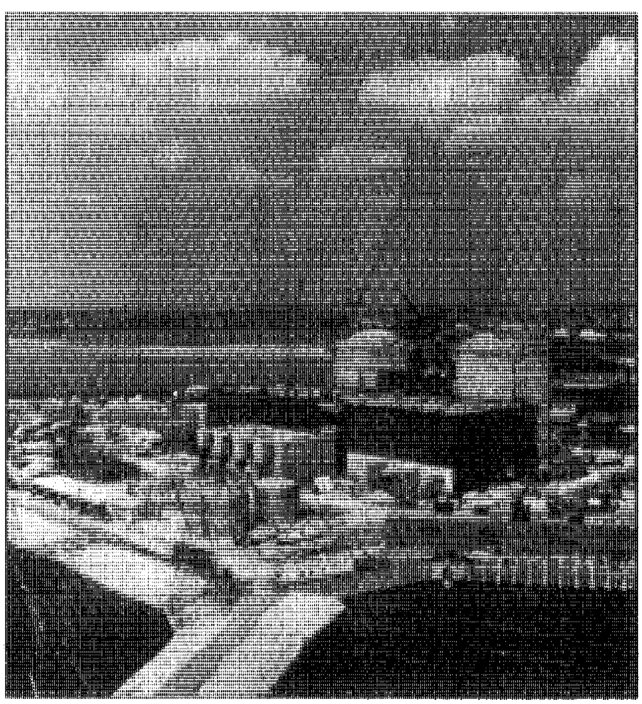
Radiation Embrittlement. It has been found that neutron radiation embrittlement of reactor vessels is higher in many plants than previously thought. The NRC's regulatory documents are being updated to reflect this reality. And research is being performed to examine the factors that control neutron radiation embrittlement and to develop additional data useful in updating the regulatory documents. As a related effort, the effects of low-temperature, low-flux irradiation on the integrity of reactor pressure vessel supports is being evaluated.

The PTS rule, 10 CFR 50.61, was amended on May 15, 1991, to make the methodology for evaluating reference temperature consistent with that in Regulatory Guide 1.99, Revision 2. Specifically, the amended rule uses the same formula used in the regulatory guide and permits the use of "credible" surveillance data in evaluating the Reference Temperature/Pressurized Thermal Shock (RTPTS) values for each plant. With the amended rule, the RTPTS value for some plants will increase, and may decrease for others. In some cases, the new value may exceed the PTS screening criterion prior to the end of the licensed life for the plant. In these cases, the licensees will be considering making the analyses and physical changes to the plant needed to satisfy the regulatory requirements.

Because the number of variables that could have a significant influence on embrittlement was so large, and the factors so inter-related, an empirical approach cannot completely resolve the issue. For that reason, emphasis has been increasingly given to study of the underlying mechanisms of neutron radiation and the resulting embrittlement. While this work will not be completed for several years, there has been significant progress in the recent past through the use of high-resolution devices, such as the "field ion atom micro-probe" and "small angle neutron scattering." This progress has improved confidence in interpreting the empirical results and in defining additional test reactor irradiation programs. An international group of experts formed to cooperate on these problems has provided valuable discussion and interaction on the subject.

The mechanisms research has made significant progress in identifying mechanisms that seem to control the embrittlement process, partially clearing the way for developing a predictive model that can replace the empirical approach currently used in evaluating irradiation damage. Production of that predictive model is the ultimate goal of this research. While the results of past research have contributed significantly to achieving it, these studies have also identified many interactions that must be understood before a comprehensive predictive model can be completed. The research has demonstrated that the dominant irradiation embrittlement mechanism for pressure vessel steels is the accelerated formation of extremely small (1–2 nanometers, or billionths-of-a-meter) copper-rich precipitate in the microstructure of the steel. Secondary microstructural changes have also been shown to contribute to the irradiation embrittlement of steels.

A comprehensive collection of radiation embrittlement data from surveillance reports and other published reports of commercial power reactors has been compiled in a computerized data base, the Power Reactor Embrittlement Data Base (PR–EDB). The data base provides the information needed to update Regulatory Guide 1.99 and supports other embrittlement research projects. The Electric Power Research Institute (EPRI), reactor vendors, utilities, and research institutions are using this data base to help solve embrittlement problems. Representatives from several foreign countries have indicated an interest in exchanging embrittlement data information and possibly establishing an international data base of this type.



Embrittlement studies have provided data related to fracture toughness, crack initiation and crack arrest which are useful in defining and promulgating industry standards in these areas. Research has begun to determine the toughness properties of reactor vessel weld metal from the canceled Midland (Mich.) plant, shown above while under construction.

The embrittlement research, coupled with the material properties research, has furnished the fracture-toughness data base used by the ASME Code Sections III and XI in developing the crack initiation and crack arrest toughness curves. These curves are used in every pressure vessel integrity analysis and are essential to ensuring safe operation of nuclear reactor pressure vessels. Recent results from test reactor irradiations suggest that the ASME Code approach to shifting the fracture toughness curves to account for irradiation damage may not completely account for such damage. It appears that the Code's procedure may under-predict the actual shift in the fracture toughness curves, eroding the anticipated margin of safety in many regulatory analyses. To assist in predicting margins of safety, research has been initiated to study the toughness properties of reactor vessel weld metal from the canceled Midland Unit 1 (Mich.) nuclear plant. During fiscal year 1991, work was started on the fabrication of test specimens and irradiation capsules, for the purpose of assessing the effects of neutron radiation on the material. Irradiation tests to be conducted in 1992–1993 should provide an enhanced basis for safety analysis of a number of operating reactor vessels having similar weld metal.

The embrittlement research 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 have been supported by industry efforts and by research performed in the former U.S.S.R. and exchanged under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety. The combined results of these efforts provide reasonable assurance that thermal annealing is a practical method for mitigating the effects of irradiation damage. Steps are planned to improve techniques for predicting annealing recovery and re-embrittlement rates. While much more work is needed to provide appropriate regulatory guidance, the underlying principle has been demonstrated.

The embrittlement validation research using decommissioned reactor pressure vessels is a relatively new undertaking. The only decommissioned pressure vessel material that has been examined to date is material obtained from the Gundremmingen plant in the Federal Republic of Germany. Several other decommissioned reactors are being considered for study.

Surveillance Dosimetry. An important aspect of the surveillance program to determine the degree of embrittlement in the pressure vessel of an operating nuclear power plant is the prediction of the amount of neutron radiation exposure (neutron fluence) of the vessel. Fluence determinations are made by calculations of the fluence, dosimetry measurements at key surveillance locations, and a consolidation of the measurements and calculations to reduce uncertainties of predictions at critical locations of the vessel. These predictions must be reasonably accurate to ensure that the plant is operating in conformance with NRC safety regulations.

Dosimetry research has led to new values for the cross-section for inelastic scattering of iron atoms that have been included in the Evaluated Nuclear Data Files. Reevaluation of experiments using these data files has resulted in significant improvements in the correlation between calculations and measurements for predicting reactor vessel fluences. Finally, dosimetry research has provided the basis for improvements in dosimetry measurements, including ex-vessel dosimetry measurements. The work will culminate in a regulatory guide on dosimetry expected to be published in fiscal year 1992.

Steam Generator Integrity

Results from a recently completed NRC research program on the reliability of eddy current (ET) inspection techniques to detect and characterize steam generator tube degradation have indicated a need for improvements in the ET inspection process. To address this need, the NRC is funding research at the Pacific Northwest Laboratory to develop ET performance demonstration qualification requirements.

In fiscal year 1991, work focused on participation in the ASME Section XI Special Working Group on ET Examination (SWGET) for development of generic performance demonstration qualification requirements. Toward the end of fiscal year 1990, a draft appendix was prepared and submitted to the SWGET for consideration during fiscal year 1991. An important addition was the adoption of regression methods for grading ET system performance on probability of detection tests.

The NRC has also been active in the international program for the Inspection of Steel Components (PISC). A major task of the program is to conduct an international round-robin on the effectiveness of steam generator tube inspection techniques in which seven U.S. teams are involved.

Piping Integrity

Environmentally Assisted Cracking and Degradation. "Residual life" assessment reviews (RLA reviews, see discussion under "Aging of Reactor Components," later in this chapter) for light-water reactors (LWRs) indicate that low-cycle fatigue is a potentially significant degradation mechanism in LWR primary piping. Current fatigue design for austenitic stainless steel piping is based on the ASME Section III fatigue design curves. These curves give the design life in terms of the number of cycles at a given stress (or strain) level. The design curves are obtained by adding a safety factor—which accounts for frequency, environment, temperature, surface finish, heat-to-heat variation, etc.—to a mean data curve. The objective of the current work is to provide better information on the effects of operating temperature and environment on the fatigue behavior of austenitic piping steels, in particular the Type 316NG stainless steel that has been used for replacement piping in boiling-water reactors (BWRs) affected by intergranular stress corrosion cracking.

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 ferritic material is exposed to the reactor coolant. During fiscal year 1991, testing was continued to ensure that the existing predictive equations, which are based on test data from unclad materials, adequately predict crack growth rates in clad materials. Initial results suggest that the crack growth rates are higher in the clad materials, but that the differences are relatively small. The work continues in order to better quantify the effects on the cladding.

Current procedures for estimating fatigue life are based on the ASME Code Section III and its fatigue design curves. These curves, developed about 20 years ago, were obtained by adding a safety factor to a mean data curve that was based on tests of smooth, polished specimens tested in a room-temperature air environment. The safety factor was intended to account for several effects, including the effect of loading rate, the effect of the water coolant environment, the effect of operating temperature, the effect of surface roughness, and normal material variability.

Based on results obtained during fiscal year 1991, as well as results obtained from earlier work in the United States and abroad, it is now clear that the margins in the Code are smaller than intended for some situations. Since no consensus on fatigue life estimation procedures is available, data from ongoing tests and from the literature and programs in Europe and Japan are being evaluated to develop interim procedures that adequately account for the effects of operating temperature and the water coolant on fatigue life.

Cast duplex austenitic-ferritic stainless steels are used extensively in the nuclear industry in pump casings and valve bodies, and in primary coolant piping in pressurized-water reactors (PWRs). Recent investigations suggest that embrittlement of the ferrite phase in these steels may occur after 10–to–20 years at reactor operating temperatures. This could potentially affect the structural integrity of pressure boundary components during high strain-rate loading (e.g., seismic events). The potential concern is greatest in PWRs, where slightly higher temperatures are typical and cast stainless steel piping is widely used.

Research on this subject has been ongoing since 1982. During fiscal year 1991, procedures and correlations for estimating fracture toughness and tensile properties of these materials have been validated with experimental data from materials removed from a decommissioned nuclear power plant and from materials subjected to accelerated aging in the laboratory. Conservative estimates of fracture toughness can be made for cast stainless steels of unknown chemical composition; progressively more accurate estimates can be made based on the information that is known about the material. These procedures and correlations provide an experimentally validated engineering tool that can be used to estimate the service-induced degradation in properties for cast stainless steels. The research is continuing to examine potential degradation in properties for stainless steel welds.

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 leak-or-break alternatives have 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 broader questions regarding "leak-before-break" phenomena for all piping.

The NRC has funded research into several aspects of pipe fracture, including analysis of material properties and full scale pipe fracture experiments. The NRC's primary piping fracture research program had been the Degraded Piping Program, conducted by Battelle's Columbus (Ohio) Division. This program, initiated in 1984, was completed in 1988, and the final report was issued in 1989. The Degraded Piping Program has, among its many contributions to an understanding of piping fracture technology, identified several areas that call for deeper study. Three particularly important areas are the effects of anisotropic material properties, the effects of short cracks, and the effects of seismic or dynamic loading.

During fiscal year 1991, 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 piping that contains flaws 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 will provide experimental data for validating and improving pipe fracture analysis methods. Other efforts in this study examine the fracture behavior of bi-metallic welds and the significance of material property variability. The study is expected to continue through fiscal year 1995.

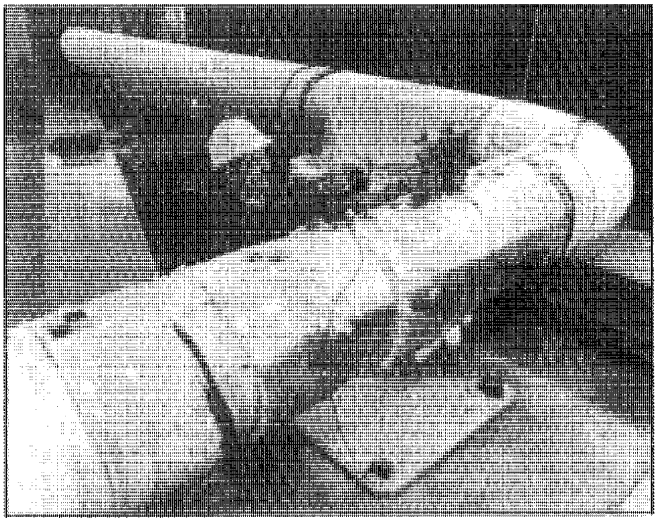
During fiscal year 1991, the NRC completed the first International Piping Integrity Research Group (IPIRG–1) program to evaluate the effects of seismic and dynamic loading and other piping integrity issues. The research group is a consortium of nine governmental and industrial organizations that are jointly funding this research. The work involved performing fracture experiments on a typical piping loop, made with 16-inch-diameter pipe that was one-inch thick. Intentionally cracked test sections were welded into the loop at a high stress location. The tests were performed at typical PWR pressures and temperatures (2,250 pounds-per-square-inch and 550F); the loading was intended to simulate seismic events.

In general, the results support the NRC's pipe fracture analysis approach used in leak-before-break analyses and the fracture analysis approach used by Section XI of the ASME Boiler and Pressure Vessel Code in developing flaw evaluation procedures. However, issues were identified that warrant further study.

The success of the IPIRG–1 program, and the progress made by the IPIRG participants toward an international consensus on the pipe fracture technology, led the participants to form a second jointly funded program, the IPIRG–2 program. That work will consider more representative seismic loading histories and will include short cracks and cracks in fittings. The program will be completed in approximately three years.

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, wrought and cast



A pipe system test facility, designed and built by Battelle Laboratory's Columbus (Ohio) Division, was the site of fracture experiments on a typical piping loop, with a pipe one-inch thick and 16 inches in diameter. At left, one of a number of test sections has been deliberately cracked and welded into the loop at a high stress location. On the right, the piping is subjected to typical pressurized water reactor pressures and temperatures (2,250 pounds-per-square-inch and 550xF). The research was conducted under the International Piping Integrity Research Group program.

stainless steel piping, and pressure vessels. It also 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. In December 1990, the SAFT system was used to inspect a reactor pressure vessel, as part of an international study assessing the effectiveness of advanced ultrasonic technologies. Results showing the accuracy of the SAFT–UT inspection are not available because these "blind test" round-robins have not yet been completed and evaluated. In fiscal year 1991, the SAFT technology was transferred to General Electric for incorporation into their next generation reactor pressure vessel inspection system. Discussions were also held with other major nuclear industry vendors concerning the transfer of the SAFT technology to them.

Inservice Inspection System Qualification. Research that included both national and international studies and field experience over the last several years has shown that inservice inspection, as currently practiced, is not sufficiently reliable or effective. NRC research indicates a need for qualification of the entire inservice inspection (ISI) process—including personnel, procedures and equipment—as described in the *1987 NRC Annual Report,* pp. 115 and 116.

With the acceptance by Section XI of the ASME Code of mandatory appendices on personnel training and qualification and on criteria for performance demonstration, the NRC and its research contractor have been working with industry to review and discuss industry's plans to implement these appendices. Close coordination is being maintained with the industry group Performance Demonstration Initiative (PDI) through the Nuclear Utility Management and Resources Council (NUMARC) to monitor progress and critique plans.

Other work in progress is concerned with assessing the overall effectiveness of current code requirements for ISI in order to ensure operational safety of the reactors. Theoretical modeling and empirical testing conducted from 1988-to-1991 will form the technical basis for the new criteria for overcoming identified shortcomings. A major improvement is expected in conjunction with an ASME research task force to develop ISI guidelines based on risk, with a systematic setting of priorities among reactor components.

Additional code requirements were prepared and submitted to the ASME Section V Subcommittee to fulfill a need for code rules to cover computerized UT imaging systems that are being used by the NDE/ISI industry for examining important nuclear power plant components. The ASME Section XI rules and procedures for flaw evaluation in reactor pressure vessel welds were examined under a new initiative to determine if these criteria adequately consider the impact of degraded material properties.

Continuous Monitoring for Crack Growth and Leak Detection. NRC-funded research has produced technology in support of the application of continuous acoustic emission (AE) monitoring to detect the initiation and growth of cracks in nuclear reactor components as they might occur during reactor operation. The same technology also provides a very sensitive coolant leak detection capability, developed under NRC sponsorship, at the Argonne National Laboratory. The background of this effort was described in the *1990 NRC Annual Report,* p. 141, and results from the research are documented in the NRC report NUREG/CR-5645. The benefits expected include increased safety through detection and evaluation of crack growth as it occurs, improved capability to detect and locate coolant leaks as they initiate, and reduced personnel exposure to radiation through reduced need for manual inspection of reactor components.

The program has produced AE monitoring technology and methodology proven in off-reactor tests and application guidance in ASTM Standard E 1139 and ASME Code Case N-471; field validation currently is being cooperatively performed in an application with Philadelphia Electric Company (PECO) at the Limerick Unit 1 (Pa.) power reactor.

As described in the *1990 NRC Annual Report,* p. 141, AE monitoring is being applied to detect growth in a flaw indication identified in 1989 at Limerick Unit 1 (Pa.), during normal ultrasonic testing (UT) prior to refueling the reactor. Safety analysis indicated that the flaw could remain in place during another fuel cycle without compromising safety. The licensee for the facility, however, elected to apply AE monitoring on a test basis and a crack-arrest-verification (CAV) specimen technique, to give added assurance that the crack would not grow during operation without detection. AE monitoring at Limerick Unit 1 during the fuel cycle following detection (May 1989 to September 1990) has been completed, and the results have been analyzed. A relationship developed earlier in the AE program to relate AE data to crack growth rate was used to interpret the AE data in terms of estimated crack growth. There was partial correlation between the crack growth indicated by AE and that indicated by follow-up UT performed at the end of the fuel cycle, but the AE also indicated crack growth in locations not indicated by the UT. That finding is not necessarily inconsistent, when examined in light of the nominal detection threshold of about 20 percent of wall thickness for UT. The crack growth indicated by AE was small in most locations and inspection of the weld by UT is particularly difficult in this case, because of the geometry of the weld. Comparing the three surveillance methods on a common basis of *maximum crack growing rate-per-year,* UT and AE agreed within about 25 percent, while the CAV prediction was about an order-of-magnitude lower.

AE monitoring of the weld at Limerick Unit 1 has been continued for a second fuel cycle to maximize the reliability of the field validation. This activity started in December 1990 and will be completed about May 1992. The final factual evaluation of the AE monitoring results will be achieved by destructive examination of the weld.

International Reliability Studies. The NRC has been a leader in an international program called the Programme for the Inspection of Steel Components (PISC) that is assessing the effectiveness of technologies and procedures for the inservice inspection of nuclear reactor components. The output from this program will aid regulators and code bodies in establishing technical bases for improving inspection requirements. The NRC has been in a leadership role in developing the PISC program objectives. And the NRC has supported the Pacific Northwest Laboratory in the design of the PISC studies, fabrication of flawed specimen, implementation of the testing, and data analysis of the comprehensive data bases. Specific PISC studies include the addressing of human influence on inspection reliability, pressure vessel inspection capability using SAFT-UT, inspection of stainless steel piping, and inspection of steam generator tubing mockups and nozzles and dissimilar metal welds. All the studies are in full progress with results starting to become available. The PISC program is to be completed at the end of 1993 and the results, analysis, and interpretation from all the studies will begin to be released to the participants over the next two years. As these results become available, they will be used to develop and support new inspection requirements.

Support to NRC Regulatory Oversight. NRC research helps NRC regional and headquarters staff meet their oversight responsibilities by assisting in training the staff in understanding the new and developing technologies that are being applied to inservice inspection. During the past year, these efforts included selecting various computer-based ultrasonic inspection systems for detailed review and evaluation; conducting a seminar for NRC staff, entitled "An Introduction to Computer-Based Inservice Inspection"; developing guidelines for reviewing ultrasonic field procedures; and constructing a steam generator tube bundle mockup. A draft report was prepared describing the general functions of computer-based ultrasonic equipment and provided a review of selected systems.

The design for the steam generator tube bundle mockup was completed during fiscal year 1991. Fabrication was started on mockup structural elements, as well as on wastage and fatigue-crack-degraded tube samples. Additional efforts are under way to contract for fabrication of chemically degraded tube samples, including intergranular attack, stress corrosion cracking, and pitting, and for characterization of cracked tube samples by computed tomography.

AGING OF REACTOR COMPONENTS

Aging Research

Aging is a key concern with currently operating plants and is clearly a critical consideration in 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-degraded 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 degradation processes—such 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 and structures.

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 the information and the 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 1991, the NPAR consisted of 16 separate but related projects concerned with the study of the effects of aging on 23 individual mechanical and electrical components and on 17 systems comprising such components.

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.-U.S.S.R. 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 1991, phase 1 aging assessments were completed on the following special topics and safety-related components and systems:

(1) Instrument and Control Systems

(2) Reactor Core Internals

(3) Control Rod Drive System (PWR)

(4) Reviews of Industry Reports

(5) Degradation Modeling of Component Aging.

Reports were issued on the above-mentioned Phase 1 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 the Equipment Qualification Advisory Group of EPRI and by 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) Solenoid Operating Valves

(2) Auxiliary Feedwater System

(3) Auxiliary Feedwater Pumps

(4) Diesel Generators

(5) Circuit Breakers and Relays

(6) Reactor Protection System.

Residual Life Assessment of Major LWR Components. Intrinsic to the general exploration of reactor aging is the residual life assessment (RLA) of major components and structures. The objective of the RLA, as an element of the NPAR program, is to develop technical bases and criteria by which to assess methods for mitigating the effects of aging on major components and structures when considering possible license renewal. The approach is to gauge the degradation of the major LWR components and structures by the synergistic influences of radiation embrittlement thermal fatigue, corrosion fatigue, environmental attack, metallurgical changes, microbiologically and otherwise induced corrosion, moisture intrusion, erosion, and so forth.

Research completed in this area in 1991 focused on developing models and procedures for estimating aging damage in specific LWR components, to ensure continued safe operation. The studies included the evaluation of advanced inspection, surveillance and monitoring methods for characterizing the aging damage. The results will be useful to the NRC licensing process by establishing policies and guidelines for making license renewal decisions. The components assessed or being assessed are LWR reinforced-concrete containments, PWR pressure vessels, LWR metal containments, PWR steam generator tubes, and cast stainless steel components. Results with respect to PWR pressure vessels, LWR metal containment, PWR steam generators, tubes, and cast stainless steel components are documented in NUREG/CR-5314, Volume 1 (draft), Volume 5 (draft), and Volumes 3 and 4, respectively.

Technical Bases for License Renewal. A rulemaking process to formulate a license renewal rule is under way and is expected to lead to a technical and procedural rulemaking by early fiscal year 1992. Besides a final rule, more detailed regulatory guidance addressing the technical safety issues related to aging is needed, both to implement the rule and to advise licensees on license renewal application requirements. An interim guidance document is expected to be completed by the middle of fiscal year 1992.

A draft regulatory guide, setting forth the standard format and technical content required in applications to renew nuclear power plant operating licenses, was issued for public comment. The regulatory guide is being revised to reflect the changes in the final rule (10 CFR Part 54) and to accommodate public comments. The purpose of the regulatory guide is to establish a uniform format and content acceptable to the NRC staff for structuring and presenting the technical information to be compiled by an applicant for a renewed nuclear power plant operating license and submitted by the applicant as part of an application for a renewed license. The regulatory guide identifies the content of, and provides technical criteria for, the compiled technical information.

PRA-Based Priorities Among Risk Contributions and Maintenance. A second report (revision to NUREG/CR-5587) was issued on PRA-based priorities 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 consistent with procedures for setting priorities. The second report also describes various approaches for transforming a baseline PRA into an age-dependent PRA, and it gives answers to questions that are likely to arise when applying an aging PRA. The report also incorporates work developed under a different task for including the effects of aging on passive components in the baseline PRA. This latter work is described in more detail under the section entitled "Aging of Passive Components," below.

Aging of Passive Components. A report (draft NUREG/CR-5730) was issued on the development of a methodology to include the effects of aging on passive components (pipes, structures, 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 when subjected to the stresses caused by the loadings on the components. The failure calculation can be substituted into a PRA for the plant that will calculate the effects of this failure on plant risk. The method is demonstrated in the report for application to a pipe weld in which a crack occurs. When the crack grows because of the pipe loadings, it will eventually reach a depth that is unsatisfactory for the assurance of safety. At that point, the pipe has failed and the effect on plant risk is determined. A computer program has been developed to aid in making these calculations.

A procedure was also developed for identifying those aged passive components having the most impact on plant risk. Effective inspection and maintenance actions can be taken on these components that will control the effects of the aging and reduce the risk.

Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components. Eight selected regulatory instruments, e.g., NRC regulatory guides and the *Code of Federal Regulations,* were reviewed for safety-related information on three additional major LWR components: cables, containment, and basemat. 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 safety-related regulatory instruments do provide implicit guidance for aging management, but that there is room for improvement with explicit guidance. A draft NUREG/CR report was prepared and was under review at the close of the report period.

Inspection Integration. The NPAR program has the potential to support the ongoing inspection effort conducted by the Regions in accordance with the NRC inspection program. One objective of the inspection effort is to ensure that safety systems and safety-related components have not been measurably degraded as a result of any cause, including aging.

A review of NRC inspection procedures suggests that the information requirements of inspectors are vast; the NPAR data base can assist the inspectors in focusing their attention on those components and systems most likely to affect the plant safety as the plant ages. And the NPAR-developed data and research results can provide the inspector with criteria for judging the validity of findings and the completeness of the licensee's responses.

In the light of the information needs identified by the inspectors, NPAR reports for selected components and systems were reviewed, and information of potential use to NRC inspection activity was excerpted and published in two documents—an "aging report summary" and an "aging inspection guide." The summary for each equipment type and system studied in the NPAR program includes identification of aging-related problems, highlights of the operating experience, solutions to aging problems, and references likely to be available to the inspectors. The aging inspection guide is the more concise document, focused on the specific inspections to be considered when assessing the operational readiness of the component or system. The guide also contains visual inspection techniques for detecting aging degradation, including external and internal indicators, and important operating parameters.

Degradation Modeling of Components. Efforts related to developing component degradation modeling approaches in the study of aging and maintenance effects on components are continuing. Application of degradation modeling approaches to residual heat removal (RHR) pumps, service water pumps, and air compressors have demonstrated that inclusion of degradation states in reliability modeling provides a better understanding of aging and maintenance effects. The model provides a quantitative means of characterizing aging effects, evaluating maintenance effectiveness, and assessing component reliability. The model being developed is applicable to both standby and continuously operating components.

Since age-related failures generally include a degradation phase, the degradation rate serves as a precursor of the failure rate. Increasing aging trends in the degradation rate can signal future increasing aging trends in the failure rate. In the case of compressors, the failure rate, which is significantly lower than the degradation rate in the first three years, increases faster in the later years, reaching approximately the same values as the degradation rate at the end of the 10 years of operation. This behavior indicates the ineffectiveness of maintenance in preventing degradation from leading to failure, as the air compressors age. Another finding related to RHR pumps indicates a lag time of two years for degradation to affect failure occurrence.

The model is being extended to explicitly show the reliability effects of different maintenance and test intervals, different maintenance and test efficiencies, and different repair times. Further developments will include time-dependent "Markov approaches," multiple degradation studies to model progression of degradation, sub-component level modeling, and potential for using the degradation modeling approaches of PRA models of the plant to determine the core damage frequency.

Components, Systems, and Facilities

Component Cooling Water Systems. The component cooling water (CCW) system has been identified as one of the support systems that is important to plant safety. An increase in CCW unavailability can adversely impact plant risk (as discussed in studies such as NUREG-1150 and the TIRGALEX study). NRC Generic Safety Issue 65 relates to the high probability of core melt from CCW system failure. Because of its importance, an aging assessment of this system was completed in fiscal year 1991. The objectives were to identify and characterize the aging degradation mechanisms relevant to the system, to assess their impact on system unavailability, and to provide recommendations on the available methods for the detection and mitigation of aging in the CCW system. Aging degradation contributes to over 70 percent of the failures, with the most common aging mechanism being "wear." Fifty percent of the failures resulted in degraded performance of the system, while 27 percent caused a loss of redundancy. Component failure rates can increase with time because of the effects of aging, which can lead to an increase in system unavailability as the plants age. To prevent this, good detection and mitigation practices are required. Basic inspection, surveillance, monitoring, and maintenance (ISM&M) practices are good, but they are not comprehensive enough to completely manage the effects of aging. These basic methods include ways to detect incipient aging degradation before failures occur, as well as maintenance practices to mitigate the effects of aging. But they do not typically address all aging mechanisms. Supplemental measures are available that can improve the system reliability. These were identified based on manufacturers' recommendations or past plant experience. Some of the supplemental steps include thermography examination of pumps to detect hot spots and eddy current testing of heat exchanger tubes to detect cracks or flaws. Each of the supplemental actions was correlated with the aging mechanism it helps to detect or mitigate. The study recommends that various supplemental practices be added to the basic practices to formulate an effective ISM&M program for detecting and mitigating aging.

Control Rod Drive Systems for PWR Plants. The PWR control rod drive (CRD) system positions the control rods within the core to control reactivity changes encountered during operation and to provide a sufficient source of negative reactivity to ensure a rapid reactor shutdown. The aging study of this system examines the design, construction, operation and maintenance of the system to assess its potential for degradation as the plant ages. The extent to which aging can affect the safety objectives of the system is also included in this study. One such consideration includes component failures in the CRD system resulting in plant transients, which unnecessarily challenge other safety systems. The Westinghouse and Combustion Engineering (CE) plants use similar "magnetic jack" mechanisms, with the exception of Palisades (Mich.) and Fort Calhoun (Neb.), which use a rack-and-pinion drive mechanism. Babcock & Wilcox (B&W) plants use a "roller nut" type of mechanism. The magnetic jack and roller nut mechanisms are actuated by externally mounted stator coils, while an electric motor is used to drive the rack-and-pinion mechanism. All the mechanisms use similar magnetically actuated reed switches to provide actual rod position indication. Forced air cooling systems are used by Westinghouse and CE, while B&W uses a water-cooled system. Inspections of design differences, consideration of certain design modifications and improved maintenance techniques are being provided as they become evident from operating experience. Some of the most significant problems identified for the Westinghouse CRD include unexpected wear of control rod cladding surfaces, the susceptibility of certain cast drive mechanism pressure housings to leakage because of embrittlement, the vulnerability of electronic components to elevated ambient temperatures, corrosion and wear of operating coil stack connectors, and a potential generic concern related to inaccurate control rod position information. With regard to CE and B&W designs, corrosion associated with primary coolant leakage (seal degradation, housing cracks and vent valve leaks), and failures of power and control system components (power supplies) were the most prevalent aging degradation mechanisms. Because of the inherent design features in the system and careful maintenance performed by the utilities, this system has not exhibited any system failure as a result of component degradation.

A wide variation exists between utilities' preventive and predictive maintenance programs, which has impacted their ability to identify and mitigate aging. This study has recommended an increased emphasis on inspections and root cause analysis. Some of these activities include the use of advanced monitoring techniques—such as infrared thermography for electronic components, motor current signature analysis to detect proper control rod operation, and electronic characterization and diagnostics—as possible alternatives for assessing electrical integrity.

Control Rod Drive Systems for BWR Plants. A project was undertaken by the Oak Ridge National Laboratory (ORNL), under the aging research program, to collect and evaluate data on the past performance failure mechanisms and aging of BWR control rod drive (CRD) systems. A workshop was organized and attended by BWR utility personnel to review and collect information on CRD maintenance. A substantial number of system problems were found to result from the failure of hydraulic control unit components including the accumulator and various scram valves. The leading causes of CRD mechanism degradation was found to be embrittlement and fatigue fracture of the graphiton seals. Recommendations for improved maintenance practices were provided to minimize some of the failures observed. (The results are documented in NUREG/CR-5699.)

Heat Exchanger. Heat exchangers are vital components of nuclear power plants, serving as interfaces between both safety-related and non-safety-related systems and components, and as the ultimate heat sink to provide for safe operation in the event of a plant-transient or accident, as well as to mitigate the effects thereof. A review of nuclear plant operating experience by ORNL 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 air coolers or from gasket failures, accounted for about 35 percent of the total. In most cases, inter-fluid or external leakage is more of a nuisance than 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. These types of problems may not be readily recognized because the exchangers 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—to which the study provides data—should provide definitive guidance in detecting degraded capability.

Friction in Motor-Operated Valves. A report on the effects of aging on internal valve components was issued during the report period (draft NUREG/CR- 5735). The intent of this work is to determine whether the moving internal valve components can be affected by corrosion buildup brought about by fluid conditions during normal plant operations. In the valve experiments reported under "Reactor Equipment Qualification," it was demonstrated that these internal friction forces are under-predicted. To assure that the valves are able to operate as they should, it is necessary that the effects of friction on them be better understood. The information contained in the draft report cited above identifies the main aging mechanisms (corrosion, deposition, and erosion) that can influence the friction values. The follow-up results of nuclear plant reactor trips, where valves were disassembled for inspection, are also reported and show that stainless steel valves are likely to be less affected than either carbon or low alloy valves. Although only a small number of valves were observed, this important work is being supplemented with friction experiments to account for other metal-fluid environment interactions.

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. Accelerated aging and accident survival tests of cable products have been completed at the SNL Low Intensity Cobalt Array (LICA) facility during which cables were aged to the equivalent of 20, 40 and 60 years of operation. During the aging process, the condition of the cables was monitored, using both electrical and mechanical measurements—including insulation resistance, polarization index at three different voltages, capacitance and dissipation factors over a range of frequencies, elongation profiling, cable indenter modulus measurements, and hardness and density measurements. The most effective of the condition monitoring methods was elongation-at-break. Hardness, indenter modulus, and density also correlated with aging for some cable insulation and jacket materials. Neither tensile strength nor any of the electrical measurements exhibited a consistent trend with aging. Most of the cables were found to be functional throughout the 60-year aging and the loss-of-coolant-accident tests that followed aging.

Snubbers. The research results provide information relevant to recent operating experience for both hydraulic and mechanical snubbers, particularly in regard to age-related influences. Methods were identified that are useful in monitoring the service life of snubbers. Recommendations are being developed for the Subsection ISTD of the ASME-O&M Code. The principal findings of this research are:

- The primary environments that contribute to aging degradation in snubbers are temperature, vibration, moisture, and dynamic transients.
- Based on the eight nuclear power plants investigated, approximately 47 percent were mechanical functional test failures and 52 percent of test failures were service-related.
- Hydraulic snubber seal life is primarily a function of operating temperature. Seal life limits originally proposed by snubber manufacturers are generally conservative.

Service Water Systems. The objective of the service water system aging study was to identify and characterize the principal aging degradation mechanisms relevant to the service water system, to assess their impact on operational readiness, and to provide a methodology for the mitigation of aging in the service water system. The following regulatory applications evolved from the aging assessment of the service water system:

- Technical basis for the implementation of Generic Letter 89–13.
- Support for NRR in the modification of the standard technical specifications addressing service water systems.
- The development of a draft research information letter on service water systems.

To satisfy the need for a formal procedure to identify the cause of age-related degradation of service water systems, a root-cause method of analysis was developed. A positive outgrowth of the service water system aging assessment was the transfer of the methodology related to root cause analysis and to the use of artificial intelligence to a Department of Defense facility.

Low Flow Operation of Safety-Related Pumps. Bulletin 88–04 was issued by the NRC requiring utilities to examine their safety-related pump operation to determine the potential for dead-heading pumps in parallel operation and the adequacy of the minimum flow rate. ORNL evaluated industry responses under the aging research program and made several site visits to review the detailed calculations of the utilities. It was found that low flow operation can degrade pumps, and that there are no generic guidelines for determining acceptable pump operation in all modes. The minimum low flow was found to be inadequately addressed at some nuclear plants. It was recommended that pump qualification criteria and new diagnostic techniques providing more meaningful information on pump degradation be developed. Parallel pump dead-heading problems were identified in the residual heat removal systems at some plants, where the pump discharge miniflow line originates downstream of the pump discharge check valve. Results and recommendations are provided in NUREG/CR-5706.

Fire Safety. The NRC is currently sponsoring a research program at SNL on the "Vulnerability of Aged Electrical Components to Fire" and is also participating in a large scale cable fire test program, sponsored by Germany at the HDR reactor facility located in Kahl, Germany. Two separate studies on the impact of aging on the performance of electric cables in a fire were completed at SNL during 1991. The effect of cable thermal aging on material flammability and cable vulnerability to fire-induced electric failure were examined. It was found that cable material flammability was significantly reduced as a result of aging. Cable thermal vulnerability to fire-induced electric failure was only slightly increased by thermal aging. Thus, cable aging does not appear to significantly increase fire risk.

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 to be run in December 1991, involving a large scale cable tray fire in a lower elevation room in the containment building. The NRC is providing 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. Activity under this program also includes participation in an international fire computer code validation comparison, using the HDR fire test data. The fire computer code models to be evaluated include those frequently used in the fire risk assessment for U.S. nuclear power plants, such as COMPBRN.

Structural Components. During fiscal year 1991, three reports were generated by the structural aging research program. The first, an annual report, updated the progress and status of the overall program, as well as giving projections and details of work still to be done. The second report was a single-volume sample of a future four-volume set of data on how structural materials change as they age. (These data will eventually also be accessible electronically via personal computers.) The third report used three different, but typical, U.S. nuclear power plant types to develop an aging assessment methodology. The methodology uses relative weighting factors to rank concrete structures in nuclear power plants by the importance of their structural elements, safety significance, environmental exposure, and the influence of degradation factors.

Mechanisms for strength degradation attributable to corrosion of steel reinforcement and the detensioning of pre-stressing tendons were incorporated into the reliability analyses being developed for reinforced concrete structures. A considerable amount of concrete aging data was acquired for input to the structural materials aging data base; 13 formal technical presentations were also given.

REACTOR EQUIPMENT QUALIFICATION

Experiments were completed in early fiscal year 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, not only because of steam release outside containment, but also because other emergency equipment may be exposed to the harsh water and steam environment and may fail.

A total of six different valves were tested, three having six-inch and three with 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 plant.

In the *1990 NRC Annual Report,* findings from the valve experiments were disclosed. The main findings were that one six-inch valve did not fully close because it had undergone significant damage to internal parts during closure. The five other valves were capable of stopping the high-velocity flows in all of the closing experiments. However, one 10-inch valve also underwent significant damage to its internal parts, although the damage did not prevent the valve from closing. (It should be recalled that all valve actuators that provide power to close the valves had been set to deliver larger thrusts than would normally be the case for in-plant operation to ensure closure during the experiments.) Other important findings from the experiments that were reported last year showed that the valve internal friction forces that must be overcome by the actuators are under-predicted. This latter finding leads to under-predicting the required closing thrusts and ultimately may lead to under-sizing actuators for these valve applications. All of the above findings and all of the test results have been made available to the nuclear industry for use in improving the overall reliability of valves.

The results reported above have raised concerns at the NRC about the capabilities of similar in-plant valves to accomplish their intended safety functions when necessary. Therefore, the NRC has notified the nuclear power plant owners that a valve evaluation program should be instituted at each plant to ensure valve operability over the remaining life of each plant or valve as the case may be. Most plants have already started to comply with the NRC request, and NRC inspectors have been auditing plants over the past several months to evaluate the respective valve programs. The inspectors have required training, technical information, and criteria for performing their evaluations. Thus, during fiscal year 1991, most of the research effort has been devoted to analyzing the test data obtained from the valve experiments (identified earlier), in order to develop the needed information and criteria and to provide training for the NRC inspector. Some of the areas where the research effort during fiscal year 1991 has made important inroads in advancing valve technology are:

- A modified valve thrust formula for bounding closing thrust requirements for gate valves was developed. The formula reflects the effects of friction, measured temperatures and pressure, and fluid conditions from the experiments. The formula includes terms that had not been previously identified in the industry standard formula.

- Progress has been made toward quantifying the effects of corrosion, wear, and fluid lubrication, as well as the importance of design parameters on valve operability. Clearances between valve internal parts are important for identifying whether a gate valve will experience damage during high-flow operation.

- A computer program has been developed for the use of NRC inspectors in evaluating valve calculations at nuclear power plants. The program provides a consistent set of criteria and methods the inspectors can use when performing these difficult evaluations. The program has also been made available to parties outside the NRC for their use in predicting valve performance.

The nuclear industry is also contributing to improving valve reliability. An extensive program, including experiments on other valves, has been developed by the Electric Power Research Institute (EPRI). The program started in April 1991 and will continue for approximately three years. Some foreign countries also have valve programs under way. Although the NRC valve research effort will continue, the level of effort will decrease in fiscal years 1992 and 1993. During these years and in subsequent years, NRC efforts will focus on reviewing, evaluating and confirming the EPRI program results and, where possible, the results of foreign programs. These efforts will enable the agency to supply NRC inspectors with additional technical information for evaluating nuclear power plant valve programs.

SEISMIC RESEARCH

The primary goal of the NRC seismic research program is to define the potential for earthquakes at nuclear power plant sites and in the regions surrounding them, and to determine the possible effects of earthquakes on the plants and their safety systems.

Earth Sciences

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 uncertainty in estimating plant hazards. Although recent NRC (NUREG/CR-5250) and EPRI (NP-6395) studies have advanced the methodology for characterizing seismic hazards at nuclear reactor sites, further seismic hazard research will be needed. The goal of the RES earth science program is to reduce uncertainties in hazard estimates by continued research into the causes and distribution of seismicity. Successes of past research programs together with applications of newly developed methods promise to significantly reduce uncertainties in seismic hazard estimates within the next decade.

Seismographic Networks. The National Seismographic Network (NSN), established through a cooperative agreement with the U.S. Geological Survey (USGS), has progressed to the operational stage and was officially dedicated on April 3, 1991. The NRC is providing 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 Golden, Colo.

In return, the USGS will operate the network and provide seismographic data to the NRC. This particular agreement has been cited as an example of unusually effective cooperation between two agencies of the Federal Government.

At present, a few stations are operational, together with the satellite link and processing facilities in Golden. However, with the details of the instrumentation having been worked out, installation of new stations should progress rapidly, and a substantial portion of the network is expected to be in place by the end of fiscal year 1992. Completion of the full network is expected to occur in fiscal year 1993.

With its dual range, three-component seismometers, this network will carry out the functions of both a microseismic 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.

NRC support for regional seismographic networks in the central and eastern United States was continued during this fiscal year, again at a funding level that was somewhat reduced from the previous year. This support function will be continued until the end of fiscal year 1992. At that time, sufficient data will be available from the NSN, and all regional network support from the NRC will end.

Additional investigations were conducted by the Geological Survey of Canada in the area of ground rupture during the December 25, 1989, magnitude 6.0 earthquake near Ungava,Quebec, and strong motion recorders deployed last fall were retrieved during July 8 through 22, 1991. The investigations revealed that the main rupture extended two kilometers farther north than reported last year and many other secondary ruptures were identified. Sub-bottom acoustic profiling was conducted to detect seismically induced deformation in lake sediment, in several lakes in the area. The results revealed isolated pockets of disturbed soil. The aftershock activity was not large enough to trigger the strong-motion recorders installed last year.

Northeastern Neo-tectonics. Last year, investigations at Ferland, Quebec—the site of the November 25, 1988 Saguenay earthquake and in the vicinity of the 1727 Cape Ann earthquake at Newbury, Mass.—identified liquefaction features induced by these earthquakes and also paleoliquefaction features caused by prehistoric earthquakes. In fiscal year 1991, following up on these studies, new investigations were started at Newbury and Moodus, Conn., and Ossippi, N.H., sites of historic and ongoing seismicity. These studies are focusing on identifying geological evidence for prehistoric earthquakes, including seismically induced liquefaction features, such as occur in fluvial soils, glacial outwash deposits and lacustrine sediments, and landslides, rockfalls and slumps. This research will also identify similar features that were caused by other than tectonic phenomena and compare their characteristics with those that were tectonically induced.

Paleoliquefaction Studies in Southeast. During fiscal year 1991, field and laboratory work was conducted in connection with a study extending the search for paleoliquefaction features from the coastal areas of South Carolina inland and into the southern Appalachians. The work was concentrated on fluvial and lacustrine deposits along the Savannah and Edisto Rivers, on Carolina bays in the coastal plain, and on the Bowman and Union County areas. Some preliminary work in the seismic zones of the southern Appalachian area of Tennessee and Giles County, Va., was also carried out.

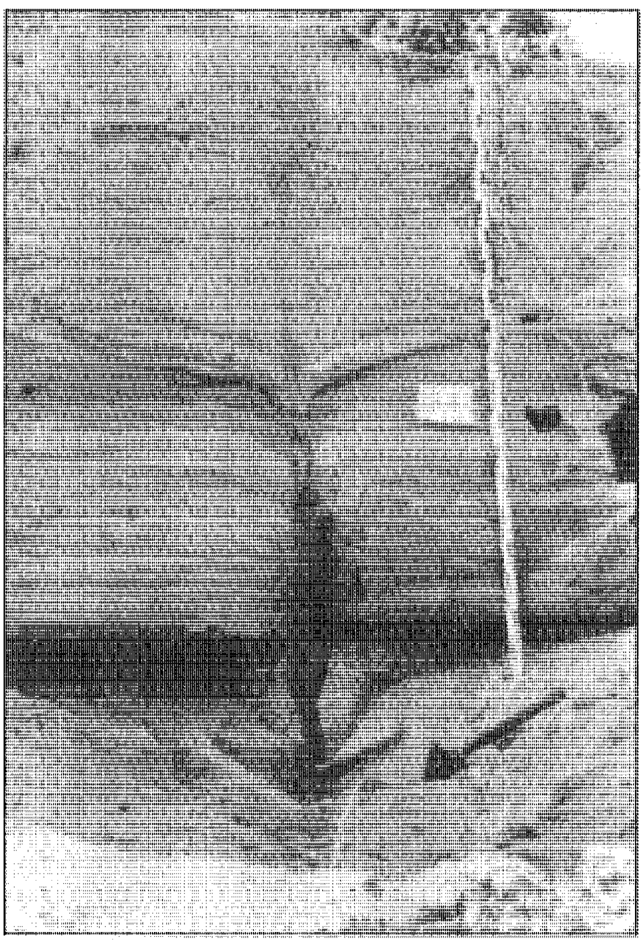
To date, no clear neo-tectonic features that are not related to the Charleston source area have been found. Preliminary results suggest that the Bowman area is not connected with the Charleston source area and has not been the source of earthquakes with a magnitude greater than 6.0 for the past several thousand years. Progress was made in establishing valid criteria for recognizing neo-tectonic features in these non-coastal areas.

Union County, located in the South Carolina Piedmont, was the site of an intensity 7–8 earthquake in 1913. Liquefiable fluvial sands have been identified, but no liquefaction features have been found so far. The investigation of this area will continue in winter, when lower water levels will expose additional outcrops. The Appalachian area has very few deposits that are liquefiable. Those that are available will be investigated, but other neo-tectonic methods are needed to obtain information on possible paleoseismic events. Seismically induced landslides and ground fissures and cave deposits have been identified as possible sources of information. While a single line of evidence in this area may not be conclusive, it is expected that multiple lines of evidence may permit firmer conclusions.

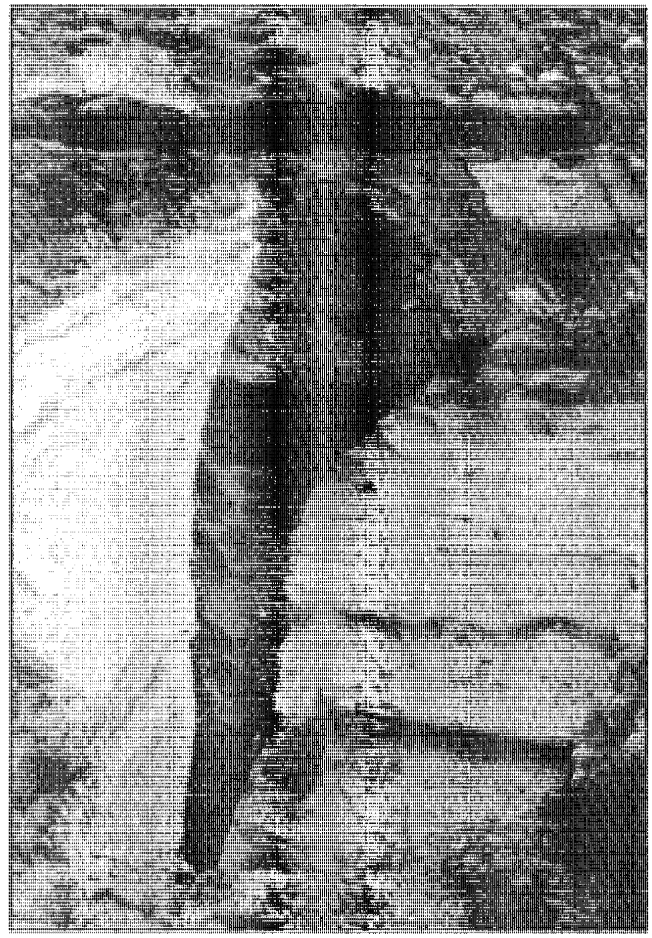
Central Virginia Seismic Zone. A seismic reflection survey that was completed in 1987 included a traverse along the Roanoke River, from Bedford to Brookneal, in a generally non-seismic zone, and several lines in the central Virginia area near Richmond. Final interpretation of this survey, together with reprocessing and reinterpretation of the I–64 seismic traverse acquired by the USGS and new geological data, has led to new conclusions concerning the structure and seismicity of this area. Most significant is the interpretation that the Piedmont and Blue

Ridge region of the central and southern Appalachians contains only one terrane boundary, namely the Taconic suture. Previous interpretations had assumed two or three terrane boundaries. The Taconic suture is repeated at the surface by faulting and folding, and it passes through the lower crust and lithosphere somewhere east of Richmond. The rupture is spatially associated with seismicity in the central Virginia seismic zone, but it is not comformable with earthquake focal planes and seems to have little causal relation to them.

In central Virginia, the metamorphic Piedmont and Blue Ridge plate is nine kilometers thick, whereas its thickness in the aseismic area of the Roanoke River traverse is only three kilometers. However, the plate may be more extensively broken by high angle normal faults in the central Virginia seismic zone. Thus, greater infiltration by ground water may reduce the strength of the fault planes present and lead to a higher rate of seismicity.



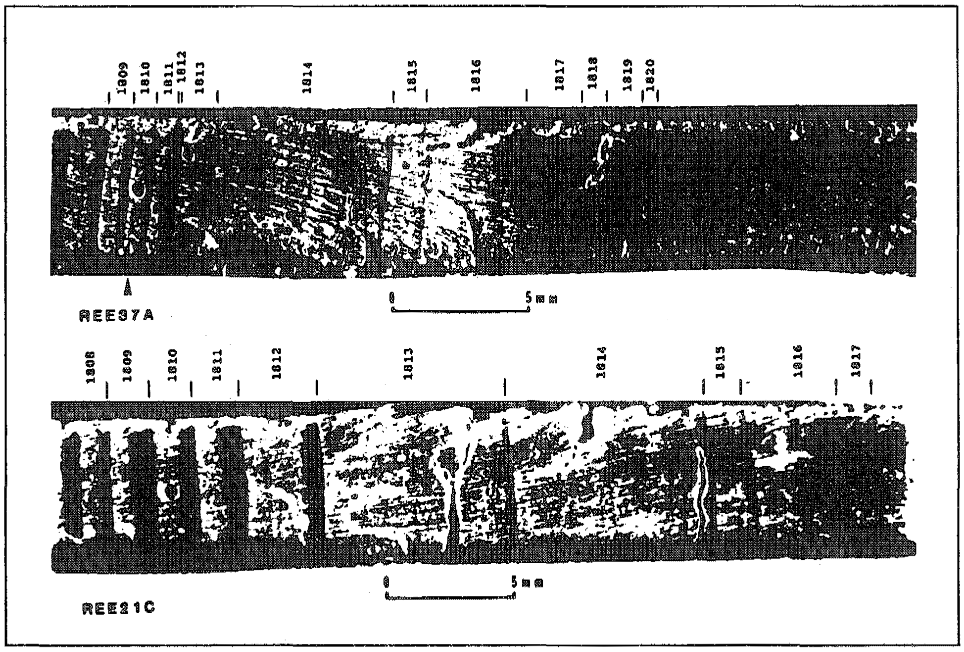
Shown above is an "ice wedge" feature, which is similar to a seismically induced liquefaction feature but was caused by the freezing and thawing, with soil incursion, of a fracture. This particular example is in the Hain Brother's Pit in east-central Connecticut.



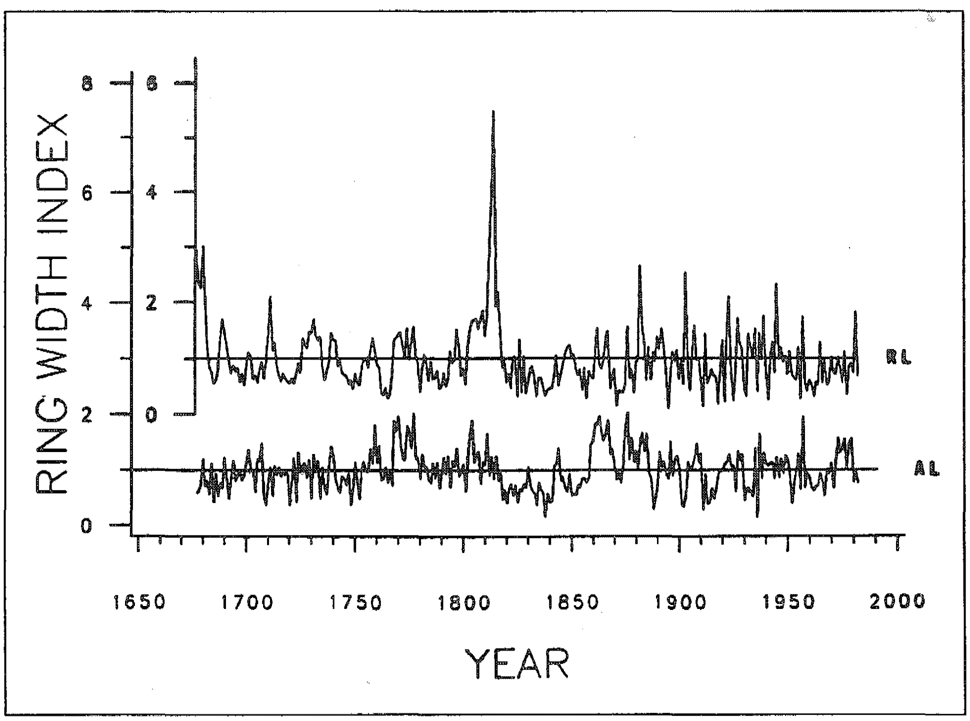
A buried ice mass was present to the left of the structure shown in the center of the photograph. As the ice melted, soils slumped down into the "kettle" cavity, and the over-lying glacio-fluvial deposits filled the open fissure. The feature exhibits characteristics similar to seismically induced liquefaction. The fault is located at Taftville, Conn.

New Madrid Seismic Zone. Evidence obtained from an array of three-component seismometers (the PANDA array of Memphis State University) deployed in the area has led to new insights into the structure of the New Madrid area and has proved the value of three-component digital recording methods. The new technology has made it possible to compute single-event focal mechanisms, compared to the composite mechanisms used in the past. As a result it was determined that, in the central segment of the seismic zone, earthquakes define a narrow, 30-kilometer-long fault zone that dips in the range of 33°–52' to the southwest. It was also found that a thin low-velocity layer exists at upper crustal depths in the area. This is interpreted to be a clastic sediment layer.

Tree-ring analyses were performed on drilled cores from bald cypress trees in Reelfoot Lake, Tenn. Reelfoot Lake is reported to have been formed during the earthquakes of 1811–1812, and growth responses in bald



The exposed bald cypress cores shown above clearly evidence the effect on tree growth of the 1811–1812 earthquakes near New Madrid, Mo. These earthquakes included the most severe shocks ever generated east of the Rocky Mountains; the area is still the source of considerable seismic activity. The bald cypresses, which tolerate inundation well, survived the quakes which created Reelfoot Lake in Tennessee, resulting in the accelerated growth between 1812 and 1815, visible in the tree cores above and strikingly in the graph below. The graph gives tree ring chronologies from Reelfoot Lake (top) and Allred Lake in Alabama. Ring-width index shows tree growth for each year.



cypress confirm the co-seismic subsidence of the Reelfoot Lake basin. While most of the hardwood trees in the area of the lake were killed, bald cypress tolerate inundation well, and many specimens up to 800 years in age remain. The bald cypress show a large growth surge after 1811, attributed to the altered hydrologic regime. In comparison to other areas, Reelfoot Lake is the only area in the mid-continent that shows a growth surge in bald cypress at this time. A second growth anomaly in the Reelfoot Lake area is a sharp decline in the density of latewood, that has lasted from 1812 until now. This also is ascribed to altered hydrologic conditions. A third characteristic of the Reelfoot cores is the presence of numerous cracks in pre-1812 portions that may indicate physical damage sustained by the trees during the earthquakes.

During fiscal year 1991, the Bootheel lineament, a 113-kilometer-long lineament, identified by satellite photographic analysis in the New Madrid seismic area, was investigated by geologic mapping, seismic reflection profiling, and trenching. The investigations revealed that the lineament is a complex zone of strike-slip deformation, consisting of multiple flower structures and fractured rock, with deformation at least as young as the base of the Quaternary Period.

Paleoseismicity in Southern Oklahoma. The northwest-striking Meers-Duncan-Criner fault zone lies along the northeastern border of a structural trough, the southern Oklahoma aulocogen separating it from a series of crustal uplifts to the southwest, such as expressed by the Wichita Mountains. (An aulocogen is a trough formed by a rift that has failed to develop.) The fault zone, which is aseismic, consists of at least five segments. Two of the segments, the Meers and Criner faults, show evidence of recent (late Quaternary) activity.

Paleoseismic studies along the Meers fault were completed in October 1989. Detailed trench logging and geologic mapping indicate left-lateral oblique slip, down to the southwest, on a steeply northwest dipping to nearly vertical fault. Analyses of these data and radiocarbon dating show that there have been at least two surface faulting events during the past 3,200 years that were probably associated with earthquakes ranging in magnitude from 6.75–to–7.25. The latest displacement occurred about 1,500 years ago. Analysis of faulted alluvial terraces along the Meers fault suggests that a period of quiescence lasting many tens of thousands of years preceded the faulting events.

After completion of the Meers fault investigation, a study of the Criner fault was begun. Geological reconnaissance and studies of aerial photographs suggest that the Criner fault may also have experienced late Quaternary displacement. It is downdropped to the southwest and is located about 80 kilometers southeast of the Meers fault. Morphologic evidence suggests young displacement and cross-cutting relationships between the fault, and late Quaternary terrace deposits at one location suggest that the last displacement occurred between 10,000 and 20,000 years ago. Detailed investigations similar to those carried out on the Meers fault are being conducted to assess the seismic hazard potential of the Criner fault, after a long delay in obtaining access to properties containing critical exposures of the fault.

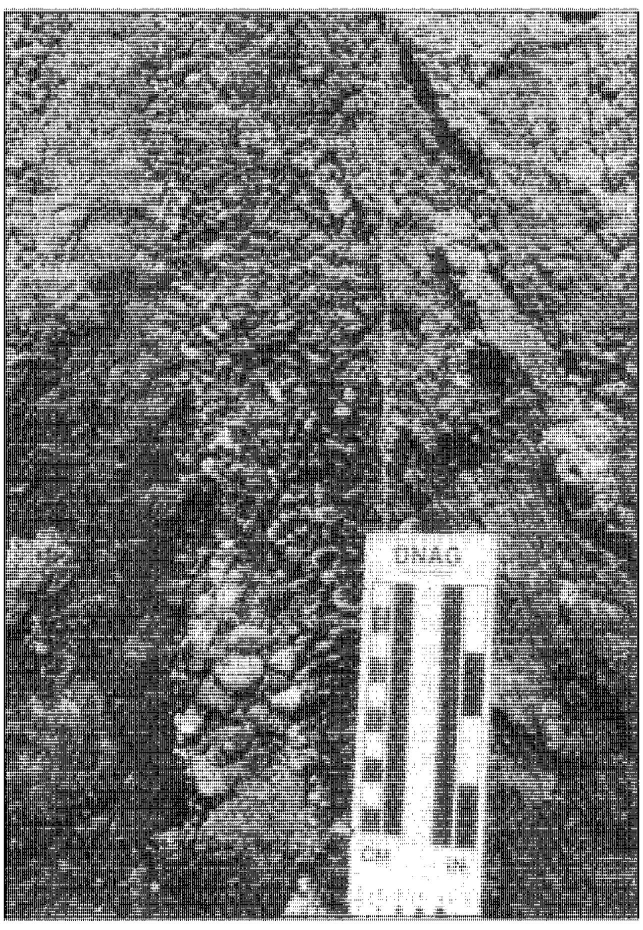
Paleoseismicity of Southern Illinois and Indiana. Southeastern Illinois has had seven significant events during the 200-year historical record. There has been considerable debate on the geological structures responsible for this seismic activity, but none has been identified with any confidence. For example, the continuity of the seismicity belt, along with geophysical evidence, has led to the interpretation that the fault system in southernmost Illinois and Indiana is a northeastern extension of the New Madrid seismic zone. An alternative explanation is that the earthquakes originate in a complex transition zone connecting two tectonic regimes. Partly because of a lack of knowledge about a causative mechanism of the earthquakes, an investigation was begun in fiscal year 1991 to identify and analyze paleoseismic evidence along the Wabash River and its tributaries.

Mapping and analysis of large dikes and lateral spreads exposed on the banks of the Wabash and White Rivers, and other drainages, suggest that a large earthquake, centered near Vincennes, Ind., occurred between 2,500 and 7,500 years ago. Comparing the sizes, distribution, and other characteristics of these features with seismically induced features at Charleston, S.C., and New Madrid, Mo., suggests that this earthquake was larger than the 1886 Charleston earthquake (magnitude of about 7.0) but smaller than the 1811–1812 New Madrid earthquakes (magnitude of about 8.0).

Pacific Northwest. Underlying the Pacific Northwest is the Cascadia subduction zone, in which the oceanic Juan de Fuca plate is being subducted beneath the North American plate. This region is an enigma, in that, while the geological and geophysical evidence indicate active subduction, there have been no historic large-thrust earthquakes along the plate interface, a phenomenon observed in other subduction zones around the rim of the Pacific Ocean.

The USGS has been conducting a major study of the geology and tectonics of this region for the last five years. The NRC is partially funding two neo-tectonic research projects under the program, one in southwestern Washington and the other in central Oregon. These efforts are continuations of investigations that revealed geologic evidence suggesting the occurrence of several large prehistoric and Holocene earthquakes. The evidence lies in marsh and shallow marine sediments, which indicate several cycles of normal stratigraphic deposition abruptly terminated by catastrophic events. These events are interpreted by most researchers to indicate large subduction zone earthquakes. At least five events are in evidence in southern Washington. The ongoing research is to better define the ages of these events, determine their regional extent, and estimate their recurrence intervals, using precise radiocarbon dating techniques of subsidence-killed Sitka spruce trees to reduce the errors inherent in the conventional technique of dating. Another study aiming to accomplish this is the analysis of diatom fossils and sand found on top of several buried peat layers. This analysis is expected to determine whether these materials were deposited by tsunamis following the earthquakes.

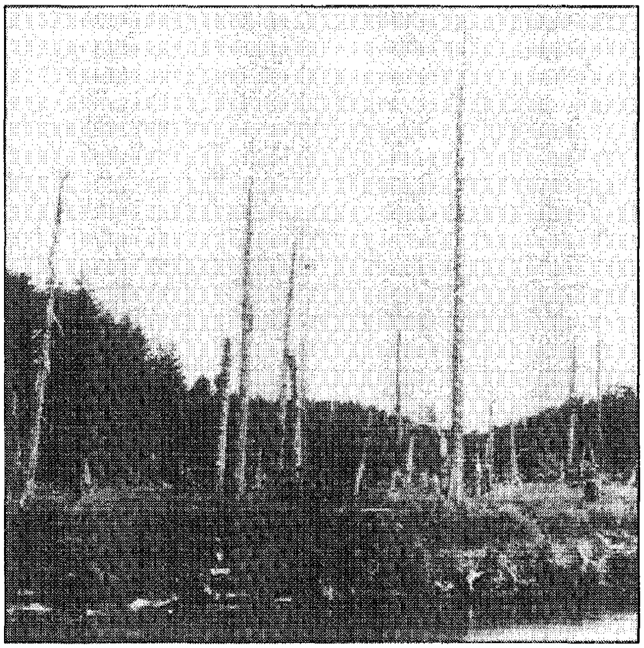
In conjunction with these studies, an investigation is underway to identify and define seismically induced paleoliquefaction features in this region. Thus far, reconnaissance along the Chelahis River and other nearby drainages has not identified evidence of seismically induced deformation features. One paleoliquefaction feature was identified near the Copalis River, and detailed investigations were conducted. The ages of the feature, which was determined to have been formed 11,000 years ago, do not coincide with the ages of any of the subsidence events. Additional river reconnaissance will be carried out, and lake bed sediment will be examined for earthquake-induced deformation.



Shown here is a seismically induced liquefaction injection feature consisting of an irregularly shaped mass of sediment that has been thrust upward through the soil profile. The location is near the Wabash River in Posey County, Ind.

Fault Segmentation Studies. The size of the maximum or characteristic earthquake that a fault can produce and the location of that earthquake along the length of a fault are major means of estimating design ground motions. Fault rupture length is a key parameter for constraining the size of future earthquakes on a fault. The constraint can be obtained from studies of the rupture length versus the magnitude or energy release of previous earthquakes. During the past decade, fault segmentation has emerged as a field of earthquake research that has important implications and applications for evaluating seismic hazard. It is based on the common observation that fault zones, especially long ones, do not rupture over their entire length during a single earthquake. A variety of structural and paleoseismic studies and investigations of historical earthquakes clearly indicate that the location of rupture is not random, that there are physical controls in a fault zone that define the extent of rupture and divide a fault into segments, and that segments can persist through many seismic cycles. Inherent in the concept of segmentation is the idea of persistent barriers that control rupture propagation. The recognition and identification of rupture segments hold the potential to provide new insights into characterizing seismic sources and an understanding of the controls of rupture initiation and termination.

Segmentation for selected faults is being evaluated using paleoseismic recurrence data and information on slip-per-event and slip rate. The data base is small but there are a number of faults that have the potential to yield information on long term segmentation. The data collection and analysis currently under way include (1) timing of the most recent and prior events along the length of the fault, (2) slip distribution during the historical event and slip during paleo-earthquakes on the same segment (repeated similar slip would imply fixed segment lengths; variable slip would indicate variability in segmentation), (3) slip rates at different locations on the fault, and (4) structural geology and geophysics of the fault zone. Although some data are available in the published literature, much information is being obtained from unpublished files and paleoseismic studies in progress. Part of the data collection has involved field visits for on-site evaluation of published information and, in some cases, development of new data, such as slip-per-event for paleo-earthquakes on segments that have had historical ruptures.



Shown above is a seismically induced paleoliquefaction feature formed 11,000 years ago near the Copalis River, in the Pacific Northwest, where snags of western red cedar protrude through the brackish tidal marsh.

Studies during fiscal year 1991 have focused primarily on the Rodgers Creek–Hayward fault zone, the segment of the San Andreas fault that ruptured during the 1989 Loma Prieta earthquake, and the Wasatch fault zone. Possible segmentation boundaries have been identified along the Rodgers Creek and Hayward faults and are being investigated. The two faults are separated by a six-kilometer-wide releasing bend. The last rupture on the Rodgers Creek fault occurred in 1808, and there is no current seismicity or creep occurring. The Hayward fault, on the other hand, is experiencing creep and seismicity.

Geomorphic reconnaissance and trenching in the Santa Cruz Mountains show that, although the Loma Prieta rupture did not reach ground surface, repeated surface faulting has occurred on this segment of the San Andreas fault through time. This indicates that variable modes of rupture may occur on this segment.

Analysis of intra-plate segmentation and its application to hazard focused on comparisons of the relationships between long term slip rates, slip-per-event, and paleoseismically determined timing of earthquakes, by using the geologic history over the last 10,000 years of the Wasatch fault zone.

Strong Ground Motion Studies. The NRC supports several strong ground motion studies related to both the eastern United States and California. A study of soil dynamics at Garner Valley (near Anza, Cal.) employs an array of wide-band strong motion seismometers, placed at various depths in boreholes, to gain information on soil dynamics and amplification of earthquake motion. The study, performed by the University of California at Santa Barbara, is one of many research programs that demonstrate effective cost sharing between the NRC and other agencies. The study is being funded in cooperation with the USGS and the U.S. Army Corps of Engineers, and with support from the Commissariat a L'Energie Atomique (CEA) of France.

Although many theoretical and laboratory studies have explored the effect of near-surface soil layers on the amplification of ground motion, very few direct measurements are available to confirm the predictions made by the more theoretical methods. The Garner Valley array is located between the San Jacinto and San Andreas faults in an area of high seismicity. The site is underlain by soil and weathered granite, over a granitic basement at about a 45-meter depth. Seismometers were placed at the surface and in boreholes at various depths ranging to 220 meters. In less than a year, 125 earthquakes were recorded, with magnitudes ranging from 1.2–to–4.7. Analysis of the data shows that, at low frequencies, amplification from bedrock to surface is by a factor of six, over a wide range of magnitudes. The lower frequencies are also those that have the highest damage potential for engineered structures.

During fiscal year 1991, records from 17 earthquakes within 20 kilometers of the instrument array, with magnitudes of 2.0 or greater, were selected as the bases for study of the amplification as a function of frequency. Then, to assess the effect of layering, the acceleration amplitude spectrum for S waves at various depths for two earthquakes, magnitudes 4.2 and 2.5, were considered. The results showed that the spectrum for relatively unweathered granite bedrock at 220-meters depth has a substantially greater amount of high frequency energy than materials at shallower depths. Weathered granite amplified low frequencies but attenuated high frequencies, and soil amplified the overall spectrum.

Other ground motion research supported by the NRC is being performed by the USGS and includes analysis of strong ground motion teleseismic records of large intraplate earthquakes and estimates of high-frequency ground motions for earthquakes in the eastern United States.

Crustal Strain Measurements. The original set of Global Positioning System (GPS) measurements for the 45-station crustal strain network covering the central and eastern United States (measured during the winter of 1987– 988) has been recomputed, using newer software for improved accuracy. This set of new measurements forms the baseline for future measurements. Experience gained, particularly from this first set of GPS measurements, has led to better survey procedures. That fact—together with improvements in instrumentation, software, and satellite availability—is expected to lead to further improvements in accuracy for the network (already at the level of a few parts-per-100 million) for a substantial portion of the first measurement set. The improved survey procedures, in particular, are expected to provide a more uniform distribution of accuracies over the entire region surveyed.

Probabilistic Seismic Hazard Assessments. Probabilistic seismic hazard assessments (PSHAs) began 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, which is in progress, will put substantial emphasis on PSHAs as part of the investigation required for 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 balanced 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; the other was performed by the EPRI and sponsored by utilities in the Seismicity Owners Group. The two studies proceeded by similar methodologies and produced hazard curves with similar characteristics; they also produced consistent relative hazard rankings for plant sites in this region. A difficulty arises, however, from the fact that, at certain sites, absolute hazard levels may differ by as much as two orders of magnitude.

Because more consistent absolute hazard levels will be needed in the future to resolve questions of power plant design and licensing, a plan was formulated for a study that will analyze the two existing methodologies to identify the sources of discrepancy and attempt to mitigate the differences between the LLNL and EPRI approaches. From previous analyses, it is known that certain input parameters—such as seismic parameters and ground motion models—cause some of the differences. It appears that the computer codes used to do the calculations will give approximately the same results for a given input, although the validity of that statement also needs to be more fully verified. The planned study is aimed at a consolidated methodology that will provide more uniform results and can be used as a basis for PSHAs for the next decade or so. A peer review by a panel appointed by the National Academy of Sciences is planned to ensure the impartiality and objectivity of the study.

Seismic Engineering Research

Besides the earth science research discussed above, the NRC seismic research program includes several engineering-oriented programs to ascertain the effect of earthquakes on nuclear plant structures and safety systems.

Implementation of Executive Order 12699. Executive Order 12699, "Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction," was issued by the President, on January 5, 1990, to implement certain provisions of the Earthquake Hazards Reduction Act of 1977. The Executive Order applies to all Federal agencies that (1) are responsible for the design and construction of new Federally owned buildings; (2) are responsible for construction and lease of new buildings for Federal use; (3) assist in the financing, through grants or loans, of newly constructed buildings; (4) guarantee the financing, through loan or mortgage insurance programs, of newly constructed buildings; or (5) are responsible for regulating structural safety of new buildings. Agencies responsible for construction projects of the first two types must demonstrate compliance for all projects for which development of detailed plans and specifications is initiated subsequent to the date of the order. Agencies administering the other types of programs have three years from the date of the order to establish an appropriate seismic hazard reduction program.

During the past year, the staff performed a careful review of the Executive Order and NRC requirements for the design and construction of buildings associated with nuclear power reactors and with other activity. The other activity included Class 104 licenses for medical therapy and research and development facilities, processing of uranium ores in milling operations, high-level-waste repository licensing, on-site spent fuel storage, licensing of plutonium processing and fuel fabrication plants, and license application reviews for uranium enrichment facilities. It was concluded that NRC's current practice meets the requirements of the Executive Order and that no new regulatory action is necessary.

Individual Plant Examination for Seismic Events. A major task in the seismic engineering area concluded during the report period with the publication of the NRC's final guidance for conducting an individual plant examination for seismic events, pursuant to implementation of the Commissions' Severe Accident Policy. (For a description of the development of draft guidance, see the *1990 NRC Annual Report,* pp. 153 and 166.) A number of changes were made to the draft guidance documents as a result of an NRC- sponsored workshop and written responses from the public. Further clarifications of staff guidance in the seismic area were made during a question-and-answer session that was part of the Nuclear Utility Management and Resources Council (NUMARC) workshop. (For further information, see "External Events," under Severe Accident Implementation, later in this chapter.) A program to develop a PC-based computer code to assist in reviews of the licensee's submittals related to individual plant examinations for external events (IPEEEs) was initiated in 1991. The program will allow the staff to perform alternative parametric and sensitivity studies and to develop generic regulatory insights. This program can also be used for seismic risk and margin studies for advanced reactors.

Revision of Appendix A to 10 CFR Part 100. Starting early in 1991, the staff began a major rulemaking effort, i.e., the revision of the seismic and geologic siting criteria for nuclear power plants, Appendix A to 10 CFR Part 100. The revision was undertaken in order to (1) benefit from the experience gained in applying the existing regulation, (2) resolve interpretative questions, (3) provide needed regulatory flexibility to incorporate state-of-the-art improvements in the geosciences and earthquake engineering, (4) simplify the language to a more "plain English" text, and (5) acknowledge various internal staff and industry comments. Criteria not associated with the selection of the site or establishment of the safe shutdown earthquake ground motion will be placed in Part 50. This action is consistent with the location of other design requirements in Part 50.

Several issues are being addressed by the staff in conjunction with the revision of the regulations. In the geosciences area, the emphasis on deterministic and probabilistic assessments, along with guidance on how the two methods should be merged, if applicable, is being evaluated. In the earthquake engineering area, the proper role of the "operating basis earthquake" in future plant design is being assessed.

The revision of the geologic, seismic, and earthquake engineering criteria is being performed in conjunction with the revision of the reactor siting criteria, 10 CFR Part 100. (See "Regulatory Applications of New Source Terms," later in this chapter.)

Seismic Component Fragilities. The NRC-approved guidance document for individual plant examinations of external events has endorsed the use of the seismic component fragilities developed by Brookhaven National Laboratory. The components included are motor control centers, switchgears (low and medium voltage), panelboards, switchboards, power supplies, instrumentation and control panels, transmitters, indicators, switches, transformers, batteries, battery chargers, inverters, motors, and electrical penetration assemblies. The seismic fragilities are expressed in terms of medium and standard deviations of spectral acceleration capacities to allow the NRC staff to understand the conservatisms associated with the estimates of seismic fragility. While the initial use of this work was for the resolution of Unresolved Safety Issue A-46 and for IPEEE, the NRC staff is now evaluating seismic PRAs for advanced reactors using the results of this study.

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, Taiwan, that historically has had 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, and new members Korea Power Engineering Company (KOPEC) and Korea Electric Power Corporation.

The duration of the Hualien project is expected to run for 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. The facility is now scheduled for full operation in the fourth quarter of 1992.

Confirming Safety of Nuclear Waste Disposal

The NRC's waste management research seeks to develop and verify methods for predicting and assessing the performance of waste disposal facilities; evaluate and confirm the data bases used in such performance assessments; provide technical support to the licensing staff in their interactions with the Department of Energy (DOE) and the States (see Chapter 7); and develop regulatory standards to support the licensing of facilities and methods for the disposal and management of high-level and low-level radioactive wastes.

During 1990–1991, research program plans for both high-level waste (HLW) and low-level waste (LLW) were further developed in an effort to ensure the usefulness of the program in meeting the needs of the licensing staff.

High-Level Waste

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 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 in Nevada, currently under consideration by the DOE as as a permanent HLW repository. 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 are 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.

Geohydrology. Since transport by ground water is the most likely path by which radionuclides from disposed waste might reach the environment, the NRC is actively studying the movement of ground water in the unsaturated fractured media currently under consideration by DOE. An experimental site has been located in unsaturated fractured tuff (the same rock type as the repository host rock) in Arizona, where field and laboratory testing is being conducted by the University of Arizona. The objective of the field study is to determine what types of measurements are needed to properly characterize the hydrology of fractured rock and how measurement data should be analyzed to model ground-water flow. This work currently entails an appraisal of techniques and methods for measuring rock properties in place, and for assessing infiltration and movement of water in rock formations. The project is using numerical calculations of flow and transport to judge the importance of site features, appropriateness of fracture models, and theories and measurements of flow-controlling properties and processes.

Investigators at the Center for Nuclear Waste Regulatory Analyses (CNWRA) in San Antonio, Tex., are examining methods to perform stochastic hydrologic analyses for repository scale systems. The validity of the models used to describe ground-water flow and radionuclide transport is being evaluated in an international project called INTRAVAL. The NRC staff and research contractors from the CNWRA, the University of Arizona, Sandia National Laboratories, Massachusetts Institute of Technology, and the Battelle Pacific Northwest Laboratory are participating in the 13-country validation effort.

Cooperative experiments and data analyses being done under a cooperative agreement between NAGRA (Switzerland) and the NRC, negotiated during fiscal year 1987, continue to augment the field testing program cited above.

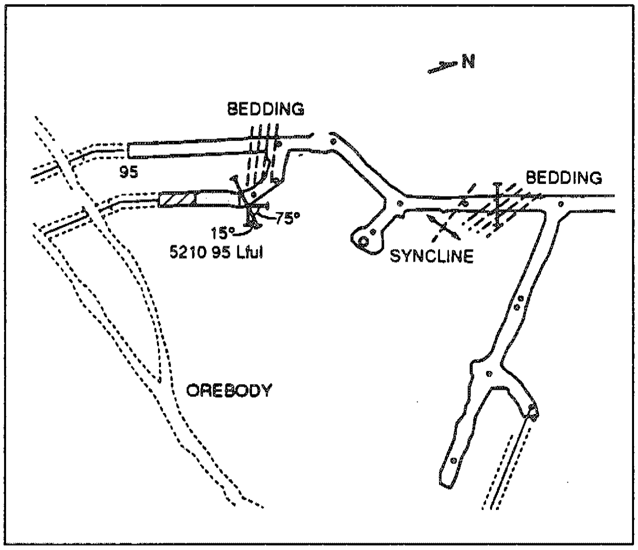
Stability of Underground Openings. When specifying suitable site conditions for a repository, Federal law (10 CFR Part 60) specifically requires consideration of natural phenomena and site conditions that could adversely affect achievement of the prescribed performance objectives. An important phenomenon that could affect both the short and long term performance of a repository is ground motion resulting from seismic activity. Similarly, ground motion caused by underground nuclear explosions at the Nevada Test Site needs to be evaluated. Ground motion from either source could cause rock displacement, rise in water tables, etc., which could violate established repository performance objectives.

To understand the effects of seismicity on the underground openings for an HLW repository, the NRC is sponsoring research at the CNWRA. Initial results from the studies indicate that structural damage at depth can occur; a field site in an existing mine is being instrumented to assess these effects.

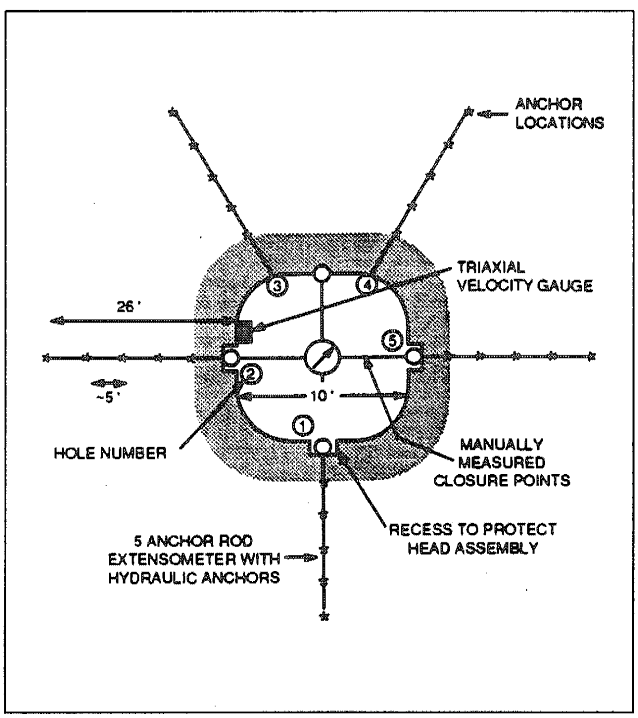
Sealing of Boreholes and Shafts in Tuff. The isolation of nuclear waste in deep geological repositories may require that penetrations in the geological host rock barrier—such as shafts, drifts, ramps, and boreholes in the vicinity of the repository—be sealed to prevent the creation of potential pathways for the movement of radionuclides to the accessible environment.

To evaluate the performance of seals in the unsaturated HLW tuff environment, the NRC has supported research studies at the University of Arizona. Both laboratory and field tests of seals were conducted for a variety of potential seal materials. Characterization testing confirmed that tuff is an extremely non-uniform rock with highly variable properties and extremely low hydraulic conductivity.

Geochemistry. Knowledge and application of geochemistry is important to an understanding of many aspects of 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 1991, the chemistry of the natural waters at Yucca Mountain was evaluated at the CNWRA by geochemical models, in the context of



The Center for Nuclear Waste Regulatory Analysis is conducting an instrumented study of the seismic effects at the 5,210-foot level of the Lucky Friday mine near Coeur d'Alene region of Idaho. Above is the system of ramps in the mine at this level. Below is a cross section of the tunnel and instruments at the site. These kinds of study have application to the planned high-level waste repository at Yucca Mountain, Nev.



the variable compositions of minerals that rapidly react with ground water. The evolution of water as it is heated while moving toward the waste packages was modeled at the CNWRA and used as input for waste package performance testing. The state of the art in measuring and modeling physical-chemical processes that retard radionuclide transport has been explored by the CNWRA. The NRC is participating in an international field study at the Koongarra ore body in northern Australia, observing the actual movement of radionuclides. That study is providing a basis for validating performance assessment models to be used in HLW repository licensing. The fourth year of the study has seen the completion of data collection and the undertaking of in-depth hydrologic and geochemical modeling. The results of simple transport models have been compared with site data, and more sophisticated transport modeling is continuing. A study at Johns Hopkins University to develop a coupled thermo-hydrogeochemical transport model has successfully completed model development, and research has been started to test it against data from natural systems such as the Koongarra ore body.

Rulemaking. A proposed guide published for public comment in November 1990 provides the information needed by the NRC to review DOE's license application for the HLW repository. The NRC continued to closely monitor EPA's development of a revised high-level radioactive waste standard. The NRC staff has commented on EPA working drafts of the standard. In July 1990, a petition was received from the States of Washington and Oregon asking the NRC to undertake a rulemaking regarding the classification of some radioactive wastes at the DOE facilities at Hanford, Wash. During fiscal year 1991, the NRC completed internal review of this petition, and formal resolution is expected early in fiscal year 1992.

Low-Level Waste

NRC research in support of licensing activity for low-level 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 November 1989. That document identifies issues, regulatory needs, 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 attended by State representatives, such as the conference called "Waste Management '91," and the Annual DOE LLW Management Conference.

Engineered Enhancements and Alternatives to Shallow Land Burial. There is great interest on the part of States and State compacts in alternatives to shallow land burial for the disposal of low-level radioactive waste. Concrete is expected to play an important role in engineered alternatives to shallow land burial. In 1991, the National Institute of Standards and Technology (NIST) continued investigating, for the NRC, the durability of concrete in engineered alternatives to shallow land burial, while the Idaho National Engineering Laboratory continued to develop a mathematical model describing concrete performance. NIST has published a report on modeling transport processes in concrete and the diffusion of chloride ions in concrete.

LLW Waste Forms. Low-level radioactive waste collected from operating nuclear power stations and solidified in cement is being tested at the Idaho National Engineering Laboratory. The studies are aimed at ensuring that radionuclide and chemical leaching characteristics, as well as the compressive strength of the solidified waste, are consistent with NRC technical positions and the requirements of 10 CFR Part 61 for waste form stability. Under examination is the stability of decontamination waste obtained from nuclear reactors, having undergone commercial decontamination processes and been solidified in cement. Field studies 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. A report has been issued on the release of radionuclide and chelating agents from cement-solidified LLW (NUREG/CR-5601).

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 3.) 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 resistive layer barrier (compacted clay) over a conductive layer barrier. The second system has functioned perfectly since its completion in January 1990. However, its long term performance remains to be assessed.

Performance Assessment. Research is continuing on a performance assessment methodology. Emphasis is being given to engineered enhancements to shallow land burial. The Sandia National Laboratories are assessing the validity of performance assessment models, and the Pacific Northwest Laboratory (PNL) is examining mathematical models for radionuclide transport through concrete. The Massachusetts Institute of Technology (MIT) has been investigating the use of stochastic methods for dealing with large scale non-uniformity of site hydrologic characteristics. The University of Arizona and New Mexico State University are working cooperatively with MIT by providing a field test at Las Cruces, N.M., of MIT's theoretical work.

LLW Source Term Modeling. Development of the LLW source term code, BLT (breech, leach, transport) continued during fiscal year 1991. The Brookhaven National Laboratory has refined and expanded the transport submodel to consider geochemistry and gas transport. To provide confidence in the model predictions, the BLT code continues to be benchmarked against lysimeter experiments using saltstone waste forms at the Savannah River Laboratory and using 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. This work represents a first attempt at quantification of source terms for use in performance assessment.

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 MIT—is intended to confirm the reliability of unsaturated flow and transport models of LLW disposal facilities. This work is a part of the INTRAVAL international study that deals with model validation of groundwater flow and transport models.

Rulemaking. Final amendments to 10 CFR Part 40 that provide licensing for the custody and long term care of uranium and thorium mill tailings disposal sites were published in the *Federal Register* in October 1990 (55 FR 45591).

A Petition for Rulemaking (PRM-61-1) from the North Carolina Chapter of the Sierra Club was filed. The petitioner requested the Commission to adopt a regulation to permit the design and construction of a zero-release low-level radioactive waste disposal facility in a saturated zone. The petitioner stated that the regulation was necessary in order for the General Assembly of North Carolina to consider a waiver of a North Carolina statute that requires that the bottom of a low-level waste facility be at least seven feet above the "season-high" water table. The petition was rejected; a Denial of Petition was published in the *Federal Register* in July 1991 (56 FR 34035).

Resolving Safety Issues And Developing Regulations

GENERIC SAFETY ISSUES

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 forth in the *1982 NRC Annual Report* for determining the priority of GSIs. In December 1983, a comprehensive list of the issues was published in "A Prioritization of Generic Safety Issues" (NUREG-0933), and the list has been updated semi-annually since (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 1991, the NRC identified 29 new GSIs, established priorities for 12 issues (Table 1), and resolved four GSIs (Table 2). Table 3 contains the schedules for resolution of all unresolved GSIs.

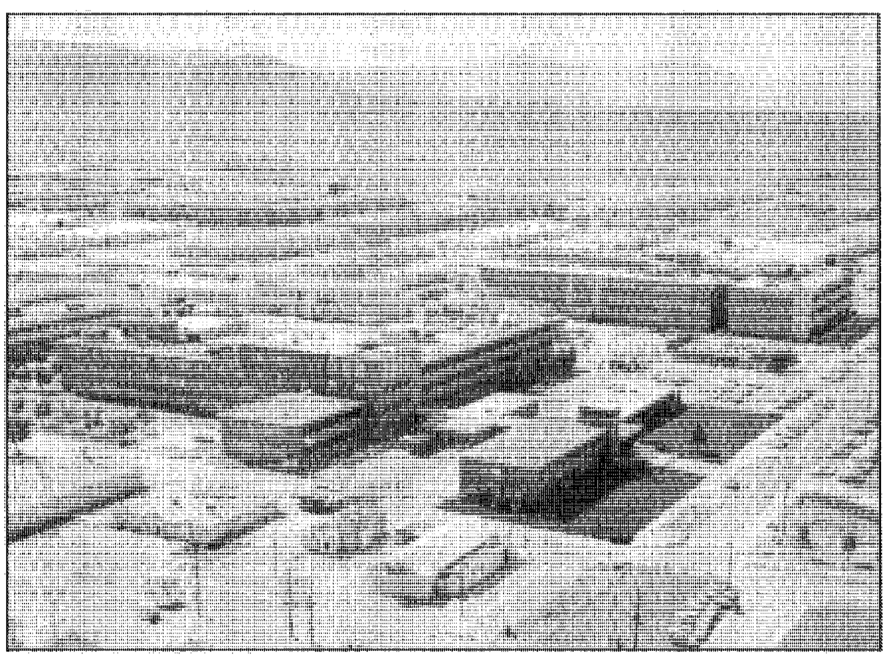
ADVANCED REACTORS

Research programs in support of anticipated design certification of advanced reactors were reorganized in fiscal year 1991 to correspond to related organizational changes in the NRC's Office of Nuclear Reactor Regulation (NRR). The tasks of completing the pre-application reviews of the advanced liquid metal reactor and of the modular high temperature gas reactor was transferred to NRR to make room for an expansion of support programs dedicated to other designs under consideration.

Eight different designs are being considered for potential design certification under the relatively new Part 52 of Title 10 of the Code of Federal Regulations. These designs fall into two groups of four. One group comprises evolutionary and passive light-water reactors (ABWR, SYSTEM 80+, AP600, and SBWR); the other group covers other concepts (PIUS, CANDU, ALMR, AND MHTGR).

The two evolutionary plants (ABWR and SYSTEM 80+) are similar enough to current designs that little additional research is needed beyond that already under way in relation to current operating reactors. However, a major effort is being made pursuant to certification of the passive designs (AP600 and SBWR), since they are different from current LWR designs and are likely to be certified relatively soon. High priority efforts were made to

The Sandia National Laboratories, shown at the right, have been the site of extensive NRC-sponsored research over the years into manifold facets of nuclear power plant safety. Besides the work carried out at Sandia on core-melt progression, fuel-coolant interaction and plant component-aging, cited earlier in this chapter, the National Laboratories are currently participating in high- and low-level waste repository analyses, on behalf of the NRC. Sandia National Laboratories are located near Albuquerque, N.M.



review vendor test programs and to identify research needs requiring early attention; for example, the Westinghouse experimental program has been reviewed, and areas for possible enhancement identified. Chief among the latter was the need for a properly scaled integral test facility to investigate possible system interactions of the major AP600 components under gravity flow conditions. Another concern calling for an integral facility was determined to be the interaction between pumped non-safety systems and gravity-driven safety systems when the latter may be hampered by hardware failures.

Some of the initial work performed by the NRC to provide an independent capability for reviewing applicant submittals included development of a thermal-hydraulic model for the AP600 design for use with the NRC's RELAP5 computer code, permitting the staff to independently evaluate safety system performance. Some preliminary analysis of the small-break LOCA in the Westinghouse AP600 design showed cases where the fourth stage of the automatic depressurization system might not adequately promote flow from the in-containment refueling water storage tank into the vessel. Westinghouse personnel were informed of this result and have improved the design of the automatic depressurization system. A considerable amount of research is also under way on digital instrumentation and controls and on control room design, since these new features will be present in the next generation of power reactors.

Systematic studies of a more preliminary nature are being initiated for the other advanced reactor concepts. They include systems engineering studies to identify important accident sequences and safety systems, as well as more detailed analyses of accident sequences thus identified. These studies will lead to the identification of research needs requiring early attention and to the development of an independent analytic capability available to the NRC staff in its future review of these designs for certification.

DEVELOPING AND IMPROVING REGULATIONS

A final rule (10 CFR Part 71) on modifying NRC's transportation regulations has been delayed until the Department of Transportation is prepared to issue a companion rule. Public comments on the proposed rulemaking have been evaluated, and the final rule is in preparation, with publication expected sometime in fiscal year 1992. The rule proposes limitations on the shipment of low-specific-activity materials and maximizes compatibility between NRC and International Atomic Energy Agency (IAEA) regulations.

A final rule (10 CFR 50.73) on access authorization at nuclear power plants and an accompanying regulatory guide were published in the *Federal Register* in April 1991 (56 FR 18997). The rule requires a nuclear power reactor licensee to have an access authorization program in its site physical security plan. This measure would provide greater assurance that persons granted unescorted access to protected and vital areas are trustworthy and do not pose a threat to commit radiological sabotage.

A proposed rulemaking (10 CFR Part 74) on the material control and accounting requirements for uranium enrichment plants and an associated regulatory guide were published for public comment in the *Federal Register* in December 1990 (55 FR 51726). The rulemaking is following an accelerated schedule because a license application has been 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 licensing of such a facility. A final rulemaking is expected to be published in the *Federal Register* early in fiscal year 1992.

In a program initiated in 1985 and continued through 1991, the NRC staff undertook to evaluate existing regulatory requirements in terms of their risk effectiveness, and to eliminate or modify those requirements with only a marginal safety importance. A three-volume research report (NUREG/CR–4330) provided detailed technical assessments of requirements associated with a limited number of topics. In a follow-on effort, a set of regulatory requirements were identified as candidates for possible elimination or modification. Work was begun in 1990 to evaluate the safety significance of these candidate regulations in order to identify those of marginal safety significance, for which modification or elimination may be in order. Final recommendations were forwarded to the Commission in July 1991; implementation of final recommendations will begin in fiscal year 1992.

A final rule amending regulations in 10 CFR Parts 20, 30, 40 and 70, to revise licensee reporting requirements with respect to notifications of incidents related to radiation safety, was published in the *Federal Register*, in August 1991 (56 FR 40757). The rule will ensure that significant occurrences at facilities operated by material licensees are promptly reported to the NRC. The Commission will be able to determine whether a licensee has taken the actions necessary to protect public health and safety and whether generic safety concerns that may require prompt NRC actions are being identified.

Table 1. Issues Prioritized in FY 1991

Number	*Title*	*Priority*
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM
38	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	DROP
72	Control Rod Drive Guide Tube Support Pin Failures	DROP
73	Detached Thermal Sleeves	NEARLY RESOLVED
100	Once-Through Steam Generator Level	DROP
120	On-Line Testability of Protection Systems	MEDIUM
143	Availability of Chilled Water Systems and Room Cooling	HIGH
150	Overpressurization of Containment Penetrations	DROP
151	Reliability of Anticipated Transient Without Scram Recirculation Pump Trip in BWRs	MEDIUM
153	Loss of Essential Service Water in LWRs	HIGH
A–19	Digital Computer Protection System	LICENSING ISSUE
B–22	LWR Fuel	DROP

Table 2. Generic Safety Issues Resolved in FY 1991

Number	*Title*
128	Electrical Power Reliability
130	Essential Service Water System Failures at Multi-plant Sites
135	Steam Generator and Steam Line Overflow
II.J.4.1	Revise Deficiency Report Requirements

A final rule amending 10 CFR Part 50 to require the licensee to implement the NRC-approved Emergency Response Data System (ERDS) at all nuclear power plants was published in the *Federal Register* in August 1991 (56 FR 40178). (The proposed rule had been published in the *Federal Register* for public comment in October 1990 (55 FR 41695).) The rule would supplement the voice transmission over the existing Emergency Notification System and require that a direct electronic data link between the licensee's computer and the NRC's Operation Center be activated by the licensee during an alert of higher emergency condition, so as to transmit timely and accurate updates of critical information on plant conditions. This measure would allow the NRC to perform its primary role during an emergency at a licensed nuclear power facility—monitoring the licensee to ensure that appropriate recommendations are made with respect to necessary off-site actions to protect public health and safety.

A final rule was published in the *Federal Register* in July 1991 (56 FR 34101) to amend the 10 CFR Part 35 regulations that apply to the medical use of byproduct material. The amendments require medical-use licensees to implement quality management (QM) programs and revise misadministration reporting requirements. Implementation of the new performance-based requirements is supported by the issuance of a regulatory guide that includes specific guidance for QM programs and an approach acceptable to the NRC for meeting the requirements of the final rule. The rule provides a high confidence that byproduct material and radiation from byproduct material will be administered as directed by the authorized user physician. The feasibility of this approach was evaluated during a pilot program involving 70 medical-use licensees and subsequent discussion with professional associations and Agreement States.

A proposed rule, 10 CFR Parts 31 and 32, on requirements for the possession of industrial devices containing byproduct material was submitted to the Commission for approval in August 1991. This rule would require general licensees to provide the NRC with specific information about radioactive material used under the provisions that establish general domestic licenses for byproduct material. The proposed action would improve public health and safety by reducing the likelihood for unnecessary radioactive exposures from radioactive materials, by ensuring that generally licensed devices are properly accounted for and disposed of. The proposed rulemaking is expected to be published in the *Federal Register* for public comment early in 1992.

A proposed rulemaking (Appendix H to 10 CFR Part 73) on weapons-firing qualifications and physical fitness programs for security personnel at category I fuel cycle facilities was submitted to the Commission for approval in September 1991. The proposed rule would amend the Commission's regulations to include day-firing qualification courses in each type of required weapon, as well as a standardized physical fitness training course, and fitness standards, for security personnel. Standardization of day-firing courses, making them consistent with those established for night-firing, is needed to provide for a uniform, enforceable program. The proposed rulemaking is expected to be published in the *Federal Register* for public comment early in fiscal year 1992.

The Commission is considering a proposed rulemaking (10 CFR Part 50) on training and qualification of nuclear power plant personnel. The proposed rule would amend the Commission's regulations requiring that each applicant for and holder of a license to operate a nuclear power plant establish and use a "systems approach" in developing training programs for management, supervisory, professional and technical workers who have an impact on

Table 3. Generic Safety Issues Scheduled for Resolution

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
15	Radiation Effects on Reactor Vessel Supports	HIGH	03/94
23	Reactor Coolant Pump Seal Failures	HIGH	06/92
29	Bolting Degradation or Failures in Nuclear Power Plants	HIGH	/91
87	Failure of HPCI Steam Line Without Isolation	HIGH	03/92
105	Interfacing Systems LOCA at LWRs	HIGH	04/92
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	HIGH	09/92
121	Hydrogen Control for Large, Dry PWR Containments	HIGH	11/91
143	Availability of Chilled Water Systems and Room Cooling	HIGH	01/94
153	Loss of Essential Service Water in LWR's	HIGH	TBD
B-56	Diesel Reliability	HIGH	09/92
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	10/92
HF5.1	Local Control Stations	HIGH	11/92
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	06/92
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	TBD
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	12/92
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	10/91
106	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	03/92
120	On-Line Testability of Protection Systems	MEDIUM	12/92
142	Leakage Through Electrical Isolators	MEDIUM	06/94

Table 3. Generic Safety Issues Scheduled for Resolution
(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
151	Reliability of Anticipated Transient Without Scram Recirculation Pump Trip in LWR's	MEDIUM	TBD
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	01/92
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	02/95
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	TBD
I.D.3	Safety System Status Monitoring	MEDIUM	TBD
73	Detached Thermal Sleeves	NEARLY RESOLVED	TBD
83	Control Room Habitability	NEARLY RESOLVED	12/91
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	10/91
I.D.5(3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	10/91

the health and safety of the public. Licensees and applicants would also be required to establish qualification requirements for these personnel. The objectives of the proposed rule are to codify existing industry practices related to personnel training and qualification and to meet the directives in Section 306 of the Nuclear Waste Policy Act of 1982 (Public Law 97–425). The proposed rulemaking is expected to be published in the *Federal Register* for public comment early in 1992.

The Commission issued a denial of a petition for rulemaking (PRM–50–50) from Charles Young for publication in the *Federal Register* in January 1991 (56 FR 1749). The petitioner requested that the Commission amend its regulations to prevent nuclear power plant operators from deviating from license conditions or technical specifications during an emergency. The petitioner believes that nuclear power plants should be operated in accordance with the operating license and appropriate technical specifications, and that requiring a senior operator to follow the technical specifications during an emergency enhances plant safety.

A final rule was published in the *Federal Register* in January 1991, amending the 10 CFR Parts 20 and 50 regulations that apply to the Operations Center Area Code telephone numbers. The amendment provides the correct commercial telephone number for licensees to contact the NRC Operations Center.

Summary of Rulemaking Actions

During fiscal year 1991, 91 rulemaking actions were processed, of which 23 rules were formally published, 16 were terminated/withdrawn, and 52 are ongoing (see Table 4). Besides the 52 ongoing rulemaking actions, there are 49 potential rulemaking actions, and it is estimated that in fiscal year 1992 there will be approximately

Table 4. Rulemaking Actions Processed During FY 1990

Rulemaking Activities	*Number*
Final Rulemakings Published	23
Rulemakings Terminated/Withdrawn	16
Ongoing Final Rulemaking Actions	14
Ongoing Proposed Rulemaking Actions	25
Rulemakings on Hold	13
Total Rulemakings	91

15-to-20 new rulemaking requests calling for RES review and approval by the Executive Director for Operations.

Regulatory Analysis

The NRC conducts regulatory impact analyses (RIAs) in support of certain regulatory actions (e.g., rulemakings, backfits, generic safety issues, regulatory guides). The NRC is in the process of updating and revising the "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 concerning 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 also to update the methods and information bases for performing regulatory impact analyses to reflect experience gained over the past several years. To aid NRC analysts in preparing RIAs, work has begun on updating replacement energy costs and estimating 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, which are both 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 assessing 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.

Maintenance Rule and Regulatory Guide

In March 1988, the Commission issued a Policy Statement on the Maintenance of Nuclear Power Plants. In this statement, the Commission indicated its intention to pursue a rulemaking on maintenance. In developing the proposed rulemaking, the staff had extensive contact with U.S. industry (airline and nuclear) and studied foreign nuclear maintenance programs and practices. A three-day public workshop was held in July 1988 to solicit comments on rulemaking options. The information gathered was used in formulating the proposed rule and its supporting regulatory guide. The Commission issued the proposed rule for public comment in November 1988 and the supporting draft regulatory guide in August 1989. In December 1989, the Commission issued a revised policy statement to restate its views with respect to maintenance and to indicate its intention to hold rulemaking in abeyance for a period of 18 months. During the 18-month time interval, the Commission monitored industry initiatives and progress in maintenance improvements and re-evaluated the need for issuing a final rulemaking. Based on its evaluation, the Commission concluded that a regulatory framework should be in place to provide a mechanism for evaluating the overall continuing effectiveness of licensee maintenance programs. Accordingly, the Commission issued a final rule, "Monitoring the Effectiveness of

Maintenance at Nuclear Power Plants" (10 CFR 50.65), on July 10, 1991.

The purpose of the maintenance rule is to require commercial nuclear power plant licensees to monitor the effectiveness of maintenance for safety-related and certain non-safety-related plant equipment, as defined in 10 CFR 50.65, in order to minimize the likelihood of failures and events caused by the lack of effective maintenance. The rule requires that licensees monitor the performance or condition of certain structures, systems and components (SSCs) against licensee-established goals, in a manner sufficient to provide reasonable assurance that those SSCs will be capable of performing their intended functions. Such monitoring would take into account industry-wide operating experience. Where monitoring proves unnecessary, licensees would be permitted the option of relying upon an appropriate preventive maintenance program.

The NRC staff will produce a regulatory guide on the subject, a draft of which will is expected to be released to the public document by spring of 1992. The nuclear industry (NUMARC) is producing a consensus guidance document for monitoring the effectiveness of maintenance in nuclear power plants, in parallel with the NRC staff effort. It is expected that this document will be available, at least in draft form, for NRC evaluation by the end of March 1992. The NRC will make a preliminary decision at that point whether to proceed with the staff-developed regulatory guide or endorse the industry guidelines. A final decision will be made at the end of July 1992.

Safety Goal Implementation

In 1986, the Commission published its Safety Goal Policy Statement. On June 15, 1990, the Commission directed the staff to routinely consider the safety goals in reviewing and developing regulations and regulatory practices. To realize this directive, plans have been established to develop a formal mechanism to ensure that future regulatory initiatives are evaluated for conformity with the safety goal policy requirements. While the Commission recognizes that consideration of the safety goal in assessing regulatory actions will initially engender a variety of results, the Commission also believes, as detailed guidance is developed and experience gained, that this variation will be minimal. In support of the implementation plan for the Safety Goal Policy Statement, a definition of a "large release" is being developed. When complete, this definition will provide a "sub-tier" element for use in reviewing and developing regulations and regulatory practices.

License Renewal

The NRC has been considering what requirements should be placed on nuclear power plants in the event that licenses to operate 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 the second as part of an advance notice of proposed rulemaking (published August 29, 1988). The advance notice requestedcomments on "Regulatory Options for Nuclear Plant License Renewal" (NUREG–1317, August 1988). Comments were summarized and analyzed in a "Survey and Analysis of Public Comments on NUREG-1317: Regulatory Options for Nuclear Plant License Renewal" (NUREG/CR–5332), issued in March 1989. The third time occurred when the NRC published the proposed rule for nuclear power plant license renewal on July 17, 1990 (55 FRN 29043). The final rule (10 CFR Part 54), with appropriate supporting documents, is expected to be published in late 1991.

As part of a separate rulemaking, the NRC has undertaken a generic environmental study with the purpose of defining the scope and focus of the 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 FRN 29964). 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 also issued (55 FRN 29967). The NRC published the proposed rule and draft GEIS for comment on September 17, 1991 (56 FRN 47016). It was also announced at that time that a public workshop would be conducted to review the technical basis of the proposed rule. The workshop was held in the fall of 1991.

The following support documents were issued with the proposed rule:

(1) NUREG–1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Draft for Comment, Volumes I and II, August 1991. This document provides the technical basis for the proposed rule.

(2) NUREG–1440, "Regulatory Analysis of Proposed Amendments to Regulations Concerning the Environmental Review for Renewal of Nuclear Power Plant Operating Licenses," Draft for Comment, August 1991. This regulatory analysis examines alternatives to the proposed rule and provides information that supports the alternative chosen.

(3) DG–4002, "Draft Regulatory Guide, Proposed Supplement 1 to Regulatory Guide 4.2, Guidance for

the Preparation of Supplemental Environmental Reports in Support of an Application to Renew a Nuclear Power Station Operating License," August 1991. This guide details the information that should be included in an application for license renewal.

(4) NUREG–1429, "Environmental Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Draft for Comment, August 1991. This report provides a framework for the NRC staff to determine whether or not environmental issues important to license renewal have been identified and the impacts evaluated and provides acceptance standards to help the reviewers comply with the National Environmental Policy Act.

The final Part 51 rule and GEIS are expected to be published in 1992.

SEVERE ACCIDENT IMPLEMENTATION

In the 10 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 the 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

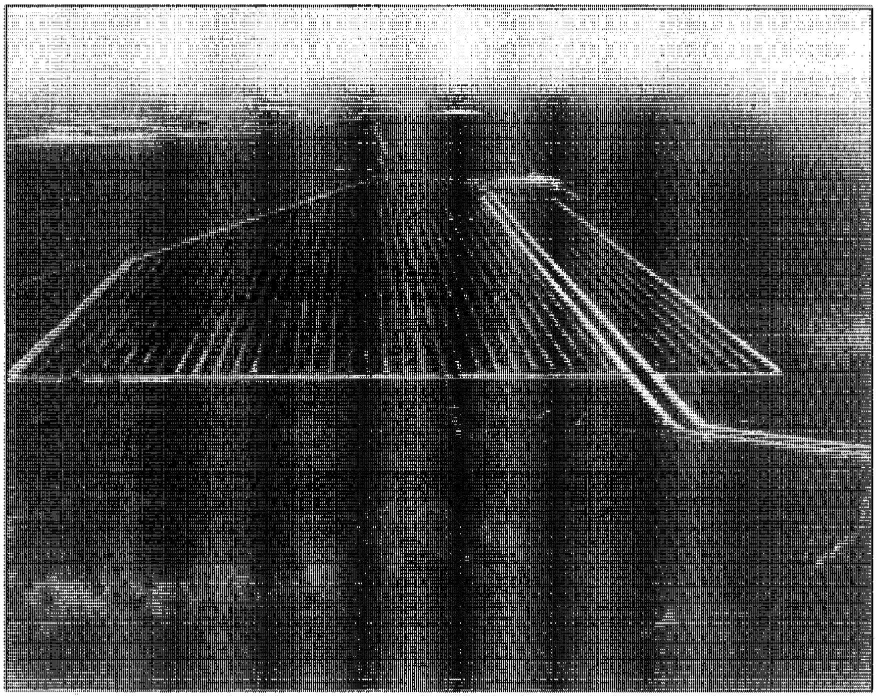
Severe accident research has generated a number of insights concerning containment performance during a severe accident. These have included both strengths and weaknesses of existing containment designs. In some cases, identified containment weaknesses or uncertainties in containment performance have 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 to identify containment vulnerabilities and to devise measures to correct those vulnerabilities. Because of concerns about Mark I containments, the CPI program initially studied these containments, leading to a requirement for BWR Mark I plants to 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 examinations (IPEs) and accident management programs. 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 starting the plant-specific backfit of the hardened vent 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 only remaining activity under this program is to finish and issue a series of NUREG/CR technical reports documenting the analyses and evaluations done by the staff and its contractors in assessing 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. To date, 11 out of the planned 12 reports have been issued, with the remaining report scheduled for issuance by November 1991.

Regulatory Application of New Source Terms

Consideration of source terms entered the regulatory process because the Commission's reactor site criteria (10 CFR Part 100) require that an accidental fission product release from the reactor core into the containment should be an assumed occurrence (for safety design purposes) and that its radiological consequences should be evaluated on the assumption that the containment leaks at its "expected demonstrable leak rate." The criteria for the release into the containment is derived from a 1962 report, TID–14844, which assumed an instantaneous release of fission products. Although this source term is



Pursuant to the Commission's Severe Accident Policy Statement, the NRC has required individual plant examinations (IPEs) of all existing plants to identify any plant-specific vulnerabilities to severe accidents. Among the new IPE submittals during fiscal year 1991 was that for the Turkey Point nuclear power plant, a two-reactor (PWR) facility in Dade County, Fla. Shown at left is the plant's cooling canal system, a layout of parallel canals comprising a total 168 miles. The plant (at the top center) and canal system are located about 25 miles south of Miami, on the western shore of Biscayne Bay.

included in the Commissions 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 gained. As a result, it has been recognized that a number of areas of regulatory activity may benefit from changes introduced as a result of source term and severe accident research. In fiscal year 1991, work continued on a replacement to TID–14844. It is expected that a draft report will be sent to the Commission in February 1992.

Update of Siting Regulations. In fiscal year 1991, the staff initiated a rulemaking which would decouple siting from plant design, for the purpose of more directly incorporating requirements related to acceptable site characteristics. It was expected that the proposed rule would be ready for Commission consideration by mid-1992.

Emergency Planning Regulations. In fiscal year 1991, the staff initiated rulemaking to add emergency planning requirements to 10 CFR Part 72 for independent storage of spent nuclear fuel and high-level radioactive waste. It is expected that a proposed rule will be sent to the Commission by mid-1992. Also in fiscal year 1991, a revision to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," was initiated to revise the approach for the development of Emergency Action Levels. A draft was expected to be issued for public comment in early 1992.

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, preparation of a generic letter to plant operators requesting the IPE, and development of review plans, and eventual review of the results of the IPE submittals. Imposition of any requirement to correct identified plant-specific vulnerabilities not voluntarily corrected would be governed by the backfit rule. Accident management is not part of the IPE process but can be affected by results derived from the process. Consideration of severe accident vulnerabilities from external hazards (earthquakes, flood, wind, etc.) is discussed below.

Major steps in the IPE effort during fiscal year 1991 have involved review of IPE submittals and completion of a procurement process to obtain contractual assistance for the IPE reviews. Three new IPE submittals were received from licensees for the Oconee (S.C.), Seabrook (N.H.), and Turkey Point (Fla.) facilities. The draft safety evaluation report was completed for the Yankee-Rowe (Mass.) submittal. To support the IPE reviews, three contracts were awarded to allow for a more in-depth review of select IPE submittals. Because the Turkey Point IPE submittal was the first one not based on a previously reviewed PRA, it is the first selected for the more in-depth review.

External Events

In December 1987, the NRC established an External Event Steering Group (EESG) to make recommendations concerning the individual plant examinations for vulnerabilities to severe accidents initiated by external events (e.g., earthquakes, floods, fires). Recommendations were needed with respect to (1) what external events need consideration in the IPE; (2) what methods can be used in the examination; and (3) how the IPE for external events (IPEEE) can be coordinated with other ongoing regulatory activity involving external events, particularly in the seismic area.

Three subcommittees were established in April 1988 to make recommendations in the areas of (1) seismic, (2) fires, and (3) high winds, flood, and others (e.g., man-made hazards such as nearby transportation and military and industrial facilities). During 1989, the three subcommittees completed their studies and made recommendations for the IPEEE to the EESG.

In May 1990, the staff completed work on a draft generic letter and draft guidance document (NUREG-1407), to be sent to licensees, which describes the scope, acceptable methods, and reporting requirements for the IPEEE. The draft documents were issued for public comment on July 25, 1990. In September 1990, the staff conducted a workshop on the draft generic letter and on NUREG-1407 to solicit comments and answer questions concerning their content. Approximately 210 representatives from industry, State agencies, and the public attended the workshop. The staff revised the generic letter and NUREG-1407 to clarify and incorporate changes resulting from feedback received at the workshop and subsequently issued the final generic letter, GL 88-20, Supplement 4, and NUREG-1407, in June 1991. The generic letter requires licensees to submit their plans and schedules for performing their IPEEEs in December 1991, with completion of their IPEEEs by June 1994.

In August 1991, the staff completed its review of an EPRI/NUMARC fire evaluation methodology, "Fire Vulnerability Evaluation Methodology (FIVE)," and issued an evaluation report endorsing the use of FIVE as a viable alternative to fire PRA in the IPEEE process.

The staff is currently developing a review plan for the IPEEE submittals. It is expected that the approach for review of the IPEEE will follow closely that developed for review of the internal-event IPE submittals.

RADIATION PROTECTION AND HEALTH EFFECTS

The NRC maintains a program of research and standards development in radiation protection dedicated to ensuring continued protection of workers and the public from radiation and radioactive materials in connection with licensed activity. 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 licensees. The program also contributes to the monitoring of licensee performance in such tasks as controlling occupational dose through the use of new dose reduction techniques.

The primary focus of the health effects research program is to reduce the uncertainty associated with estimating health effects from exposure to radiation. Currently, besides conducting its own studies, the NRC staff reviews research funded by other agencies—such as the Department of Energy (DOE) and the Department of Health and Human Services—and attempts to improve understanding of this critical area. Improved risk estimations are needed for establishing radiation protection policy and standards, for assessing severe accident consequences, and for implementing agency safety goals.

Radiation Protection Issues

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 during the report period. (ALARA is an acronym referring to the regulatory goal of reducing radiation exposures to a level "as low as reasonably achievable.") BNL has published a series of reports (NUREG/CR-3469) that abstracts material from 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 1991, BNL focused on providing guidance to high dose worker groups and developing an international dose reduction data base.

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. This effort becomes even more important with the implementation of the new Part 20, making ALARA a requirement.

In 1991, 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. This work will provide a technical base for future NRC regulatory decisions regarding further changes in worker dose limits.

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 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 July 1, 1991, a total of 67 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. A draft regulatory guide that will discuss methods of meeting the NVLAP procedures for processor accreditation will be published for comment early in fiscal year 1992.

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 early in 1992. Tests against the revised standard (HPSSC P/N 13.32) have begun, and testing is expected to continue through June 1992. Twenty-four facilities are expected to participate. Should the tests indicate that the revised standard is a suitable criterion for testing, appropriate rulemaking will be initiated to require extremity dosimeters to be processed by processors certified under the NVLAP procedures in use at NIST.

New Skin Dose Computer Code. A new computer code for calculating dose to the skin from radioactive materials on the skin will be published in 1992. The code will replace the VARSKIN code, in use since 1986. The new 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 dose from different shapes of particles and particles separated from the skin by clothing. The code will calculate the dose from both gamma and beta radiations.

Self-Powered Photon Detector. Research under contract 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. The contractor has applied for a patent application for the use of this detector. A final report on this research will be published shortly, as NUREG/CR–4833.

Tissue Equivalent Thermoluminoscent Dosimeters. Research under contract to develop a gamma-ray spectrometer/dosimeter has begun. The purpose 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 feasibility of the concept using a four-detector cadmium telluride assembly, but some detector leakage problems arose that prevented making low dose measurements. The problems have now been corrected, and it is projected that the Phase II research will provide adequate measurements, leading to development of a commercial prototype under Phase III.

"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 contractual research. Under Phase II of the contract, a prototype of a system for surveying clothing has been successfully demonstrated. It is expected that this system will be marketed for commercial use. The system has potential for reducing radiation exposure of personnel who may wear "clean" protective clothing and be unaware that the clothing bears particulate radioactive material.

Health Effects Research

Embryo/Fetal Dose from Maternal Intake. A study to improve understanding of the contribution of maternal radionuclide burdens to pre-natal radiation exposure was continued in fiscal year 1991, with significant progress. The NRC has published for comment a report entitled "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses" (NUREG/CR–5631). The report provides a methodology for calculating internal doses to the embryo/fetus and a data base for selected radionuclides. Work is currently under way on re-issuance of NUREG/CR–5631, with an expanded data base that will include uranium and other isotopes of previously described elements, such as strontium-89, cesium-134, or plutonium-238. Research that will permit inclusion of other radionuclides, such as technetium, molybdenum, americium and other transuranic elements, is planned. The methods and data developed under this project will be used by the NRC in preparation of an interim regulatory guide 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 methodology will also be used to calculate doses in cases of accidental releases of radioactive materials.

Improvement of Health Effects Models. A revision to the document "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis" (NUREG/CR–4214, Revision 1), 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, entitled "Modification of Models Resulting From Recent Reports of Health Effects of Ionizing Radiation," was published in August 1991. The documents that led to the revision of models presented in the 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).

Cellular and Molecular Biology. Based on the discovery of oncogenes, the development techniques of recombinant DNA molecular biology, and the progress that has been made in the characterization of certain human cancers in genetic terms, the NRC sponsored a feasibility study aimed at reduction of uncertainties in risk coefficients. The results of this study were reported in "Cellular and Molecular Research to Reduce Uncertainties in Estimates of Health Effects from Low-Level Radiation" (NUREG/CR–5635).

The study concluded that it is feasible to reduce uncertainties of radiation-induced health effects by mounting a program of radiation research directed at the mechanism(s) of radiation-induced cancer, with special reference to risk of neoplasia associated with protracted, low doses of sparsely ionizing radiation. The study has been distributed to Federal agencies, the National Academy of Science, the Radiation Effects Research Foundation, and individual scientists.

Chemical Toxicity of Uranium Hexafluoride Compared to Radiation Doses (NUREG–1391). This staff report compared the chemical toxicity of uranium hexafluoride with the acute effects of a radiation dose of 25 rems to the whole body (the value used in Part 100, dealing with reactor siting criteria). The work will be used to support development of licensing requirements for commercial uranium enrichment facilities. The draft report was published for comment in April 1990; a final report is scheduled for publication in 1992.

Development of Rules and Regulatory Guides

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 1989 revealed that about 203,000 persons were monitored, of whom about 53 percent received measurable doses. The workers received a collective dose of approximately 36,200 person-rems or an average annual dose of about 0.33-rem-per-worker among those receiving a measurable dose. These figures are about 10–15 percent lower than those found for 1987. 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. Preliminary compilations of the exposure data reported by nuclear power plants for calendar year 1990 are not significantly changed. (One additional reactor was reported on during this period.)

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 575,000 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 75,400 persons terminating employment during 1988 revealed that 3,622 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.

Revision of Part 20 Radiation Standards. The Commission has approved using 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). The revision updates the Commission's regulations to incorporate recommendations made by the ICRP, the NCRP, and the revised Federal Radiation Guidance for Occupational Exposure issued in 1987. The new standards represent a significant change from the methods previously employed to assess and control radiation doses. The new Part 20 will result in a reduction of the permissible annual occupational dose from a possible 17 rems (three-rem/quarter external plus five-rem annual internal) to a total effective dose of five rems-per-year. The dose limit for members of the general public is reduced from an implicit 0.5 rem-per-year in the present rule to an explicit value of 0.1 rem-per-year. The new Part 20 contains appendices that give the radionuclide concentration limits for air, water and sewage.

Proposed Rule on Large Irradiators. A proposed rule for large irradiators was published for public comment in the *Federal Register* in December 1990 (55 FR 29043). A two-day public workshop to discuss the proposed rule was held in Rockville, Md., in February 1991. 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. A final rule on the subject is scheduled for publication in fiscal year 1992.

Certification of Industrial Radiographers. A final rule that would recognize a third-party certification program of the American Society for Nondestructive Testing (ASNT) was published in the *Federal Register* in March 1991 (56 FR 11504). The rule would give licensees the option of using the ASNT program in lieu of describing their training program to NRC. The certification program is expected to improve both training and safety performance in the workplace.

Air Sampling in the Workplace. A proposed regulatory guide, "Air Sampling in the Workplace," to meet the requirements of the new Part 20 was published in the *Federal Register* (56 FR 52078) for public comment in September 1991. The guide deals with such issues as what a licensee should do to demonstrate that samples are representative of the air inhaled by workers, and what measurements are necessary to be able to adjust derived air concentrations to account for particle size. The guide is accompanied by a technical manual, "Air Sampling in the Workplace," describing how the recommendations in the guide can be met. Both documents are scheduled to be issued in final form in fiscal year 1992.

Fuel Cycle. A proposed rule, published in the *Federal Register* (56 FR 46739) for public comment in September 1991, would amend the Commission's regulations 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. The principal effect of these amendments is that uranium enrichment facilities would be licensed subject to the provisions of the 1954 Act pertaining to source material and special nuclear material rather than under the provisions pertaining to a production facility. The Commission is currently reviewing a license application by the Louisiana Energy Service Corporation to construct and operate a commercial uranium enrichment facility. (See "Uranium Enrichment," under Fuel Cycle Licensing and Inspection, in Chapter 4.)

The staff is continuing to follow emerging technologies for uranium enrichment and other fuel cycle facilities for potential radiological, chemical, and criticality safety concerns. Technical support was provided for LLW and other non-fuel-cycle areas. "Ice-Condenser Aerosol Tests" (NUREG/CR-5768) was published and presents the results of an experimental investigation of aerosol particle transport and capture using a full scale height and reduced scale cross-section test facility, based on the design of the ice compartment of a PWR ice-condenser containment system. Results of 38 tests encompasses thermal-hydraulic as well as aerosol particle data.

Decommissioning

A proposed rule was published in the *Federal Register* (56 FR 50524) for public comment in September 1991 to amend the Commission's decommissioning regulations and require holders of a specific license for possession of byproduct material, source material, special nuclear material, and for 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 with an earlier GAO report.

A Notice of Receipt of Petition for Rulemaking was published (56 FR 4845) on a joint petition by the General Electric Company and the Westinghouse Electric Corporation, requesting that the Commission amend its decommissioning regulations and provide a means for self-guarantee of decommissioning funding costs by certain non-electric utility reactor licensees, who meet stringent financial assurance and related reporting and oversight requirements. As requested by the Commission, the notice also solicits public comments on other self-guarantee criteria, if any, and the basis for the criteria, with more information on self-guarantee.

Four reports associated with reactor decommissioning technology and costs have been published. These are "Radionuclide Characterization of Reactor Decommissioning Waste and Spent Fuel Assembly Hardware" (NUREG/CR-5343); "Re-evaluation of the Cleanup Cost for the Boiling Water Reactor (BWR) Scenario 3 Accident from NUREG/CR-2601" (NUREG/CR-2601, Addendum 1); "Report on Waste Burial Charges" (NUREG/CR-1307, Revision 2); and "Comparison of Two Decommissioning Cost Estimates Developed for the Same Commercial Nuclear Reactor Power Station" (NUREG/CR-0672, Addendum 4). The final regulatory guides on standard format and content of plans for reactor decommissioning and reactor decommissioning record-keeping are in preparation.

A Commission Paper, "Decommissioning Costs" (SECY-91-164), was completed on May 31, 1991. Staff work in developing information on the safety, costs and wastes related to the decommissioning of light-water reactors (LWRs) and other nuclear facilities is progressing according to schedule. As stated in SECY-91-164, the staff expects the completion of revised cost estimates for LWRs by October 1993.

NATIONAL STANDARDS PROGRAM

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 1991, the NRC staff continued its active participation in the national standards program, particularly with respect to setting priorities. NRC participation derives from a need for national standards to define acceptable ways of implementing the NRC's basic safety regulations. Approximately 193 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 1991, the NRC published a proposed rule in the *Federal Register* that would amend paragraph 50.55a to update references to ASME Code Section III and Section XI for the purpose of incorporating improved rules for the construction, inservice inspection, and inservice testing of nuclear power plant components. The proposed rule also would expedite implementation of certain new Section XI requirements for examination of the reactor vessel.

ASME Code Cases provide alternatives to the rules specified in the ASME 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 1991. Revisions were also initiated to Regulatory Guide 1.36 on non-metallic 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.

Proceedings And Litigation

Chapter

9

This chapter covers salient activities, proceedings and decisions involving the NRC's Atomic Safety and Licensing Boards, as well as noteworthy Commission decisions in cases under litigation. The chapter closes with a review of significant litigation involving the NRC during the report period, in cases both pending and decided.

Office of the Secretary. The Secretary of the Commission maintains the official NRC adjudicatory and rulemaking dockets for the Commission. The adjudicatory dockets contain the filings of all parties to the Commission's licensing and enforcement proceedings; transcripts of the adjudicatory hearings held in each case; and all Orders and Decisions issued by the Commission, or the Commission's Atomic Safety and Licensing Boards. The rulemaking dockets contain the comments of members of the public on newly proposed agency rules and rule amendments, as well as comments on specific petitions for rulemaking and NRC/State Agreements on which the NRC seeks views before taking final action.

The Docketing and Service Branch also serves Orders of the Commission and the Atomic Safety and Licensing Boards on parties to proceedings and certifies indexes of the dockets to the courts.

ATOMIC SAFETY AND LICENSING BOARDS

Adjudicatory hearings at the Nuclear Regulatory Commission are conducted by administrative judges sitting alone or in three-member Licensing Boards. The judges are drawn from the Atomic Safety and Licensing Board Panel created by the Commission in 1962 under the authority of Section 191 of the Atomic Energy Act. The panel's judges hold both legal and technical expertise in a variety of disciplines.

The Atomic Energy Act requires that a hearing precede every issuance of a construction permit for a nuclear power plant or related facility. The Commission's nuclear power plant licensing proceedings have been characterized as among the most complex and controversial administrative hearings conducted by the Federal Government. Under the Act, or by Commission rules, an opportunity for a hearing must be provided on such matters as antitrust issues, enforcement actions, civil penalties, operating licenses or other matters the Commission directs to be heard. These hearings are the Commission's principal public forum in which individuals and organizations can voice their concerns and have them adjudicated by an independent tribunal. They also provide a means for NRC license holders to contest Commission actions.

While Licensing Boards consisting of three administrative judges are required for cases concerning commercial nuclear power reactors and related facilities, a broad range of other matters may be heard by a single administrative judge or administrative law judge from the panel. Commission appointments to the panel are based upon recognized experience, achievement and independence in the appointee's field of expertise. The Commission or the panel's Chief Administrative Judge assigns individual judges to those hearings where their professional expertise will assist in resolving the particular technical and legal matters at issue. During fiscal year 1991, the panel comprised 41 administrative judges (15 full-time and 26 part-time). By profession, they included 12 lawyers, 10 public health and environmental scientists, 16 engineers or physicists, and three medical doctors. (See Appendix 2 for the names and disciplines of fiscal year 1991 panel members.)

ASLBP Caseload. During the fiscal year ending September 30, 1991, the panel's proceedings involved a total of 48 cases. Twenty-five cases were related to nuclear power plants or related facilities, and 23 cases dealt with other Commission licensees. Twenty-four proceedings were closed and 30 new proceedings were docketed during the report period.

Automation. In fiscal year 1991, the panel continued intensive efforts to thoroughly automate the hearing process. Driven by continued restrictions in the number of available support personnel, as well as by the panel's on-going program to reduce delays in the licensing process, the panel has moved rapidly in recent years to achieve an "electronic" office, especially in management of its voluminous and complex hearing records. To fully exploit its computer resources in 1991, the panel continued to employ the code INQUIRE, an electronic docket conceived, developed and maintained by the panel. INQUIRE contains an adjudicatory data base and a companion search-and-retrieval interface which operate on an IBM 9370 mini-computer. Among other things, ASLBP decisions are entered into INQUIRE the day of issuance and are thus immediately available throughout the agency. The panel also upgraded some personal computer support equipment, completed word processing standardization, and developed a number of ways to expedite various computer operations. Finally, the panel completed the software research needed to duplicate the mainframe INQUIRE functions on a personal computer system and began testing of a prototype.

Case Management. Besides these measures to computerize the litigation process, the panel continued to apply traditional case management tools and techniques in an effort to streamline, focus, and resolve contested licensing matters. Licensing Boards frequently structure their hearing schedules into distinct phases, each dealing with discrete groupings of related issues. In complex proceeding involving several topics and multiple issues, the panel frequently creates separate, parallel Licensing Boards and assigns one or more discrete topics to each board. Not only do those parallel adjudications save time, but also panel members' expertise can be more precisely matched to the issues to be resolved.

Licensing Boards continued to take an active role in shaping the issues before them by such methods as consolidating admissible contentions, monitoring the discovery portion of the proceeding, and fostering a free exchange of views among the parties conducive to a possible settlement of disputed issues. By these means, the vast majority of proposed contentions are resolved prior to hearing.

Licensing Boards also have had considerable success in aiding in the settlement of docketed cases before final adjudication. During fiscal year 1991, significant litigation expenses were avoided by settlements of cases involving *Cambridge Medical Technology Corporation,* Order of October 19, 1990; *Cleveland Electric Illuminating Company* (Perry, Unit 1), LBP–90–39, 32 NRC 368 (1990); *American Radiolabeled Chemicals, Inc.,* Order of November 5, 1991; *St. Mary Medical Center,* LBP–90–46, 32 NRC 463 (1990); *Northern States Power Company* (Prairie Island, Units 1 and 2), LBP–91–8A, 33 NRC 210 (1991); *Cintichem, Incorporated,* Order of March 14, 1991; *Tennessee Valley Authority* (Sequoyah, Units 1 and 2), LBP–91–10, 33 NRC 231 (1991); *Barnett Industrial X-Ray,* LBP–91–16, 33 NRC 274 (1991); *Vermont Yankee Nuclear Power Corporation* (Vermont Yankee nuclear power plant), Order of September 3, 1991; and *Arizona Public Service Company* (Palo Verde Units 1, 2 and 3), LBP–91–37, __ NRC__ (1991).

As the need for proceedings on initial operating license for power reactors remains in abeyance, the panel has turned its attention to the increasing number of enforcement and informal proceedings on its docket. This caseload derives from the agency's regulatory responsibility for the more than 100 nuclear plants in operation in the United States, as well as a demand for NRC staff oversight of over 7,000 materials licensees. Informal proceedings, usually involving materials licenses, generally rely on the decisions of a single administrative judge, in creating and shaping the record of the proceeding. In such proceedings—e.g., informal proceedings under 10 CFR Part 2, Subpart L—a hearing is conducted only as to those issues that the administrative judge cannot resolve on the basis of the parties' written submissions, or to develop additional information deemed relevant by the judge.

The panel has continued the policy, in such proceedings, of assigning a legal or technical administrative judge from the panel as an assistant to the presiding administrative judge, so that, while the benefits of an informal procedure are preserved, so is the the availability of expertise associated with the traditional three-member Licensing Boards.

Shoreham Nuclear Power Plant

During fiscal year 1991, a Licensing Board issued a number of decisions related to the Shoreham (N.Y.) nuclear plant, responding to motions and hearing requests filed by the Shoreham-Wading River Central School District and the Scientists and Engineers for Secure Energy. These petitioners, who have wanted to see the facility brought into operation, expressed opposition to certain actions of the NRC, invoking an agreement between the licensee for the Shoreham, the Long Island Lighting Company (LILCO), and the State of New York that LILCO would not operate Shoreham and would sell Shoreham to the Long Island Power authority for subsequent decommissioning.

LILCO had not requested decommissioning during fiscal year 1991, but it did request and was granted certain license amendments and a "possession only" license by the NRC, based on a "no hazard" determination that allowed Shoreham to be maintained more economically. The petitioners opposed these amendments and the "possession only" license, contending that such licensing actions would constitute *de facto* decommissioning, requiring the NRC to prepare environmental impact statements. They also claimed that any environmental impact statements must consider the operation of Shoreham as one of the cost benefit options. Based on several interim rulings by the Commission indicating, among other things, that resumed operation of Shoreham was not an option, the Licensing Board rejected most of petitioners' requests, including all of contentions submitted for litigation during this period (*Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant), LBP–91–1,

33 NRC 15 (1991); LBP-91-7, 33 NRC 179 (1991); LBP-91-23, 33 NRC 430 (1991); LBP-91-26, 33 NRC 537 (1991); LBP-91-32, 34 NRC 132 (1991)). However, the board did find that one of the petitioners had standing to intervene (see discussion on organizational standing, below).

Experiments with Americium and Plutonium

In a proceeding concerning a university testing facility, the presiding officer granted a license for experiments with unencapsulated americium and plutonium (*Curators of the University of Missouri* (Trump S Project, LBP-91-31, 34 NRC 29 (1991)). To assure safety, the presiding officer ordered that fire extinguishers be installed and that the licensee's procedures be modified to reduce the risk of a serious fire that might disperse nuclear materials.

Late Intervention: Ignorance of the Law

In a proceeding involving a Massachusetts company engaging in the conversion of depleted uranium, the presiding officer dismissed as untimely a petition for a hearing, since the petitioners had not requested a hearing within 30 days of receiving actual notice of the application and had not demonstrated an adequate excuse for untimeliness, as required by the regulations (*Nuclear Metals, Inc.*, LBP-91-27, 33 NRC 548 (1991).) The petitioners argued that they lacked notice because the public information about the application did not include notice of the right to oppose the application. In dismissing the petition, the presiding officer held that the principle that "ignorance of a law is not an excuse for violating it" applies to the regulatory timeliness provisions, particularly where a petitioner has sufficient knowledge to inquire further. He also noted that, in this case, the petitioners had received actual notice of the application eight months prior to filing their request for hearing.

Shown above is a view of the Shoreham (N.Y.) nuclear power plant, looking south from the waters of Long Island Sound. Years of NRC Licensing Board and Commission decisions, as well as numerous petitions and extensive litigation involving the controversial plant, have eventuated in the conclusion that the facility will not be put into service. (See separate discussions in this chapter under "Atomic Safety and Licensing Boards," "Commission Decisions," and "Judicial Review.")

Standing

Presumption of Standing Based on Close Proximity to a Facility. For purposes of determining whether an intervenor has standing, the possibility of injury has traditionally been presumed to extend, in NRC proceedings, to intervenors living within 50 miles of the nuclear facility. In a license amendment proceeding involving the Palo Verde (Ariz.) nuclear facility, a party contended that the 50-mile presumption should only apply to construction permit or operating license proceedings which involve wide-ranging activity that can potentially affect a large geographic area. Because license amendment proceedings are usually much more limited in scope, the party claimed that the petitioner, a resident of Tempe, Ariz., must allege the specific injury that could occur from the affected activity. The Licensing Board disagreed and found that the petitioner did not have to show specific injury if there was a potential for off-site consequences. The board found this potential present at Palo Verde because the license amendment involved changes to several systems which were important to safety (*Arizona Public Service Company* (Palo Verde, Units 1, 2 and 3), LBP–91–4, 33 NRC 132 (1991)).

Organizational Standing. In *Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant, LBP 91–32, 34 NRC 132 (1991)), the Licensing Board found that a New York organization had standing to intervene in a license amendment proceeding involving the Shoreham facility. Historically, an organization establishes standing in an NRC proceeding after some of its members, who could be injured by the action in question, authorizes it to represent their interests. However, the board concluded that this organization had standing on its own, based on its organizational function of disseminating information to its members. Specifically, this organization was unable to act on information essential to its activity when the NRC failed to issue environmental impact statements for several licensing actions. In granting standing, the board recognized that NEPA's purpose of ensuring well informed government decisions and stimulating public comment on agency actions effectively lowers the threshold for establishing injury to informational interests.

Inferred Standing. In a license amendment proceeding involving the *Georgia Power Company* (Vogtle Units 1 and 2, LBP–91–33, 34 NRC 138 (1991)), a local intervenor had participated in an earlier NRC proceeding involving the same nuclear facility. The board did not require this intervenor to again establish standing, because standing had already been established in the earlier case.

Technical Specification Amendments

In *Georgia Power Company* (Vogtle, Units 1 and 2, LBP–91–21, 33 NRC 419 (1991)), a local organization contended that a technical specification amendment, involving a plant modification by a licensee, should not be allowed, because there was a better way of making the modification. The licensee's method comported with current NRC regulatory guidance. In dismissing the contention, the board concluded, as a matter of law, that if regulatory requirements were met, the board could not limit a licensee's choice of actions, even if one method was clearly better than the other.

Civil Penalties

In *Fewell Geotechnical Engineering, Ltd.,* (LBP–91–29, 33 NRC 561, (1991)), the staff ordered a radiographer to be suspended from his job for three years for violating operating procedures and not being truthful. The Licensing Board modified the order by reducing the period of suspension to nine months and requiring the radiographer to serve three months as a radiographer's assistant, before resuming work as a radiographer. In reducing the penalty, the board differentiated between types of willful misconduct. The willfulness in question here—lying while in a panicked and in stressed state of mind—was not deemed as culpable as in those cases where individuals have intentionally plotted to deceive the NRC.

Written Testimony

In the *Tulsa Gamma Ray, Inc.* (LBP–91–25, 33 NRC 535 (1991)) civil penalty proceeding, a party requested that the licensee, an Oklahoma radiography company, be required to file written testimony, as opposed to being able to use live testimony, at the hearing. The Licensing Board held that the licensee in a civil penalty case has a right to present live testimony where credibility is a significant factor.

Inspection Fees

In a show-cause proceeding seeking license revocation for failure to pay an NRC inspection fee, a Missouri-based byproduct material licensee had requested a waiver of that fee on the ground that its licensed equipment was used exclusively for government projects (*Rhodes-Sayre & Associates, Inc.*, LBP–91–15, 33 NRC 535 (1991)). The Licensing Board considered this request and also took up the question whether the staff should have imposed some lesser sanction than license revocation. The board concluded that there was no abuse of staff discretion in either instance, and also found that the enforcement actions taken were consistent with other similar NRC actions and with the Commission's regulations.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1991 are discussed below.

Yankee-Rowe Nuclear Power Plant

In June 1991, the Union of Concerned Scientists and the New England Coalition on Nuclear Pollution filed a Petition for Emergency Enforcement Action and Request for Public Hearing with the Commission, seeking the immediate shutdown of the Yankee-Rowe (Mass.) nuclear plant. They asserted that the plant was in violation of NRC requirements for pressure vessel integrity and that, therefore, the Commission could have no reasonable assurance that the facility did not pose undue risk to public health and safety. Petitioners also asserted that the NRC staff had acquiesced in the licensee's noncompliance with NRC requirements, for which reason petitioners requested emergency action from the Commission itself, rather than requesting action from the Executive Director of Operations, under 10 CFR 2.206. They also asked the Commission to refrain from further ex parte contact with the staff and with Yankee-Rowe Atomic Energy Company and to order an adjudicatory hearing to determine facility compliance with NRC regulations.

In *Yankee Atomic Electric Company* (Yankee-Rowe nuclear power plant, CLI- 91-11, 34 NRC 3 (1991)), the Commission held that it would rule directly on this petition, since it always retains the power to take jurisdiction in any petition before it, and because the question had sufficient public importance to warrant this action. The Commission denied the request on *ex parte* communications, noting that its rules on *ex parte* do not formally apply until a notice of hearing is filed.

The Commission held that the NRC staff had been correct in denying petitioners' initial request for immediate shutdown, in view of staff's conservative analysis of the risk posed by Pressurized Thermal Shock (PTS) events, and because the licensee was in compliance with 10 CFR Part 50, Appendix G, and with the safety considerations of Appendix H. The Commission also agreed with the staff's conclusion that it was imprudent to permit continued operation beyond the end of Cycle 21 (approximately April 1992) pending resolution of PTS uncertainties.

The Commission noted that an immediate plant shutdown would not contribute to a quicker resolution of the PTS uncertainties. In light of the circumstances, as a matter of prudent regulatory judgment, the Commission ordered that the following actions be taken: (1) that the licensee submit to the NRC its evaluation and plan of modifications to its operating conditions that will provide a greater safety margin against reactor vessel failure from a PTS challenge, by a factor of 5–to–10, through a mix of hardware modifications, human resource allocations, and operating procedure modifications; (2) that the NRC staff review the licensee's evaluation and promptly report the results to the Commission; if the staff concludes that the modifications are acceptable, it shall prepare a confirmatory order, to be issued upon Commission approval, directing the licensee to make the modifications in plant operating procedures within two weeks of the order; (3) that, if the licensee or staff determines that proposed modifications in procedure would not be effective, the staff return to the Commission for further guidance; (4) that the licensee submit its plan to resolve uncertainties in the chemical and metallurgical characteristics of the reactor vessel; (5) that the staff monitor licensee's implementation of the test plan and advise the Commission of the earliest date that the tests may be run; (6) that the licensee make monthly reports to staff and petitioners; (7) that petitioners be kept informed of developments, have access to all relevant documents, and attend all meetings; and (8) that in no event would operation of Yankee-Rowe continue past April 15, 1992, without Commission approval.

State of Illinois Request

In 1989, the State of Illinois requested that its Agreement with the NRC be amended to provide Illinois with regulatory authority over uranium and thorium mill tailings. In support of its request, Illinois submitted its radiation control program for NRC staff assessment. The staff published its assessment for public comment on March 28, 1990 (55 FR 11,459). The Kerr-McGee Chemical Corporation holds an NRC license for the West Chicago Rare Earths Facility, which contains a large quantity of thorium mill tailings. Kerr-McGee filed comments objecting to the proposed amended agreement and also filed a motion demanding a full adjudicatory hearing on the effect on operation of the West Chicago facility of any Illinois requirements which differ from NRC requirements.

In *State of Illinois* (Amendment Number One to the Section 274 Agreement between the NRC and Illinois), CLI–90–09, 32 NRC 210 (1990)), the Commission denied Kerr-McGee's petition for a hearing and announced it had decided to enter into an amended agreement with Illinois. The amended agreement became effective on November 1, 1990 (55 FR 46,591). In denying Kerr-McGee's motion for a hearing, the Commission agreed that section 274o of the Atomic Energy Act requires the Commission to provide an opportunity for hearing on the effect of Illinois' differing requirements at the West Chicago site but found that this site-specific obligation would arise only later, when and if Illinois seeks to impose such differing requirements. Kerr-McGee's subsequent petition for reconsideration was also denied.

A separate, but related, adjudicatory proceeding, which concerned an amendment to Kerr-McGee's NRC license which would permit Kerr-McGee to bury the tailings waste on-site, remains pending before the Commission. On February 28, 1991, the Appeal Board issued a decision in this proceeding vacating the license amendment which had been authorized by the Licensing Board (*Kerr-McGee Chemical Corporation,* ALAB-944, 33 NRC 81 (1991)). Kerr-McGee petitioned for Commission review of this decision. On July 3, 1991, the Commission, upon joint motion of the parties, issued an order holding this proceeding in abeyance for six months to allow the parties time to reach a negotiated settlement.

Shoreham Nuclear Power Plant

The Commission issued a number of significant decisions during the report period pertaining to the Shoreham (N.Y.) nuclear power plant.

In March 1990, the NRC staff had issued a Confirmatory Order Modifying License prohibiting the Long Island Lighting Company ("LILCO") from "placing any nuclear fuel in the Shoreham reactor vessel without prior approval of the NRC" (55 FR 12,758 (April 5, 1990)). The staff had also published notices allowing changes in the physical security plan for the plant (55 FR 10,528 (March 21, 1990)) and easing off-site emergency preparedness standards (55 FR 12,076 (March 30, 1990)). Scientists and Engineers for Secure Energy (SE2) and Shoreham-Wading River School District (Shoreham-Wading) filed a Petition to Intervene and Request for Hearing, in response.

Petitioners sought various remedies, including an order directing the staff to prepare an Environmental Impact Statement (EIS) on the proposed decommissioning and, in the EIS, to consider resumed operation as an alternative to decommissioning. They claimed that the actions taken by LILCO and the staff amounted to a *de facto* decommissioning, triggering the imposition of requirements set forth in the National Environmental Policy Act (NEPA).

In *Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant, CLI-90-8, 32 NRC 201 (1990)), the Commission held that the Atomic Energy Act and NEPA did not require the NRC to consider the plant's resumed operation as an alternative to decommissioning in an environmental review. The Commission noted that basic NEPA principles require that an agency consider "reasonable" alternatives (*NRDC v. Morton,* 458 F.2d 827, 834 (D.C. Cir. 1972)), but that the Commission was not "to devote itself to extended discussion of the environmental impact of alternatives so remote from reality as to depend on, say, the repeal of the antitrust laws." The Commission also stated that "[u]nder NRC regulations, the NRC must approve of a licensee's decommissioning plan...but nowhere in our regulations is it contemplated that the NRC would need to approve of a licensee's decision that a plant should not be operated." The Commission also noted that, except in highly unusual circumstances, not present here, "the NRC lacks authority to direct a licensee to operate a licensed facility." The Commission also found that "resumed operation" was not a "reasonable" alternative to decommissioning, under the NEPA "rule of reason."

SE2 and Shoreham-Wading next petitioned the Commission for a hearing when LILCO requested an amendment that would change its license from one to "possess, use, and operate" Shoreham to one to "possess, use, but not operate the facility" (*Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant), CLI-91-1, 33 NRC 1 (1991)). Petitioners claimed this requested amendment constituted a "possession only" license, which the Commission must deny, because LILCO had not yet submitted its decommissioning plan pursuant to 10 CFR 50.82(a).

The Commission agreed with petitioners that granting LILCO's request would convert its license to a "possession only" license. However, the Commission stated that neither regulations, NEPA, nor policy considerations required a decommissioning plan to be submitted in conjunction with a "possession only" license application. The Commission then forwarded petitioners' request to intervene, and its own guidance regarding the "possession only" license, to the Licensing Board.

SE2 and Shoreham-Wading then filed a petition for reconsideration of CLI- 90-8, 32 NRC 201 (1990). *Amicus curiae* briefs were filed by the Secretary of Energy (DOE), the Council on Environmental Quality, the Long Island Power Authority (LIPA), and the State of New York. The Commission denied this petition because petitioners failed to demonstrate any legal flaw in CLI-90-8 *Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant), CLI-91-2, 33 NRC 61 (1991)).

In view of world events taking place at the time (armed conflict in the Persian Gulf), the Commission decided to issue guidance to the parties regarding potential requests for NRC action under sections 108, 186(c), or 188 of the AEA. The Commission stated that, under the Energy Reorganization Act of 1974, prior action by DOE is necessary before the NRC can act under any of the three sections. Under section 108, once Congress declares war or a national emergency, DOE must issue a finding that it is necessary to the common defense and security to order operation of the plant. Under section 186(c), after the revocation of any license, DOE must issue a finding that operation of the facility is "of extreme importance to the national defense and security" of the United States. Finally, under section 188, again after revocation of a license, DOE must issue a finding that plant operation is necessary for DOE's "production program," file a petition with the NRC requesting ordering of operation and explaining who would bear the expenses for just compensation to the utility for its expenses.

Finally, in *Long Island Lighting Company* (Shoreham Unit 1 nuclear power plant, CLI-91-8, 33 NRC 461 (1991)), the Commission declined petitions to reconsider CLI-90-8 and CLI-91-2; it further denied a request for the NRC staff to cease review of pending matters and to hold all future Shoreham proceedings in abeyance pending completion of proceedings in the New York Court of Appeals.

The Commission acknowledged there was a "non-trivial" possibility that the Settlement Agreement would be modified or vacated by the court. However, the Commission held that such a modification or vacation would not have an adverse impact on the primary holdings in both its rulings, i.e., that the decision not to operate Shoreham was a private one and that NEPA only requires the NRC to consider alternative methods of decommissioning. The Commission stated "[t]hus there is nothing before the New York Court of Appeals that is central to our decision."

Acting on the staff's recommendation that Shoreham's license be converted to a "possession only" license, the Commission approved that recommendation, subject to an administrative stay to allow petitioners time to seek a possible judicial stay.

JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1991 is summarized below.

Pending Cases

Critical Mass Energy Project v. NRC. (No. 90-5120 (D.C. Cir.).) This is a protracted Freedom of Information Act suit (pending since 1984) in which plaintiffs seek access to "SEE-IN" documents prepared by the nuclear industry's Institute of Nuclear Power Operations (INPO) and shared with the NRC. In March 1990, the District Court granted summary judgment to the NRC on the ground that SEE-IN documents were exempt from disclosure under FOIA's exemption 4 (protecting "confidential" commercial information). The court reasoned that the documents warranted protection because their disclosure might disrupt the NRC's beneficial relationship with INPO.

The Court of Appeals reversed the District Court's judgment and remanded the case. The court began with the proposition, first stated in *National Parks & Conservation Association v. Morton* (498 F.2d 765 (D.C. Cir., 1974)), that exemption 4 applies only where disclosure would jeopardize one of two interests: (1) the government's need for continuing access to commercial data, or (2) the need to safeguard submitters of commercial information from competitive harm. All concede that the second interest was not at stake.

The court then flatly rejected the District Court's asserted basis for applying exemption 4: the loss of agency efficiency and effectiveness because of hostile relations with INPO resulting from FOIA disclosure. The court held that NRC's fear of making INPO "unhappy" was insufficient to justify withholding INPO documents from public disclosure, particularly where INPO conceded that it would continue preparing the documents, regardless of their availability under FOIA, and where the NRC conceded that it had the authority to compel INPO's members to share the documents with the agency.

The court also rejected on the current record the alternative NRC argument (not embraced by the District Court) that FOIA disclosure would make INPO's SEE-IN documents less useful. The NRC had contended that the sources for the INPO documents—frequently workers and officials at reactors—would be less candid in their evaluations should public disclosure become the norm. The problem with this argument, according to the court, was that it rested on "self-interested speculations" by NRC and INPO officials, rather than on testimony from "working level employees." It thus remanded the questions for further fact-finding.

The Department of Justice (DOJ) sought rehearing *en banc* on the exemption 4 question. On September 6, 1991, the full Court of Appeals vacated the panel decision and granted rehearing *en banc*. The full court is prepared to consider the question raised in the DOJ rehearing petition: whether FOIA exemption 4 requires an agency to show that its access to data actually would be impeded by disclosure. DOJ's reading of exemption 4—which was suggested by two of the panel judges in their concurrence—is that it offers blanket protection to all commercial documents submitted to agencies by outsiders, provided that the outsider itself treats the documents as confidential. There is no need, under this reading of exemption 4, to inquire whether non-disclosure is necessary to protect agency access to information.

Kerr-McGee Chemical Corp. v. NRC. (No. 90–1534 (D.C. Cir.).) This lawsuit challenges the Commission's decision to amend the existing agreement between the NRC and the State of Illinois to permit Illinois to assume regulatory jurisdiction over uranium and thorium mill tailings. (See "Commission Decisions," above). Petitioner is the owner of a contaminated site in the City of West Chicago, Ill., that falls within the State's jurisdiction under the new agreement.

After petitioner filed its brief in the Court of Appeals, and while the NRC was preparing its own, it was learned that petitioner and the State were engaged in settlement negotiations concerning the further disposition of the contaminated West Chicago site. The parties then jointly moved the Court of Appeals to hold the case in abeyance for a six-month period pending the possible settlement. On June 15, 1991 the Court of Appeals granted the motion, and issued an order requiring status reports at 60-day intervals.

Nuclear Information and Resource Service, et al. v. NRC. (No. 89–1381 (D.C. Cir.).) In December 1990, the NRC and NUMARC sought panel rehearing or *en banc* rehearing of the D.C. Circuit decision invalidating (in part) the Commission's standardization rule (10 CFR Part 52). On March 27, 1991, the D.C. Circuit Court issued an order granting the suggestions for rehearing *en banc.* The original panel, while upholding the rule's provisions for early site selection, standardized designs, and combined construction permits and operating licenses, had struck down the part of the rule limiting the right to a formal post-construction hearing to one issue: whether the acceptance criteria in the combined license have been met.

The full court issued an order setting a briefing schedule and asking the parties to address specific questions concerning the validity of the abbreviated Part 52 licensing procedures under the Atomic Energy Act, the continued validity of a 1984 decision *(Union of Concerned Scientists v. NRC)* in light of *Chevron v. NRDC,* and the reviewability of post-construction safety determinations under Part 52.

Shoreham-Wading River School District v. NRC. (No. 91–1140 & 91–1301 (D.C. Cir., July 19, 1991).) These two cases are the third and fourth in a series of lawsuits seeking to undo the shutdown of the Shoreham nuclear power plant on Long Island, N.Y. (They were consolidated by the District of Columbia Circuit.) The actions primarily challenge the issuance of a "possession only" license for Shoreham. Petitioners sought an emergency stay of the license. In a very unusual move, the Department of Justice, which normally joins in NRC pleadings, filed its own memorandum supporting petitioners' stay request. DOJ argued, as the Department of Energy had before the Commission, that Shoreham should remain in operational readiness, at least until the NRC performs environmental studies of its shutdown.

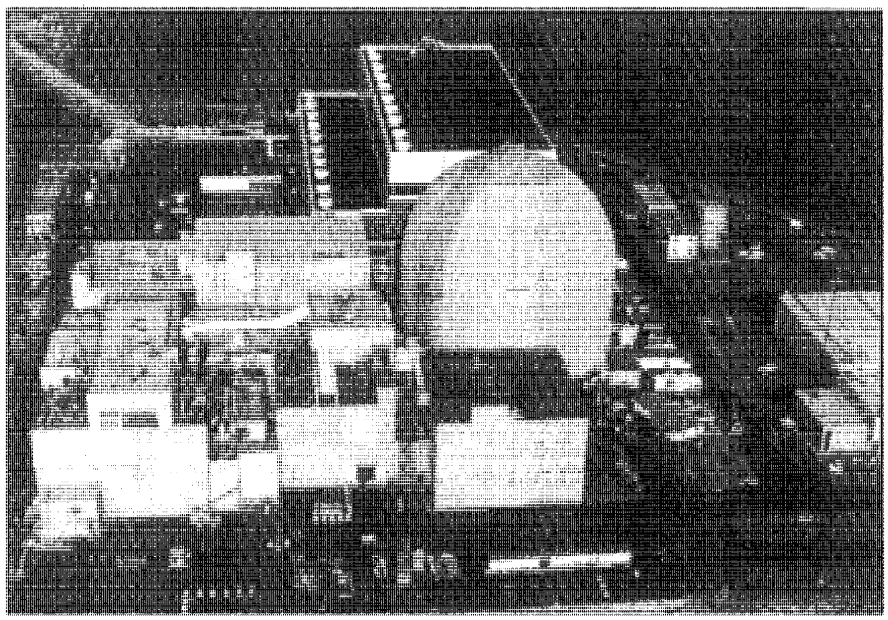
The NRC filed an opposition to petitioners' motion. The Court of Appeals issued an order denying the stay. The terse court order stated simply that "[p]etitioners have not demonstrated satisfaction of the stringent standards necessary for a stay." The court also denied petitioners' request to expedite the appeal. Petitioners unsuccessfully sought a stay in the Supreme Court. The case was being briefed on the merits at the close of the report period.

Significant Judicial Decisions

Commonwealth of Massachusetts v. NRC. (924 F.2d 311 (D.C. Cir. 1991), *cert. denied,* 60 USLW 3265 (US 1991).) A panel of the District of Columbia Circuit unanimously upheld the Commission's decision giving "immediate effect" to a Licensing Board decision authorizing a full-power license for the Seabrook (N.H.) nuclear power plant. With one exception, the Court of Appeals rejected the various arguments made by the Commonwealth of Massachusetts and other challengers to Seabrook's license.

Initially, the court agreed with NRC's argument that the court had jurisdiction over the Commission's "immediate effectiveness" decision only (and decisions subsumed in it), and not the full range of Seabrook decisions rendered by the Commission's regulatory bodies. The court then turned to the fundamental question in the case: whether petitioners were correct that the Commission's emergency planning regulations "require [the] Commission to judge an emergency plan in terms of the actual dose of radiation received by a particular EPZ [Emergency Planning Zone] population in a hypothetical accident scenario." The court rejected petitioners' reading of the regulations, and deferred to the NRC's view that compliance with the 16 planning standards in 10 CFR 50.47(b) satisfied the regulatory goal of "reasonable assurance" of "adequate protective measures." The court found nothing in the text of the regulations, in the Commission deliberations adopting them, or in the NRC's enabling legislation that required the "actual dose" approach demanded by petitioners.

The court also upheld the Commission's decision, in its "immediate effectiveness" determination, that Seabrook could begin operations despite the Appeal Board's remand of certain emergency planning issues to the Licensing Board. The court accepted the Commission's rationale that these issues were "not significant for the plant in question" (10 CFR 50.47(c)).



A panel of the District of Columbia Circuit Court upheld the Commission's decision giving "immediate effect" to a Licensing Board decision authorizing a full-power license for the Seabrook (N.H.) plant, shown here. The Circuit Court also upheld the Commission's decision that Seabrook could begin operations despite an Appeal Board's remand of certain emergency planning issues to the Licensing Board. The Supreme Court denied certiorari to petitioners in the case, in October 1991.

Finally, the court considered two issues that had come up in connection with the low-power license, but in the court's view remained relevant to the full-power license. First, it upheld as "sensible" the Licensing and Appeal Board's rejection of petitioners' claim that their contention focusing on aquatic blockage of cooling systems should be construed to cover aquatic corrosion as well.

Second, it remanded for "reasoned decision-making" the question whether it was permissible to reject petitioners' late-filed contention on the June 1988 full-participation exercise of the emergency plan. The court upheld as "reasonable" the Appeal Board's conclusion that petitioners had waited too long to proffer their contention, and therefore did not satisfy the "good cause" for lateness element of the five-factor "late-filed" test, but the court felt that the Appeal Board had failed to consider the "materiality" of the contention in addressing the other four elements. The court was not entirely clear as to whether it viewed "materiality" as implicit in the existing five-factor test or as a factor imposed by section 189a of the Atomic Energy Act, as construed in *Union of Concerned Scientists v. NRC* (735 F.2d 1437 (D.C. Cir. 1984), cert. denied, 469 U.S. 1132 (1985)). The court did comment that the NRC's five-factor test seemed "not well suited," and "odd," in the context of exercise contentions. The court also rejected as too "terse" the Appeal Board's alternative explanation that petitioners' contention did not meet the Commission's "fundamental flaw" standard for exercise contentions.

Despite the remand for further explanation on the exercise contention, the court did not vacate Seabrook's license. The court decided "against imposing an immensely disruptive interim status quo" where there seemed to be no "ongoing flaws" and where "a clean record" in Seabrook's most recent full-participation exercise (in December 1990) "will likely moot this issue."

The Supreme Court denied *certiorari* in this case on October 7, 1991.

Public Citizen v. NRC. (940 F.2d 679 (D.C. Cir. 1991).) This lawsuit was brought by Public Citizen and other organizations to challenge the Commission's 1990 issuance of a "Below Regulatory Concern" (BRC) policy statement. That statement articulated an approach, setting numerical ranges, for deregulating (as "below regulatory concern") activity involving exposures to extremely low levels of radiation. Petitioners argued that the BRC policy statement amounted to a substantive rule, issued improperly without notice and comment and without the required environmental review under the National Environmental Policy Act. Petitioners also attacked the substance of the BRC policy as lying outside the general consensus of expert opinion on safe levels of low-level radiation exposures.

The Court of Appeals issued a decision rejecting petitioners' suit as unripe. On the question of whether the policy statement amounted to an improperly issued binding rule, the court concluded that the "statement sends mixed messages." The court pointed to "some unequivocal language" in the statement, and to "as many indications cutting the other way." The court concluded, therefore, that with the "statement's own signals being in conflict, only Commission practice under the policy can make the issue determinable and thus fit for review." Since the Commission had as yet taken no action under the BRC policy, the court decided not to review it at that time.

Similarly, the court cited the "ripeness" issue to avoid addressing the NEPA questions. The court agreed with the NRC argument that the "BRC policy is not mature enough to constitute a 'proposal' for 'action.'" The court characterized as "premature" petitioners' concern that the Commission would review BRC environmental concerns piecemeal. The court stated that "the very existence of the policy statement will (ironically) give petitioners an argument that BRC exemptions are so 'related' as to require a programmatic Environmental Impact Statement once the Commission actually confronts a specific request for exemption."

Finally, the court, taking note of the Commission's recently declared moratorium on implementation of the BRC policy, concluded that its judgments regarding ripeness passed over any question as to "whether the new developments create additional grounds for deferring review."

Shoreham-Wading River School District, et al. v. NRC. (931 F.2d 102 (D.C. Cir. 1990).) This is the second lawsuit brought by the Shoreham-Wading River Central School District and the Scientists and Engineers for Secure Energy against the NRC for failure to perform an environmental review of the consequences of decommissioning the Shoreham (N.H.) plant. The D.C. Circuit dismissed the first suit for lack of a reviewable final order. Petitioners returned to the D.C. Circuit claiming that the NRC improperly allowed Shoreham's owners an exemption from full insurance coverage and was about to relax Shoreham's emergency preparedness and physical security requirements without preparing the environmental impact statement required by NEPA. Petitioners sought an emergency stay of the NRC's actions. The NRC argued primarily that petitioners had shown no irreparable injury and that the NRC had not yet authorized any "irreversible" step toward decommissioning that might trigger the agency's NEPA duties. On May 11, 1990, the court denied petitioners' request for a stay and denied their motion for an expedited appeal. The case was argued on March 11, 1991, and the court issued its decision on April 30, 1991.

In its decision, the court described the Shoreham plant as "all dressed up with nowhere to go." The court was not persuaded by petitioners' arguments (1) that the NRC had unreasonably delayed disposition of petitioners' 2.206 petition, (2) that a Commission Confirmatory Order (requiring NRC permission to restart Shoreham) should not have been made immediately effective, (3) that the Commission ought not have granted Shoreham an exemption form the insurance requirements applicable to operating plants, and (4) that the Commission was required to perform a full environmental analysis of Shoreham's decommissioning in connection with the Confirmatory Order or the insurance exemption. On this last point, the court reasoned that "[n]either action commits LILCO or the Commission to decommissioning or constrains their choices one whit."

In closing, the court recognized "that petitioners are here primarily because of a commitment to nuclear energy," but admonished them that "their track is almost certainly counter-productive." The court pointed out that, by making it more costly to exit from the nuclear industry, petitioners were, in effect, discouraging entrance into the industry. Petitioners did not seek *certiorari* in the Supreme Court.

State of New York v. United States. (No. 90-6031, 91-6033, 91-6035 (2d Cir., August 9, 1991).) This case involved an appeal of a District Court decision dismissing the constitutional challenge by New York State and Allegany and Cortland Counties to the Low Level Waste Policy Amendments Act of 1985, 42 USC 2021 et seq. The State and counties had argued that the Act violated the Tenth Amendment's guaranty of State sovereignty. The U.S. Court of Appeals for the Second Circuit issued an opinion affirming the District Court judgment.

The Court of Appeals agreed with the NRC's argument that *Garcia v. San Antonio Metropolitan Transportation Authority* (469 U.S. 528 (1985)), and *South Carolina v. Baker* (485 U.S. 505 (1988)), govern this case. *Garcia* and *Baker* allow the courts to review Congressional enactments for infringing state sovereignty only where there is a defect in the political process or where constitutional equality among the states is jeopardized. In support of its analysis, the court quoted with approval a recent law review article written by Dan Berkovitz, a former NRC attorney. Petitioners have sought review in the Supreme Court.

Union of Concerned Scientists v. NRC. (920 F.2d 50 (D.C. Cir. 1990).) This lawsuit challenged a 1989 NRC rule heightening the "threshold" pleading standards in licensing proceedings. Petitioner did not challenge the heightened pleading standard alone but argued that the NRC ought not be permitted to impose tougher threshold pleading standards while, at the same time, adhering to its traditional approach on "late-filed" contentions. That approach rests on a test balancing five factors: (1) good cause for lateness, (2) the availability of other means to protect the late petitioner, (3) the assistance to be expected from the late petitioner, (4) the extent that the new petitioner's interest is protected by existing parties, and (5) the extent that the new petitioner will broaden the issues or delay the proceeding (10 CFR 2.714(a)). Petitioner argued that late intervention should be automatic, without applying any balancing test, when the NRC staff releases safety or environmental documents containing new information. Otherwise, according to petitioner, the NRC would abrogate the hearing guaranty contained in section 189a of the Atomic Energy Act.

The Court of Appeals issued a decision rejecting petitioner's position and upholding the NRC's new threshold contention rule and its traditional late-filed contention rule. The court started with the proposition that petitioner's "challenge to the NRC's procedural rules faces a steep uphill climb," because "the Act itself nowhere describes the manner in which this 'hearing' is to be run." The court rejected petitioner's argument that an earlier D.C. Circuit case, *Union of Concerned Scientists v. NRC* (735 F.2d 1437 (D.C. Cir. 1984), *cert. denied,* 469 U.S. 1132 (1985)), should be read "to require that a licensing proceeding embrace anything new revealed in the SER or the NEPA documents" or "that the NRC consequently may not employ the balancing test to preclude consideration of new 'information.'" The earlier UCS decision, ruled the court, does not guarantee a hearing on all new *evidence,* but holds merely that the NRC cannot refuse a hearing altogether on an *issue* that the NRC itself agrees is material to a licensing decision.

Management and Administrative Services

Chapter 10

This chapter deals with internal events and activities of the NRC, such as changes in Commission membership, consolidation of NRC offices in a single locale, noteworthy aspects and initiatives in personnel management, the NRC's 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 and initiatives carried out under the Federal Women's Program.

Changes Within the Commission

Two changes occurred on the Commission during the year. On July 2, 1991, Dr. Ivan Selin was sworn in as Chairman of the Nuclear Regulatory Commission for a five-year term, extending to June 30, 1996. Chairman Selin succeeded Chairman Kenneth M. Carr, whose term expired on June 30, 1991. Dr. Selin had most recently served as Under Secretary of State for Management, the principal advisor to the Secretary of State on all matters involving the allocation of State Department resources. (See Chapter 1.)

Filling the vacancy created in June of 1990, with the expiration of Commissioner Thomas Roberts' second term, E. Gail de Planque was sworn in as the newest member of the Commission, on December 16, 1991, after the close of the report period. Commissioner de Planque had formerly served as Director of the Environmental Measurements Laboratory of the Department of Energy. (Other changes and appointments at the senior staff level are reported in Chapter 1.)

Headquarters Facility

At the end of October 1991, a new conference center opened on the lobby level of NRC Headquarters, at One White Flint North, in Rockville, Md. The center contains three medium-sized conference rooms and a large conference room which can be divided in half. The conference center at the lobby level facilitates meetings with NRC visitors because its location obviates the need for standard security and escort procedures.

Groundbreaking for construction of the second building at the White Flint venue took place on September 12, 1991, and construction is scheduled for completion in late 1993. The agency's consolidation plan calls for occupancy of the new facility beginning in 1994. (See below.)

Consolidation of NRC Headquarters

During the first half of fiscal year 1991, the Government and the developer agreed on lease terms and conditions for the construction of the second building of the two-building NRC headquarters complex, in Rockville, Md. Agreement also was reached among the parties concerning Montgomery County (Md.) restrictions on the site plan and traffic management. At the close of the report period, the first stages of site clearing and excavation for the second building had begun. The first building was purchased in 1986 and occupied in 1988.

In addition to providing office space for more than 1,400 people, the second building, Two White Flint North, will be equipped with a state-of-the-art Emergency Operations Center, central computer facility, a multi-purpose auditorium with a capacity of up to 300 seats, a day-care center for infants and toddlers, and expanded staff training facilities and amenities for the use of the 2,450 people who will work in the two-building complex. The building is expected to be ready for occupancy by early 1994. (See Chapter 1.)

PERSONNEL MANAGEMENT

NRC Staff Ceilings

During fiscal year 1991, the NRC expended a total of 3,300 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 329 employees and lost 174 permanent full-time employees, the latter figure representing an attrition rate of 4.9 percent. During the period, the agency participated in 94 recruitment trips, or "job fairs." The recruitment effort generated approximately 3,958 applications. Recruitment

Commissioner E. Gail de Planque was sworn in as a member of the U.S. Nuclear Regulatory Commission on December 16, 1991, bringing the Commission to its full complement of five members. Commissioner de Planque had formerly served as Director of the DOE's Environmental Measurements Laboratory in New York City.

during the year depended on three key mechanisms—advertisements, recruitment trips, and an applicant inventory/tracking system.

Awards and Recognition

In fiscal year 1991, 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 May, the NRC presented eight NRC Distinguished Service Awards and 34 Meritorious Service Awards, while recognizing those employees who had, during the year, received 665 Special Achievement Awards, 380 High Quality Performance Salary Increases, five Suggestion Awards, and 137 Certificates of Appreciation. Two NRC Executives received Presidential Distinguished Executive Rank Awards, 12 received Presidential Meritorious Executive Rank Awards, 88 received Senior Executive Service (SES) bonuses, and 12 received SES Pay Level Increases. Besides these NRC citations, 16 NRC employees and one NRC office were nominated for awards by outside organizations, and one of the employees and the nominated office received awards.

Labor Relations

The NRC and the National Treasury Employees Union (NTEU) began negotiating a new Collective Bargaining Agreement during the report period, reaching agreement on 14 articles that included these subjects: Grievances, Training and Development; Annual Leave; Day Care; and Official Time. Significant issues still open at the end of the fiscal year included: Hours of Work; Performance Awards; Reassignment of Resident Inspectors; and Union Representation. The open issues were pending before the Federal Services Impasse Panel for resolution at the close of the report period.

Training and Development

The NRC provides more than 60 different on-site courses in reactor-related technology, probabilistic risk assessment, end-user computer applications, and the development of executive, management, supervisory and administrative skills. During the fiscal year, NRC employees also participated in a wide variety of training and developmental programs conducted at colleges and universities, at other Government agencies, and in the private sector to improve performance and to assure up-to-date technical proficiency.

In its effort to provide ample developmental opportunity to all the staff, the NRC sponsors a number of special programs and activities. 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. Other programs sponsored by the agency during the period 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, and the Congressional Fellowship Program. In addition, a Graduate Fellowship Program designed to develop technical expertise in engineering, health physics, and specialized scientific disciplines was offered for the first time.

The Probabilistic Risk Assessment (PRA) Technology Transfer Program continued to undergo redesigning and restructuring during the fiscal year. The objective of the revisions is to formulate curricula, develop innovative training aids, and to conduct training courses for PRA practitioners, NRC managers, inspectors and others with a need for general or in-depth knowledge of PRA techniques. The general goal of the PRA Program is to provide training which will facilitate the wider use of PRA on-the-job.

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. Several courses on equal employment opportunity and cultural diversity are also offered to new and experienced managers.

During the report period, the NRC Individualized Learning Center continued to provide opportunities for improving staff knowledge and skill through the most current methods and media. The Center is designed to give employees convenient access to a wide variety of instruction, using the latest in audio/video, computer-based, and multi-media programming. An option to borrow training programs was initiated during the fiscal year to further expand employee opportunity to schedule extra training. The Learning Center provides 150 programs in a broad spectrum of subjects, including secretarial skills, project management, communication, management and supervision, computer skills and employee assistance.

Rotational Assignments

During fiscal year 1991, the NRC increased the 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 162 rotational assignments during the period.

Executive Leadership Development

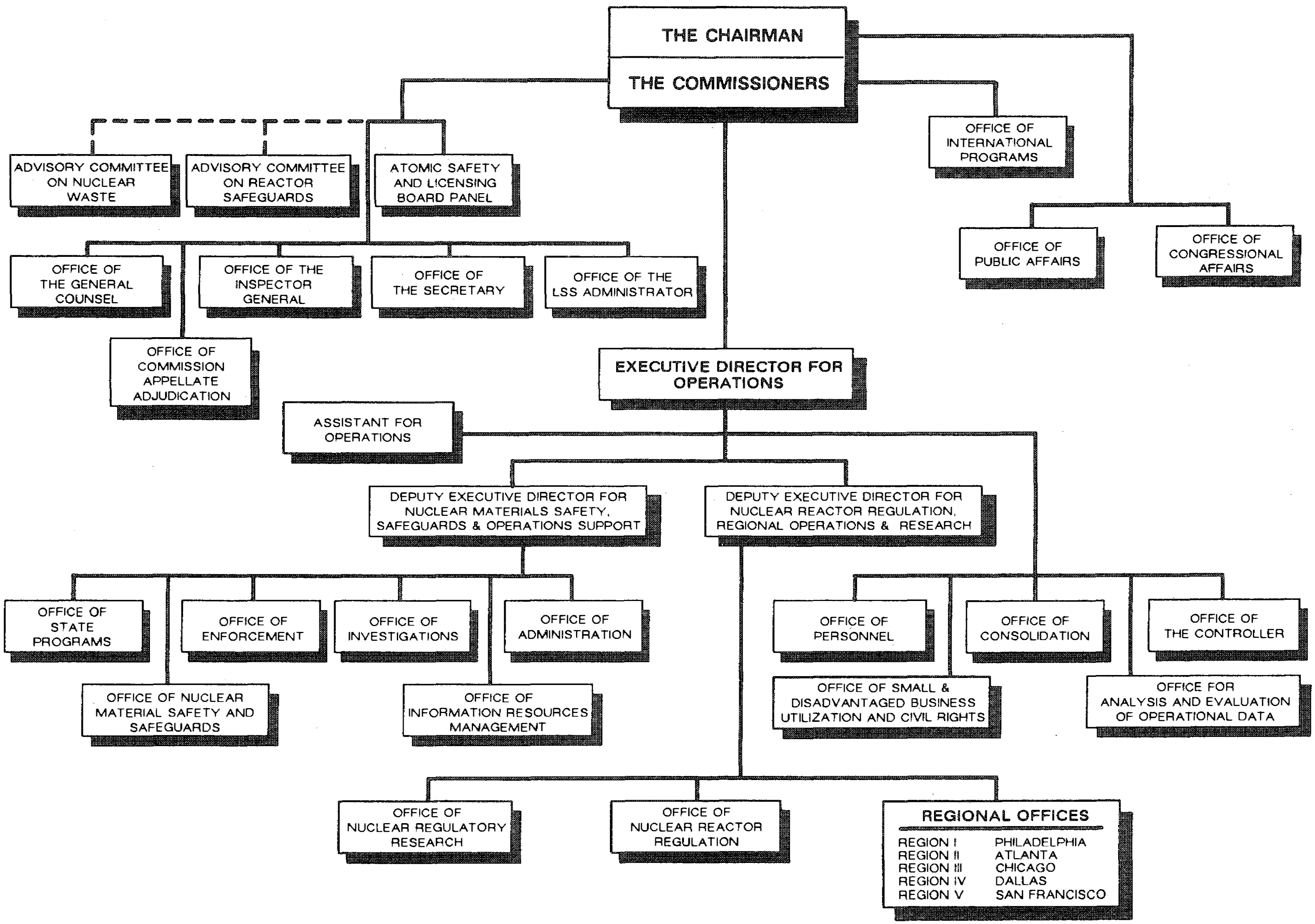
Members of the Senior Executive Service (SES) continued participation in an active rotational assignment program. Their assignments, which are intended to develop greater understanding of all aspects of the agency's operations, involved exchanges of assignments between Headquarters and the Regions, as well as inter-office exchanges within Headquarters. To provide a forum for wide-ranging discussion of vital technical and administrative issues facing the NRC, the fourth annual NRC SES conference for all senior management personnel was conducted. During the report period, 14 executives attended Brookings Institution Education Programs and 22 attended the Federal Executive Institute.

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 1991, a total of 16 NRC employees received voluntary leave donations from fellow employees.

Consolidation of all NRC Headquarters personnel continued to progress during fiscal year 1991, with construction of the second headquarters building, White Flint Two North, now under way, following several delays. In the artist's rendering, the new building is depicted to the right of the existing building, One White Flint North; the comples is located at 11545 Rockville Pike in North Bethesda, Md., and currently houses the Commission offices and most of the headquarters staff.

UNITED STATES NUCLEAR REGULATORY COMMISSION



Employee Assistance and Health Programs

During the fiscal year, the NRC Employee Assistance Program (EAP) staff continued to give individual counseling and referral assistance to NRC personnel with such problems as chemical dependency, job stress, chronic illness, and family issues. The agency entered into an agreement with the Public Health Service to make EAP services readily accessible to regional and field personnel. 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, conducted by the American Lung Association.

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; screenings for diabetes, glaucoma, high blood pressure, and cancer; immunizations; and health awareness programs on topics such as gastro-intestinal disorders, AIDS, and auto-infusion.

New Program Initiatives

The NRC has begun developing policies and procedures to implement appropriate provisions of the Federal Employees Pay Comparability Act of 1990 (FEPCA). These include a Senior Level System, Relocation and Recruitment Bonuses, Retention Allowances, and Advances in Pay. The Senior Level System will parallel the Senior Executive System (SES) and offer an alternative career development path for the NRC's non-supervisory technical, legal and administrative professional employees.

NRC INFORMATION RESOURCES

NRC Office Automation

During fiscal year 1991, the NRC successfully completed the first year of a three-year project to improve office automation. The Agency Upgrade of Technology for Office Systems (AUTOS) project will replace outdated and aged IBM 5520 and Displaywriter word-processing equipment with microcomputers configured into local area networks and wide area networks. State-of-the-art office assistance tools—such as word-processing, documentation transmission, electronic mail, calendar scheduling, and spreadsheets—will characterize this new environment. By the end of fiscal year 1993, all NRC staff will have a personal computer and be part of the AUTOS network.

Nuclear Documents System

The NRC's Nuclear Document System (NUDOCS) is the agency's centralized document search and retrieval system for information associated with the licensing and inspection of nuclear reactors and materials, as well as for documentation related to nuclear regulatory, adjudicatory, and high-level and low-level waste issues. During fiscal year 1991, in order to enhance system capabilities and improve performance, the NUDOCS data base was moved to a larger, more powerful Data General MV/40000 minicomputer. The upgrade to the MV/40000 has dramatically enhanced system capabilities and performance and has resulted in a higher level of user satisfaction. NUDOCS users now have access to such enhanced capabilities as customized searches, which facilitate the retrieval of documentation related to nuclear plant licensing. The upgrade has also resulted in faster response time in preforming searches, and has increased the number of ports available for NUDOCS users to access the system. Throughout the report period, the NRC continued to accommodate requests for access to the publicly available portion of the NUDOCS data base from such sources as the utilities, foreign entities, representatives of the news media, National Laboratories, universities and State governments.

NRC Emergency Telecommunications System

The regulatory requirements for emergency communications are set forth in 10 CFR 50.47(b)(6) and 10 CFR Part 50, Appendix E, IV.E.9d. Licensees are required to establish provisions for prompt communications among principal emergency response organizations, to emergency personnel and to the public. Currently, the NRC uses the public switched network (PSN) and a dedicated, single line network for the Emergency Telecommunications System (ETS). Serious concerns were raised about blockage of the PSN, in case of an emergency and reduced reliability, and reduced reliability and maintenance problems caused by aging and obsolete dedicated network equipment. Orders were issued to American Telephone and Telegraph to install Federal Telecommunications System (FTS) 2000 services at 119 nuclear power plants and emergency operations facilities. Installations at all locations are expected to be completed by spring of 1991.

OFFICE OF THE INSPECTOR GENERAL

In accordance with the Inspector General Act of 1978, the NRC Office of the Inspector General (OIG) was established, on April 15, 1989. It is one of 26 such statutory entities created within the Executive Branch. The Inspector General is appointed by the President of the United States, with the advice and consent of the Senate.

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 agency's programs and operations. The office also has oversight responsibilities regarding the conduct of NRC employees and contractor personnel. In executing its mission, the OIG carries out agency audits, inspections and investigations, and makes recommendations to NRC management as appropriate.

During the fiscal year 1991, the OIG (1) completed 17 audits of the NRC's operations and programs, (2) reviewed 175 contract audit reports, (3) performed one contract audit, and (4) closed out 72 investigations.

OIG Fiscal Year 1991 Audits

A review of NRC's Emergency Planning Regulations. In 1980, the NRC amended its regulations to require that "no operating license for a nuclear power reactor will be issued unless a finding is made by the NRC that the state of on-site and off-site emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency." Before that time, the NRC believed that the nuclear power plant's safety systems were adequate to protect the public from a release of radiation. The accident at the Three Mile Island (Pa.) plant in 1979 demonstrated that a nuclear plant's safety systems alone were not adequate to assure protection of the public.

At the request of Congress, the OIG initiated a review of the NRC's emergency planning regulations. OIG found that the NRC's stated objective of emergency planning was not consistent with the objective set forth in NRC regulations. It was also learned that the NRC had determined that it could conduct its own review of off-site emergency plans. The OIG questioned the NRC's authority in this respect, because the President had assigned the lead role for off-site emergency planning to the Federal Emergency Management Agency (FEMA). The OIG recommended that the NRC (1) modify its stated objective of emergency planning and (2) determine whether NRC authority provides for the review of off-site emergency planning. In a February 22, 1991 memorandum responding to the OIG recommendations, the NRC General Counsel concluded that the NRC has the authority to conduct its own review of off-site emergency plans.

NRC's Management of Industry Reports. In passing the Energy Reorganization Act of 1974, the Congress imposed requirements that the nuclear industry detect and report on any defective parts supplied for use in their facilities. Under Section 206 of the Act, a vendor or licensee is required to notify the NRC promptly whenever a defective part that could pose a substantial safety hazard has been supplied to an NRC licensee.

To satisfy this legislative requirement, the NRC issued a regulation entitled "Reporting of Defects and Noncompliance," Title 10, *Code of Federal Regulations*, Part 21. Following the Three Mile Mile (TMI) accident, the NRC reaffirmed its intention to the President's Commission on TMI to improve its regulations by clarifying the reporting guidelines. The objective was to introduce reporting uniformity and to provide for an earlier identification of defective parts and an earlier correction of problems associated with such parts. The NRC committed itself to making these regulatory revisions by August of 1980.

The OIG conducted an audit to determine whether the NRC's regulations and proposed revisions were adequate to ensure compliance with Section 206 of the Act. The OIG also set out to ascertain whether the NRC's management of industry reports was adequate to ensure prompt resolution of any hazardous condition created by reported defects. The OIG audit revealed that the NRC needed to improve its management of reports filed by licensees and vendors and to provide a more accurate, reliable and effective system of review and response. Furthermore, the OIG concluded that the NRC needed to fulfill its commitment to the President's Commission on TMI by completing the promised regulatory revisions.

Seabrook Allegations Handled Appropriately. The OIG reviewed the manner in which the NRC handled allegations made by the Employees Legal Project (ELP) concerning the safe operation of the Seabrook (N.H.) nuclear power plant. The ELP is an independent group whose mission is to provide legal assistance and also anonymity to employees and former employees who have allegations regarding nuclear power plants located in New England.

The review was undertaken to determine whether the NRC staff had addressed the ELP allegations in accordance with agency policy and procedures. The review also focused on whether the Commission had adhered to established procedures in resolving these allegations before voting on a full-power operating license for the Seabrook facility.

In the conduct of its investigation of the ELP allegations, the NRC staff expended over 1,000 inspection and managerial hours. The NRC staff concluded that the allegations were not material to the licensing of Seabrook and reported its findings to the Commission. The OIG analysis of the allegations and of the manner in which the staff handled them led to the conclusion that the NRC staff had received, evaluated and resolved the allegations in accordance with agency policies and procedures. The OIG also found that the Commission had followed established agency procedures in reviewing the allegations and had resolved them before voting on the full-power license for the Seabrook plant.

An Audit of NRC's License Fee Billing Process. The NRC is one of the few Federal agencies required by law to collect approximately 100 percent of funds to attain its budget authority from fees for services rendered. The NRC collects fees for such services as its review of applications for new licensees and its inspection of licensee operations.

The OIG reviewed the NRC's license fee processing (Part 170) to determine the adequacy of the NRC's billing procedures and of related internal controls for these fees. The OIG examined the costs of services to determine their accuracy and to judge whether they were being billed in a timely manner.

The OIG found that, although the NRC was taking steps to improve its billing operations through increased automation, additional corrective measures were needed to achieve full compliance with Treasury Department requirements. The time-frames established by the NRC for its billing processes significantly exceeded those specified by the Treasury Department. The OIG also found that fundamental deficiencies existed in the accumulating, editing and processing of cost data. These deficiencies added to the time required to produce an accurate bill. Program managers agreed with the OIG's recommendations to correct these and other billing inadequacies.

The Debt Collection Act. In 1982, the Congress enacted the Debt Collection Act to facilitate debt collection within the Federal Government. Under the authority of the Act, the Office of Management and Budget (OMB) issued Circular A–129, entitled "Managing Federal Credit Programs," which prescribes policies for collecting Federal loans and other debts.

The NRC is required to collect fees for services provided to its licensees in accordance with Title V of the Independent Offices Appropriation Act of 1952. The OIG reviewed the NRC's debt collection process for delinquent accounts, in order to determine whether the agency was in compliance with certain follow-up provisions of the Debt Collection Act and Circular A–129. The inquiry revealed that the NRC was not in full compliance with the Act or the OMB circular. The NRC had not computerized its system for collecting debts and did not routinely compute and apply to interest accruals, penalties and other administrative charges to its accounts.

The NRC's practice of manual bookkeeping and collection of delinquent accounts was not adequate to achieve full compliance with the Act or the OMB circular. The OIG recommended that the NRC develop an automated system for ensuring adequate follow-up on overdue accounts and for computing all applicable interest and other charges.

The Prompt Payment Act of 1982. The OIG assessed the timeliness of the NRC's payments to vendors and evaluated the agency's compliance with the Prompt Payment Act of 1982, as amended. Congress adopted this legislation to give incentives to the Federal Government to pay its bills promptly and to ensure that the Government pays its contractors in a predictable and timely manner.

In its review, the OIG found that the NRC was not fully complying with this Act. The NRC failed to take advantage of available discounts. The OIG recommended that the NRC fully automate its payment system, provide its staff with proper training, and document procedures to implement the Act.

OIG Fiscal Year 1991 Investigations

Alleged Threat Against Nuclear Power Plant Employee. The OIG received information that a nuclear power plant employee had been verbally threatened by a co-worker. At the time of the alleged threat, the employee had been providing information to the NRC and to the OIG about certain hazards at the plant. The employee interpreted the alleged threat as a message from plant managers that the employee should no longer cooperate with NRC investigators.

Insufficient information was developed to substantiate that the alleged offender had intended to threaten or harm the employee or to convey any message from the plant managers. It was later determined, however, that, during the OIG's investigation, the alleged offender made false statements to Federal investigators. That case has been referred to the U.S. Attorney's office.

Licensee Failed to Provide Requested Documents to NRC Inspectors. The OIG received an allegation that plant managers at a nuclear facility had deliberately withheld documents from the NRC that revealed serious problems of a technical nature. The OIG investigation disclosed that the NRC inspector had not received certain pertinent documents that the plant managers had promised to provide. It was also concluded that plant managers had been negligent in failing to ensure that the NRC inspector received all requested documents. The OIG staff found that on two previous occasions NRC inspectors had not received documents that they had requested. The report was referred to NRC management for appropriate action.

Abuse of Federal Telecommuncations System Calls. The OIG received an allegation that an NRC employee had used the NRC Federal Telecommunications System (FTS) to place frequent personal long-distance calls. An OIG investigation confirmed that, over a five-year period, the employee had used the FTS to place more than 200 personal calls. The matter was referred to the U.S. Attorney's office.

The NRC Failed to Investigate Allegation of Licensee Wrongdoing. During an investigation at a nuclear power plant, the OIG received information that NRC regional staff may have been informed of possible licensee wrongdoing but failed to properly record and investigate the allegation.

The OIG determined that the allegation had not, in fact, been entered into the NRC management tracking system, because the staff had failed to follow procedures. Consequently, the allegation concerning licensee wrongdoing had not been controlled, evaluated or resolved by the NRC staff. An investigation of the licensee wrongdoing was subsequently undertaken.

Investigation of False Labor Charges. NRC managers advised the OIG that a subcontractor appeared to have submitted false labor charges to the NRC, over a three-month period. The OIG's investigation revealed that the owner and an employee of a subcontracting firm had billed the NRC for more than $40,000 for hours not worked. The case was referred to the U.S. Attorney's office.

Conflict-of-Interest Laws Violated. The OIG received an allegation that two former NRC supervisors had been involved in a matter possible involving a conflict of interest. The OIG investigation showed that, following termination of their employment with the NRC, the supervisors had provided an affidavit supporting an electric utility's rebuttal of an enforcement action proposed by the NRC. This post-employment activity is prohibited by Title 18, Section 207, of the *United States Code*. The case was referred to the U.S. Attorney's office.

FINANCIAL MANAGEMENT

Contracting

In fiscal year 1991, NRC contracting with commercial firms for technical assistance, research and general purchases totaled approximately $96,509,512. Contracts under the Small Business Innovative Research Program came to $475,289, and grants and cooperative agreements with education and non-profit institutions totaled $3,034,469.

NRC License and Annual Fees

The Omnibus Budget Reconciliation Act of 1990 (Public Law 101–508) requires that, in fiscal year 1991, 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 1991, a total of $465 million was appropriated to the NRC (Public Law 101–514), of which $19,650,000 was derived from the Nuclear Waste Fund. Of the remaining $445,350,000, approximately 98 percent, or $438,610,118, was collected through license fees and annual charges. The net amount appropriated to the NRC in fiscal year 1991 was $6,739,882. Table 1 shows the amounts collected through license and annual fees in fiscal year 1991.

New Fee Schedules

The Commission adopted revised fee schedules in 10 CFR Parts 170 and 171, based on the fiscal year 1991 budget. These schedules became effective August 9, 1991. The revisions were made to implement Public Law 101–508, passed by the Congress on November 5, 1990. For fiscal year 1991, the law requires that the NRC recover approximately 100 percent of its budget authority, which is $465 million, less the amount appropriated from the Nuclear Waste Fund, by assessing license, inspection and annual fees.

Major changes to the fee regulations 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 $92-per-hour to $115-per-hour.

- Increase all flat fees for radioisotope programs by 25 percent, using the increased hourly rate as a basis.

Table 1. License and Annual Fee Collections – FY 1991

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR Part 170	$75.5 million	$7.6 million	$83.1 million
10 CFR Part 171	$321.1 million	$34.4 million	$355.5 million
TOTAL FEES	$396.6 million	$42.0 million	$438.6 million

- Discontinue the deferral of license review fees for standardized reactor designs and assess fees for these reviews from the effective date of the revised regulation.

- Remove the ceiling of $50,000 previously established for the review of topical reports.

- Add inspection fees for inspections related to cases, packages, shipping containers and "Part 71" vendor Quality Assurance programs and for the inspection of manufacturers and initial distributors of sealed sources and devices.

- Add an application fee of $600 and an inspection fee for Agreement State licensees working in non-Agreement States under a reciprocity general license.

- Revoke the existing exemption provisions in 170.11(a)(1), (2), (8), (9) and (11) and assess licensing fees for import and export licenses and inspection fees for State and local government agencies, Indian Tribes and Indian organizations, and holders of licenses authorizing the use of depleted uranium as shielding only in devices and containers.

Changes in Part 171

- Increase the Part 171 annual fees assessed to operating power reactors.

- Establish annual fees for non-power reactors that are not owned or operated by non-profit educational institutions.

- Establish annual fees for materials licensees—including fuel fabrication facilities, uranium recovery facilities, transportation and spent fuel storage cask users, and other small materials licensees. The holders of registrations for Quality Assurance programs, as well as government agencies which are licensed by the NRC, will also be charged an annual fee.

- Establish a maximum annual fee of $1,800-per-licensed-category for those licensees that qualify as "small entities" under the NRC's standards.

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 noted above, and the revisions to 10 CFR Parts 170 and 171 became effective August 9, 1991. Three lawsuits have been filed with the U.S. Court of Appeals for the District of Columbia Circuit, petitioning the court to review the final fee regulations.

Chief Financial Officers Act of 1990

The Chief Financial Officers (CFOs) Act of 1990 (Public Law 101-576) was intended to inaugurate a new era in Federal management and accountability and to bring about improved financial control throughout the Federal Government. The CFO Act is the most comprehensive financial improvement legislation since the Budget and Accounting Procedures Act of 1950. The Act provided that Chief Financial Officers be appointed in 14 departments and nine agencies, including the NRC. It also requires each agency to have a Deputy CFO.

The legislation requires that CFOs report to the head of the agency regarding financial matters; oversee all agency financial management activities; develop and maintain an integrated agency accounting and financial management system; direct, manage and provide policy guidance and oversight of agency financial management personnel, activities and operation; prepare and submit an annual financial report to the Chairman and to the Director of the Office of Management and Budget (OMB); monitor the financial execution of the agency's budget and submit performance reports to the Chairman, and biannually review agency fees for services and recommend revisions to reflect costs incurred.

The Act requires each agency to submit to the Director of OMB a proposal that consolidates its accounting, budgeting and other financial management activities under the agency CFO. It also establishes a CFO Council to advise and coordinate the activities of the member agencies.

Specific reporting requirements are laid down by the CFO Act. Under its terms, the OMB is required to submit to Congress a financial management status report and a government-wide Five-Year Financial Management Plan, within 15 months after enactment and annually thereafter. OMB is required to report to Congress, within 180 days of enactment, as to which agencies perform commercial functions for which financial statements are to be prepared under Section 3515. And each agency head must submit to the Director of OMB, by March 1992, a financial statement for fiscal year 1991, with annual statements to be submitted thereafter. Beginning in fiscal year 1992, each financial statement must be audited according to accepted government standards. The NRC has requested an exemption for the fiscal year 1991 statement.

The annual report to the Chairman and to OMB, which is due by August 31 of each year, comprises a description and analysis of agency financial management; a copy of the annual NRC financial statement; a copy of the annual Inspector General's audit report; and other documents.

New Accounting System at NRC

The NRC has undertaken to upgrade its financial management system. The Office of the Controller reveiwed the options and decided to replace the existing accounting system with accounting computer software called the Federal Financial System (FFS), developed by American Management Systems, Inc. The software operates on an IBM 3090 mainframe operated by the Financial Management Service of the Department of the Treasury. The new system—which accommodates the recording and tracking of commitments, obligations and expenditures, among other features—is expected to become operational by October 1, 1992.

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 1991.

- It was estimated that $58,000,000 in total prime contracts would be awarded during fiscal year 1991. The actual total for prime contract dollar awards was $79,104,146.

- It was estimated that small business prime awards would be $27,500,000, or 47.41 percent of the total estimate. The actual achievement for small business prime awards was $34,029,897, or 43.02 percent of the actual dollar awards, reflected in the previous item.

- The NRC estimated that awards to "8(a) firms" would be $9,500,000, or 16.38 percent, in fiscal year 1991. Awards to "8(a) firms" were actually $14,889,937, or 18.82 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 $525,000, or 0.91 percent. The actual achievement was $971,911, or 1.23 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 $2,500,000, or 4.31 percent. Awards to such firms came to $2,124,964, or 2.69 percent of the total dollar amount of all prime contracts, regardless of dollar value.

- The goal for subcontract awards to small business was $2,050,000, or 68.33 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $2,900,000, or 78.38 percent of total subcontracts awarded. The NRC's total subcontract dollar awards goal in fiscal year 1991 was $3,000,000. The NRC's total subcontract dollar awards were $3,700,042.

- The goal for subcontract awards to small disadvantaged businesses was $350,000, or 11.67 percent. Subcontracting awards to small disadvantaged businesses totaled $480,000, or 12.97 percent of total subcontract dollars awarded.

During the report period, 120 interviews were conducted with firms wanting to do business with the NRC, and 40 follow-up meetings were arranged with NRC technical personnel. The staff of the NRC 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 Week observance, in May 1991, and Minority Enterprise Development Week, in October 1991.

Civil Rights Program

During fiscal year 1991, the Commission was briefed—on December 17, 1990 and on July 16, 1991—concerning the NRC's Equal Employment Opportunity (EEO) and Affirmative Action 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 Senior Executive Service Performance Review Board.

The Civil Rights Program staff sponsored a three-day training seminar for for Equal Employment Opportunity counselors from NRC Headquarters and NRC Regional Offices. The event, which was held in Hunt Valley, Md., was well attended and well received.

An initiative of NRC's Executive Director for Operations, James M. Taylor, encouraging minority employees in the "Engineering and Physics" occupational series to prepare Individual Development Plans (IDPs) has elicited a substantial response. The IDP program is designed to assist employees in establishing both short and long term training and development objectives, and to monitor and guide their progress in fulfilling individual development plans.

Federal Women's Program

As part of the agency's continuous effort to heighten awareness of women's contributions and opportunities, National Women's History Month was observed throughout the NRC during March 1991, with outstanding speakers, receptions, exhibits and presentation of awards. In recognition of the crucial role women were fulfilling in the Persian Gulf conflict, a special program was held in NRC Headquarters, featuring Brigadier General Myrna Williamson, U.S Army, as keynote speaker. The program commenced with a performance by the Colonial United States Drum and Fife Corps and the presentation of flags and yellow ribbons for each of the four hundred-plus in attendance. The observance included a slide presentation entitled, "Proud To Be An American," citing and commending all NRC employees taking part in the conflict.

A special program recognizing the crucial role women were fulfilling in the Persian Gulf conflict was presented in March 1991, National Women's History Month. Shown here, at the right, is the keynote speaker for the event, Brigadier General Myrna Williamson, with Era Marshall, the NRC's Federal Women's Program Manager. The program was repeated several times at NRC Headquarters.

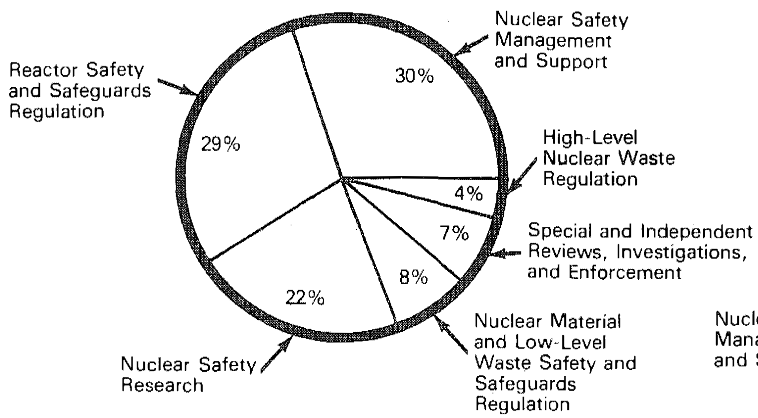
The presentation was very well received and was repeated several times at NRC Headquarters. National Secretaries Week was again celebrated at the agency with luncheons and training seminars for secretaries, and Women's Equality Day was commemorated with the presentation of speakers and the film, "How We Got the Vote." Regional Federal Women's Program Coordinators and the Federal Women's Program Advisory Committee for the Washington Headquarters offices sponsored several career opportunity and development seminars.

Forty-eight women were hired during the report period. Women continue to make gains in mid- and senior-level positions. The number of women in grade–13 increased by 4.7 percent, in grade–14 by 19 percent, and in grade-15 by more than 10 percent. The number of women in the Senior Executive Service rose to 10. Women represent 34 percent of the total NRC work-force.

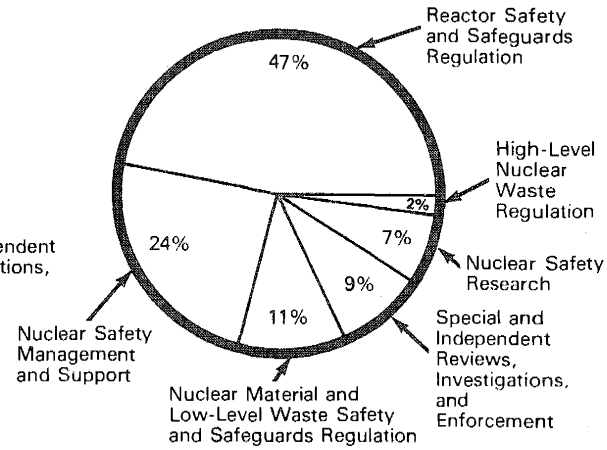
In order to address concerns regarding sexual harassment in the Federal workplace, the NRC's Federal Women's Program Manager, Era Marshall, assisted by Dennis Dambly, NRC's Assistant General Counsel for Administration, and by Marvin Itzkowits, NRC's Special Counsel for Personnel, Labor and Civil Rights, conducted Prevention of Sexual Harassment Training for the Regional Offices.

The Annual Training and Planning Conference of the Federal Women's Program took place this fiscal year in Denver, Colo., in conjunction with the Federally Employed Women's National Training Program.

NRC RESOURCES
FY 1991
Actual

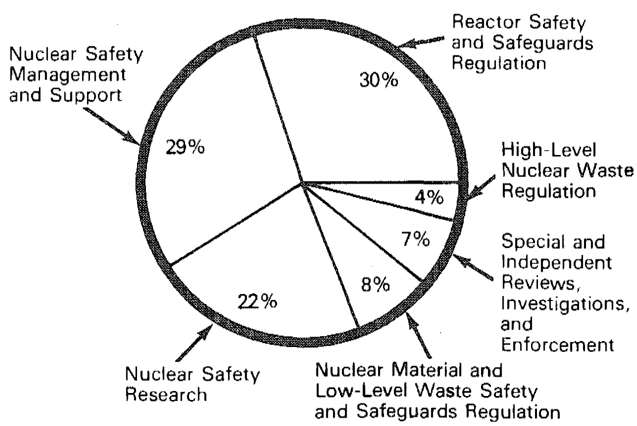


FUNDS $467.7 MILLION

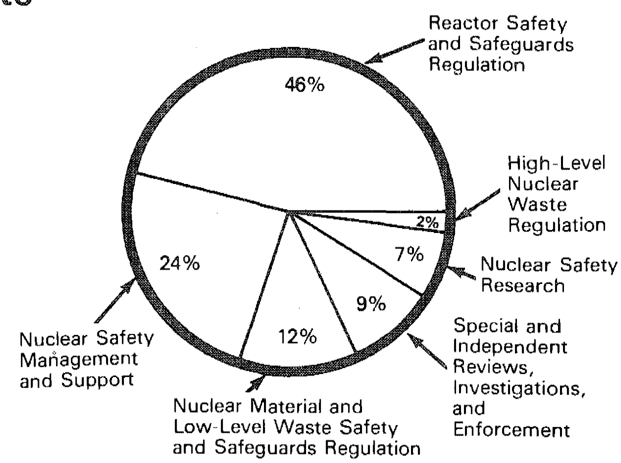


PERSONNEL 3, 300
(Full Time Equivalent)

NRC RESOURCES
FY 1992
Estimate



FUNDS $512.5 MILLION



PERSONNEL 3, 335
(Full Time Equivalent)

Appendix 1

NRC Organization

(As of December 31, 1991)

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, David A. Ward, 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. Bernaro, 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 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 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.

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; 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 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/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 Office of the Licensing Support System Administrator (LSSA) and the Department of Energy (DOE) on selected aspects of the design, development and operation of the support system (see Office of the Licensing Support System Administrator, above).

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

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.

As of September 1991, the members were:

CHAIRMAN: MR. DAVID A. WARD, Research Manager, retired, E.I. du Pont de Nemours & Company, Savannah River Laboratory, and Consulting Engineer, North Augusta, S.C.
VICE-CHAIRMAN: DR. PAUL G. SHEWMON, Professor, Metallurgical Engineering Department, Ohio State University, Columbus, Ohio.
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.
DR. WILLIAM KERR, Professor Emeritus of Nuclear Engineering, University of Michigan, Ann Arbor, Mich.
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. 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 Reguatory Commission, Washington, D.C.
DR. CHESTER P. SIESS, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.
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

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, 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 JOHN H FRYE, III (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, Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE THOMAS S. MOORE (Legal), 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 JAMES P. GLEASON (Legal), Silver Spring, Md.
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, Washington, D.C.
JUDGE GUSTAVE A. LINENBERGER, JR., Physicist (retired), U.S. Nuclear Regulatory Commission, Hagerstown, Md.
JUDGE EMMETH A. LUEBKE, Physicist (retired), U.S. Nuclear Regulatory Commission, Chevy Chase, Md.
JUDGE KENNETH A. McCOLLOM, Electrical Engineer (retired), Oklahoma State University, Stillwater, Okla.
JUDGE MARSHALL E. MILLER, (Legal; retired), U.S. Nuclear Regulatory Commission, Daytona Beach, Fla.
JUDGE PETER A. MORRIS, Physicist (retired), U.S. Nuclear Regulatory Commission, Potomac, Md. JUDGE
RICHARD R. PARIZEK, Geologist, Pennsylvania State University, University Park, Pa.
JUDGE HARRY REIN, Medical Doctor, Longwood, Fla.
JUDGE LESTER S. RUBENSTEIN, Nuclear Engineer (retired), U.S. Nuclear Regulatory Commission, Oro Valley, Ariz.
JUDGE DAVID R. SCHINK, Oceanographer, Texas A&M University, College Station, Tex.
JUDGE GEORGE 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

The Licensing Support System Advisory Review Panel (LSSARP) was established in 1989 to advise the NRC Office of the Licensing Support System Administrator and the Department of Energy on selected aspects of the design, development and operation of the Licensing Support System. The panel held two public meetings during fiscal year 1991. As of September 1991, the members were:

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, Nev., Comprehensive Planning Department.
STEVE BRADHURST, Nye County, Nev., Board of Commissioners.
BARBARA CERNY, U.S. Department of Energy.
DAVID COPENHAFER, U.S. Securities and Exchange Commission.
PETER CUMMINGS, Las Vegas, Nev., City Manager's Office.
PETE GOICOECHEA, Eureka County, Nev., Commissioner.
CHRISTOPHER HENKEL, Edison Electric Institute.
ELGIE HOLSTEIN, Nye County, Nev., Board of Commissioners.
FELIX KILLAR, U.S. Council for Energy Awareness.
STEVEN KRAFT, Edison Electric Institute.
JOHN LAMPROS, White Pine County, Nev.
CORINNE MACALUSO, U.S. Department of Energy.
LORETTA METOXEN, National Congress of American Indians.
MALACHY MURPHY, State of Nevada.
JAY SILBERG, Utility Nuclear Waste Management Group.
LENARD SMITH, Lincoln County, Nev., Commissioner.
HARRY SWAINSTON, State of Nevada.

OTHER NRC ADVISORY GROUPS

Advisory Committee on the Medical Uses of Isotopes

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. As of September 1991, the members were:

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.
DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.
DR. JACK K. GOODRICH, Nuclear Medicine Radiology Associates of Erie, Erie, Pa.

DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.
DR. NILO E. HERRERA, Director, Department of Laboratory Medicine, Danbury Hospital, Danbury, Conn.
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

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.

As of September 1991, the members were:

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.
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.
DR. MARTIN J. STEINDLER, Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Ill.

Advisory Panel for the Decontamination of Three Mile Island Unit 2

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.
THOMAS GERUSKY, Director of the Pennsylvania Bureau of Radiation Protection, Department of Environmental Resources, Harrisburg, 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 Associate 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. T
THOMAS SMITHGALL, Resident of Lancaster, Pa.
ANN TRUNK, Resident of Middletown, Pa.
NEIL WALD, Professor of Radiation Health, Department of Radiology, University of Pittsburgh, Pittsburgh, Pa.

Nuclear Safety Research Review Committee

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. As of December 1991, the members were:

CHAIRMAN: DR. DAVID L. MORRISON, Technical Director, Energy Resource and Environmental Systems Division, MITRE Corporation, McLean, Va.
DR. E. THOMAS BOULETTE, 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
  Houston1-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. Bess Chen, Librarian
  Martin Luther King Regional Library
  7340 24th Street Bypass
  Sacramento, Cal. 95822
  Rancho Seco nuclear plant

- Mr. Chi Su Kim, 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. Ramona Scott, Librarian
  Charles S. Miley Learning Resources Ctr.
  Indian River Community College
  3209 South Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie nuclear plant

- Ms. Esther B. Gonzalez, Librarian
  Urban and Regional Documents Collection Library
  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
  120 West Johnson 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. Nancy Barbour, Librarian
  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

- Ms. Jan Brown
  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. Bea Rodgers, 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
  S. Commerce at Washington Street
  P.O. Box 1406
  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

- Mr. Bill Olbrich
  Government Publications Librarian
  John M. Olin Library
  Washington University
  One Brookings Drive
  St. Louis, Mo. 63130
  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 Juanita 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. Eileen M. Disbrow
  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. Thomas Larson
  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

- Ms. Dawn Hubbs,
  Documents Librarian
  J. Murrey Atkins Library
  University of North Carolina at
  Charlotte—UNCC Station
  Charlotte, N.C. 28223
  William B. McGuire nuclear
  plant

- Ms. Janet Virnelson, Head
  Adult Services
  Cameron Village Regional Library
  1930 Clark Avenue
  Raleigh, N.C. 27605
  Shearon Harris nuclear plant

- Mrs. Arlene Hanerfeld
  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. Ann Freed, Reference Librarian
  Perry Public Library
  3753 Main Street
  Perry, Ohio 44081
  Perry nuclear plant

- Mrs. Julia Baldwin, Documents Librarian
  Government Documents Collection
  William Carlson Library
  University of Toledo
  2801 West Bancroft Avenue
  Toledo, Ohio 43606
  Davis-Besse nuclear plant

OKLAHOMA

- Ms. O.J. Grosclaude
  Stanley Tubbs Memorial Library
  101 E. Cherokee St.
  Sallisaw, Okla. 74955
  Kerr-McGee Sequoyah

OREGON

- Mr. Robert Lockerby
  Engineering 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
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  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. 22901
  North Anna nuclear plant

- Mr. Alan Zoellner
  Documents Librarian
  Swem Library
  College of William and Mary
  Williamsburg, Va. 23185
  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, and 4
  Richland low-level waste burial site

WISCONSIN

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  Government Documents Section
  Cofrin Library
  University of Wisconsin
  2420 Nicolet Drive
  Green Bay, Wis. 54301
  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 1991

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Submitting Applications for the Licensing of Test and Research Reactor Operators Directly to Headquarters—Part 55

On October 11, 1990 (55 FR 41334), the NRC published an amendment to its regulations that requires test and research facility applications for operator and senior reactor operator licenses to be submitted to the responsible Headquarters Office. This amendment, effective November 13, 1990, is necessary to improve efficiency and consistency in the examination and licensing of test and research reactor operators by having a central office monitor the issuance and renewal of licenses.

Interim Procedures for Agency Appellate Review—Part 2

On October 24, 1990 (55 FR 42992), the NRC published an amendment to its regulations, effective October 25, 1990, that implements a transition plan which provides that, with certain exceptions, the Commission, rather than an appeal board, will provide agency appellate review for appellate matters filed in the interim period between October 25, 1990, and the effective date of a final appellate review rule.

Custody and Long Term Care of Uranium and Thorium Mill Tailings Disposal Sites—Part 40

On October 30, 1990 (55 FR 45591), the NRC published an amendment to its regulations issuing general licenses that permit NRC to license the custody and long term care of reclaimed or closed uranium or thorium mill tailings sites after remedial action or closure under the Uranium Mill Tailings Radiation Control Act has been completed. This amendment, effective November 29, 1990, is necessary to meet the requirements of Titles I and II of the Uranium Mill Tailings Radiation Control Act.

Statement of Organization and General Information; Minor Amendments— Parts 0 and 1

On November 15, 1990 (55 FR 47740), the NRC published an amendment to its regulations, effective immediately, to reflect the establishment of the Office of Inspector General (OIG) by formally removing references to the Office of Inspector and Auditor (OIA) from its regulations. The authority and responsibility for OIA functions have been transferred to the OIG.

Enforcement of Nondiscrimination on the Basis of Handicap in Federally Assisted Programs

On December 19, 1991 (55 FR 52136), the NRC published an amendment to its regulations, effective January 18, 1991, that added a cross-reference to the Uniform Federal Accessibility Standards to the provisions implementing Section 504 of the Rehabilitation Act of 1973. This amendment is necessary to diminish the possibility that recipients of Federal financial aid would be subject to conflicting enforcement standards.

Operations Center Area Code Telephone Number Change—Parts 20 and 50

On January 10, 1991 (56 FR 994), the NRC published an amendment to its regulations, effective immediately, to change the current area code telephone number at the NRC Operations Center from (202) to (301). This action is necessary to implement changes initiated by the C&P Telephone Company to accommodate the increasing demand for telephone numbers in the metropolitan Washington, D.C. area.

Access Authorization Fee Schedule for Licensee Personnel—Parts 11 and 25

On February 14, 1991 (56 FR 5926), the NRC published an amendment to its regulations, effective immediately, to revise the fee schedule for background investigations of licensee personnel who acquire access to National Security Information and/or Restricted Data and access to or control over Special Nuclear Material. This action is necessary to inform the public of the changes to the fee schedules in the NRC's regulations.

Procedures Applicable to Proceedings for the Issuance of Licenses for the Receipt of High-Level Radioactive Waste at a Geologic Repository—Part 2

On February 26, 1991 (56 FR 7787), the NRC published an amendment to its regulations, effective March 28, 1991, concerning the Rules of Practice for the licensing of high-level radioactive waste at a geologic repository. This action enhances the Commission's ability to comply with the schedule for the Commission's decision on the construction authorization for the repository contained in Section 114(d) of the Nuclear Waste Policy Act of 1982, as amended, while providing for the thorough technical review of the license application and the fair treatment of the parties to the hearing.

Assistance to Prospective Petitioners—Part 2

On March 12, 1991 (56 FR 10359), the NRC published an amendment to its regulations, effective immediately, concerning its procedures for filing a petition for rulemaking with the NRC. This action is necessary to clarify the type of assistance that the NRC may provide to a prospective petitioner.

ASNT Certification of Industrial Engineers—Part 34

On March 19, 1991 (56 FR 11504), the NRC published an amendment to its regulations, effective April 18, 1991, concerning radiographic operations to provide license applicants and licensees the option to affirm that individuals acting as radiographers will be certified in radiation safety by the American Society for Nondestructive Testing (ASNT) prior to commencing duties as radiographers. The intent of this rulemaking is to encourage industrial radiography licensees and license applicants to participate in the ASNT program because the Commission believes that this program can contribute significantly to improved safety.

Access Authorization Program for Nuclear Power Plants—Part 73

On April 25, 1991 (56 FR 18997), the NRC published an amendment to its regulations to require an access authorization program for individuals requiring unescorted access to protected and vital areas at nuclear power plants. This amendment, effective May 28, 1991, will minimize the likelihood that unescorted access to protected and vital areas will be given to individuals whose background, psychological profile, or changes in behavioral pattern indicate a potential for committing acts that are, or could be, detrimental to the public health and safety. A confirmation of effective date for information collection requirements for this rule was published on September 20, 1991 (56 FR 47671).

Change in Commercial Telephone Number for Region V—Parts 20, 21, and 73

On April 26, 1991 (56 FR 19253), the NRC published an amendment to its regulations, effective immediately, to indicate a change in the commercial telephone number for the NRC's Region V Office, located in Walnut Creek, Cal. These amendments are necessary to inform the public of these administrative changes to NRC regulations.

Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events—Part 50

On May 15, 1991 (56 FR 22300), the NRC published an amendment to its regulations for light-water nuclear power plants to change the procedure for calculating the amount of radiation embrittlement that a reactor vessel receives. This amendment, effective June 14, 1991, establishes a screening criterion that limits the amount of embrittlement of a reactor vessel beltline material beyond which the plant cannot continue to operate without justification based on a plant-specific analysis. This amendment also prescribes the procedures that must be used for calculating the amount of embrittlement for comparison to the screening criterion.

Standards for Protection Against Radiation—Parts 2, 19, 20, 30, 31, 32, 34 35, 39, 40, 50, 61 and 70

On May 21, 1991 (56 FR 23369), the NRC published an amendment to its regulations revising its standards for protection against ionizing radiation. This amendment, effective June 20, 1991, is necessary to incorporate updated scientific information and to reflect changes in the basic philosophy of radiation protection. The revision conforms the Commission's regulations to the Presidential Radiation Protection Guidance to Federal Agencies for Occupational Exposure and to recommendations of national and international radiation protection organizations.

Return of Topaz II Reactor to Soviet Union—Part 110

On May 31, 1991 (56 FR 24682), the NRC published an amendment to its regulations pertaining to import and export of nuclear equipment and material to permit the return of the Topaz II Reactor System to the Union of Soviet Socialist Republics (U.S.S.R). This rulemaking action, effective immediately, permits the export of Topaz II, which is owned by the Government of the U.S.S.R., without issuance of a license by the NRC.

Procedures for Direct Commission Review of Decisions of Presiding Officers— Parts 0, 1 and 2

On June 27, 1991 (56 FR 29403), the NRC published an amendment to its regulations, effective July 29, 1991, to establish a new system for agency appellate review of decisions and actions of presiding officers in all formal and informal agency adjudications. The new system provides for discretionary review by the Commissioners of the NRC of most partial and final initial decisions, referred rulings, and certifications of questions.

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants—Part 50

On July 10, 1991 (56 FR 31306), the NRC published an amendment to its regulations, effective July 10, 1996, to require commercial power plant licensees to monitor the effectiveness of maintenance activities for safety significant plant equipment in order to minimize the likelihood of failures and events caused by the lack of effective maintenance.

Revision of Fee Schedules; 100 Percent Fee Recovery—Parts 52, 71, 170 and 171

On July 10, 1991 (56 FR 31472), the NRC published an amendment to its regulations governing the licensing, inspection, and annual fees charged to its licensees. The amendments, effective August 9, 1991, are necessary to implement Public Law 101-508, passed by the Congress on November 5, 1990, which mandates that the NRC recover approximately 100 percent of its budget authority ($465 million) in fiscal year 1991, and the four succeeding years.

Operators' Licenses—Parts 2 and 55

On July 15, 1991 (56 FR 32066), the NRC published an amendment to its regulations to specify that the conditions and cutoff levels established pursuant to the Commission's Fitness-for-Duty Programs are applicable to licensed operators as conditions of their licenses. This amendment, effective August 14, 1991, provides a basis for taking enforcement actions against licensed operators: (1) who use drugs or alcohol in a manner that would exceed the cutoff levels contained in the fitness-for-duty rule; (2) who are determined by a facility medical review officer to be under the influence of any prescription or over-the-counter drug that would adversely affect his or her ability to safety and competently perform licensed duties; or (3) who sell, use or possess illegal drugs.

Duplication Fees—Part 9

On July 15, 1991 (56 FR 32070), the NRC published an amendment to its regulations that revises the charges for copying records publicly available at the NRC Public Document Room in Washington, D.C. This amendment, effective immediately, is necessary in order to reflect the change in copying charges resulting from the Commission's award of a new contract for the copying of records.

Standards for Protection Against Radiation: Monitoring Reports—Part 20

On July 15, 1991 (56 FR 32071), the NRC published an amendment to its regulations concerning the submittal of radiation exposure monitoring reports. The amendment, effective immediately, changes the address to which the licensee submits reports on an individual's exposure to radiation and radioactive material to the NRC.

Quality Management Program and Misadministration—Parts 2 and 35

On July 25, 1991 (56 FR 34104), the NRC published an amendment to its regulations, effective January 27, 1992, governing therapeutic administrations of byproduct material and certain uses of radioactive sodium iodide to require implementation of a quality management program to provide high confidence that the byproduct material or radiation from byproduct material will be administered as directed by an authorized user physician.

Criteria and Procedures for the Reporting of Defects and Conditions of Construction Permits—Parts 21 and 50

On July 31, 1991 (56 FR 36081), the NRC published an amendment to its regulations on the reporting of safety defects. The amendments, effective October 29, 1991, will reduce duplicate reporting of defects, clarify the criteria for reporting defects, and establish uniform time periods for reporting and uniform requirements for the content of safety defect reports.

Imports from South Africa—Part 110

On August 13, 1991 (56 FR 38335), the NRC published an amendment to its regulations pertaining to the import of source material or special nuclear material from South Africa to permit uranium manufactured or produced in South Africa to be imported into the United States under general license. This amendment, effective immediately, is necessary to conform the Commission's regulations to Executive Order 12769, issued by the President on July 10, 1991, which, among other things, terminates the prohibition on nuclear trade with South Africa in Section 309 and 311 of the Comprehensive Anti-Apartheid Act of 1986.

Emergency Response Data System—Part 50

On August 13, 1991 (56 FR 40178), the NRC published an amendment to its regulations to require licensees of all operating nuclear power facilities except Big Rock Point (Mich.) to participate in the Emergency Response Data System (ERDS) program. This amendment, effective September 12, 1991, requires licensees to submit to the NRC timely and accurate data on a limited set of parameters whose values indicate the condition of the plant during a declaration of an alert or higher emergency classification.

Revisions to Procedures to Issue Orders; Deliberate Misconduct by Unlicensed Persons—Parts 2, 30, 40, 50, 60, 61, 70, 72, 110 and 150

On August 15, 1991 (56 FR 40664), the NRC published an amendment to its regulations, effective September 16, 1991, that establishes procedures to be used in issuing order to licensed and unlicensed persons to provide reasonable assurance that licensed activities will be conducted in a manner that will protect the public health and safety. The NRC is also revising its Enforcement Policy to reflect these new amendments.

Notification of Incidents—Parts 20, 30, 31, 34, 39, 40 and 70

On August 16, 1991 (56 FR 40757), the NRC published an amendment to its regulations to revise material licensee reporting requirements for byproduct, source, and special nuclear material regarding the incidents related to radiation safety. This amendment, effective October 15, 1991, is necessary to ensure that significant occurrences at material licensee facilities are promptly reported to the NRC so that the Commission can evaluate whether the licensee has taken appropriate action to protect the public health and safety and whether prompt NRC action is necessary to address generic safety concerns.

Change in Commercial Telephone Number for Region V—Parts 20, 21 and 73

On August 21, 1991 (56 FR 41448), the NRC published an amendment to its regulations, effective September 2, 1991, to indicate a change in the commercial telephone number for the NRC's Region V Office, located in Walnut Creek, Cal.

Fitness-for-Duty Programs—Part 26

On August 26, 1991 (56 FR 41922), the NRC published an amendment to its regulations governing fitness-for-duty programs that are applicable to licensees who are authorized to construct or operate nuclear power reactors. This amendment, effective September 25, 1991, clarifies the NRC's intent concerning the unacceptability of taking action against an individual that is based solely on the preliminary results of a drug screening test and to permit, under certain conditions, employment actions, up to and including the action of temporary removal from unescorted access or from normal duties, based on an unconfirmed positive result from an initial screening test for marijuana or cocaine.

Program Fraud Civil Remedies Act—Part 13

On September 18, 1991 (56 FR 47132), the NRC published an amendment to its regulations, effective October 18, 1991, to establish the procedures the Commission will follow in implementing the provisions of the Program Fraud Civil Remedies Act of 1986 (the Act) and to specify the hearing and appeal rights of persons subject to penalties and assessments under the Act. The Act authorizes certain Federal agencies, including the Nuclear Regulatory Commission, to impose, through administrative adjudication, civil penalties and assessments against any person who makes, submits, or presents a false, fictitious, or fraudulent claim or written statement to the agency.

REGULATIONS AND AMENDMENTS PROPOSED

Emergency Response Data System—Part 50

On October 9, 1991 (55 FR 41095), the NRC published a notice of proposed rulemaking that would require licensees to participate in the Emergency Response Data System (ERDS) program and to set a definite schedule for its implementation. The ERDS is a direct electronic data link between computer data systems used by licensees and the NRC Operations Center.

Options and Procedures for Direct Commission Review of Licensing Board Decisions—Part 2

On October 24, 1991 (55 FR 42947), the NRC published a notice of proposed rulemaking that would provide rules of procedure for direct Commission review of the decisions of presiding officers in all formal and informal adjudicatory proceedings. These regulatory changes are necessitated by the Commission's decision to abolish the Atomic Safety and Licensing Appeal Panel.

Licenses and Radiation Safety Requirements for Large Irradiators—Parts 19, 20 21, 30, 36, 40, 51, 70 and 170

On December 4, 1991 (55 FR 50008), the NRC published a notice of proposed rulemaking that would establish a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in large irradiators.

Material Control and Accounting Requirements for Uranium Enrichment Facilities Producing Special Nuclear Material of Low Strategic Significance—Parts 2, 40, 70 and 74

On December 17, 1991 (55 FR 51726), the NRC published a notice of proposed rulemaking that would establish new performance-based material control and accounting requirements that would be applicable to uranium enrichment facility licensees who produce significant qualities of special nuclear material of low strategic significance. The proposed amendment would impose additional requirements to ensure that enrichment facilities would produce only enriched uranium of low strategic significance as authorized.

Codes and Standards for Nuclear Power Plants—Part 50

On January 31, 1991 (56 FR 3796), the NRC published a notice of proposed rulemaking that would incorporate 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, with a specified modification. The proposed amendment would impose augmented examination of the reactor vessel shell welds, and would separate in the regulations the requirements for inservice testing from those for inservice inspection, by placing the requirements for inservice testing in a separate paragraph. Revision of Fee Schedules; 100 Percent Fee Recovery— Parts 71, 170 and 171

On April 12, 1991 (56 FR 14870), the NRC published a notice of proposed rulemaking that would amend provisions governing the licensing, inspection, and annual fees charged to its licensees. These changes are necessary to implement Public Law 101-508, passed by the Congress on November 5, 1990, which mandate that the NRC recover approximately 100 percent of its budget authority in fiscal year 1991.

NRC Licensee Reinvestigation Program—Part 25

On July 31, 1991 (56 FR 36113), the NRC published a notice of proposed rulemaking that would require a reinvestigation program for NRC licensee personnel with "Q" and "L" access authorizations and to amend the fee schedule to cover investigative costs. This amendment is necessary to achieve a higher level of assurance that licensee personnel with access to Restricted Data or National Security Information remain eligible for such access.

Decommissioning Funding for Prematurely Shutdown Power Reactors—Part 50

On August 21, 1991 (56 FR 41493), the NRC published a notice of proposed rulemaking that would amend its regulations on the timing of the collection of funds for decommissioning for those nuclear power reactors that have shut down before the expected end of their operating lives. The proposed rule would require 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 plant.

Uranium Enrichment Regulations—Parts 2, 40, 50, 51, 70, 75, 110, 140, 150 and 170

On September 16, 1991 (56 FR 46739), the NRC published a notice of proposed rulemaking concerning the licensing of uranium enrichment facilities that would reflect changes made to the Atomic Energy Act of 1954, as amended (Act) by the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990. The proposed rule would ensure that uranium enrichment facilities will be licensed subject to the provisions of the Act pertaining to source material and special nuclear material rather than under the provisions pertaining to a production facility.

Environmental Review for Renewal of Operating Licenses—Part 51

On September 17, 1991 (56 FR 47016), the NRC published a notice of proposed rulemaking that would establish new requirements for environmental review of applications to renew operating licenses for nuclear power plants. The proposed rule would define the number and scope of environmental impacts that would need to be addressed as part of a license renewal program.

DOE-L or DOE-Q Reinvestigation Program for NRC-R Access Authorization Renewal Requirements—Part 11

On September 30, 1991 (56 FR 49435), the NRC published a notice of proposed rulemaking that would allow an exception to NRC-R access authorization rnewal requirements. The proposed rule is intended to reduce administrative and investigative costs to the licensee and administrative costs to the Federal Government.

Appendix 5

Regulatory Guides—Fiscal Year 1991

NRC regulatory guides describe methods acceptable to the NRC staff of implementing specific parts of the Commission'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 October 1, 1990, to September 30, 1991.

Division 1—Power Reactor Guides

1.17 Withdrawn. Protection of Nuclear Power Plants Against Industrial Sabotage (Revision 1)

1.58 Withdrawn. Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Revision 1)

1.64 Withdrawn. Quality Assurance Requirements for the Design of Nuclear Power Plants (Revision 2)

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 27)

1.85 Materials Code Case Acceptability—ASME Section III, Division 1 (Revision 27)

1.88 Withdrawn. Collection, Storage, and Maintenance of Nuclear Power
Plant Quality Assurance Records (Revision 2)

1.123 Withdrawn. Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Revision 1)

1.144 Withdrawn. Auditing of Quality Assurance Programs for Nuclear Power Plants (Revision 1)

1.146 Withdrawn. Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 8)

Division 2—Research and Test Reactor Guides

None

Division 3—Fuels and Materials Facilities Guides

None

Division 4—Environmental and Siting Guides

None

Division 5—Materials and Plant Protection Guides

5.66 Access Authorization Program for Nuclear Power Plants

Division 6—Product Guides

None

Division 7—Transportation Guides

7.11 Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of Four Inches (0.1 m)

7.12 Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater Than Four Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)

Division 8—Occupational Health Guides

None

Division 9—Antitrust and Financial Review Guides

None

Division 10—General Guides

None

DRAFT GUIDES

Division 1

DG-1008 Reactor Coolant Pump Seals

DG-1009 Standard Format and Content of Technical Information for Applications To Renew Nuclear Power Plant Operating Licenses

Division 3

DG-3003 Format and Content for the License Application for the High-Level Waste Repository

Division 4

DG-4002 Proposed Supplement 1 to Regulatory Guide 4.2, Guidance for the Preparation of Supplemental Environmental Reports in Support of an Application To Renew a Nuclear Power Station Operating License

Division 5

DG-5002 Material Control and Accounting for Uranium Enrichment Facilities Authorized To Produce Special Nuclear Material of Low Strategic Significance

SG301-4 Withdrawn. Standard Format and Content Guide for Access Authorization Plans for Nuclear Power Plants

Division 8

DG-8003 Proposed Revision 1 to Regulatory Guide 8.25, Air Sampling in the Workplace

Appendix 6

Civil Penalties and Orders—Fiscal Year 1991

CIVIL PENALTY ACTIONS IN FISCAL YEAR 1991 (Organized According to Enforcement Action Numbers)

Licensee, Facility, and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 91*	*Summary*
Illinois Power Company (Clinton) EA 86-143	$50,000 proposed in FY 87; withdrawn in FY 91	Discrimination against whistleblower.
Eastern Testing and Inspection, Inc. (Pennsauken, NJ) EA 87-079	$6,500 proposed in FY 87; imposed in FY 89; payments being made over time	Violations involving use of dosimeters, audits, transportation of source.
Basin Testing Laboratory, Inc. (Williston, ND) EA 88-265	$5,000 proposed in FY 89; imposed in FY 90; payments being made over time	Use of unqualified person to perform radiography, transportation of licensed materials, providing inaccurate information to NRC.
Minnesota Mining and Manufacturing (3M) (Minneapolis, MN) EA 89-028	$160,000 proposed in FY 90; $117,500 imposed and paid in FY 91	Violations involving control and reporting of leaking static eliminators.
Process Technology (Rockaway, NJ) EA 89-080	$13,000 proposed and paid in FY 91	Violations of interlocks at irradiator.
C&R Laboratories (Pearl City, HI) EA 89-101	$1,500 proposed and paid in FY 91	Failure to survey and creation of false survey records.
T.V.A. (Watts Bar) EA 89-201	$240,000 proposed and imposed in FY 90; paid in FY 91	Discrimination against whistleblowers.
Cambridge Medical Diagnostics (Billerica, MA) EA 89-233	$8,000 proposed and imposed in FY 90; withdrawn in FY 91 upon discontinuation of operations	Airborne releases in excess of regulatory limits.
SYNCOR Corporation OH) EA 90-053	$20,000 proposed in FY 90; imposed and paid in FY 91	Willful failure to follow NRC requirements; (Blue Ash, falsification of test records.
Dr. G. Anthony Doener (Freehold, NJ) EA 90-061	$1,000 proposed in FY 90; imposed and paid in FY 91	Breakdown in control of radiation safety program.

Licensee, Facility, and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 91*	*Summary*
PX Engineering Company (Boston, MA) EA 90–065	$7,500 proposed and imposed in FY 91; pending	Providing inaccurate information to NRC.
University of Puerto Rico (San Juan, PR) EA 90- 076	$12,500 proposed in FY 90; imposed and paid in FY 91	Violations involving control of material, calibrations, survey, leak tests, inventories, and annual reviews.
Mississippi X-Ray Service (Wesson, MS) EA 90–095	$7,500 proposed in FY 90; imposed and later withdrawn in FY 91	Radiography violations, including failure to maintain surveillance and failure to survey.
University of Wisconsin (Madison, WI) EA 90–098	$7,500 proposed in FY 90; imposed and paid in FY 91	Violations involving therapy misadministrations; untrained operators; unverified treatment plans.
Barnett Industrial X-Ray (Stillwater, OK) EA 90–102	$7,500 proposed in FY 90; imposed in FY 91; payments being made over time	Overexposure.
High Mountain Inspection (Mills, WY) EA 90–104	$2,500 proposed in FY 90; imposed and paid in FY 91	Radiography violations, including failure to survey and to supervise.
Newman Memorial Hospital (Shattuck, OK) EA 90–106	$5,000 proposed in FY 90; imposed and paid in FY 91	Inadequate oversight by radiation safety committee; failures by radiation safety officer.
Illinois Power Company (Clinton) EA 90–108	$112,500 proposed and paid in FY 91	Failure to promptly identify and correct degraded conditions in service water system.
Professional Service (Pittsburgh, PA) EA 90–112	$16,000 proposed, $14,000 imposed and paid in FY 91	Falsification of NRC-required records.
Consolidated Edison Co. (Indian Point) EA 90–114	$62,500 proposed and paid in FY 91	Failure to follow procedure; falsification of test documents.
Arizona Public Service (Palo Verde) EA 90–121	$125,000 proposed and paid in FY 91	Violations of fire protection requirements.
Georgia Power Company (Vogtle) EA 90–129	$40,000 proposed and paid in FY 91	Late emergency notifications.
Carolina Power & Light (Brunswick) EA 90–130	$62,500 proposed in FY 90; paid in FY 91	Potential overexposure.

Licensee, Facility, and EA Number	Civil Penalties Proposed, Imposed and/or Paid in FY 91	Summary
Tripler Medical Center (Honolulu, HI) EA 90-132	$5,000 proposed, $2,500 imposed and paid in FY 91	Exposure of nursing infant.
General Motors (Saginaw, MI) EA 90-137	$875 proposed in FY 90; paid in FY 91	Lost gauge.
Alt & Witzig Engineering (Indiannapolis, IN) EA 90-141	$2,500 proposed and paid in FY 91	Failure to control access to licensed material; transportation violations.
Tri-State Associates (Woodbridge, VA) EA 90-142	$7,500 proposed in FY 90; $3,750 imposed in FY 91; payments being made over time.	Overexposure and failure to report.
SYNCOR Corporation (Folcroft, PA) EA 90-144	$12,500 proposed and paid in FY 91	Unsecured and unattended materials in unrestricted areas.
Arizona Public Service (Palo Verde) EA 90-147	$75,000 proposed in FY 90; imposed and paid in FY 91	Inadequate control of licensed operator medical examination program.
G.P.U. (Oyster Creek) EA 90-148	$75,000 proposed and paid in FY 91	Failure to properly implement operator requalification program.
Midwest Inspection Service (Green Bay, WI) EA 90-152	$10,000 proposed, $8,751 imposed in FY 91; pending	Breakdown in control of radiation safety program.
Pennsylvania Power & Light (Susquehanna) EA 90-156	$25,000 proposed and paid in FY 91	Resolution of nonconformance reports regarding environmental qualification of Limitorque valve actuators.
North Detroit Hospital (Detroit, MI) EA 90-160	$2,500 proposed and paid in FY 91	Breakdown in control of radiation safety program.
Roche Professional Services (Philadelphia, PA) EA 90-161	$7,500 proposed and paid in FY 91	Providing false information to NRC; unauthorized use of licensed material.
Cabell Huntington Hospital (Huntington, WV) EA 90-163	$3,750 proposed and paid in FY 91	Breakdown in control of radiation safety program.
Commonwealth Edison Co. (Dresden) EA 90-168	$37,500 proposed and paid in FY 91	Inadequate temporary containment air sampling.

Licensee, Facility, and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 91*	*Summary*
West Shore Hospital (Manistee, MI) EA 90-172	$4,375 proposed and paid in FY 91	Diagnostic misadministration; breakdown in control of radiation safety program.
Arkansas Power & Light (ANO) EA 90-175	$50,000 proposed and paid in FY 91	Inoperable control room emergency ventilation system.
Power Authority of NY (Indian Point) EA 90-178	$50,000 proposed, imposed, and paid in FY 91	Inattentive control room operators.
Milwaukee County Medical (Milwaukee, WI) EA 90-181	$3,750 proposed and paid in FY 91	Breakdown in control of radiation safety program.
McCallum Testing Lab. (Chesapeake, VA) EA 90-183	$800 proposed and paid in FY 91	Unsecured gauge left in unattended vehicle by employee.
Baltimore Gas & Electric (Calvert Cliffs) EA 90-186	$12,500 proposed and paid in FY 91	Shift supervisor instructed guards to turn off metal.
Georgia Power Company (Vogtle) EA 90-196	$50,000 proposed and paid in FY 91	Inadequate protection of Safeguards Information.
Indiana & Michigan Electric (D.C. Cook) EA 90-194	$150,000 proposed and paid in FY 91	Fire protection violations.
Fewell Geotechnical Engineering, Ltd. (Pearl City, HI) EA 90-196	$20,000 proposed in FY 91; pending	Failure to survey and to lock source; false information.
Albert Einstein Medical Center (Philadelphia, PA) EA 90-197	$2,500 proposed and paid in FY 91	Control of licensed material.
T.V.A. (Sequoyah) EA 90-200	$30,000 proposed and paid in FY 91	Multiple failures to follow overtime procedures.
Southern California Edison Company (San Onofre) EA 90-201	$150,000 proposed and paid in FY 91	Containment sump valve left open, steam-driven auxiliary feedwater
Commonwealth Edison Co. (Quad Cities) EA 90-203	$50,000 proposed and paid in FY 91	Reactivity control event.

Licensee, Facility, and EA Number	Civil Penalties Proposed, Imposed and/or Paid in FY 91	Summary
Commonwealth Edison Co. (Braidwood) EA 90-208	$87,500 proposed and paid in FY 91	Loss of reactor coolant system inventory during residual heat removal testing.
V.A. Medical Center (Albany, NY) EA 90-209	$3,750 proposed and $3,333 paid in FY 91	Breakdown in control of radiation safety program.
Muskogee Medical Center (Muskogee, OK) EA 90-212	$1,250 proposed and paid in FY 91	Therapy misadministration.
Western Stress, Inc. (Houston, TX) EA 90-213	$15,000 proposed and paid in FY 91	Overexposure resulting from failure to survey; removal of personal dosimetry; providing false information.
Virginia Electric & Power Company (Surry) EA 90-215	$50,000 proposed and paid in FY 91	Recirculation spray heat exchangers inoperable.
Northeast Utilities (Millstone) EA 90-219	$50,000 proposed and paid in FY 91	Containment integrity violations.
University of Cincinnati (Cincinnati, OH) EA 91-001	$8,750 proposed, imposed, and paid in FY 91	Breakdown in control of radiation safety program.
Wolf Creek Nuclear Operating Corporation EA 91-003	$25,000 proposed and paid in FY 91	Safety injection pumps inoperable due to frozen recirculation line.
Nuclear Fuel Services (Erwin, TN) EA 91-004	$10,000 proposed and paid in FY 91	Failure to adhere to criticality control rules.
Portland General Electric (Trojan) EA 91-005	$50,000 proposed and paid in FY 91	Operator medical qualification.
Louisiana Power & Light (Waterford) EA 91-006	$37,500 proposed and paid in FY 91	Inoperability of control room ventilation system.
Commonwealth Edison Co. (Dresden) EA 91-014	$100,000 proposed and paid in FY 91	Loss of containment integrity due to leaking valve.
Texas Utilities Generating Company (Comanche Peak) EA 91-015	$50,000 proposed and paid in FY 91	Fire watch log falsification.
Northeast Utilities (Millstone) EA 91-016	$37,500 proposed and paid in FY 91	Degraded service water system.

Licensee, Facility, and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 91*	*Summary*
Commonwealth Edison Co. (Quad Cities) EA 91-018	$112,500 proposed and paid in FY 91	Loss of reactor coolant event due to inadequate control of post-maintenance testing.
Youngstown State Univ. (Youngstown, OH)	$625 proposed and paid in FY 91	Management breakdown and radiation safety violations.
Carolina Power & Light (Brunswick) EA 91-023	$50,000 proposed and paid in FY 91	Failure to adhere to maintenance procedures.
Upjohn Company (Kalamazoo, MI) EA 91-024	$5,000 proposed and paid in FY 91	Radiation safety violations..
Georgia Power Company (Hatch)	$50,000 proposed and paid in FY 91	Potential overexposure; unauthorized traversing incore probe operation.
Laramie County Memorial Hospital (Cheyenne, WY) EA 91-033	$2,500 proposed and paid in FY 91	Multiple violations of 10 CFR Part 35.
Northeast Utilities (Millstone) EA 91-034	$50,000 proposed and paid in FY 91	Degradation of service water system and failure of shift supervisor to trip circulation water pumps.
McDowell & Associates (Ferndale, MI) EA 91-040	$375 proposed and paid in FY 91	Lost moisture density gauge.
Yankee Atomic Electric Company (Yankee Rowe) EA 91-042	$50,000 proposed and paid in FY 91	QA & QC deficiencies during emergency diesel generator replacement.
T.V.A. (Sequoyah) EA 91-043	$75,000 proposed and paid in FY 91	Failure to respond to diesel air pressure alarm.
Carolina Power & Light (Brunswick) EA 91-045	$87,500 proposed and paid in FY 91	Failure to adhere to procedures during emergency diesel generator maintenance.
Power Authority of New York (Fitzpatrick) EA 91-048	$137,500 proposed and paid in FY 91	Unmonitored radioactive release to an unrestricted area.
Gulf States Utilities (River Bend) EA 91-059	$37,500 proposed and paid in FY 91	Repetitive violations of high radiation area boundaries.
Chemetron Corporation (Newburgh Heights, OH) EA 91-060	$7,500 proposed in FY 91; pending	Contamination of site; loss of control of material.

Licensee, Facility, and EA Number	Civil Penalties Proposed, Imposed and/or Paid in FY 91	Summary
Industrial NDT Company (N. Charlestown, SC) EA 91-061	$5,000 proposed and paid in FY 91	Temporary loss of radiography device.
Flordia Power & Light (St. Lucie) EA 91-062	$37,500 proposed and paid in FY 91	Closed cooling water outlet valve.
Rutgers University (New Brunswick, NJ) EA 91-070	$6,250 proposed in FY 91; pending	Breakdown in control of radiation safety program.
Houston Light & Power (South Texas) EA 91-074	$75,000 proposed and paid in FY 91	Anticipated Transient Without Scram system reliability problems.
Carolina Power & Light (Shearon Harris) EA 91-076	$50,000 proposed and paid in FY 91	Auto trip channel inoperable.
Construction Engineering (Pittsburgh, PA) EA 91-077	$1,250 proposed in FY 91; pending	Failure to have alarm rate dosimeter.
Materials Testing & Inspection (Ft. Wayne, IN) EA 91-078	$1,750 proposed and paid in FY 91	Radiation safety violations, including overexposure, lack of radiation safety officer, and training.
Vermont Yankee (Vermont Yankee) EA 91-081	$75,000 proposed and paid in FY 91	Failure to follow procedures in responding to alarms.
T.V.A. (Browns Ferry) EA 91-083	$75,000 proposed and paid in FY 91	Containment integrity violation due to open airlocks.
Cleveland Clinic Foundation (Cleveland, OH) EA 91-084	$7,500 proposed and paid in FY 91	P-32 contamination; breakdown in control of radiation safety program.
Cotton Houston Services (Houston, TX) EA 91-087	$2,500 proposed and paid in FY 91	Failure to have alarming rate meter.
Public Service Corp. of Colorado (Ft. St. Vrain) EA 91-088	$62,500 proposed and paid in FY 91	Repetitive radiation protection violations.
University of Puerto Rico (San Juan, PR) EA 91-089	$6,250 proposed in FY 91; pending	Breakdown in control of radiation safety program.

Licensee, Facility, and EA Number	*Civil Penalties Proposed, Imposed and/or Paid in FY 91*	*Summary*
American Fibrit (Battlecreek, MI) EA 91-090	$1,500 proposed and paid in FY 91	Breakdown of fixed gauge program.
Alabama Power Company (Farley) EA 91-102	$25,000 proposed and paid in FY 91	Technical specification violation; turbine-driven auxiliary feedwater pump recirculation valve misalignment.
St. Luke's Hospital (Aberdeen, SD) EA 91-109	$3,750 proposed and paid in FY 91	Breakdown in control of radiation safety program.
Stone Container Corp. (Coshocton, OH) EA 91-112	$1,000 proposed and	Lost gauge.

ORDERS ISSUED IN FISCAL YEAR 1991 (Organized According to Enforcement Action Numbers)

Licensee, Facility, and EA Number	*Summary*
Fewell Geotechnical Engineering, Ltd. (Pearl City, HI) EA 90-190	Order Modifying License (effective immediately). Radiography violations, including failure to survey and lock source, and providing false information.
Tumbleweed X-Ray Co. (Greenwood, AR) EA 90-210	Order Modifying License (effective immediately). Overexposure to radiographer's assistant.
C&R Laboratories (Pearl City, HI) EA 90-216	Order Modifying License. Falsification of survey records.
Western Stress, Inc. (Houston, TX) EA 90-218	Order Modifying License (effective immediately). Overexposure to right hand and to whole body.
Tumbleweed X-Ray Co. (Greenwood, AR) EA 91-012	Order Suspending License (effective immediately). GeneralOverexposure.
Power Authority of New York (Fitzpatrick) EA 91-053	Order Modifying License (effective immediately). Reactor operator refused to cooperate with Fitness for Duty Program.
Power Authority of New York EA 91-054	Order Suspending License (effective immediately) and an Order to Show Cause Why License Should Not be Revoked. Reactor operator (individual operator) refused to cooperate with Fitness for Duty Program.
Midwest Inspection (Green Bay, WI) EA 91-085	Order Modifying License (effective immediately). Deliberate use of Service unqualified radiographer.

Appendix 7

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1991)

The following is a listing of the 120 nuclear power reactor electrical generating units which were in operation or under construction in the United States as of December 31, 1991, representing a total capacity of approximately 110,000 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 120 reactor units listed, 82 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, 1991. At that time, there were 112 commercial nuclear reactors in the United States with operating licenses; these units had been operating for a cumulative 1,486 reactor-years (an additional 107 reactor-years had been accumulated by reactors now permanently shut down). At the end of 1991, 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

ARIZONA (continued)

Wintersburg	Palo Verde Unit 3 nuclear power plant	1,304	PWR	OL 1987	Arizona Public Service Co.	1988

ARKANSAS

Russelville	Arkansas Nuclear One Unit 1 nuclear power plant	836	PWR	OL 1974	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One Unit 2 nuclear power plant	858	PWR	OL 1978	Arkansas Power & Light Co.	1980

CALIFORNIA

San Clemente	San Onofre Unit 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
Clay Station	Rancho Seco Unit 1 nuclear power plant	873	PWR	OL 1974	Sacramento Municipal Utility District	1975

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

FLORIDA (continued)

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
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

ILLINOIS (continued)

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	Wolf Creek nuclear power plant	1,150	PWR	OL 1985	Kansas Gas & Electric Co.	1985
LOUISIANA						
Taft	Waterford nuclear power plant	1,151	PWR	OL 1984	Louisiana Power & Light Co.	1985
St. Francisville	River Bend Unit 1 nuclear power plant	934	BWR	OL 1985	Gulf States Utilities Co.	1986
MAINE						
Wiscasset	Maine Yankee Atomic Power	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						
Rowe power plant	Yankee nuclear Electric Co.	175	PWR	OL 1960	Yankee Atomic	1961
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 power plant	Palisades nuclear	635	PWR	OL 1971	Consumers Power Co.	1971
Laguna Beach	Fermi Unit 2 nuclear power plant	1,093	BWR	OL 1985	Detroit Edison Co.	1988
Bridgman	Cook Unit 1 nuclear power plant	1,044	PWR	OL 1974	Indiana & Michigan Electric Co.	1975
Bridgman	Cook Unit 2 nuclear power plant	1,082	PWR	OL 1977	Indiana & Michigan Electric Co.	1978

MINNESOTA						
Monticello	Monticello nuclear power plant	525	BWR	OL 1970	Northern States Power Co.	1971
Red Wing	Prairie Island Unit 1 nuclear power plant	503	PWR	OL 1973	Northern States Power Co.	1973
Red Wing	Prairie Island Unit 2 nuclear power plant	500	PWR	OL 1974	Northern States Power Co.	1974
MISSISSIPPI						
Port Gibson	Grand Gulf Unit 1 nuclear power plant	1,250	BWR	OL 1982	Mississippi Power & Light Co.	1985
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

NEW YORK (continued)

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

PENNSYLVANIA (continued)

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 & Light Co.	BWR	OL 1984	Pennsylvania Power 1985	

SOUTH CAROLINA

Hartsville	Robinson Unit 2 nuclear power plant	665	PWR	OL 1970	Carolina Power & Light Co.	1971
Seneca	Oconee Unit 1 nuclear power plant	860	PWR	OL 1973	Duke Power Co.	1973
Seneca	Oconee Unit 2 nuclear power plant	860	PWR	OL 1973	Duke Power Co.	1974
Seneca	Oconee Unit 3 nuclear power plant	860	PWR	OL 1974	Duke Power Co.	1974
Broad River	Summer Unit 1 nuclear power plant	900	PWR	OL 1982	So. Carolina Electric & Gas Co.	1984
Lake Wylie	Catawba Unit 1 nuclear power plant	1,145	PWR	OL 1984	Duke Power Co.	1985
Lake Wylie	Catawba Unit 2 nuclear power plant	1,145	PWR	OL 1986	Duke Power Co.	1986

TENNESSEE

Daisy	Sequoyah Unit 1 nuclear power plant	1,128	PWR	OL 1980	Tennessee Valley Authority	1981
Daisy	Sequoyah Unit 2 nuclear power plant	1,148	PWR	OL 1981	Tennessee Valley Authority	1982
Spring City	Watts Bar Unit 1 nuclear power plant	1,165	PWR	CP 1973	Tennessee Valley	1988
Spring City	Watts Bar Unit 2 nuclear power plant	1,165	PWR	CP 1973	Tennessee Valley Authority	1989

TEXAS

Glen Rose	Comanche Peak Unit 1 nuclear power plant	1,150	PWR	OL 1990	Texas Utilities	1988
Glen Rose	Comanche Peak Unit 2 nuclear power plant	1,150	PWR	CP 1974	Texas Utilities	1989

TEXAS (continued)						
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 Supply System	Indef.
Richland	WPPSS No. 2 (Hanford) nuclear power plant	1,103	BWR	OL 1983	Wash. Public Power Supply System	1984
Satsop	WPPSS No. 3	1,242	PWR	CP 1978	Wash. Public Power Supply System	Indef.
WISCONSIN						
Two Creeks	Point Beach Unit 1 nuclear power plant	495	PWR	OL 1970	Wisconsin Electric Power Co.	1970
Two Creeks	Point Beach Unit 2 nuclear power plant	495	PWR	OL 1971	Wisconsin Electric Power Co.	1972
Kewaunee	Kewaunee nuclear power plant	515	PWR	OL 1973	Wisconsin Public Service Corp.	1974

U.S. Nuclear Power Plants with Operating Licenses

(Plant – type – MWe – cp – ol)*

Arkansas 1 = pwr, 836, 12/68, 5/74.
Arkansas 2 = pwr, 858, 12/72, 12/78.
Beaver Valley 1 (Pa.) = pwr, 810, 6/70, 7/76.
Beaver Valley 2 = pwr, 833, 5/74, 8/87.
Big Rock Point (Mich.) = bwr, 69, 5/60, 5/64.
Braidwood 1 (Ill.) = pwr, 1120, 12/75, 7/87.
Braidwood 2 = pwr, 1120, 12/75, 5/88.
Browns Ferry 1 (Ala.) = bwr, 1065, 5/67, 12/73.
Browns Ferry 2 = bwr, 1065, 5/67, 8/74.
Browns Ferry 3 = bwr, 1065, 5/67, 8/76.
Brunswick 1 (N.C.) = bwr, 790, 2/70, 11/76.
Brunswick 2 = bwr, 790, 2/70, 12/74.
Byron 1 (Ill.) = pwr, 1105, 12/75, 2/85.
Byron 2 = pwr, 1105, 12/75, 1/87.
Callaway (Mo.) = pwr, 1145, 4/76, 10/84.
Calvert Cliffs 1 (Md.) = pwr, 825, 7/69, 7/74.
Calvert Cliffs 2 = pwr, 825, 7/69, 11/76.
Catawba 1 (S.C.) = pwr, 1129, 8/75, 1/85.
Catawba 2 = pwr, 1129, 8/75, 5/86.
Clinton (Ill.) = bwr, 930, 2/76, 4/86.
Comanche Peak 1 (Tex.) = pwr, 1150, 12/74, 4/90.
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.
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.
Rancho Seco (Cal.) = pwr, 873, 10/68, 8/74.
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.page 12
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.
Yankee-Rowe (Mass.) = pwr, 167, 11/57, 12/63.
Zion 1 (Ill.) = pwr, 1040, 12/68, 10/73.
Zion 2 = pwr, 1040, 12/68, 11/73.

Total as of 12/31/91 = 112.

Reactor projects for which construction permits were in effect** as of 12/31/91 (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 12/31/91 = 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.

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