ANNUAL REPORT 1989

United States Nuclear Regulatory Commission

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NRR• NMSS• OE• GPA• OP• OGC• OIG• OI• OC• IRM• REGIONS• ACRS• ACNW

UNITED STATES NUCLEAR REGULATORY COMMISSION

July 27, 1990

The President
The White House
Washington, DC 20500

Dear Mr. President:

This Annual Report for 1989 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 1989, with additional treatment of events after that period where circumstances warranted.

Respectfully,

Kenneth M. Carr

Kenneth M. Carr
Chairman

ANNUAL REPORT 1989

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 June 1989

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Table of Contents

Chapter 1—1989 Highlights/Licensing Summary
(1-3)

Chapter 2—Nuclear Reactor Regulation
(5-37)

Chapter 5—Safeguards and Transportation
(81-87)

Chapter 6—Waste Management
(89-99)

Chapter 7—Communicating with Government and the Public
(101-117)

Chapter 8—Nuclear Regulatory Research
(119-167)

Chapter 9—Proceedings and Litigation
(169-184)

Chapter 10—Management and Administrative Services
(185-198)

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 5; 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 contingency 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 NONPROLIFERACTION ACT OF 1978

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

1989 Highlights / Licensing Summary

Chapter 1

This is the 15th annual report of the U.S. Nuclear Regulatory Commission (NRC), covering events and activities occurring in fiscal year 1989 (the year ending September 30, 1989), with some treatment of events, where warranted, from the last quarter of the calendar year.

The NRC came into being under the Energy Reorganization Act of 1974 as an independent agency of the Federal Government. The five NRC Commissioners are nominated by the President and confirmed by the U.S. 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—as in the operation of nuclear power 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 through the licensing and regulatory oversight of nuclear reactor operations and other activities involving possession and use of nuclear materials, including the transport and disposal of nuclear materials and wastes; through the safeguarding of nuclear materials and facilities from theft and sabotage; through the issuance of rules and standards; and through inspection and enforcement actions.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1989 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 gives Commission and senior staff changes during the report period, a summary of licensing actions, and the status of agency consolidation.

Changes Within Commission and Senior Staff

Two changes occurred on the Commission during the year. In July, Commissioner Kenneth M. Carr began his term as Chairman of the NRC, succeeding Chairman Lando W. Zech, Jr., whose term had expired. Chairman Carr was first appointed to the Commission in August 1986. The vacancy created by the retirement of Chairman Zech was filled after the close of the fiscal year, when Dr. Forrest J. Remick, former Chairman of the Advisory Committee on Reactor Safeguards, was sworn in on December 1, 1989, bringing the Commission back to its full complement of five. (See Appendix 1 for listing of Commissioners and other senior NRC officials.)

In January 1989, the Commission appointed a second Deputy Executive Director for Operations (DEDO), to report directly to the Executive Director. Thus two DEDO positions were created, one designated DEDO for Nuclear Reactor Regulation, Regional Operations, and Research, the post to be filled by James M. Taylor (formerly Deputy Executive Director, and subsequently appointed Executive Director for Operations (see below)); and DEDO for Materials Safety, Safeguards and Operations Support, the post to be filled by Hugh L. Thompson, Jr. (formerly Director of the Office of Nuclear Materials Safety and Safeguards).

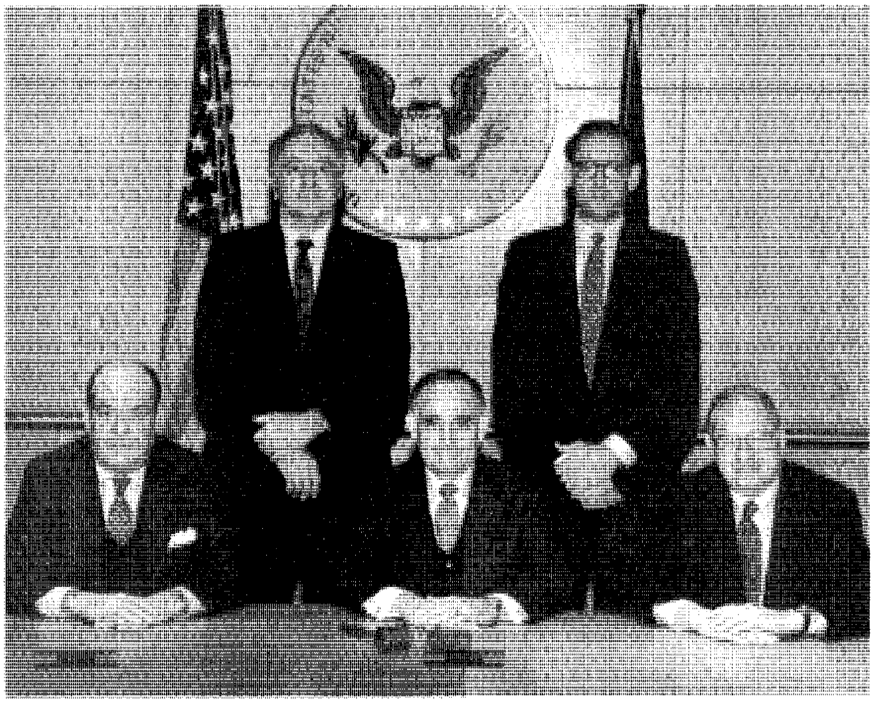
In February 1989, the Office of Administration and Resources Management (ARM) was abolished and the three constituent offices reinstated, namely,

- The Office of Administration, Patricia G. Norry, Director
- The Office of the Controller, Ronald M. Scroggins, Controller
- The Office of Information Resources Management, Joyce A. Amenta, Director.

In July 1989, the new Office of the Licensing Support System Administrator began operations (see Chapter 6), Lloyd J. Donnelly, Administrator.

On November 22, 1989, David C. Williams was sworn in as the NRC's first Inspector General (see Chapter 10). The Office of the Inspector General, created in April 1989, supplanted the former Office of Inspector and Auditor.

On December 1, 1989, James M. Taylor was named the NRC's Executive Director for Operations (EDO), succeeding Victor Stello, Jr., who joined the senior management of the Department of Energy. Mr. Taylor had been Acting EDO since July 1989.

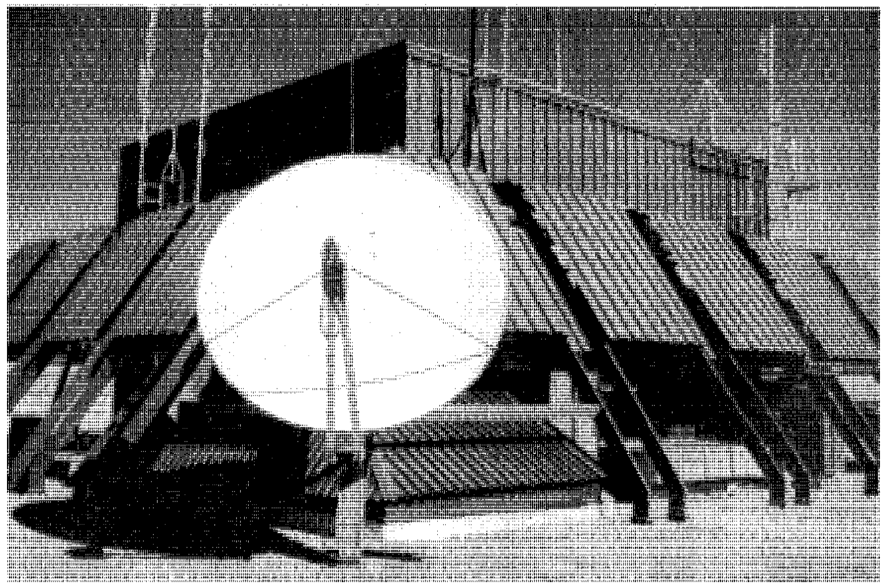


The full Commission is shown above. Seated, from left to right, are Commissioner Thomas M. Roberts, Chairman Kenneth M. Carr, and Commissioner Kenneth C. Rogers. Standing at left is Commissioner Forrest J. Remick and at right is Commissioner James R. Curtiss.

Chairman Carr, who had been a member of the Commission since August 1986, became the seventh Chairman of the NRC in July 1989. The position vacated when former Chairman Lando W. Zech, Jr., completed his term was filled when Dr. Remick was sworn in as Commissioner in December 1989.

In the photo below, assembled for his swearing-in ceremony, are newly appointed Commissioner Forrest J. Remick, at left, with members of his family at center and the Chairman and other Commissioners to the right. Dr. Remick, who has served as Chairman of the Advisory Committee on Reactor Safeguards, was on the NRC's Atomic Safety and Licensing Board Panel for 10 years. Dr. Remick has held a number of posts at the Pennsylvania State University, most recently that of Associate Vice President for Research and Professor of Nuclear Engineering.

Shown at right is the satellite dish installed on the roof of the NRC Region IV office building in Arlington, Tex. During fiscal year 1989, the NRC Satellite Network was completed, making videoconferencing available between and among all five Regional Offices and NRC Headquarters. (See Chapter 10).



The Office of Special Projects—which was established in February 1987 to deal with the particularly complex regulatory problems that had emerged at the Tennessee Valley Authority nuclear power plants and the Texas Utilities Electric Company units at Comanche Peak (Tex.)—was transferred on January 1, 1989, to the Office of Nuclear Reactor Regulation, as the Associate Directorate for Special Projects.

Power Reactor Licensing In Fiscal Year 1989

During the fiscal year, the NRC issued four low-power licenses and four full-power licenses (three of them to the recipients of the low-power licenses). There was one fuel-loading license issued (Limerick Unit 2 (Pa.)) during the report period. The addition of four full-power licenses brings the number of reactors licensed to operate at full power in the United States to 111, plus one facility with an operating license for less than full power operation—Seabrook (N.H.), licensed for low-power operation only. This brings the total of facilities licensed to operate to 112, excluding plants licensed but permanently shut down, as of September 30, 1989. At that time, there were 10 plants for which construction permits had been issued, but most of these are projects which have been halted and/or deferred. (See Appendix 7.)

Fuel Cycle and Byproducts Licensing

The NRC currently administers approximately 8,200 licenses for the possession and use of nuclear materials in medical and industrial applications. The 29 Agreement States administer about 16,000 additional licenses. The NRC Regional Offices administer all material licenses, with the exception of exempt distribution licensing and sealed-source and device design reviews, which are carried out at NRC Headquarters.

More than 120 licensing activities dealing with fuel cycle plants and facilities—such as fuel fabrication and fuel storage facilities—were carried out during fiscal year 1989. More than 5,500 licensing actions were taken on applications for new byproduct materials licenses and renewals of existing licenses. About 2,800 fuel facility and material licensee inspections were conducted during the period, and three team assessments of major materials licensees were performed.

Consolidation of NRC Headquarters

The first phase of the NRC Headquarter's consolidation effort was completed in April 1988 when the Commissioners moved into One White Flint North, located at 11555 Rockville Pike, Rockville, Md. The second phase, consisting of the construction and occupancy of a second building adjacent to One White Flint North, has been delayed. Completion and occupancy were previously scheduled for 1991. Montgomery County zoning, site plan and building permit review processes have taken longer to complete than expected. (See Chapter 10.)

Nuclear Reactor Regulation

Chapter 2

The Office of Nuclear Reactor Regulation (NRR) has responsibility for the licensing and regulatory oversight of nuclear reactors in the civilian sector. These include both nuclear power reactors operated by the electric utilities and non-power research reactors, such as those operated by the various universities. Not included are the reactors operated by the Department of Energy (DOE) for the purpose of furnishing fissionable materials used in nuclear weapons.

The licensing activities of NRR begin 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," later in this chapter.)

In recent years, the steady increase in the number of licensed operating nuclear plants and the corresponding decrease in the number of plants still under construction have brought about a substantial shift in NRC activity. Staff energies are currently directed mainly to the safety regulation of the 112 nuclear power plants now licensed for operation in the United States.

Regulatory activities related to nuclear power plants during fiscal year 1989 are treated in this chapter under the following headings:

- Status of Licensing
- Plant License Renewal
- Special Projects
- Improving the Licensing Process
- Inspection Programs
- Performance Evaluation
- Quality Assurance
- Operator Licensing
- Emergency Preparedness
- Safety Reviews
- Antitrust Activities
- Indemnity, Financial Protection, and Property Insurance

NRC Regulatory Information Conference. In April 1989, NRR planned, coordinated and conducted the NRC Regulatory Information Conference in April 1989. The conference was designed to provide a forum for non-confrontational communication between NRC management and senior managers in the nuclear industry. Utility managers and supervisors rarely have the opportunity to interact with NRC managers in a context where technical issues and regulatory philosophy can be discussed freely and openly.

Five hundred registered individuals attended the conference. Of the registrants, 60 percent represented nuclear utilities; 35 percent represented private companies associated with the nuclear power industry, including manufacturers, vendors, law firms, and contractors; and 5 percent represented other government agencies, national laboratories, foreign organizations and research organizations. Representatives from Canada, England, Italy, Mexico, Spain, Japan, Taiwan, and Yugoslavia were in attendance.

The conference generally met its objectives, and it is expected that such events in the future will contribute significantly to the more efficient and effective implementation of the NRC's safety directives and initiatives by the nuclear industry.

STATUS OF LICENSING

License Applications and Issuances

During fiscal year 1989, the NRC issued four low-power operating licenses (for South Texas Unit 2 (Tex.), Vogtle Unit 2 (Ga.), Limerick Unit 2 (Pa.), and Seabrook Unit 1 (N.H.)) and four full-power licenses—three for the same South Texas Unit 2, Vogtle Unit 2, and Limerick Unit 2 facilities and one other (Shoreham (N.Y.)) which had received its low-power license in fiscal year 1985. One fuel loading license (Limerick Unit 2) was issued during the report period. The addition of the five units authorized to operate at low or full power brings the total of licensed power reactors in the United States to 112, as of September 30, 1989. (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

LICENSING THE NUCLEAR POWER PLANT

The first step in the nuclear power plant licensing process is the filing with the NRC of an application by a utility for a construction permit. The application usually follows considerable consultation between the utility and the NRC staff and comprises many volumes of data, covering both safety and environmental aspects of the intended operation, in accord with NRC requirements and guidance. The next phase encompasses various safety, environmental, safeguards (from theft or sabotage), and antitrust reviews undertaken by the NRC staff. Thereafter, as required by law, the independent Advisory Committee on Reactor Safeguards, or ACRS, carries out an assessment of the proposed project and of the results of the earlier reviews and makes its recommendations. The fourth phase is a mandatory public hearing on the matter conducted by a three-member Atomic Safety and Licensing Board, or ASLB, which makes an initial decision as to whether a construction permit should be granted. This decision is subject to appeal by any person or group with standing in the proceeding to an Atomic Safety and Licensing Appeal Board, or ASLAB, and could be carried 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 ASLAB.

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.

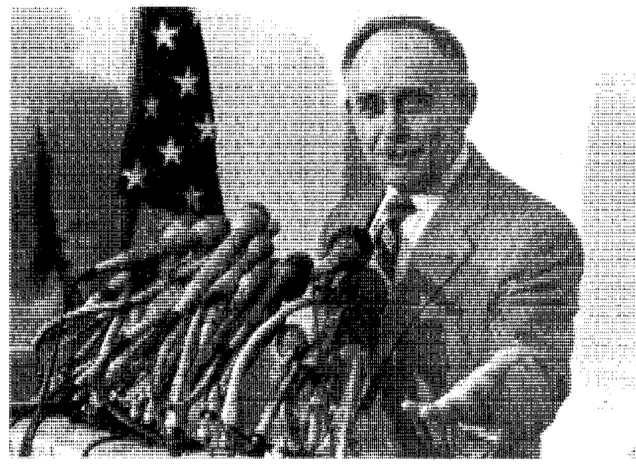
The NRC staff examines the applicant's PSAR to determine whether the plant design is safe and consistent with NRC rules and regulations; whether valid methods of calculation were employed and accurately carried out; whether the applicant has conducted its analysis and evaluation in sufficient depth and 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.



The controversial Seabrook (N.H.) nuclear power plant was licensed for low-power operation during fiscal year 1989. After the close of the report period, the Commission announced the issuance of a full-power license to the utility, the Public Service Company of New Hampshire, evoking an organized demonstration (left) outside NRC Headquarters in Rockville, Md., and a press conference within, at which Chairman Kenneth M. Carr (right) explained the regulatory rationale for the licensing action and answered questions on the decision.

operating licenses or construction permits, and no construction permits or manufacturing licenses were issued during this fiscal year. At the close of fiscal year 1989, there were 10 nuclear power plants still technically under construction in the United States, although some of them are delayed indefinitely.

Table 1 is a numerical summary of NRR activity in power reactor licensing during fiscal year 1989, and Table 2 identifies the licensee and facilities licensed, and gives additional information.

Licensing Actions for Operating Power Reactors

As noted, there were 112 power reactors licensed to operate at the end of fiscal year 1989. After operations begin, both routine activities and unexpected events at these facilities can result in a need for "licensing actions" on the part of the NRC. Routine post-licensing activities affecting the reactor operations include license amendment requests and any related public hearings, requests for exemption from regulations, new regulations requiring backfit modifications to operating reactors, orders for modification of a license, new generic activities, petitions for action under 10 CFR 2.206 by members of the public, or review of information supplied by a licensee for the resolution of technical issues. In recent years, routine activities have also included plant-specific actions needed to deal with allegations of violations or other post-licensing concerns. These activities, and the growth in the number of operating reactors, have resulted in a relatively large number of new actions. During fiscal year 1989, NRR completed about 3,000 licensing actions. About 76 percent of these actions 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 3,600 to 2,530 licensing actions under review.

Special Cases

Shoreham. During fiscal year 1989, the NRC issued a full-power operating license to the Long Island Lighting Company (LILCO) for the Shoreham Unit 1 (N.Y.) nuclear power plant. However, the licensee has entered into an agreement with the State of New York to transfer ownership of the Shoreham Station to an agency of the State for decommissioning. In return, the State is to approve a series of rate increases for the licensee. The utility also expects to be able to take a Federal tax write-off in connection with the proposed decommissioning of Shoreham.

The licensee has removed all fuel from the reactor vessel and, at the close of the report period, was holding the fuel in the spent-fuel pool. The company has also reduced staffing levels at Shoreham to a level consistent with a protracted defueled condition at the plant.

The NRC has received a written commitment from the licensee that the plant systems will be preserved from degradation until the NRC approves the transfer of its license to an entity of New York State for decommissioning. Also, the licensee has committed itself to maintaining the reduced facility staffing level constant and near its current level by either transferring experienced and trained personnel back to the Shoreham site from other parts of its organization or hiring

Table 1. Power Reactor Licensing By Category—FY 1989

Fuel-load and Pre-critical Test Operating License issued	1
Low-Power Operating Licenses issued	4
Full-Power Operating Licenses issued	4
Operating License applications under review	10

suitably trained consultants. The NRC will continue to closely monitor the licensee's equipment preservation program and its staffing status until issues involving Shoreham are resolved.

Although the NRC had not, at fiscal year's end, received an application for the transfer of the Shoreham operating license to a New York State agency for decommissioning of the facility, the NRC has received correspondence from several organizations, including the DOE, expressing oppositionto the proposed decommissioning of the Shoreham station. Thus it appears that implementation of the licensee's settlement agreement with the State of New York may involve substantial litigation.

Fort St. Vrain Shutdown. Fort St. Vrain is a high-temperature gas-cooled reactor facility operated by Public Service Company of Colorado. The plant was permanently shut down on August 18, 1989, following a failure of the control rod drives and degradation of the steam generator ring headers. The decision to prepare for decommissioning came somewhat earlier than expected, inasmuch as the licensee had planned to operate the plant until 1990.

On June 30, 1989, the licensee submitted a comprehensive preliminary decommissioning plan. The plan included the removal of all of the spent fuel from the reactor vessel, with one-sixth of it to be stored in the fuel storage wells and five-sixths of it to be sent to a Department of Energy (DOE) facility in Idaho for reprocessing, under an existing contract. The licensee planned either to construct an independent spent fuel storage installation (ISFSI) or to arrange for DOE to reprocess the remaining one-sixth of the core. The maximum storage capacity of the fuel storage wells is one-third of the core.

After submittal of the preliminary decommissioning plan, the Governor of Idaho informed DOE that the State of Idaho would not accept any additional spent fuel from the Colorado reactor plant. The licensee began defueling on November 27, 1989, and completed the removal of one-third of the core (the maximum capacity of the fuel storage wells) in early 1990. In light of the difficulties with the original plan, the licensee has proposed that the ISFSI be constructed to store all of the plant's spent fuel. The licensee was also evaluating plans for converting the Fort St. Vrain facility into a conventional, gas-fueled power station. The high-pressure, superheated secondary side of the gas-cooled nuclear plant lends itself to such conversion more readily than do the steam systems associated with more typical U.S. nuclear facilities.

Rancho Seco Voted Closed. The Rancho Seco (Cal.) nuclear power plant is owned and operated by the Sacramento Municipal Utility District (SMUD), a public utility. For various reasons, among them the increase in electric rates over a period of years, the ratepayers of Sacramento put a referendum on the ballot to let the voters decide whether or not SMUD should continue to operate Rancho Seco as a nuclear facility. On June 6, 1989, the referendum was voted on, and the outcome was 53.4 percent (111,867) against continued operation and 46.6 percent (97,460) in favor of continued operation. On June 7, 1989, the plant was shut down. Attempts to sell the facility fell through, and, as of the end of fiscal year 1989, SMUD was proceeding with plans to decommission the plant.

PLANT LICENSE RENEWAL

According to a projections of the Electric Power Research Institute, the nation's installed electric generating capacity will have to increase from 600 gigawatts in 1984 to about 1,100 gigawatts in the year 2010 to meet increasing demand. Complicating the

Table 2. Licenses Issued For Operation Of Nuclear Power Plants—FY 1989

Applicant	*Facility*	*Low-Power*	*Full-Power*	*Location*
Houston Lighting Power	South Texas 2	12/16/88	3/28/89	12 miles SSW and of Bay City, Tex.
Georgia Power	Vogtle 2	2/9/89	3/31/89	26 miles from Augusta, Ga.
Philadelphia Electric	Limerick 2	7/10/89	8/25/89	21 miles from Philadelphia, Pa.
Long Island Lighting	Shoreham	7/3/85	4/21/89	7.1 miles from Brookhaven, N.Y.
Public Service Company of NH	Seabrook 1	5/26/89	—	13 miles from Portsmouth, N.H.

country's ability to meet this expected demand is the fact that, within the next 20 years, many commercial nuclear power plants will have reached the 40-year term of their operating licenses, a limit imposed by the Congress in the Atomic Energy Act of 1954, as amended. The first currently active operating license will expire in the year 2000, and the licenses of approximately 43 percent of all operating plants in the United States will expire by the year 2010.

In order to maintain an adequate energy supply in the early 21st century, some utilities are looking to extend the useful life of certain nuclear power plants beyond 40 years. The NRC has instituted a program to develop the regulatory requirements for license renewal sufficiently early that nuclear utilities can make timely and informed decisions concerning future generating capacity. The NRC is working closely with the industry, as represented by the Nuclear Management and Resources Committee (NUMARC), on this program. The program comprises a number of major activities—including a rulemaking proceeding, regulatory guidance development, industry technical report reviews, and lead plant reviews.

Rulemaking

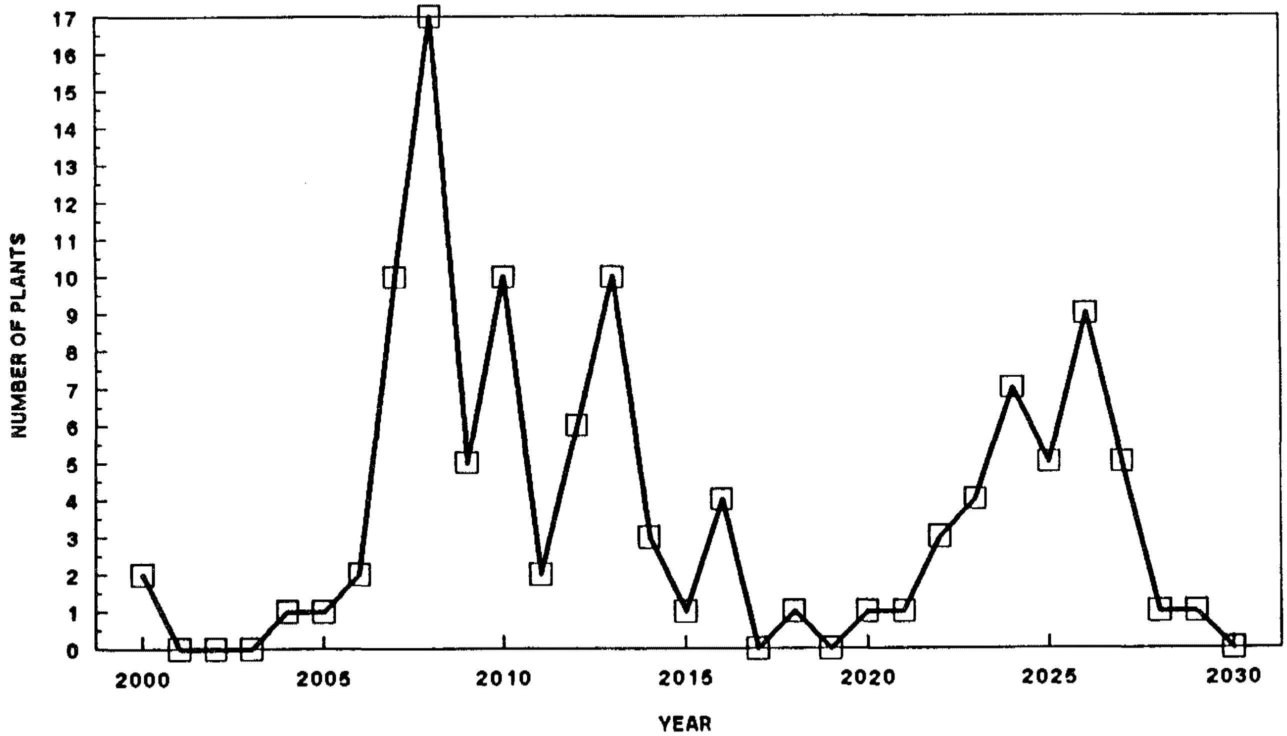
The NRC has prepared a framework discussion of the regulatory philosophy for license renewal and a conceptual basis for the regulatory language embodying this philosophy. This information was published in the *Federal Register* and submitted for discussion and comment in a public workshop on license renewal held in Reston, Va., on November 13-14, 1989. The schedule calls for publication of a proposed rule in May of 1990. The final rule is scheduled to be published in April 1992.

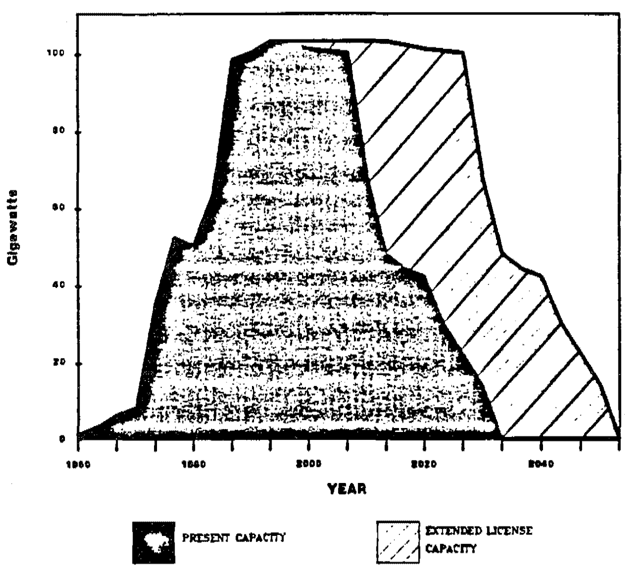
Regulatory Guidance Development

The regulatory guidance development activity will proceed in parallel with the rulemaking proceeding and industry technical report and lead plant reviews. Currently the NRC is planning to publish key draft regulatory guidance for comment beginning in late 1990. The staff anticipates completion of final regulatory guidance by early 1994.

Industry Technical Report Reviews

The industry, in parallel with the NRC's efforts and through NUMARC, is preparing a series of technical reports that focus on potential age-related degradation mechanisms associated with a given system or component and that identify preventive, corrective, and mitigative actions necessary to a program for renewing plant licenses. The NRC is presently reviewing several of these reports for the purpose of approving relevant portions thereof for use in the license renewal process.





The nation's need for increased electrical generating capacity in the coming decades is projected to reach about 1,100 gigawatts by the year 2010, which means hundreds of new 1,000 megawatt plants or their equivalent will be needed between now and then. Many of the current nuclear plants, which today provide about 20 percent of the nation's electricity, will be reaching the expiration of the 40-year term of their initial licenses during this period. For these reasons, license renewal for nuclear plants is an important matter, and the NRC is preparing requirements and procedures in advance to facilitate industry decisions on seeking license renewal.

The graph above shows year-by-year the number of nuclear plants whose original 40-year licenses will be running out after the turn of the century. At left is a graph showing the nuclear contribution, in gigawatts, to the national electrical capacity as it has been since 1960 and as it will be—without license renewal and (hatching) with extended operation.

Lead Plant Reviews

Northern States Power Company and Yankee Atomic Electric Company, in cooperation with NUMARC and with the support of the DOE, have agreed to allow their Monticello (Minn.) and Yankee Rowe (Mass.) plants to serve as "lead plants" in the NRC's license renewal program. The utilities' current schedules for submitting their license renewal applications are June 1991 for Yankee Rowe and December 1991 for Monticello. The staff expects to complete its review of each of these applications in approximately two years from the date each application is received.

SPECIAL PROJECTS

The NRC Office of Special Projects (OSP) was established in February 1987 to provide a strengthened and integrated staff organization to deal with particularly complex regulatory problems that had emerged at the Tennessee Valley Authority (TVA) nuclear power plants and the Texas Utilities Electric Company (TU Electric) units at Comanche Peak (Tex.). On January 1, 1989, the OSP was transferred in its entirety to NRR as the Associate Directorate for Special Projects. With the licensing and restartactivities related to these projects stabilized, the staff is planning full integration of these projects back into the mainstream of regulatory oversight early in 1990. The activities associated with the TVA plants and Comanche Peak will be integrated into the existing NRC organizational structure.

Some background on, and the current status of, the affected plants is discussed below.

TVA Projects

In 1985, the NRC staff issued a letter to the Chairman of the TVA Board of Directors indicating that there were significant and 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 on its own accord and had made commitments to the NRC that the plants would not be restarted without NRC concurrence. The multitude and complexity of the issues were not limited to operating reactors, since questionable construction practices had also surfaced at the TVA's Watts Bar Unit 1 (Tenn.) project.

Sequoyah. Sequoyah Unit 1 achieved criticality on November 6, 1988, following NRC authorization for restart. On November 18, 1988, Unit 1 went into automatic shutdown from a 72 percent of full-power level, following an electrical ground in the main generator stator. Repairs were completed, and the unit was restarted on December 25, 1988. Following another automatic shutdown on December 26, 1988, Unit 1 achieved criticality on December 27, 1988, and proceeded to full-power operation. Unit 1 has since achieved a record run of more than 224 days of continuous event-free operation.

Sequoyah Unit 2 was shut down for a refueling outage on January 19, 1989, following 210 continuous days of operation. In April 1989, during the restart from the outage, Unit 2 experienced three automatic shutdowns during the transition from auxiliary feedwater to main feedwater control. TVA was required to perform a post-shutdown review and root-cause analysis and to discuss its findings with the NRC staff prior to any restart. This was done and, following a meeting on April 23, 1989 with the NRC, the reactor was restarted on April 26, 1989, and proceeded to full-power operation.

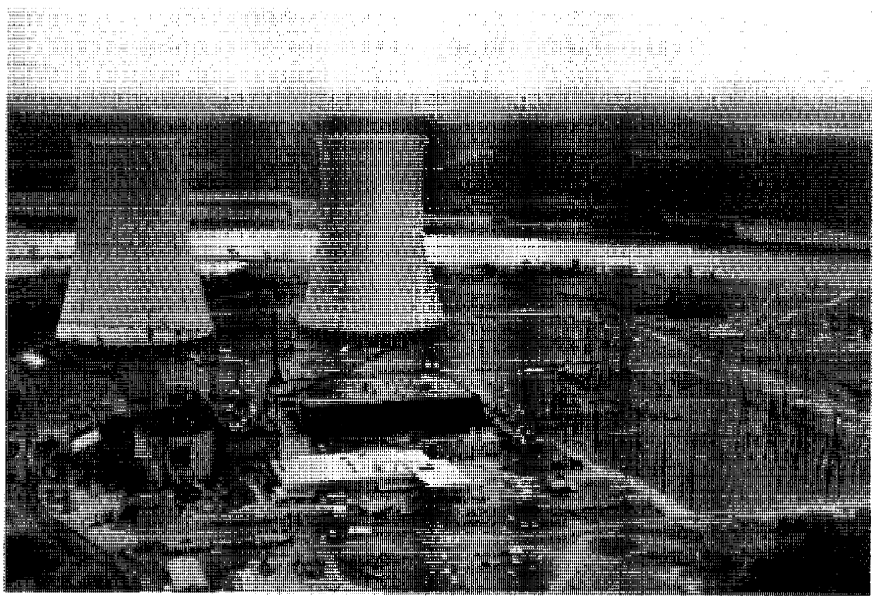
In June 1989, NRC senior management decided to remove the Sequoyah site from the category of plants requiring special attention.

Browns Ferry. All three units at Browns Ferry remained shut down and Units 1 and 3 remained defueled throughout the fiscal year. Fuel was reloaded into Unit 2 in January 1989. Units 1 and 3 have been shut down since March 1985, and Unit 2 since September 1984. The TVA has projected a restart of Unit 2 for the second quarter of 1990 but has not established a schedule for restarting Units 1 and 3.

The staff issued a Safety Evaluation Report (SER) in April 1989 covering portions of the Browns Ferry Nuclear Performance Plan for Unit 2 restart. Two supplements to the SER are planned before restart is authorized. Also, the staff plans to undertake major inspection activities to support restart of Unit 2.

Watts Bar. The TVA had announced that the priorities for the startup of its facilities would be in this order: Sequoyah, Browns Ferry and Watts Bar. Having restarted Sequoyah, the TVA has stepped up activity at Watts Bar and established a December 1990 target date for Unit 1 fuel loading. No completion schedule has yet been adopted for Unit 2. In May 1989, the TVA 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 (WBPT), made up of TVA personnel assisted by nationally recognized nuclear power experts, was formed in November 1987. The WBPT was responsible for the WBNPP and for defining the scope of necessary corrective actions and 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. A significant portion of the review of these program plans was initiated during fiscal year 1989. The NRC staff has prepared the master licensing and inspection plan and is closely monitoring the TVA's implementation of corrective activities.



The Watts Bar (Tenn.) facility is scheduled for fuel loading by the end of 1990, after a prolonged period of corrective actions to resolve deficiencies identified through various allegations, express employee concern, and independent assessments. The two-unit TVA plant is shown at an early stage of construction.

Bellefonte. In July 1988, the TVA informed the NRC that the TVA Board of Directors had deferred the construction of Bellefonte Nuclear Plant, 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 TVA's effort to hold electric rates constant for a specific period of time. The construction permits expire on July 1, 1994, for Unit 1 and July 1, 1996, for Unit 2. The TVA identified and provided the description of various licensing activities that will be performed during the deferral period.

TU Electric's Comanche Peak Project

During the report period, TU Electric continued implementing a comprehensive corrective action program addressing deficiencies discovered by the Comanche Peak Response Team (CPRT). Their effort included reanalysis, revision or updating of existing design calculations; physical reinspection of as-built hardware; and actual physical hardware changes and reconstruction. The NRC staff approved, with several conditions, the TU Electric CPRT and corrective action program plans. By the end of calendar year 1988, the staff issued evaluations of the individual elements of the corrective action program, namely, large- and small-bore piping and pipe supports, conduit supports, cable trays, cable tray hangers, equipment qualification, instrumentation and controls, as well as electrical, mechanical, civil/structural, and heating, ventilation, and air conditioning systems.

At the end of fiscal year 1989, the staff was continuing its efforts with other scheduled activities necessary for Comanche Peak licensing. According to the utility's schedule at that time, Unit 1 would be ready to load fuel during October 1989. TU Electric projected that construction of Unit 2 would be complete and the plant would be licensed about two years after these activities were complete at Unit 1. Construction of that unit was expected to be complete, with readiness to begin operations under a low-power license, some time during the first half of fiscal year 1990. The staff's inspection efforts have substantially shifted from construction activities to the preparation for plant operation. The staff planned for a two-week Operational Readiness Assessment Team independent inspection during October 1989.

In January 1989, a private party filed a motion before the U.S. Court of Appeals for the D.C. Circuit Court to overturn the Commission's memorandum and order CLI-88-12 denying the Citizens for Fair Utility Regulation's (CFUR's) late-filed petition to intervene in the Comanche Peak licensing process. In February 1989, CFUR petitioned the U.S. Court of Appeals for the Fifth Circuit, in New Orleans, to review CLI-88-12. At the end of the fiscal year, the court's decisions on these filings were still pending. A decision by the Director, NRR, on a petition filed pursuant to 10 CFR 2.206 by Cap Rock Electric Cooperative, Inc., was also pending. In its petition, Cap Rock requested that the Commission enforce Comanche Peak antitrust license conditions. Cap Rock asserted that the applicant was refusing it essential service information which would enable Cap Rock to purchase generating capacity and economic energy from other bulk power supply sources.

IMPROVING THE LICENSING PROCESS

Standardization

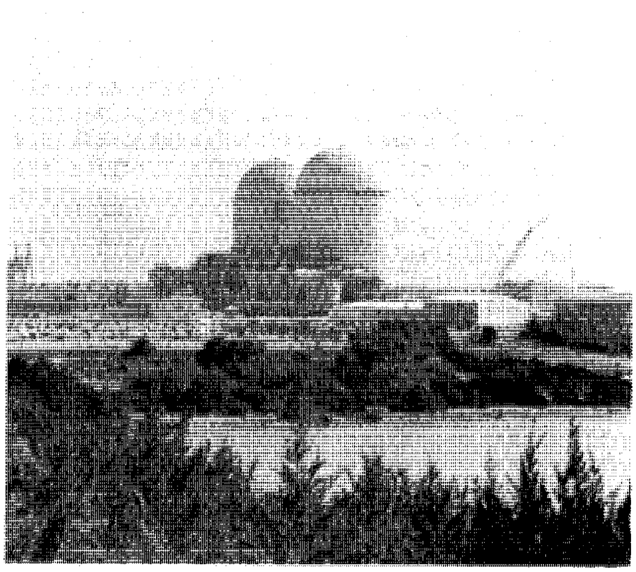
The Commission strongly endorses regulatory policies that encourage the industry to pursue standardization of power reactor designs. Standard designs are expected to benefit public health and safety in a number of ways: concentrating industry resources on common approaches to design problems that have 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 current views of the Commission and the industry. The focus of the rule is design certification, a regulatory instrument that would enable licensing concerns to be resolved before utilities had made costly investments. Subpart B of this rule provides a regulatory framework for certification through rulemaking of standard plant designs. Also, the standardization requirements of 10 CFR Part 50, Appendices M, N, and O have been subsumed under 10 CFR Part 52.

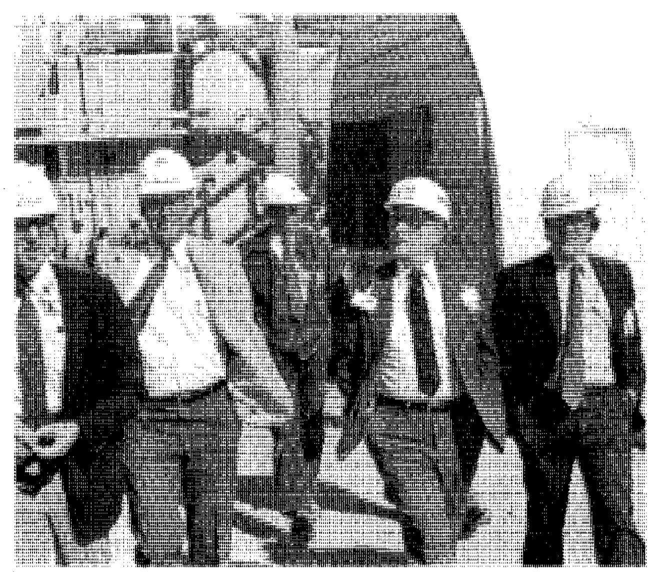
Advanced Reactors

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. To date, EPRI has submitted for NRC review 12 chapters of a 13-chapter utility requirements 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. In addition, the requirements document 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.

EPRI plans to begin submitting 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, in fiscal year 1990.

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 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 Comanche Peak (Tex.) nuclear power plant, at left, having experienced a number of delays and undergone extensive corrective action, is projected by Texas Utilities (TU) Electric to be ready for operation of Unit 1 in 1990, and two years later for Unit 2. At right, shown touring the plant early in the report period, are Chairman (then Commissioner) Kenneth M. Carr, second from right, with his Executive Assistant, Stephen Burns, at the right. On the left are T.G. Tyler and Jim Kelley of TU Electric.

To date, the NRC has received and initiated reviews of all chapters of the ABWR safety analysis report. In August 1989, the staff issued a draft Safety Evaluation Report discussing the staff's review of Chapters 1, 4, 5, 6 and 17.

Westinghouse RESAR SP/90. The NRC continued its review of the Westinghouse Electric Corporation's application for preliminary design approval (PDA) of its reference safety analysis report (RESAR) SP/90. The SP/90 design is being developed independently of the EPRI requirements document. In March 1989, the staff issued a draft Safety Evaluation Report 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 staff is reviewing the licensing review basis, which is an agreement between the NRC and the applicant providing guidance on regulatory issues in areas not addressed by the Standard Review Plan (NUREG-0800). Review of the FDA/DC application will be completed utilizing this guidance.

CANDU. Atomic Energy of Canada Limited (AECL) Technologies informed the NRC of its intent to seek design certification of the CANDU 3 nuclear power plant, under the provisions of 10 CFR Part 52, in a letter dated May 25, 1989. The staff responded in a letter dated July 6, 1989, in which it requested that AECL Technologies develop a licensing review document, submittal schedules, brief descriptions of the design, and proposed acceptance criteria. The NRC staff held preliminary meetings with AECL Technologies and the Atomic Energy Control Board, which is the Canadian regulatory body, to discuss features of the CANDU 3 design.

The CANDU 3 design is a single-loop pressurized reactor rated at 450 electrical megawatts (MWe), with two steam generators and two heat transport pumps connected in series. The design employs natural uranium fuel, heavy-water moderator, computer-controlled operation, and at-power refueling.

Severe Accidents

The Severe Accident Integration Plan provides for coordinated efforts to ensure fulfillment of the objectives of the Commission's "Severe Accident Policy Statement." Achievement of objectives of this plan constitutes a basis for assuring that the residual risks to the public from severe accidents at nuclear power plants are minimized in an effective manner. There are six main elements of the integration plan which lead to severe-accident closure for operating plants. The elements are: (1) the individual plant examination (IPE) program, (2) a containment performance improvements (CPI) program for each of the six containment types, (3) a program to improve plant operations, (4) a severe-accident research program (SARP), (5) an external events program, and (6) an accident-management program. Severe-accident closure on the objectives of the "Severe Accident Policy Statement" is achieved when the IPEs have been completed (including external events) and any appropriate changes have been implemented, a framework for an accident-management program has been developed and implemented, and generic requirements resulting from the CPI program have been implemented. Progress during fiscal year 1989 on each of the program elements is discussed below. (See Chapter 8 for discussion of the Severe Accident Research Program.)

Individual Plant Examination. Individual plant examinations are systematic analyses of each operating plant performed by the licensee to identify and remedy any potentially significant plant-specific risks not previously recognized. They will be accomplished in part through the application of probabilistic safety assessment techniques. The program objectives were identified in the Commission's November 23, 1988, Generic Letter 88-20. The plan makes available several analytical options for conducting the studies, and the staff expects that a licensee will promptly correct any significant vulnerability as it is uncovered. More-detailed NRC guidance on the analyses to be performed by licensees was issued in the August 1989 report, "Individual Plant Examination: Submittal Guidance" (NUREG-1335). Ultimately, the scope of the program will be expanded to include both internal accident initiators and such external initiators as seismic events.

Containment Performance Improvements (CPI). The need for a CPI program is predicated on the conclusion that generic severe-accident challenges exist for each light-water-reactor (LWR) containment type and should be assessed to determine whether additional regulatory guidance is warranted, or to confirm the adequacy of existing Commission policy. All LWR containment types are to be assessed starting with the boiling-water-reactor (BWR) Mark I type.

In fiscal year 1989, NRR completed work to identify enhancements for the BWR Mark I plants. A number of modifications were identified that could increase the plant's capability of preventing and mitigating the consequences of severe accidents. The Commission decided to pursue Mark I enhancements on a plant-specific basis to take into account possible unique design differences that may bear on the necessity and nature of specific safety improvements.

The several modifications proposed by the NRC included an improved capability to vent the containment to reduce pressure, avoid containment failure, and prevent core damage. This improvement is considered of special importance in enhancing a plant's capability to maintain long term decay heat removal. Because of this, the staff issued a Generic Letter encouraging utilities to consider its use. The NRC will approve installation of an improved vent under the provisions of 10 CFR 50.59 for licensees who, on their own initiative, elect to incorporate this plant improvement. For each of the remaining Mark I plants, the NRC has initiated plant-specific backfit analyses to evaluate the efficacy of requiring the installation of improved vents. The other safety improvements are to be evaluated by licensees as part of the IPE program.

Improved Plant Operations. This activity includes a number of ongoing efforts, as discussed in the 1988 Annual Report, page 16. Major progress in this area was achieved in developments described under the sections on accident management and individual plant examination.

In addition, work has continued in other areas to improve plant operations performance, particularly to improve Technical Specifications, to develop and use plant performance indicators to identify trends in operation, to continue improving plant operating procedures (including severe-accident management procedures), and to encourage licensees to put into place improved maintenance programs and operational reliability methods.

External Events. This activity seeks a technically acceptable analytical approach to accounting for severe-accident initiators such as seismic events, fire, and flood occurring outside the plant structures and systems, but with the potential to affect their operation. Activities in the report period have involved interactions with industry, through NUMARC and EPRI, to develop methods for performing the evaluations.

The staff's preliminary findings indicate that seismic vulnerabilities to severe accidents may be explored through either a probabilistic risk assessment (PRA) or a seismic "margins" approach, such as that developed by either the NRC or the EPRI. Further, a PRA considering fire initiators would be an acceptable method of examining for vulnerabilities produced by fire. NUMARC is developing other simplified methods for performing an IPE evaluation of fire initiators.

Accident Management. Accident management encompasses those actions taken during the course of an accident by the plant operating and technical staff to: (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize off-site releases. Accident management, in effect, extends the defense-in-depth principle to plant operating staff by extending the operating procedures well beyond the plant design basis into severe fuel damage regimes, with the goal of taking advantage of existing plant equipment and operator skills and creatively finding ways to terminate accidents beyond the design basis or to limit off-site releases.

"Lessons-learned" information from past PRAs, in the form of generic accident-management strategies, was developed and was to be provided to utilities in late 1989 in a supplement to Generic Letter 88-20 dealing with IPE. The Generic Letter supplement will provide licensees a set of accident-management strategies, and will request that licensees evaluate the strategies for applicability and effectiveness at their plants as part of the IPE program.

Considerable progress has also been made toward the longer term goal of developing guidance on the scope and content of a utility accident-management plan or framework for issuance in a Generic Letter on accident management in late 1990. Industry, through NUMARC, will play a key role in the process by developing guidelines for use by utilities in evaluating their accident-management capabilities for important sequences identified through the IPE. Current plans include application of the guidelines on a trial basis at several reference plants (to be determined) beginning early in 1990, and completion of a final version of the guidelines by mid-1990. The Generic Letter on accident management will address the role of these guidelines in the development of utility accident-management capabilities.

In parallel with NUMARC's activities, the NRC is developing information and guidance on accident management as part of the Accident Management Research Program. This activity will provide a technical basis and perspective for evaluating the NUMARC accident-management guidelines, and is expected to culminate in supplementary insights/guidance for inclusion in either the NUMARC guidelines or the NRC Generic Letter on accident management.

Technical Specifications Improvements

On February 10, 1987, the Commission issued an interim policy statement on improving Technical Specifications for nuclear power plants. The substantial increase in both the number of items and in the detail of the requirements contained in Technical Specifications has caused concern that those Technical

Specifications with little or no safety significance were diverting attention from matters of more immediate importance to the safe operation of nuclear power plants. In addition, the lack of clear bases for each Technical Specification had caused conflicting interpretations of Technical Specification requirements. The interim policy statement established a set of objective criteria for determining which regulatory requirements and operating restrictions should be included in the Technical Specifications that are issued as part of every power reactor operating license. On May 9, 1988, the NRC specified for each of the vendor owners groups which current Technical Specifications should be retained in their respective Standard Technical Specifications (STS) proposals. The owners groups submitted their new STS for NRC review between March and June 1989, and the staff was reviewing the proposals at the close of the report period. In addition, five operating nuclear power plants were identified to be the "lead plants" for initial adoption of the new STS. They are San Onofre Units 2 and 3 (Cal.), North Anna Units 1 and 2 (Va.), Crystal River Unit 3 (Fla.), Hatch Unit 2 (Ga.), and Grand Gulf (Miss.). These plants will be the first to convert their current Technical Specifications to the new STS for their respective vendor.

The NRC is continuing its work on specific line-item improvements to the existing Technical Specifications. During fiscal year 1989, two Generic Letters were issued on line-item improvements to Technical Specifications. The first concerned the implementation of programmatic controls for the Radiological Effluent Technical Specifications (RETS) in the "Administrative Controls" section of the Technical Specifications and relocation of the procedural details of RETS to a program outside of the Technical Specifications. The second Generic Letter concerned the elimination of a restriction that limited the combined time period between any three consecutive surveillance test intervals to less than 3.25 times the specified surveillance interval.

Supporting activities for the improvement of Technical Specifications included the issuance of an industry-proposed guidance document on the proper application of the regulations contained in 10 CFR 50.59. The implementation of the policy statement will lead to the transfer of existing Technical Specifications to license-controlled documents, such as the final safety analysis report.

This will permit subsequent changes to many of these requirements without the need for prior NRC approval, under the 10 CFR 50.59 safety evaluation process. The staff has recognized the need for improving the 10 CFR 50.59 review process and has attempted, by working with the industry, to make this regulation better understood and more effectively applied.

During fiscal year 1989, the NRC reviewed the current surveillance testing requirements contained in the Technical Specifications and identified ways to reduce surveillance testing at nuclear power plants while the reactor is in operation, or "at power." The study was prompted by a Commission request that the staff investigate the pro's and con's of continuing to require the surveillance and testing of equipment at a nuclear power plant while the reactor is at power. The goal was to ensure that the NRC does not require unnecessary tests or inspections that result in equipment disassembly or unnecessary wear which can present challenges to safety systems during such testing. The NRC also continued to develop risk-based Technical Specifications which can play an important role in overall risk management for nuclear power plants.

INSPECTION PROGRAMS

NRR is responsible for administering the reactor inspection program. The program encompasses all applicant and licensee activities conducted in the constructing and operating of nuclear facilities. Most of the inspection effort is focused on the 112 operating plants, with additional coverage of the 10 facilities with construction permits. The responsibility for developing, maintaining, and assessing the effectiveness of the reactor inspection program is shared among NRR staff.

The operating reactor program which was developed in fiscal year 1988, was implemented in fiscal year 1989. The objectives of the new inspection program are (1) to ensure that a minimum level of inspection is conducted at every plant, (2) to integrate Headquarters and Regional Office programs, (3) to provide more flexibility for the 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 activities 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 to support the Systematic Assessment of Licensee Performance (SALP) Program evaluations. In the "regional initiatives" phase of the new inspection program, Regional Offices redirected certain of their inspection resources from the plants exhibiting a high level of performance to those evincing a marginal level.

A basic element in NRC reactor regulation 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 carried out through the five NRC Regional Offices. As described later in this report, a limited number of inspection programs are conducted directly by the NRC Headquarters Office. 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 Regional Operations.

Besides the routine or planned program of inspections for reactor, fuel cycle facility, and materials licensees, the NRC conducts an aggressive program to deal with unsafe or potentially unsafe events or conditions occurring at individual plant sites or other facilities involving licensed operations (these called "reactive" inspections). In conducting reactive inspections, the NRC seeks to determine the root cause of the event or condition; evaluates the licensee management's response to it, including action to prevent recurrence; and decides whether the problem is one that could occur at other facilities. The staff then takes appropriate action on these judgments.

Reactor Inspection Program

The operating reactor inspection program is conducted by headquarters and regional inspectors. Headquarters inspectors conduct, or support the Regional Office's conduct of, inspections under the Special Team Inspection Program, discussed below. The Regional Offices conduct the majority of the required program inspections. Regional inspections are conducted by both region-based and resident inspectors. In general, region-based inspectors are specialists, and resident inspectors are generalists. Resident inspectors provide the major on-site NRC presence for direct observation and verification oflicensee activities. The work comprises in-depth inspections of control room activities; 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 for the NRC inspection efforts related to licensee event reports (LERs), events, and incidents. 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 the headquarters and regional inspection effort to be focused on those plant operations which contribute most to ensuring reactor safety and on the identification of safety problems. Minor improvements in the program were made in fiscal year 1989, based on experience and on regional assessments.

The new inspection program is made up of the following elements:

- *Fundamental Inspection Program.* This program is implemented at every plant, and consists of two parts:

(1) The Core Inspection Program provides a balanced look at a cross-section of plant activities considered important to maintaining safety.

(2) The Mandatory Team Inspection Program is a team inspection effort addressing one or more subject areas selected by an identification of an emerging safety concern, or of an area calling for increased attention because of a history of long-standing and/or recurring problems. The area of emphasis for the mandatory team inspections for fiscal year 1989 was maintenance; this effort will be continued in fiscal year 1990.

- *Regional Initiatives and Reactive Inspections.* This program provides additional inspection effort beyond that of the fundamental inspection program and is based on plant performance in specific functional areas. The Regional Administrator identifies the specific inspection activities and the plants to be inspected. Reactive inspections are generally unplanned inspections conducted at the direction of the the Regional Administrator, in response to various events or issues.

- *Special Team Inspection Programs.* Special team inspections provide an independent, in-depth, balanced assessment of licensee performance. Special team inspections are conducted by both Headquarters and Regional Offices, and are particularly useful in conducting in-depth examinations to verify the adequacy of specific engineering and operational disciplines.

- *Safety Issues Program.* This program represents the special inspection effort incorporated and implied in a "temporary instruction" (TI). A TI may be issued to ensure inspection follow-up on 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 1989, seven TIs were issued, affecting such issues as decay heat removal, storage of diesel generator fuel oil, and security.

Development and use of an innovative inspection approach to appraise the operability of safety systems at operating plants continued in fiscal year 1989. In fiscal year 1986, a new special team inspection methodology, called a safety systems functional inspection (SSFI), was introduced into the reactor inspection program for implementation by the Regions. It continues to prove a useful aspect of regional inspections, because it identifies significant safety issues that require the licensee to take corrective actions. Another special team inspection approach, the safety systems outage modification inspection (SSOMI), helps identify a need for licensees to maintain more effective controls over activities associated with the evaluation, design, procurement, installation, and testing of plant modifications. Because of their demonstrated success, these special team inspections have been made a part of the regional initiatives portion of the new inspection program. Special 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. Headquarters and regional staffs will continue to employ them in fiscal year 1990.

A Master Inspection Planning System (MIPS) is being 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, for the upcoming SALP cycle (see above and following). It also allows the Regions to redistribute and direct resources to those facilities requiring special attention. The SALP cycle for inspection varies from 12-to-18 months and is established on the basis of individual licensee performance; thus licensees who receive higher SALP ratings are inspected fewer times over a given year. Because the MIPS is electronically updated to reflect completed inspection effort, a quick and accurate record of inspection program planning and implementation will exist, once the system has been completely developed. NRR and the Regions assess the effectiveness of the inspection program through use of the MIPS and through an ongoing inspection program assessment process, which includes assessment team visits to the Regions.

NRR will continue to monitor the implementation of the new inspection program in all its phases during fiscal year 1990, to gauge its effectiveness and make adjustments as indicated.

Special Inspections

During fiscal year 1989, the headquarters special inspection staff performed 19 special team inspections of licensee performance, involving various aspects of plant design, construction, operation, and modification. Regional Office staff also performed a number of these inspections, as part of the regional initiatives and reactive inspections program.

The special team inspection program, described in detail in the *1988 NRC Annual Report*, p. 21, targets specific plants at which the NRC is concerned about licensee performance or about technical areas important to safe operation. The inspections are conducted by a team of 8-to-10 inspectors of various technical specialties, including engineers from NRC contractor organizations. The team spends from two-to-four weeks at the plant and performs in-depth, technically oriented inspections to identify any specific problems with the licensee's performance of safety responsibilities.

The headquarters special team inspection staff performed a wide variety of special inspections during the year. They dealt with such matters as safety system functionality (Nine Mile Point (N.Y.)), emergency operating procedures (Brunswick (N.C.)), safety systems outage modification inspections (SSOMIs) and/or follow-up to a prior SSOMI (Indian Point 3 (N.Y.)), Oyster Creek (N.J.), North Anna (Va.), and Fermi Unit 2 (Mich.)), independent assessments of facility design and construction (Limerick 2 (Pa.)), special cases identified by NRC management (Nine Mile Point (N.Y.)), Calvert Cliffs (Md.)), operational safety (Fort Calhoun (Neb.)), motor-operated valves (Robinson (S.C.)), design validation (Turkey Point (Fla.)), and follow-up to two NRC Bulletins (Arkansas Unit 1).

The inspections at Nine Mile Point and Calvert Cliffs sought to assess the effectiveness of licensee management oversight of operational safety activities at each plant. During the inspection, the team endeavored to determine the root causes and contributing factors in fundamental problems identified previously, and to ascertain whether licensee management had developed and implemented timely actions to correct those problems. The NRC inspection teams accomplished these objectives by interviewing management and plant personnel extensively and by directly observing activities at the plant. The two inspections differed from previous major team inspections in that greater emphasis was placed on management performance to ensure operational safety.

A new initiative was undertaken in the area of plant design documentation, during the report period. Included in the term "design documentation" are safety performance requirements for plant systems and components, design analysis and calculations, facility engineering drawings, and licensee commitments to the NRC. The licensee needs to consult this design information whenever it is considering a proposed modification to the plant, in order to determine whether the modification could adversely affect one or more of the plant safety functions.

Prior NRC special team inspections had identified modifications that some licensees had made without a sufficient engineering basis. Missing or irretrievable design documentation appeared to be a root cause of that problem. As a result of interchanges with the NRC on the subject, and on their own initiative, a number of licensees have begun the task of reconstituting the design basis for their plants and of assembling the associated documents.

Despite this activity, the NRC concluded that licensees needed to give greater attention to the subject, to make sure that the latter's decisions about facility modification and operation do not inadvertently compromise plant safety because of a lack of knowledge regarding the plant's safety design. Accordingly, the NRC established a survey program to identify the current status of the industry's ability to retrieve necessary design documents, to understand what approaches licensees were using to reconstitute the design basis for the plant, and to identify and recreate missing documents. The NRC surveyed a representative cross-section of operating plants of various vintages and one nuclear steam supply system vendor.

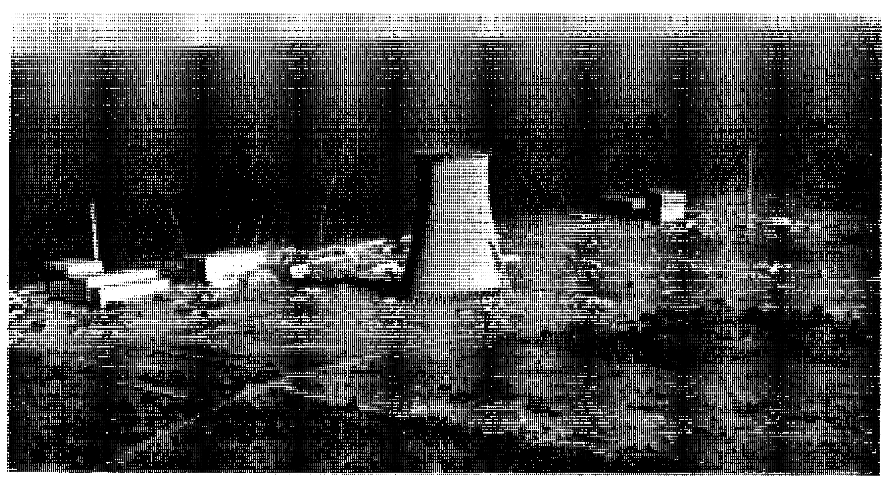
On the basis of information collected during the surveys, the NRC will issue a document that describes good practices regarding types of design documents that should be controlled and maintained, points out the strengths and weaknesses of utility-initiated design-basis reconstitution programs, and evaluates the adequacy of current NRC regulations and industry standards in this area.

Inspection of Emergency Operating Procedures

The NRC continues to implement a long term program of improving emergency operating procedures (EOPs). The early objectives of this program were to improve the technical accuracy of EOPs and to see to the incorporation of human factors principles in the procedures. Owners groups representing the four nuclear power plant vendors have satisfactorily reanalyzed relevant transients and accidents and have developed generic technical guidelines for improving their EOPs. The industry has been revising 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 methods, the plant reactor simulator was employed, when available, in conducting this assessment.

The great majority of EOP problems identified during the inspections conducted from March to October



Special team inspections bring 8-to-10 inspectors with varying technical specialties to perform in-depth inspections at a given facility over a period of weeks. A particular focus of the team inspection at Nine Mile Point (N.Y.) was safety system functionality. The facility, located on the shores of Lake Ontario, consists of two boiling water reactor units.

1988 resulted from incomplete implementation of EOP programs. The most significant programmatic problems identified were lack of a multi-disciplinary team approach in the development of EOPs, lack of independent review of the EOPs, and lack of a systematic process for ensuring that the quality of EOPs does not deteriorate over time. During the report period, meetings were held with NUMARC and the owners groups to disseminate these generic findings. They were also published, as NUREG-1358, "Lessons Learned From the Special Inspection Program for Emergency Operating Procedures."

The EOP inspection program has, as of fiscal year 1989, reached almost 60 percent of the plants. Results from the inspections indicated some improvement in the implementation of EOP programs, but the staff continued to find weaknesses in ensuring that the quality of EOPs does not diminish over time. The staff will complete inspection of all remaining plants and will return to any plants with identified problems in implementation of appropriate corrective actions.

Vendor Inspection Program

The Vendor Inspection Program is centralized in NRC Headquarters and is principally a reactive program structured to respond to vendor and licensee reports of deviations and defects in vendor-supplied products, equipment, materials, and services provided to nuclear power plants. The program involves setting priorities and other tasks to find and address issues and problems of greatest safety significance and broadest generic applicability.

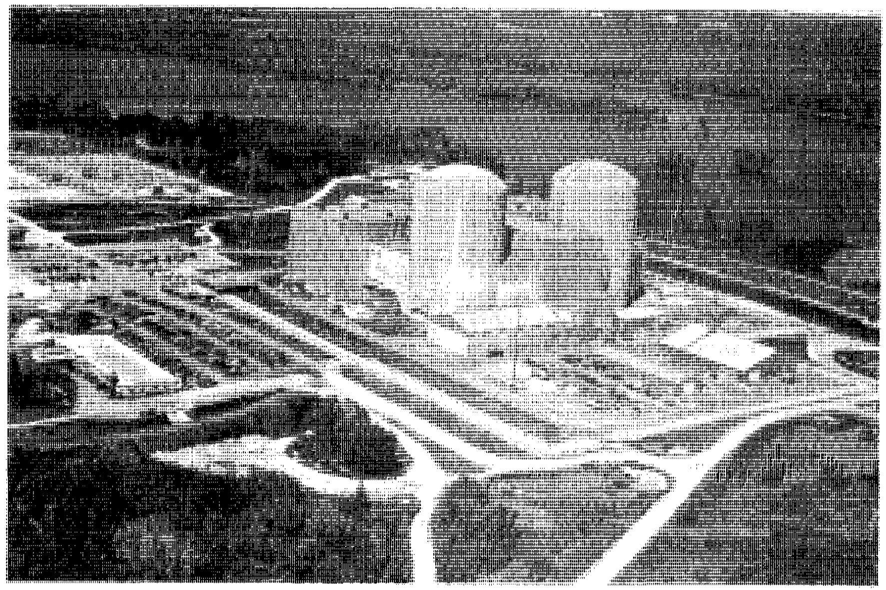
In fiscal year 1989, the NRC conducted 90 vendor inspections. These focused on vendor activities associated with nuclear plant operation, maintenance, procurements, and modifications. Inspections emphasized the quality of vendor products, the licensee/vendor interfaces, equipment problems found during operation, and corrective actions in response to identified problems. Inspections of licensees, vendors, and contractors were based on information from a variety of sources, including licensee construction deficiency and operating reactor event reports (10 CFR 50.55e, 50.72, and 50.73), vendor reports of product defects (10 CFR 21), reports of events from the NRC Regional Offices, allegations from members of the public pertaining to vendor activities, and vendor issues identified by the NRC within the framework of its inspection programs.

In response to the NRC's concerns about the possible introduction and application of misrepresented vendor products into a nuclear power plant, the staff initiated short term and long term efforts to deal with the problem. In March 1989, the NRC issued Generic Letter 89-02, "Actions To Improve the Detection of Counterfeit and Fraudulently Marketed Products," describing characteristics of an effective nuclear power plant procurement and dedication program. The Generic Letter also provided conditional NRC endorsement of an industry standard for procurement and dedication of commercial-grade equipment for safety-related applications. To address longer term corrective actions, the NRC published "Advance Notice for Proposed Rulemaking" in March 1989 requesting public comment on issues, alternative methods and requirements to improve nuclear power plant procurement, receipt inspection, testing, and dedication of equipment, materials and services. The comment period closed in July 1989, by which time the NRC had received approximately 60 responses. These are under review and will help inform a decision about the direction future rulemaking should take.

During the fiscal year, a major focus of the program involved vendor misrepresentation of products sold to nuclear power plants. The vendor inspections and reviews of misrepresented vendor products including molded case circuit breakers, metal-clad circuit breakers, trip devices, valves, valve parts, plate material, fasteners, and relays. NRC review of these issues resulted in the issuance of 15 NRC Bulletins, Information Notices, and their supplements to alert the nuclear industry and the public to the vendors, sources, and supply of those misrepresented products. The vendor inspection staff also provided extensive technical support to the NRC Office of Investigations in its effort to ascertain the extent of vendor wrongdoing and whether there were violations of Federal laws in the vendors' sales and services to the nuclear industry.

Other undertakings of the Vender Inspection Program included major on-site team inspections performed at Prairie Island (Minn.), Haddam Neck (Conn.), and Crystal River (Fla.), as part of a continuing examination of licensee practices in procurement, dedication of commercial-grade components for nuclear services, and interactions with contractors and vendors. With assistance from the headquarters vendor inspection staff, the NRC Regional Offices inspected procurement, dedication, and environmental qualification of equipment at many nuclear power plants, including Maine Yankee, South Texas, Washington Nuclear Power Unit 2, Diablo Canyon (Cal.), and Comanche Peak (Tex.).

These procurement inspections revealed that some licensees need to improve their audits of vendors, improve receipt inspection and testing, and increase the level of engineering involved in development of pro-



Major on-site team inspections under the Vendor Inspection Program were performed at a number of nuclear plants during the fiscal year, among them the Prairie Island (Minn.) facility. The two-unit facility is located near Red Wing, Minn., on the Mississippi River.

curement specifications and in verifying that the product, as received, meets the established specifications. As a result of these inspections, some licensees have been required to conduct a "look-back" program to review past procurements and to assure the suitability of some equipment and materials installed in safety-related applications.

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 SALP program (see below), enforcement actions, performance indicator tracking, trend analysis, event evaluation, operator examinations, and inspection findings. The fruition of the process comes during a semiannual meeting of NRC senior management to discuss and appraise operating plant performance. On this 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 the NRC's senior management's characterization of their overall performance. The practice of bringing senior managers together regularly to review plant performance was established following an incident at the Davis-Besse (Ohio) plant in 1985. (See the 1985 NRC Annual Report, pp. 62 and 125.)

As noted, a principal and regular source of data by which licensee performance is judged is the SALP program. Within the framework of 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 all available data related to each facility.

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

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, most recently in "Historical Data Summary of the Systematic Assessment of Licensee Performance" (NUREG-1214, Revision 5, October 1989).

Man-Machine Interface

During this fiscal year, the staff continued its review of two human factors-related TMI action items, "Detailed Control Room Design Review" (DCRDR) and "Safety Parameter Display System" (SPDS). The staff performed on-site audits of several licensee DCRDR and SPDS programs and conducted meetings and telephone conferences with a number of other licensees to evaluate the progress that licensees have made toward implementing their programs. The staff plans to complete remaining DCRDR audits by March 1990.

In April 1989, the staff issued Generic Letter 89-06, "Task Action Plan Item 1.D.2—Safety Parameter Display System—10 CFR 50.54(f)," which requested that each licensee certify whether its SPDS met NRC requirements or identify when its SPDS would meet the requirements. A guidance document, NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," was included with the Generic Letter to assist licensees in performing their self-assessments. Fourteen nuclear units were exempt from responding because the NRC had already approved their SPDS. Of the remaining licensees, approximately 36 percent have certified that their SPDS meets NRC requirements; about 47 percent indicated that they must modify their SPDS to meet requirements, and provided a schedule for doing so. The remaining 17 percent have not yet provided a certification. The staff is evaluating each response to the Generic Letter and developing a response plan, closing out this TMI action item for licensees who have satisfied or who have committed themselves to satisfying NRC requirements, and taking follow-up action for licensees not yet meeting the requirements.

Training

During fiscal year 1989, the staff continued to evaluate the results of the Institute of Nuclear Power Operations (INPO) accreditation program to determine whether the industry's voluntary efforts will suffice to ensure that the training is appropriately performance based. As part of the evaluation, NRC staff personnel are present as observers when utilities' training programs are under scrutiny by an INPO accreditation team. The staff has also conducted training inspections during the report period.

The staff has concluded that the industry is making progress in bringing about improvements in training and in implementing the Commission's policy statement. Although training improvements have been observed, training deficiencies continue to be found in both accredited and non-accredited training programs. Although the Commission continues to endorse the industry accreditation program and defers rulemaking in this area, the staff will continue to evaluate industry implementation of training and qualification programs for nuclear power plant personnel. During the reporting period, this evaluation included training inspections conducted at Susquehanna (Pa.), Turkey Point (Fla.), Nine Mile Point (N.Y.), and Ginna (N.Y.).

QUALITY ASSURANCE

This fiscal year saw completion of several quality assurance (QA) initiatives to improve NRC inspection effectiveness. One of those initiatives, the performance-based concept, was first introduced in August 1987 in SECY-87-220, "Assurance of Quality." Since then, the staff has published NUREG/CR-5151, "Performance-Based Inspection," and implemented the "Inspecting for Performance" training course for NRC inspectors. The purpose of the "Inspecting for Performance" course, and for NUREG/CR-5151 which describes the course's methodology, was to broaden the scope and increase the technical depth of NRC inspections by implementing techniques that are based on observing and evaluating activities affecting plant reliability and safety. The NRC recently decided to provide the "Inspecting for Performance" course to the staff over the next five years. Also, a course modeled after "Inspecting for Performance" has been developed and is being taught within the nuclear industry.

In order to reinforce the performance-based inspection philosophy, the NRC headquarters and regional staffs completed a series of quality verification function inspections. The first goal of these inspections was to improve the inspectors' ability to evaluate plant reliability and safety. This goal was realized by increasing the inspectors' emphasis on actual observation of ongoing facility work activities and moderating the emphasis on document and program reviews. By focusing attention on activities that are important to reliable and safe plant operations, the NRC's performance-based inspections were a model that encouraged licensees to conduct their verification efforts in a similar manner and to manage and operate their facilities with a more performance-based focus.

A revision of the NRC's "Light-Water Reactor Inspection Program for Plant Operations" (NRC Inspection Manual Chapter 2515) was completed during the report period. The new program requires that licensees be inspected in several technical disciplines, including operations, maintenance, radiological controls, engineering, physical security, and environmental protection. It also provides additional inspection guidance to follow-up on operational events and safety issues and to investigate the root causes and corrective actions related to identified concerns. With these changes, the NRC's inspection program for operations now provides greater flexibility in applying inspection resources to deal with issues of reliability and safety importance.

Several planned activities reinforcing the performance-based concept are also in progress. They include a potential revision to the existing NRC "Standard Review Plan (SRP) for Quality Assurance" (Chapter 17 of the FSAR). The SRP, which outlines the criteria against which new Quality Assurance (QA) programs are measured, will recognize the management, performance and verification components that make up a complete QA program. Also, the staff is studying Section 6 of the Standard Technical Specifications to identify where it can be modified or clarified to more clearly reflect theperformance-based concept and general principles of quality.

In a related activity, the NRC issued a report which evaluates what the NRC is doing in the area of digital software control, what NRC guidance for software quality and QA is currently in place, and how much reliance the industry is placing on digital computers to ensure safe nuclear power plant operation. Future NRC actions related to software standards and inspection are being considered.

Maintenance

Good maintenance is a key factor in achieving and maintaining a high level of safety in plant operations throughout the life of a nuclear power plant. In order to focus attention on this area, maintenance was chosen as the special area of emphasis in the Mandatory Team Inspection Program (see above). The staff continued activities related to the evaluation of maintenance effectiveness in the nuclear power industry by conducting maintenance team inspections at commercial nuclear power plants. Team inspections were completed at 30 sites during fiscal year 1989.

The inspection teams are led by a team leader from the Regional Offices and are composed of two reactor/project engineers and a radiation specialist, also from the Regional Offices. In addition, the teams are staffed with two engineers from Headquarters. Each inspection takes six weeks and, typically, the team leader devotes a few additional weeks to the inspection. The inspection is broken down into one week of preparation, two weeks of on-site inspection, one week of in-office inspection, and two weeks of documentation and report writing.

This inspection program mounts a concentrated effort in the detailed observation of all the maintenance-related activities at a nuclear power plant. The major areas of the inspection include plant performance related to maintenance, management support of maintenance, and maintenance implementation.

Results to date indicate that all sites inspected have maintenance programs in place and all but one have been rated as satisfactory or good. Evaluation of the implementation of maintenance shows that most plants are satisfactory. Inspections are planned for every site that has an operating plant and are scheduled to be completed by April 1991.

On March 23, 1988, the Commission published in the *Federal Register* (53 FR 9430) a final policy statement on maintenance in nuclear power plants. On November 28, 1988, the Commission published in the *Federal Register* (53 FR 47822) a proposed rule on maintenance programs for nuclear power plants. In May 1989, the Commission was briefed on the results of the maintenance team inspections. In June 1989, the Commission decided to hold the maintenance rule in abeyance, in order to monitor the status of industry initiatives. The staff is continuing to work on developing a Regulatory Guide and maintenance effectiveness indicators. On August 17, 1989, the Commission published in the *Federal Register* (54 FR 33988) a notice of availability of the draft Regulatory Guide entitled "Maintenance Programs for Nuclear Power Plants."

OPERATOR LICENSING

With the absence of new plant operating licenses, only replacement examinations for power- and non-power-reactor operators are currently being administered. The responsibility for administering written and operating examinations to license candidates 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 1989, the NRC issued 328 new licenses and 437 renewal licenses for reactor operators (ROs) and 328 new licenses and 784 renewal licenses for senior reactor operators (SROs). Based on the success of the requalification pilot program, the NRC reinstated its requalification program and began administering requalification examinations to currently licensed operators in October 1988. In addition to replacement examinations, the NRC administered 536 requalification examinations at 39 reactor sites. The new NRC requalification examination process is proving effective in determining the quality of the licensees' requalification program.

The NRC is in the process of implementing a national examination schedule to enable both the NRC and the facility licensees to manage their resources better by establishing a predictable schedule against which to plan their work.

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

(1) Completion of a successful pilot test of a new program for testing prospective ROs and SROs on their understanding of the generic fundamental knowledge required for operating a nuclear power plant. The NRC implemented this new process on October 1, 1989, and will administer the examination on a national scale three times a year.

(2) Pursuing methods for incorporating many of the "lessons learned" from the well accepted requalification examination process into the replacement operator licensing process. The NRC plans to implement these changes in the next 12 to 18 months.

(3) Receipt of 19 "simulation facility certifications" and plans from five plants for the use of simulation facilities that are other than solely plant-referenced simulators. This is in response to the certification requirements for simulation facilities promulgated in the latest revision to 10 CFR Part 55.

(4) Developing and implementing new examiner training and certification requirements designed specifically to enhance examiner techniques. These courses have significantly reduced the time necessary to certify new examiners.

(5) Preparation of a notice of proposed rulemaking to amend 10 CFR Part 55 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.

(6) Institution of a program to reduce the number of duplicate questions being maintained in the examination question bank located at Idaho National Engineering Laboratory in Idaho Falls, Idaho. In addition, a program has been developed for automatically generating and modifying examination outlines based on the NUREG/BR-0122, the "Examiner's Handbook for Developing Operator Licensing Examination."

(7) Plans for a revision to NUREG/BR-0122 during the first quarter of fiscal year 1990. This revision will provide specific guidance to be used in the formatting of NRC-administered operator licensing questions and examinations resulting in greater consistency between examinations.

Oversight of Regional Office performance during the fiscal year showed continued support and implementation of program office procedures. Revision 5 to the "Operator Licensing Examiner Standards" (NUREG-1021), was issued and was implemented at all Regional Offices. The revision included the definitive version of the revised requalification program evaluation. The program office has appraised the notion of centralizing all operator licensing functions associated with non-power reactors and has determined that it would be cost-effective and beneficial to administer that function from Headquarters. The Commission will evaluate this approach before implementation. Twenty-three additional contract examiners were certified to support the increase in requalification examinations necessary to meet testing requirements in the regulations. The backlog of requalification examinations that occurred during the period of pilot testing of the current approach is expected to be eliminated during the next two years.

EMERGENCY PREPAREDNESS

During the report period, evaluation of emergency preparedness in support of licensing activities continued at a high level of intensity. With respect to the controversial Shoreham (N.Y.) facility—as a result of the Commission's March 3, 1989, decision (CLI-89-02) dismissing the intervenors—all contested Shoreham proceedings were terminated. To ensure that no safety issues remained unexamined before issuance of an operating license, the Commission directed the NRC staff to report on how the contested issues, each of which involved off-site emergency preparedness, had been resolved. The staff prepared a "Director's Findings on Shoreham Emergency Planning Contentions" (April 7, 1989) and a supplement to the Safety Evaluation Report in which the staff concluded that all outstanding emergency planning contentions had been satisfactorily resolved and that the emergency plans for Shoreham provided reasonable assurance that adequate protective measures could be taken in the event of a radiological emergency. Regarding the Seabrook (N.H.) plant, support for licensing entailed evaluation of the adequacy of the applicant's vehicular alert and notification system for the Massachusetts portion of the emergency planning zone, development of a Safety Evaluation Report reflecting the level of emergency preparedness at Seabrook, and support for a successful presentation on emergency preparedness before the Advisory Committee on Reactor Safeguards. Emergency planning issues were also a significant factor in the restart of the Pilgrim (Mass.) facility. Support was provided in a number of areas, including input for a Director's Decision, communicating with State and local emergency planning officials, and coordinating with the Federal Emergency Management Agency (FEMA). These efforts culminated in a full-participation exercise, conducted on October 12 and 13, 1989. Preliminary results indicate that off-site preparedness at Pilgrim has been significantly improved from that existing at the time of the shutdown. Another licensing effort involved the review of the emergency plans for Comanche Peak (Tex.) and the preparation of a Safety Evaluation Report. The initial emergency preparedness exercise for this plant was observed and evaluated. At the remaining 70 or more reactor sites that have operating nuclear power reactors, the evaluation of the annual emergency preparedness exercise continued to be an important factor in ensuring an adequate level of safety.

A new rule, 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors," was issued in April 1989 which would allow early resolution of emergency planning and other issues that were formerly considered in operating license hearings.

Another addition to emergency preparedness was the development of implementing regulations in response to Executive Order 12657. This order governs Federal assistance in emergency planning for commercial nuclear power plants when State and local governments decline or fail to participate in the planning process. FEMA published an interim rule to implement this directive.

A number of activities were conducted to improve the emergency planning inspection process. The procedure for the observation and evaluation of emergency preparedness exercises was revised to reflect current practice. Supplemental procedures concerning the inspection of the licensee's on-site emergency planning program were revised to include information gained from past inspections.

SAFETY REVIEWS

Applications of Probabilistic Risk Assessment

Fiscal year 1989 was marked by an expanded use of probabilistic risk assessment (PRA) methods and insights in regulatory activities, both in traditional and new applications—the latter in the areas of inspection guidance, human performance, accident management, and operating plant performance.

The staff has completed its review of a number of licensee-submitted PRA studies. The staff is in the final stages of reviewing the Brunswick (N.C.) and Three Mile Island Unit 1 (Pa.) PRAs, and is in the early stages of reviewing the newly submitted South Texas PRA study. Significant progress has been made in reviewing safety improvements and PRA studies for standard plant designs, including Westinghouse SP/90, Combustion Engineering CESSAR 80+, and General Electric ABWR.

The application of PRA results and insights to licensing and inspection activities continues to be successful. PRA-based input was provided for the planning of 30 maintenance team inspections (MTIs). In addition, PRA-based guidance was developed for three safety system functional inspections (SSFIs), two safety systems outage modification inspections (SSOMIs), and two risk-based operational safety and performance assessments (ROSPAs). Additional risk-based inspection guidance documents were completed and provided to seven resident inspectors. Methodology has been completed for producing PRA-based guidance for plants that do not have PRAs. This permits use of plant-specific design data and generic risk insights to develop plant-specific risk-based guidance. The development of risk-based guidance has been revised to focus on selected plant systems, thus making the process more cost-effective.

In the area of licensing actions, PRA insights continued to be used as one of the bases in the review and evaluation of licensee submittals. In particular, the revised Standard Technical Specifications (STS) submitted by the various owners groups were reviewed from the perspective of risk-based considerations.

PRAs have also provided a vehicle for investigating the risk impact of variations in human performance in operating plants. To this end, the NRC is undertaking risk sensitivity evaluations of two state-of-the-art PRA studies for a PWR and BWR. Finally, PRA applications are now being used on a regular basis to assess the significance of operating events and inspection findings, to identify and assess severe-accident precursors, and to provide plant design-related risk insights. The integration of this information produces valuable insights about the performance of operating plants which is discussed at NRC senior management meetings and at the briefings for these meetings.

Interfacing Systems LOCA Program

The NRC has initiated a new effort regarding the resolution of various concerns related to the potential for an event involving the interface between high- and low-pressure systems at nuclear power plants. The event consists of an unisolable interfacing systems loss-of-coolant accident (ISLOCA) which bypasses the containment. Recent operating experience at foreign and domestic nuclear power plants indicates that the likelihood of an ISLOCA needs to be reassessed. Although the event was identified in the early WASH-1400 study, its evaluation in terms of risk had been limited in scope and may be subject to underestimation. For example, on several occasions, operator errors or improper procedures have created some of the conditions that could lead to an ISLOCA. In some cases, primary coolant has been released into auxiliary buildings.

In order to address these concerns, an NRC interoffice (NRR/RES/AEOD) effort was mounted. The goal of the program is to attain a high confidence that a high-consequence ISLOCA will not occur in the current generation of U.S. plants. To achieve that aim, the following areas will be examined: (1) the likelihood that an ISLOCA will not occur; (2) the likelihood that core damage, in the event of an ISLOCA, can be prevented or significantly delayed by relying on existing plant equipment, procedures, and training; and (3) the likelihood, in the event of an ISLOCA and core-melt, that equipment, procedures, and training can be used to minimize the off-site radiological consequences.

The principal elements of the interoffice program are: (1) a series of plant audits for determining the current status of a sample of representative plants; (2) a data search of the operational experience, in order to determine the types of events that may be considered as ISLOCA precursors; and (3) an analysis and evaluation of the overall risk associated with the ISLOCA event.

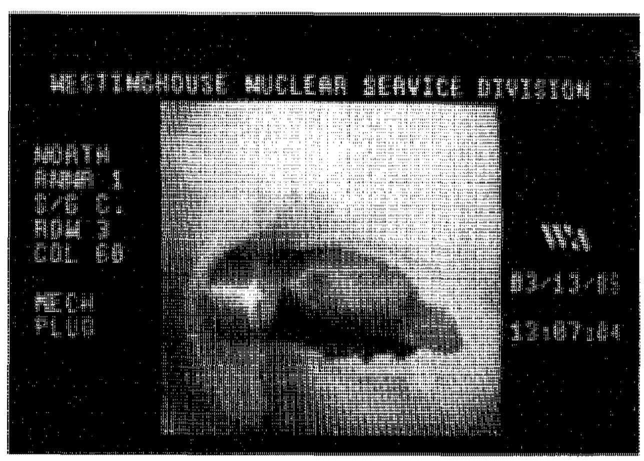
The ISLOCA Program will provide the necessary information for establishing whether plant design or operation improvements are necessary with respect to ISLOCA events. This also would include the determination of which improvements are most effective, and the degree of risk reduction that may be expected as a result of such improvements. The program is scheduled to reach a technical conclusion by the end of fiscal year 1990.

Steam Generator Tube Plug Failure

Steam generator tubes—which are an integral part of the reactor coolant system (RCS) boundary in pressurized-water reactors—are frequently subject to corrosion and/or mechanically induced degradation. When such degradation is observed during inservice inspection to exceed acceptable limits given in plant Technical Specifications, the affected tube is removed from service by installing plugs into each end of the tube. These plugs thereby replace the affected tube as part of the RCS boundary.

On February 25, 1989, North Anna Unit 1 (Va.) underwent a failure of a steam generator tube plug following a reactor trip during a feedwater isolation transient. The failed plug was a "mechanical"-type plug, supplied by Westinghouse. The plug failure caused a 75-gallon-per-minute, primary-to-secondary leak and was the subject of NRC Information Notice 89-33, "Potential Failure of Westinghouse Steam Generator Tube Mechanical Plugs." The failure mechanism involved a full circumferential severance of the top portion of the plug from the body of the plug. The top portion of the plug was propelled up the length of the affected tube by primary system pressure to a point just above the U-bend tangent point, where it punctured the outer curvature of the tube. The top portion of the plug subsequently dented an adjacent tube.

A post-event examination established that primary water stress corrosion cracking (PWSCC) was the mechanism leading to the plug failure. Numerous plants have experienced PWSCC and leaks of Westinghouse mechanical plugs, although the plug



The broken Steam Generator tube plug from North Anna Unit 1 (Va.) can be seen to the left where it lodged in the tube U-bend. Note the rupture of the pipe wall and the denting of the adjacent tube caused by the broken plug.

failure at North Anna Unit 1 is, to date, a unique event. Virtually all incidents of PWSCC to date affecting Westinghouse mechanical plugs have involved plugs fabricated from a subset of Inconel 600 material heats used for the Westinghouse mechanical plugs. The subset of Inconel 600 material heats has been found by Westinghouse to exhibit metallurgical microstructure characteristics which are not optimal for good resistance to PWSCC. This condition is attributable to the relatively low mill annealing temperatures received by these heats. Westinghouse has subsequently upgraded its procurement specifications to ensure that material heats used to fabricate new plugs will receive the proper mill annealing temperature to achieve the desired microstructural characteristics.

At the time of the North Anna Unit 1 plug failure, approximately 9,000 plugs from the so-called "most susceptible" Inconel 600 material heats were believed to have been installed at approximately two dozen plants in the United States. In response to the PWSCC problems being experienced with plugs fabricated from these material heats—including the North Anna Unit 1 plug failure—the NRC staff issued Bulletin 89-01, "Failure of Westinghouse Steam Generator Tube Mechanical Plugs." This Bulletin requested that utilities determine whether Westinghouse plugs from certain material heats of Inconel 600 (specified in the Bulletin) were installed in their steam generators and, if so, that remedial actions would be implemented as necessary to ensure that the plugs will continue to provide adequate assurance of RCS boundary integrity. Remedial action could include repair or replacement of the plugs.At the close of the report period, responses to the Bulletin had been received for all PWRs and were under NRC staff review.

Bulletin 89-01 addressed only Westinghouse mechanical plugs fabricated from Inconel 600 material heats, which are the most susceptible to PWSCC and of most immediate concern. Westinghouse studies that predict time-to-failure indicate that other Inconel 600 material heats may also become susceptible over the longer term. In addition, PWSCC has been observed in plugs manufactured by Babcock & Wilcox (B&W), although these cracks have thus far only occurred at non-critical locations which are not part of the RCS boundary. PWSCC problems with B&W plugs to date are described in NRC Information Notice 89-65, "Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Manufactured by Babcock & Wilcox." PWSCC problems with steam generator tube plugs manufactured by both Westinghouse and B&W continue to be a subject of close scrutiny by industry and the NRC staff.

Erosion/Corrosion in LWRs

The catastrophic piping component failure on December 9, 1986, at Surry Unit 2 (Va.) revealed the significance of flow-assisted corrosion (FAC) or erosion-corrosion in light-water-reactor (LWR) plants. Subsequent to the event, considerable effort by the industry, in conjunction with the NRC and the research community, has resulted in a sound understanding of the corrosion mechanism taking place in single- and two-phase piping systems. As described in the *1988 NRC Annual Report*, p. 29, significant information regarding the event at Surry Unit 2 was provided to the industry. Inspections conducted by the NRC staff and its consultants during 1988 as follow-up action to NRC Bulletin 87-01, "Pipe Wall Thinning in Nuclear Power Plants," indicated the need for the industry as a whole to establish a long term systematic program to adequately address the FAC issue. The industry and the research community have developed the necessary tools to conduct systematic evaluation of the systems in LWR plants susceptible to FAC and to evaluate those systems on the basis of the parameters that strongly affect FAC: temperature and pressure of the fluid, piping system geometry, piping component material, chemical control agent used to mitigate the effect of general corrosion, fluid flow rate, and fluid dissolved oxygen concentration. Additionally, for two-phase flow systems, the percent moisture entrained in the steam plays a significant role in the FAC phenomenon.

On May 2, 1989, the staff issued Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." The Generic Letter required all licensees to provide assurance that a program—consisting of systematic measures to ensure that erosion/corrosion does not lead to degradation of single- and two-phase high-energy carbon steel systems—has been implemented. The staff has completed an audit of the responses to Generic Letter 89-08. In the responses, it was confirmed that licensees have implemented, were implementing, or were improving existing long term programs addressing the FAC issue.

Station Blackout Rule

The bases for and the development of the "station blackout" rule (10 CFR 50.63) are set forth in the *1988 NRC Annual Report*, page 30. The discussion that follows includes a summary of this background and also provides the implementation status of the rule.

The term "station blackout" means the loss of offsite alternating current (a.c.) power to the 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 (1) 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, and (2) reduce the expected frequency of core damage resulting from station blackout events by ensuring that the plants can cope with a station blackout for a specified duration.

All licensees have submitted responses to the station blackout rule, describing the modifications and procedural changes they intend to implement to cope with a station blackout. The NRC staff is currently reviewing these responses and conducting audit reviews at selected nuclear plants for those plants considered to be most susceptible to station blackouts or those that are most likely to be undesirably affected by a station blackout. The reviews conducted to date show that the licensees have, in general, devised plausible plans for coping with a station blackout. However, the staff is finding some inconsistencies regarding the required station blackout durations involved, and in some cases, with procedures that are planned in order to meet the coping duration. The staff is working with the Nuclear Management and Resources Council to resolve these problems.

The present schedule for implementing the station blackout rule calls for completion of the staff reviews by the end of 1990 and completion of plant modifications (with a few exceptions) by the end of 1992.

TMI Action Plan Implementation Status

The accident at Three Mile Island Unit 2 (Pa.) in 1979 led to a thorough review of NRC regulatory requirements for nuclear power plants. NUREG-0737, issued in November 1980, contains all the requirements resulting from this review which were approved by the Commission for implementation at plants in operation or under construction. Supplement 1 to NUREG-0737, issued in December 1982, delineates the requirements for emergency response capabilities.

In April 1989, the NRC undertook a detailed review of the implementation status of TMI action plan items. When applied to the 112 licensed reactors, the TMI action plan resulted in a total of 13,624 applicable items for licensees to implement. The NRC review concluded that, as of May 19, 1989, only 254 items (less than 2 percent) remained to be implemented by licensees. These few remaining items are of lower priority than certain other actions also being implemented to improve safety at nuclear power plants. The NRC staff will continue to track the implementation of the remaining items by providing quarterly status reports to the Commission.

BWR Thermal-Hydraulic Stability

On March 9, 1988, a significant event occurred at LaSalle Unit 2 (Ill.), the details of which are described in the 1988 NRC Annual Report, p. 51. The LaSalle event involved power oscillations caused by neutron-flux/thermal-hydraulic instabilities of a magnitude that were not predicted by design analysis, unanticipated by the operators, and potentially in conflict with General Design Criterion (GDC) 12.

Since the event revealed that design calculations are not entirely reliable indicators that a boiling-water reactor (BWR) will not experience instability for an entire operating cycle, greater emphasis on operation procedures, surveillance, and training was deemed necessary to avoid oscillations or to suppress them if they occur. NRC Bulletin 88-07, "Power Oscillations in BWRs," was issued in June 1988 to emphasize the finding to licensees, and associated NRC inspections were initiated. Both the NRC, using national laboratory consultants, and the BWR owners group (BWROG) initiated extensive analytical studies to investigate the need for further actions to provide assurance that oscillations leading to unacceptable consequences would not occur. These studies indicated that, for some permissible operating conditions, asymmetric oscillations could produce temperatures in excess of fuel thermal safety limits, in advance of operator detection on the average power range monitors (APRMs). Corrective actions were developed and issued as Supplement 1 to Bulletin 88-07 in December 1988. All BWRs have implemented these corrective actions.

The BWROG has continued to evaluate the feasibility of long term actions involving automatic protective features to provide greater assurance of avoiding instabilities. This work and initial conclusions were discussed with the NRC in April and September 1989. The BWROG concluded that four approaches to ensure automatic protection are feasible. The concepts involve power/flux-based insertion of control rods to avoid operation in potential regions of instability or reactor protection circuits designed to detect and respond to local neutron-flux oscillations. The BWROG detailed development is continuing. It is expected that final staff review and approval of generic long term solutions will be completed in early 1990, to be followed by licensee implementation.

An additional concern related to the possibility of large-amplitude oscillations during a low-probability anticipated-transient-without-scram (ATWS) event is also being investigated. The basic questions are whether large oscillations will occur during ATWS event conditions, and if they do, will they affect operator control of parameters, such as water level, and will they significantly affect suppression pool temperature. The BWROG believes that there will be no significant effect on operator actions or the suppression pool. However, the NRC studies indicate a potential for greater power increase than the cases studied by GE. The NRC-sponsored work in this area, including improvement of the methodology, is continuing and BWROG results are under review. Resolution of the effect of possible instabilities during an ATWS is expected in 1990.

Reassessment of B&W Reactors

The NRC's Executive Director for Operations informed the Chairman of the Babcock & Wilcox owners group (BWOG), by letter dated January 24,1986, that events at B&W-designed reactors had led the NRC staff to conclude that there was a need to re-examine basic design requirements for B&W reactors. In its response, the BWOG agreed to take the lead in a concerted effort to define the factors in B&W plants causing the frequency of reactor trips or shutdowns and the complexity of post-trip response. The BWOG worked up a reassessment plan which the NRC staff reviewed, proposing certain changes that were incorporated by the BWOG. A final report by the BWOG, "B&W Owners Group Safety and Performance Improvement Program (SPIP)," Revision 5 (BAW-1919), was issued in July 1987. This effort resulted in 222 specific recommendations for improving B&W plant safety and performance.

The NRC staff reviewed BAW-1919 and presented its evaluation in "Safety Evaluation Report Related to Babcock & Wilcox Owners Group, Plant Reassessment Program" (NUREG-1231, November 1987), and in Supplement 1 to NUREG-1231, published in March 1988. Overall, the staff's evaluation was favorable. The staff concluded that the proper implementation of the BWOG/SPIP recommendations by B&W utilities should effect a reduction both in reactor trip frequency and in transient complexity, and should also result in an increase in the level of safety at B&W plants. The staff also concluded that B&W plants do not carry a core-damage risk greater than plants with pressurized-water reactors designed by other vendors.

To ensure that each utility's program would actually implement the SPIP recommendations, the staff began a program of plant-specific audits in October 1988. The audits addressed each utility's (1) program to evaluate the BWOG/SPIP recommendations, (2) implementation of selected key recommendations, and (3) responses to NRC's Office of Inspection and Enforcement (IE) Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation." This third series of audits was instituted because the staff believed that the B&W utility responses to the concerns of IE Bulletin 79-27 (which were not specifically addressed by the SPIP) needed further verification.

The first and third series of audits were completed in 1989. The staff found that each utility had established an adequate program to evaluate the SPIP recommendations for implementation and had responded adequately to IE Bulletin 79-27. (These audits were not performed for the Sacramento Municipal Utility District's Rancho Seco facility because of the licensee's decision to shut the plant down.) The second series of audits to evaluate the implementation of SPIP recommendations is planned for 1990. On the average, each utility has completed implementation on approximately 75 percent of the recommendations applicable to its facility.

Occupational Exposure Data and Dose-Reduction Studies

The 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 TMI-mandated plant upgrades imposed on all LWRs shortly after the 1979 accident. Since 1983, the annual average doses for both PWRs and BWRs have been steadily declining. In 1988, the average dose-per-unit for LWRs was 400 person-rem. This is 5 percent lower than the LWRaverage for 1987. In 1988, the average doses-per-unit for PWRs and BWRs were 336 and 529 person-rem, respectively, as compared to the 1987 averages of 371 and 513 person-rem. Thus, as compared to the 1987 averages, the 1988 average for PWRs decreased by 9 percent, and the 1988 average for BWRs increased by 3 percent. Major contributors to BWR doses in 1988 included recirculation pipe replacement or modification, work related to snubbers, installation and removal of scaffolding and shielding, and drywell decontamination. The activities that were large contributors to PWR doses in 1988 included work related to steam generators, refueling operations, installation and removal of scaffolding and shielding, and decontamination of areas and equipment.

The 1988 dose compilation includes data from 68 PWRs and 34 BWRs. This total reflects the addition of four new PWRs—Beaver Valley Unit 2 (Pa.), Byron Unit 2 (Ill.), Harris (N.C.), and Vogtle Unit 1 (Ga.)—and two new BWRs—Clinton Unit 1 (Ill.) and Perry Unit 1 (Ohio). 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.), and Indian Point Unit 1 (N.Y.) are no longer included because there are no plans to operate these plants in the future. Three Mile Island Unit 2 (Pa.), however, has been retained in the dose compilation because it is still licensed and workers there are accumulating doses during defueling and decommissioning operations.

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. Another study involves the compilation of a research data base on dose reduction projects at nuclear power plants.

Environmental Radioactivity Around Nuclear Power Plants

All licensed U.S. nuclear power plants are required by 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 expected, based on the measurement of plant effluents and the analytic modeling of the environmental exposure pathways. In turn, the studies verify that the plant is in compliance with regulations and not exceeding the amounts defined in the final environmental statements as providing very small risks to members of the public.

An extensive weekly and monthly monitoring program is required for each plant by its radiological effluent technical specifications (RETS) or by licensee effluent control procedures. A Generic Letter (89-01) issued on January 31, 1989, permits reactor licensees to relocate detailed RETS requirements to licensee-controlled documents. The overall level of effluent management and control will be required by the Technical Specifications to remain at current levels, though licensees will have the flexibility to make changes without first seeking NRC approval. (The title of Generic Letter 89-01 is "Implementation of Programmatic Controls for Radioactive Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Off-site Dose Calculation Manual or to the Process Control Program.")

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 radioiodine and particulate dusts for airborne contaminants; to milk, fish, shellfish and vegetables for contaminants that might be ingested as foods. In addition, direct radiation from each of 16 specific sectors of land surrounding the plant is measured by special radiation dosimeters that gauge the cumulative radiation dose at certain 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, 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 monitoring programs conducted by the NRC. In one, the direct radiation in the sectors surrounding each plant is measured independently by NRC dosimeters located in the same areas as the licensee's dosimeters. The results of measurements for each power reactor site from this NRC direct radiation monitoring network are published quarterly in NRC documents, also available at the LPDRs.

In addition, through the five Regional Offices, NRR sponsors contracts with 34 States to monitor the environment. The purpose of the State contracts is to establish policies and procedures for contracts and agreements with those States to independently monitor the environs of NRC-licensed facilities. Under these contracts and agreements, the States provide assistance by collecting samples or making radioactivity measurements in the environs of NRC-licensed facilities. These measurements duplicate as closely as possible certain parts of the licensee's environmental monitoring efforts, but they are carried out independently of those programs. The results of the State's monitoring efforts 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. In the future, results of the State's environmental monitoring will also be available on an annual basis in the LPDRs.

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 1989, the staff—as part of the formalized program for the assessment of major incidents—assigned augmented inspection teams to determine the facts regarding the following operating reactor events:

- Impaired shutdown cooling capability at Oyster Creek (N.J.) in October 1988.
- Electrical fire with loss of forced coolant flow at Oconee Unit 1 (S.C.) in January 1989.
- Backflow of reactor coolant through check valve in high-pressure injection line at Arkansas Unit 1 in January 1989.
- Unit 2 auxiliary transformer fault causes Unit 1 trip and equipment malfunction at LaSalle Units 1 and 2 (Ill.) in March 1989.
- Multiple equipment failures following load rejection at Palo Verde Unit 3 (Ariz.) in March 1989.
- Steam generator tube rupture at McGuire Unit 1 (N.C.) in March 1989.
- Unexpected opening of reactor core isolation cooling system valve at Pilgrim Unit 1 (Mass.) in April 1989.
- Inattentive licensee employees at Braidwood Units 1 and 2 (Ill.) in April 1989.
- Freeze plug failure in service water system at River Bend (La.) in April 1989.
- Hot water intrusion into auxiliary feedwater system at Comanche Peak Unit 1 (Tex.) in April 1989.
- Reactor operation outside bounds of test procedure at Seabrook Unit 1 (N.H.) in June 1989.
- Loss of safety system redundancy resulting in loss of control room instrumentation at Cook Unit 2 (Mich.) in August 1989.
- Inadequate net positive suction head of auxiliary feedwater pumps at Robinson Unit 2 (S.C.) in August 1989.
- Contamination of sub-basement by leaking drums at Nine Mile Point Unit 1 (N.Y.) in August 1989.
- Water spill from refueling water storage tank into auxiliary building at McGuire Unit 2 (N.C.) September 1989.

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. For one, 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 further request specific actions to be taken by utilities and require written confirmation when actions have been completed. In fiscal year 1989, the staff issued 105 Information Notices and seven Bulletins, including supplements. Generic Letters may also be issued to address operational safety matters having broad applicability. In fiscal year 1989, the staff issued 13 Generic Letters of this type, including supplements.

CLEANUP AT THREE MILE ISLAND

During fiscal year 1989, progress continued on the defueling and cleanup of the damaged reactor at Unit 2 of the Three Mile Island nuclear power plant (TMI-2) near Harrisburg, Pa. Defueling is nearing completion, as radioactive waste and fuel debris shipments have continued in parallel. General Public Utilities Nuclear (GPUN) Corporation, the licensee, de-emphasized decontamination efforts in order to concentrate on defueling. The current level of effort in decontamination is geared toward maintaining plant access and operability of systems. When defueling is complete, the licensee intends to redirect its effort to decontamination.

During fiscal year 1989, the central portions of the five layers of the lower core support assembly (LCSA) were cut away and removed. This move provided an access path to the reactor vessel (RV) lower head. Loose core debris was vacuumed from the LCSA and the lower head and loaded into defueling canisters. A large (approximately six feet in diameter by 1.5-ft. thick), once molten, resolidified mass on the RV lower head was broken up and also placed in canisters. Several hundred pounds of fine loose debris were scraped and vacuumed from the hot legs. As of September 30, 1989, approximately 283,000 pounds (94 percent) of core debris has been removed out of a total of approximately 300,000 pounds. The remaining debris is principally located behind the core baffle plates, on the RV lower head, and in the outer periphery of the LCSA. The completion of defueling was expected by November 1989.

Shipment of core debris from the TMI site to the Idaho National Engineering Laboratory (INEL) continued. During fiscal year 1989, four shipments containing a total of 75,500 pounds of fuel debris were shipped. The total shipped to date is 266,800 pounds which is 89 percent of the estimated total to be shipped.

Exposure rates to defueling crews remained low, averaging approximately 10 millirems-per-hour over the course of defueling, to date. Projected cumulative worker exposure for calendar year 1989 was approximately 850 person-rem. This is below the 1988 total of 917 person-rem.

Public hearings on the GPUN proposal to evaporate 2.3 million gallons of accident-generated water (AGW) were held by an Atomic Safety and Licensing Board (ASLB). The hearings concluded on November 15, 1988. On February 3, 1989, the board issued a decision finding in favor of GPUN on all relevant issues. On April 13, 1989 the Commission affirmed the ASLB decision without prejudice to any appeals. GPUN began to construct the evaporator in August 1989. The licensee expected to complete testing and begin operation of the evaporator in late November 1989.

A July 1989 video inspection of the RV lower head disclosed several cracks which appeared to be associated with incore instrument penetration nozzles. Higher quality color videos and a mechanical probe were used in August to obtain better information on the cracks. The cracks appeared to be up to approximately 6 inches in length, 0.25 inch wide, and more than 0.19 inch deep, but not "throughwall" wide. An international research effort, funded in part by the NRC, will obtain samples from the RV lower head, including the area containing the cracks. That effort will take place after defueling has been completed.

The 11-member Advisory Panel for the Decontamination of Three Mile Island Unit 2, which is composed of citizens, scientists, and State and local officials, was formed by the NRC in 1980 to provide input to the Commission on major cleanup issues. (See Appendix 2 for a list of current members of the panel.) During fiscal year 1989, the panel held three public meetings in Harrisburg, Pa. Principal topics discussed by the panel during these meetings were the details of the licensee's AGW disposal system, off-site radiation monitoring programs around TMI-2, and the licensee's proposal to place the facility in long term storage at the conclusion of the current cleanup effort. In October 1988, the panel met with the NRC Commissioners to discuss a variety of concerns of local individuals and other issues.

Financial Aspects of TMI-2 Cleanup

Funding by GPUN. Revenues collected by the GPUN Corporation's three operating subsidiaries in Pennsylvania and New Jersey continued to be expended on cleanup during calendar year 1989. Customer funding of the cleanup amounted to about $7.7 million in 1989 and is estimated to total approximately $255.9 million over the course of the cleanup effort. GPUN continues to provide cash advances from internal sources to alleviate any cash-flow problem related to cleanup activities. The total 1989 advance is estimated at $6.5 million. The GPUN projections provided to the NRC indicate a continuing GPUN commitment to provide such cash advances as needed. Continued improvement in the GPUN's financial condition and cash-flow position gives greater assurance that such cash advances will be made.

Cost-Sharing Plan. During calendar year 1989, the GPUN continued to receive cash payments from all suggested contributors in the TMI-2 cleanup cost sharing plan proposed by Pennsylvania Governor Richard Thornburgh in July 1981. The Edison Electric Institute's (EEI's) industry cost-sharing program paid its committed $16.3 million annual contribution in 1989, the fifth year of industry contributions through the EEI program. The NRC continues to monitor the cleanup funding situation.

ANTITRUST ACTIVITIES

As required by law since December 1970, the staff has conducted pre-licensing antitrust reviews of all construction permit 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 or operating licenses transferring ownership interest or operating responsibility in a nuclear facility are subject to antitrust review.

In fiscal year 1989, the staff completed antitrust operating license reviews associated with three power production facilities and one construction permit antitrust review of a uranium-enrichment facility. The staff also received one request to re-evaluate one of the reviews associated with a production facility. The staff ultimately concluded that there had been no significant activity on the part of the licensees that would create or maintain a situation inconsistent with the antitrust laws. Additionally, the staff initiated one operating license review of a power production facility and one construction permit review of a uranium enrichment facility.

Three license amendment requests were received from licensees involved in proposed mergers. In two of these instances, the staff concluded that there were no significant anti-competitive effects resulting from the proposed merger and, in the third, the staff determined that the competitive effects of the merger were being adequately addressed in another forum; as a result, the staff did not conduct its own competitive review.

During fiscal year 1989, the staff received three license amendment requests to add new owners as licensees. At the close of the report period, the staff was reviewing these amendment requests to determine if there are any significant anti-competitive effects associated with the proposed addition of the prospective new licensees. Moreover, the staff received four amendment requests in the past year from licensees to approve new corporate operating structures.

Although the Commission's antitrust review responsibility is primarily confined to reviews of construction permit and operating license applications, the staff is responsible for ensuring licensee compliance with and enforcement of antitrust license conditions that have been attached to many of the construction permits and operating licenses that have been issued by the Commission. During fiscal year 1989, the staff completed one outstanding compliance proceeding involving the Farley (Ala.) nuclear plant. As a result of a Notice of Violation issued by the staff to the owner of the Farley nuclear plant in 1986, the parties involved in this proceeding began extensive negotiations that were consummated in early 1989 with a settlement agreement that resulted in the complaining party withdrawing its request to the staff for an enforcement action against the Farley licensee. Another outstanding request to enforce antitrust license conditions attached to the Diablo Canyon (Cal.) operating license was under review and the staff anticipated a Director's Decision in this proceeding early in fiscal year 1990. The staff was still reviewing two outstanding requests from licensees to either suspend or delete antitrust license conditions, at the close of the report period.

The staff received three new requests to enforce antitrust license conditions attached to the licenses of the Comanche Peak (Tex.), Shearon Harris (N.C.), and Vogtle (Ga.) nuclear plants, each under review at the close of fiscal year 1989. On April 27, 1989, the Court of Appeals for the District of Columbia dismissed the suit filed in June 1988 by the Ohio Edison Company against the Commission alleging that the Commission was unable to fairly adjudicate Ohio Edison's request to suspend its antitrust license conditions attached to the Perry (Ohio) nuclear plant.

INDEMNITY, FINANCIAL PROTECTION, AND PROPERTY INSURANCE

The Price-Anderson System

Under NRC regulations implementing the Price-Anderson Act, a three-layered system was set up to pay public liability claims in the event of a nuclear incident causing personal injury or property damage.

The first layer requires all licensees of commercial nuclear power plants rated at 100 electrical megawatts (MWe) or more to provide proof of financial protection in an amount equal to the maximum liability insurance available from private sources. Currently, this amount is $200 million.

The second layer provides for a retrospective premium payment mechanism whereby the utility industry would share liability for any damages resulting from a nuclear incident in excess of $200 million. In the event of such an incident, each licensee of a commercial reactor rated at 100 MWe or more would be assessed a prorated share of damages up to the statutory maximum of $63 million-per-reactor-per-incident. At present, the secondary financial protection layer is $7.25 billion (a figure derived from the 115 power reactors rated over 100 MWe which had been licensed to operate prior to the close of the report period times $63 million-per-reactor).

The third layer, government indemnity, had formerly been fixed as the difference between the $560 million limit of liability and the sum of the first and second layers. Government indemnity for reactors was phased out for large power reactors, however, on November 15, 1982, when the sum of the first and second layers reached $560 million. The limit of liability for a single nuclear incident now increases without limit in increments of $63 million for each new commercial reactor licensed.

Renewal of the Price-Anderson Act

On August 20, 1988, after a five-year effort to renew the Price-Anderson Act, H.R. 1414 was enacted as P.L. 100-408, "The Price-Anderson Amendment Act of 1988." The Act, among other things, extends the Price-Anderson Act for 15 years, to August 1, 2002; increases the deferred retrospective premium from $5 million to $63 million-per-facility-per-incident; and requires that the President establish a "study commission" to study means of fully compensating victims of a nuclear incident where the damages exceed the limit of aggregate public liability.

On June 3, 1989, the Commission published a final rule in the *Federal Register* (54 FR 24157) amending its regulations to conform to the changes made by P.L. 100-408.

Indemnity Operations

As of September 30, 1989, 137 indemnity agreements with the NRC were in effect. Indemnity fees collected by the NRC from October 1, 1988, through September 30, 1989, total $98,100. Fees collected since the inception of the program total $23,539,234. 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 32 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 23d 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 1989 totaled $9,076,550, which is approximately 60.3 percent of all premiums paid on the nuclear liability insurance policies issued in 1979 and covers the period 1988-1989. The refunds represent 67 percent of the premiums placed in reserve in 1979.

Utility Financial Qualification and Corporate Restructuring

NRC rules (10 CFR 50.33(f) and Appendix C to 10 CFR Part 50) provide for pre-licensing financial qualifications reviews and findings regarding electric utilities that apply for power reactor construction permits. Such pre-licensing reviews and findings are not required for utilities at the power reactor operating license stage. (For background, see the *1986 NRC Annual Report*, p. 150). Non-utility applications, such as for non-power reactors, are reviewed for financial qualifications at both the construction permit and operating license stages. The NRC monitors utilities that experience severe financial difficulties at either the construction permit or the operating license stage to assure that such difficulties do not have negative safety impacts.

The NRC also reviews and approves electric utility plans for corporate restructuring to assess any impacts on licensed activities. The restructurings, actual or proposed, include (1) sale and leasebacks of nuclear power plants, (2) the formation of holding companies and utility subsidiaries, and (3) the formation of operating service companies involving outside investors.

Incentive Regulation of Electric Utilities

Economic performance incentives established by State public utility commissions (PUCs) are applicable to the construction or operation of about 75 nuclear power reactors, owned by 31 utilities in 20 States. (For background, see the *1986 NRC Annual Report*, page 150.) The NRC headquarters staff continues to monitor development of the incentives and periodically provides an updated report on all nuclear plant incentives to its Regional Offices. The staff maintains contact with the PUC staffs and the utilities responsible for implementing the incentives, in order to obtain the updated information and to consider possible safety effects of the incentives.

The Massachusetts Department of Public Utilities recently adopted a settlement agreement in which Boston Edison's charges to ratepayers are tied to the performance of the Pilgrim nuclear power plant. Increases and decreases are determined by capacity factor, SALP scores, and five performance indicators as reported by NRC and INPO. This is the only case in which these scores or indicators are used for rewarding or penalizing performance.

In this area, a new concept has been adopted for operation of the Diablo Canyon (Cal.) plant. Rather than adjusting base rates according to capital costs, the plan requires the licensee to earn revenue strictly on the amount of electricity generated by the facility. This arrangement creates more pressure to operate for short term economic benefits. It represents the only case in which profit incentives have been used directly to determine electric utility earnings.

Property Insurance

The NRC requires its power reactor licensees to carry on-site property damage insurance to provide an assured source of funding for cleanup and decontamination of a reactor plant following an accident. Such insurance is needed so that the pace and thoroughness of cleanup following an accident does not cause a threat to public health and safety because of lack of funds.

In 1987, the Commission revised its property insurance regulation to increase the amount of required insurance to slightly over $1 billion. In addition, the 1987 revision requires that any proceeds from this insurance must be expended first to stabilize, decontaminate, and clean up a reactor that has experienced an accident, when such action is required to protect public health and safety. To protect against claims from a licensee's creditors and bondholders, the insurance proceeds subject to this priority are required to be deposited with an impartial trustee, who will disburse funds for decontamination and cleanup.

After the 1987 amendments were issued, the Commission was informed by the insurers offering the property insurance that they were able neither to find anyone to act as trustee nor to incorporate the trusteeship provisions in their policy language by the October 4, 1988 deadline required by the rule. The insurers also believe that the impartial trusteeship provisions of the rule may not accomplish the intended objective of sheltering insurance proceeds from claims by bondholders and creditors. Consequently, in June 1988, the insurers and representatives of the nuclear industry submitted three petitions for rulemaking which seek to replace the trusteeship provisions of the rule with decontamination liability provisions that, petitioners believe, offer better protection of insurance proceeds from competing claims. The petitions also sought clarification of the stabilization and decontamination provisions of the rule. The Commission has extended for 18 months the implementation date of the stabilization and decontamination priority and trusteeship provisions, so as to provide adequate time to consider these petitions. On September 27, 1989, the Commission also approved publication of a proposed rule addressing petitioners' expressed concerns.

The seventh annual property insurance reports submitted by power reactor licensees indicated that, of the 75 sites insured as of April 1, 1989, 67 are covered for at least the $1.06 billion required in the rule. The remaining eight sites have sought or been granted exemptions from the full amount of required coverage because of their small size or operating status. Thirty-eight sites carry the maximum $1.725 billion currently available.

The NRC has been informed that, as of November 15, 1989, capacity provided by Nuclear Electric Insurance Limited-II will increase to $975 million, in excess of $500 million. On January 1, 1990, American Nuclear Insurers intends to increase its excess capacity above the $500 million in primary coverage, to $560 million. These actions would bring total available property insurance capacity to $2.035 billion.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS), established by statute in 1957, 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 of 1954, as revised, 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 (DOE). 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.

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 1989, 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 specific research topics:

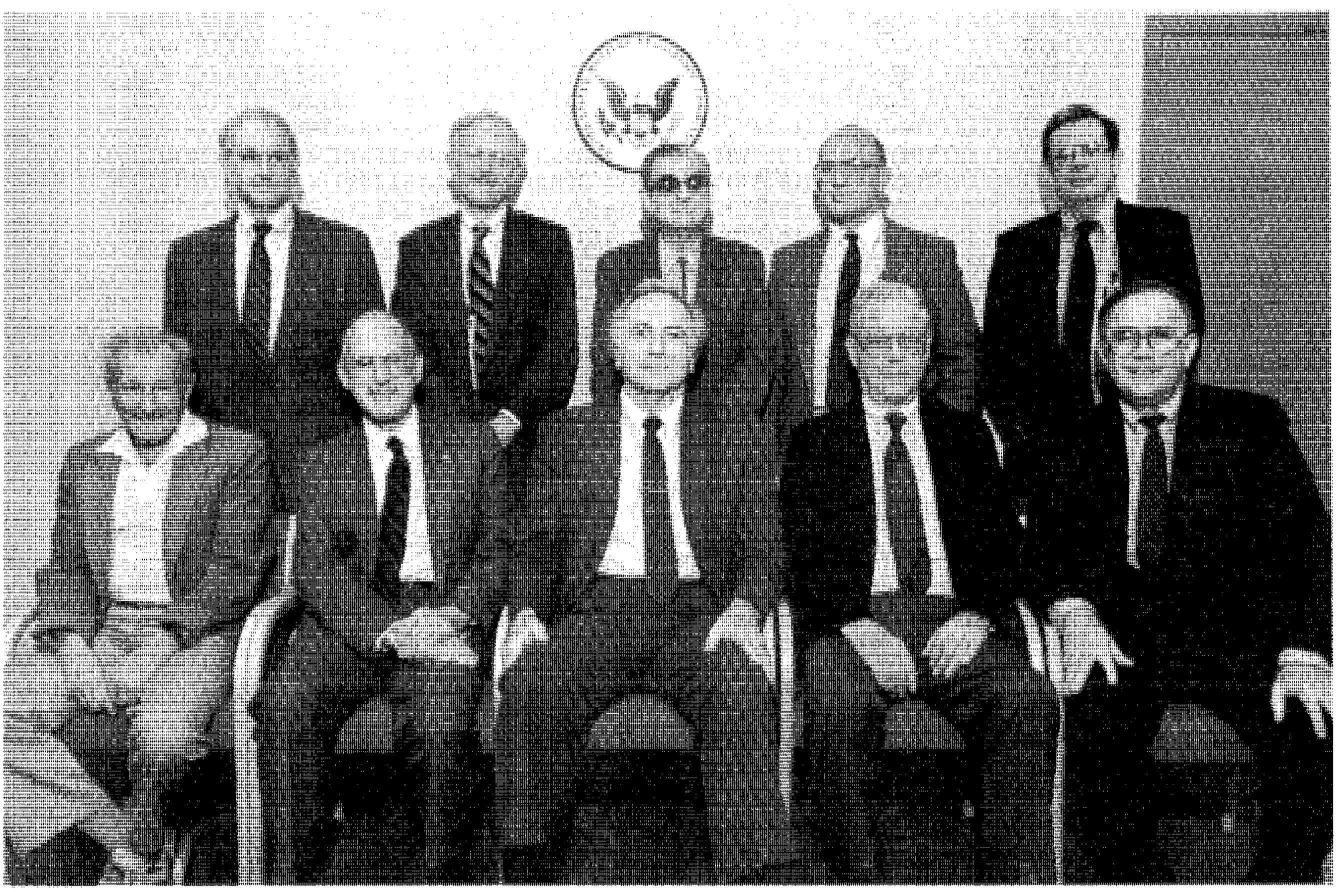
- Containment Structural Integrity
- Equipment Qualification-Risk Scoping Study
- Embrittlement of Reactor Pressure Vessel Supports
- Fire Risk Scoping Study
- Human Factors Research Program Plan
- Individual Plant Examination Program

- Inservice Inspection of Boiling Water Reactor Pressure Vessels
- Integration Plan for Closure of Severe Accident Issues
- Piping Integrity
- Safety Goal Policy Implementation Plan
- Severe Accident Management
- Thermal-Hydraulic Phenomena
- Electric Power Research Institute Advanced Light-Water Reactor Requirements Document.

The committee's activities during this period also included reports of specific licensing actions on the ALChemIE isotope enrichment facility, the Power Reactor Inherently Safe Module design, the Sodium Advanced Fast Reactor design, the Modular High-Temperature Gas Cooled Reactor, the Peach Bottom (Pa.) Atomic Power Station restart, the operating license application for the Limerick (Pa.) Generating Station, and full-power operation of the Seabrook (N.H.) Station.

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

- Experimental Program on B&W Steam Generators



As of September 1989, the members of the Advisory Committee on Reactor Safeguards were, standing, left to right: Mr. Charles J. Wylie, retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.; Dr. Paul G. Shewmon, Professor, Metallurgical Engineering Department, Ohio State University, Columbus, Ohio; Dr. Chester P. Siess, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.; Dr. Ivan Catton, Professor of Engineering, Department of Mechanical, Aerospace and Nuclear Engineering, School of Engineering and Applied Science, University of California, Los Angeles, Cal.; Mr. James C. Carroll, retired Manager, Nuclear Operations Support Department, Pacific Gas & Electric, San Francisco, Cal.;

Seated, left to right are: Dr. Harold W. Lewis, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.; Vice-Chairman Carlyle Michelson, retired Principal Nuclear Engineer, Tennessee Valley Authority, Knoxville, Tennessee, and retired Director, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.; Chairman Dr. Forrest J. Remick, Associate Vice-President for Research and Professor of Nuclear Engineering, The Pennsylvania State University, University Park, Pa. (Dr. Remick was appointed to the Nuclear Regulatory Commission in December 1989); Dr. William Kerr, Professor of Nuclear Engineering, University of Michigan, Ann Arbor, Mich.; and Mr. David A. Ward, Research Manager, retired, E.I. du Pont de Nemours & Company, Savannah River Laboratory, and Consulting Engineer, North Augusta, S.C.

- NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants"
- Implementation of the Safety Goal Policy
- Leak-Before-Break Technology
- NRC's Severe Accident Research Program Plan
- NRC's Human Factors Programs and Initiatives
- Boiling Water Reactor Core Power Stability
- Reliability and Diversity
- NRC's Thermal-Hydraulic Research Program
- Fire Risk Scoping Study.

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

- Proposed Amendment of 10 CFR 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors."
- Proposed Regulatory Guide 1.9, Revision 3, "Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units as On-site Electric Power Systems at Nuclear Power Plants."
- Proposed Regulatory Guide, Task No. EE-006-5, "Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants."
- Draft Final Rule on Standardization and Licensing Reform, 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."
- Resolution of Generic Issue 43, "Air Systems Reliability."
- Proposed Resolution of Generic Issue 99, "Improved Reliability of RHR Capability in PWRs."
- Proposed Resolution of Generic Issues 70, "Power Operated Relief Valve and Block Valve Reliability," and 94, "Additional Low-Temperature Overpressure Protection for LWRs."
- Proposed Resolution of Generic Issue 101, "BWR Water Level Redundancy."
- Proposed Final Resolution of Generic Safety Issue 103, "Design for Probable Maximum Precipitation."
- Proposed Amendment to 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."
- Proposed Resolution of Generic Issue 115, "Enhancement of Reliability of Westinghouse Solid State Protection Systems."
- Proposed Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools."
- Proposed Final Rulemaking Related to Maintenance of Nuclear Power Plants.
- Generic Letter Related to Occupational Exposure of Skin From Hot Particles.
- Generic Letter on Safety-Related Motor-Operated Valve Testing and Surveillance.
- USI A-17, "Systems Interactions in Nuclear Power Plants."
- Proposed Generic Letter Regarding Service Water System Problems Affecting Safety-Related Equipment.
- Proposed Resolution of Unresolved Safety Issue A-40, "Seismic Design Criteria."
- Proposed Resolution of Generic Issue 79, Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown.
- Proposed Resolution of Generic Issue A-29, "Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage."

In performing the reviews and preparing the reports cited above, the ACRS held 12 full committee meetings and 58 subcommittee meetings during the year.

Operational Information / Investigations And Enforcement Actions

Chapter 3

Since its formation in 1979, the NRC Office for Analysis and Evaluation of Operational Data (AEOD) has provided, as one of its primary roles, a strong, independent capability for the analysis of operational data. The office serves as the focal point for the independent assessment of operational events and manages the review, analysis and evaluation of reactor plant safety performance. It is also responsible for the NRC's Incident Response Program, Diagnostic Evaluation Program, Technical Training Center, and the Incident Investigation Program. In addition, AEOD provides support for the Committee to Review Generic Requirements (see below). The office comprises two divisions—the Division of Operational Assessment and the Division of Safety Programs—and reports to the Executive Director for Operations (EDO).

AEOD undertakes the review and evaluation of operating experience for the purpose of identifying significant events and associated safety concerns and root causes; the trends and patterns displayed by these events; the adequacy of corrective actions taken to address the concerns; and any generic applicability of these events and concerns. Pursuant to these tasks, AEOD specifically engages in the following:

- Analysis of operational safety data associated with all NRC-licensed activities and identification of safety issues which require 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 the diagnostic evaluation teams.
- Development of policy, program requirements, and procedures for NRC incident investigations of significant operational events.
- Identification of needed operational data to support safety analysis activities, and development of agency-wide operational data reporting and retrieval methods and systems.
- Development of a coordinated system for feedback of operational safety information to NRC offices, licensees, and other organizations, as appropriate, and preparation of the Abnormal Occurrence Report to Congress.
- Development in consultation with other NRC offices of the NRC policy for response to incidents and emergencies, and assessment of the NRC response capabilities and performance.
- Development of an agency-wide technical qualification program for a broad range of technical positions within the NRC staff, and providing 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 other information reported to the Operations Center, and to assure a proper NRC reaction to reported events.
- Acting as a focal point for coordination of generic operational safety information and data systems with industry, foreign governments, and other agencies involved with the collection, analysis and feedback of operational data.

GENERIC REQUIREMENTS AND OPERATIONAL ANALYSES

Committee to Review Generic Requirements

All generic requirements proposed by the NRC staff involving one or more classes of reactors, including "backfit" requirements, must be reviewed by the Committee to Review Generic Requirements (CRGR). The committee, made up of senior NRC managers from various offices of the agency, advises the EDO as to whether, in the judgment of the committee, proposed new generic requirements have merit in terms of safety and are justified in terms of cost. The CRGR membership, as of the close of the report period, was:

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

James H. Sniezek, Deputy Director, Office of Nuclear Reactor Regulation.

Denwood F. Ross, Deputy Director, Office of Nuclear Regulatory Research.

Jack R. Goldberg, Deputy Assistant General Counsel for Enforcement, Office of the General Counsel.

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

Luis A. Reyes, Director, Division of Reactor Projects, Region II.

The CRGR seeks to eliminate unnecessary demands on licensee and NRC resources. In its reviews, it 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, and/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 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 is justified. From its inception, in November 1981, through September 1989, the CRGR has held 170 meetings and considered a total of 305 separate issues. In fiscal year 1989, the CRGR held 23 meetings and considered 49 issues, including 25 generic backfits in the form of four Rules, six Regulatory Guides, nine Generic Letters, and six Bulletins. A list of those issues follows:

Proposed Rule Amendment on Containment Leak Testing
Proposed Generic Letter on the Safety Parameter Display System
Proposed Generic Letter on Motor-Operated Valves
Proposed Standard Review Plans on Fission Product Removal/Retention in Reactor Containments
Proposed Rule on the On-Site Dry Cask Storage of Spent Fuel
Information Briefing on Relaxation of Surveillance Requirements for Reactor Protection Systems in B&W Reactors
Proposed Rule Amendment on Nation Security Emergencies
Proposed NRC Bulletin on Non-Conforming Molded-Case Circuit Breakers
Proposed NRC Bulletin on Thermal Stresses in Pressurizer Piping
Proposed Advance Notice of Proposed Rulemaking on Acceptance of Products Purchased for Use in Nuclear Power Plants
Proposed Generic Letter on Severe Accident-Related Improvements in Mark I Containments
Proposed Certificate of Compliance for Spent Fuel Dry Storage Casks and Two Associated Proposed Regulatory Guides
Proposed NRC Bulletin Supplement on Power Oscillations in Boiling Water Reactors
Proposed Final Rule (Revised) on Fitness-for-Duty
Proposed Generic Letter on Administrative Improvements to Nuclear Power Plant Technical Specifications
Proposed Final Rule on Standardization and Licensing Reform
Proposed Generic Letter on Contingency Plans for Land Vehicle Bomb Threat
Information Briefing on Implementation of the Property Insurance Rule
Information Briefing on Core Operating Limit Methodology for Westinghouse Pressurized Water Reactors
Proposed Generic Letter on Non-Conforming (Including Fraudulent) Products and Dedication of Commercial-Grade Items
Proposed Generic Letter on In-Service Testing
Information Briefing on Cumulative Effects of Multiple Relaxations of Surveillance Requirements in Nuclear Power Plant Technical Specifications
Proposed Generic Letter on ASME Section III Component Replacements
Proposed Generic Letter on Occupational Exposures from Hot Particles
Proposed Regulatory Guides (Three) on Decommissioning Nuclear Power Reactors
Proposed Generic Letter on Pipe Wall Thinning
Proposed Rule and Associated Implementing Regulatory Guide on Maintenance of Nuclear Power Plants
Proposed NRC Bulletin on Failure of Mechanical Steam Generator Tube Plugs
Proposed Generic Letter on Actions To Be Taken When Equipment is Potentially Non-Conforming
Proposed Generic Letter on Service Water Systems
Proposed Generic Letter on Emergency Response Data Systems
Proposed Final Resolution for Unresolved Safety Issue A-17, "Systems Interactions in Nuclear Power Plants"
Proposed Final Resolution for Generic Issue 128, "Electrical Power Reliability"
Proposed Final Resolution of Unresolved Safety Issue A-47, "Safety Implications of Control Systems in LWR Nuclear Power Plants"
Proposed Generic Letter on NUREG-1335, "Individual Plant Examination: Submittal Guidance"

Proposed Amendments to Twenty-one Standard Review Plans to Reflect Station Blackout Rule Requirements
Proposed NRC Bulletin on Anchor-Darling Check Valves
Proposed Rule Amendments on Reporting of Defects and Non-Compliance
Discussion of Compliance and Adequate Protection Exceptions in the Backfit Rule
Proposed Rule Amendment on Calculating Radiation Embrittlement Levels in Reactor Vessel Beltlines
Proposed NRC Bulletin Supplement on Non-Conforming Molded-Case Circuit Breakers
Proposed NRC Bulletin on Hazardous Gases
Proposed Final Resolution for Unresolved Safety Issue A-40, "Seismic Design"
Proposed Final Resolution (Combined) for Generic Safety Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Safety Issue 94, "Additional Low-Temperature Overpressure Protection for Light Water Reactors"
Proposed Rule Amendment on In-Service Inspection of Metallic Containment Liners
Proposed Regulatory Guide on Criteria for Electrical System Isolation Devices
Information Briefing on Development of Guidance for Treatment of External Events in Individual Plant Examinations
Proposed Final Policy Statement on Maintenance of Nuclear Power Plants
Proposed Final Rule on Access Authorization
Proposed Generic Letter on Implementation of Fitness-for-Duty Rule Requirements

The committee also visits operating power reactors for discussions with the licensee's engineering, 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 1989, the committee visited the Fort Calhoun (Neb.) nuclear power plant, operated by Omaha Public Power District. (See the *1983 NRC Annual Report*, pp. 1-3, for background on CRGR's structure and review process.)

Analyses Of Operational Experience

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

The primary source of operational event data is the LER. In the early 1980s, a major effort was devoted to preparing the rule (10 CFR 50.73) governing content and submission of LERs. The rule clarified reporting

On the plane between Moscow and Zaporozhe in the Soviet Union are U.S. Delegation Head Jack Heltemes, Deputy Director AEOD, second from left, flanked by a translator, on the left, and Gleb Lunin, Co-leader of the Soviet Working Group. Over a weeks long period in the fall of 1989, the NRC team toured Soviet nuclear installations while a U.S.S.R. team was received at the Catawba plant in South Carolina.

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 trending. Since the implementation of the rule in 1984, the events that met the threshold test have provided a consistent basis for assessing the performance trends of the industry as a whole and those of individual licensees.

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

Foreign. AEOD also examines foreign event data in comparing and studying reactor operational experience. Reports of operational events received from the Organization of Economic Cooperation and Development's Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA)—as well as events reported through bilateral exchange programs with over 20 countries—supplement U.S. experience. The NRC continues to exchange operational data with other countries and submitted 59 reports of U.S. operating events to NEA during fiscal 1989. Several hundred foreign event reports were reviewed during this period.

Engineering Analyses of Operational Experience

In 1989, AEOD issued a number of special study reports, engineering evaluations and technical review reports, listed in Table 1. Significant attention was given to Commission-directed initiatives to develop additional performance indicators, in particular to monitor maintenance activities. Several special reports on this subject are highlighted below.

Preliminary Results of the Trial Program for Maintenance Performance Indicators (S804A). This report presents the preliminary results of AEOD's trial program for the development of maintenance indicators. The trial program used actual operational data from 13 commercial nuclear plants to assess the appropriateness of selected indicators for determining maintenance effectiveness. The report focused on three areas. The first concerns current industry practice in the use of maintenance performance indicators, based upon the operational maintenance programs for the plants that were visited in the trial program. The second phase consists of an analysis of results from the validation program for the candidate indicators. The third area addresses the capability of an existing industry component reporting system, the NPRDS (see above), to provide a data source for maintenance indicators. The preliminary results of this program indicate that the attributes of the NPRDS make it the prime data source for indicator development and implementation.

The following conclusions were drawn from the preliminary results of the trial program.

- Process indicators have merit for plant-specific monitoring and control. Plant management should continue to improve their plant-specific process indicators to support the establishment of meaningful quantitative goals toward which management can strive. However, they do not appear to provide the desired level of consistency or correlation to warrant industry-wide monitoring by the NRC.
- Indicators that are based upon actual component reliability and failure history provide the best measure of maintenance effectiveness. Such indicators need a well structured and component-oriented system to capture and track equipment history data. Reliance on an established industry-wide system; e.g., NPRDS, appears to be the only feasible near term solution to obtain needed component data for such indicators.
- The use of NPRDS to provide a data base for constructing and validating maintenance effectiveness indicators generated reasonable and encouraging results. While no specific indicators were fully validated across a number of plants, the extent of the correlations show merit for indicator use.
- The NPRDS can be used for maintenance performance monitoring; however, the need for improved timeliness and completeness in reporting and improvements in scope should be assessed.

Application of the NPRDS for Maintenance Effectiveness Monitoring (S804B). This report explores (1) the usefulness of NPRDS as a source of information for maintenance effectiveness monitoring, (2) the development of a specific indicator that is based upon the component failure reports submitted to the NPRDS, and (3) correlation of the indicator with maintenance effectiveness.

Major components in systems that have historically been significant contributors to forced outages were

Table 1. AEOD Reports Issued During FY 1989

SPECIAL STUDIES

Designation	*Subject*	*Issued*
S804A	Preliminary Results of the Trial Program for Maintenance Performance Indicators	12/88
S804B	Application of the NPRDS for Maintenance Effectiveness Monitoring	1/89
S901	Maintenance Problems at Nuclear Power Plants	2/89
	Office for Analysis and Evaluation of Operational Data 1988 Annual Report - Power Reactors NUREG-1272, Vol. 3, No. 1	6/89
	Office for Analysis and Evaluation of Operational Data 1988 Annual Report - Nonreactors NUREG-1272, Vol. 3, No.2	6/89

ENGINEERING EVALUATIONS

Designation	*Subject*	*Issued*
E806	Loss of Decay Heat Removal Due to Rapid Refueling Cavity Pumpdown	10/88
E807	Pump Damage Due to Low Flow Cavitation	10/88
E808	Operational Experience Review of Potential Large Openings in Containment	12/88
E901	Problem with Oils, Greases, Solvents and Other Chemical Substances	2/89
E902	Fires and Explosive Mixtures Resulted From Introduction of Hydrogen Into Plant Air Systems	3/89
E904	On Demand Malfunctions of HPCI and RCIC	4/89
E905	Electrical Bus Bar Failures	4/89
E906	Failure of Steam Generator Isolation Check Valves	8/89
E907	Diversion of Seal Cooler Flow for RHRPumps	9/89
E908	Excessive Valve Body Erosion at Brunswick	10/89

Table 1. AEOD Reports Issued During FY 1989
(continued)

TECHNICAL REVIEWS

Designation	*Subject*	*Issued*
T811	Degradation of Ice Condenser Containment Functional Capability	11/88
T901	Millstone Unit 1 Safety/Relief Valve Discharge Line, Vacuum Breakers Failed Open	1/89
T902	Inadvertent Reactor Trips due to RCS Flow Instrumentation Maintenance Activities	2/89
T903	Generic Implication of Browns Ferry Fire on November 2, 1987	3/89
T904	Design Deficiency of Safety Injection Block Switch	4/89
T905	Failure of 416V GE Magneblast Breaker at Fitzpatrick	4/89
T906	Broken Lifting Beam Bolts in HPCI Terry Turbine	4/89
T907	Component Degradation due to Indiscriminate Painting	4/89
T909	Operating Events Involving Dampers	5/89
T910	Investigation of Cracked Control Rod Drive Seal Housings at Palisades	6/89
T911	Evaluation of Individually Reported Safety System LERs for their Combined Significance	6/89
T912	Selected Maintenance Rework	6/89
T913	Comparison of the Proposed Maintenance Effectiveness (ME) Indicator with Catawba and Farley Nuclear Plants Regarding Inspections	7/89
T914	Overview of Design/Installation Fabrication Errors in 1988	9/89
T915	EDG Ground Fault Detection and Trip Circuit at Perry Unit 1	9/89
T916	Debris in Containment Recirculation Sumps	9/89
T918	Check Valve Failure Rates From NPRDS Data	9/89
T919	Failure of Overcurrent Protective Device at Palisades Unit 1	9/89
T920	416OV Bus Transfer Scheme at Millstone 3	10/89
T921	Proposed Use of 416OV Switchgear Beyond Their Rating at Fitzpatrick	10/89

considered. Using these data, AEOD then constructed a system-based indicator of maintenance effectiveness that monitors the increases in the failure rate within a system and provides an indication when an increase exceeds a specified value. To obtain a measure of a plant's level of maintenance effectiveness, the number of such indications is established for a number of systems and trends identified. This gives only one measure of the effectiveness of a plant's maintenance program. Other information—such as other indicators, systems reviews and operating experiences, and inspections—are needed to obtain a more complete picture of the level of the effectiveness of maintenance at any plant.

The staff based its validation of the candidate indicator, as reflective of maintenance effectiveness, upon both deterministic engineering analyses and empirical methods. For example, engineering studies of NPRDS failure records for suspect components reveal that differences in maintenance practices among plants can result in differences in failure rates. Further, root-cause analyses of the failures that constitute the indicator reveal maintenance ineffectiveness as the major cause of the failures. Empirically, the analysis shows that the indicator correlates reasonably well with other maintenance-related information derived from LERs.

Maintenance Problems at Nuclear Power Plants (S901). The objective of this study was to obtain a measure of operating nuclear plant maintenance deficiencies by reviewing operational experiences reported to the NRC over the four-year period, 1985-1989. The review focused on existing studies and reports by AEOD, generic communication documents issued by the NRC, feedback documents issued by INPO, and LERs issued by the licensees of domestic plants. The review sought to catalogue problems and to identify discernible trends in the area of maintenance. The study also addressed estimates of the cost of identified maintenance deficiencies to the commercial nuclear industry.

This review concluded that maintenance-related problems found in systems—such as the service water system and the instrument air system—and in components—such as motor-operated valves, inverters, circuit breakers, and pumps—are widespread in the industry. Problems reflect deficiencies in quality control, procedures, planning, communication and training. For example, of the 70 case studies, special studies, and engineering evaluations issued by AEOD since 1985, about 20 percent specifically addressed maintenance-related problems. A review of the 80 NRC Bulletins and Information Notices issued since 1985 also disclosed that about 20 percent dealt specifically with maintenance-related problems. The review of these documents provided conclusions very similar to those from the study reports. Similar findings were reached in the review of generic feedback documents issued by INPO.

To identify trends in reported maintenance deficiencies, data from LERs issued since 1985 were reviewed. This effort identified a gradual improvement in maintenance-related problems in the industry. Decreasing trends in forced outage rates, equipment forced outages, and reactor trip rates, all pointed to a gradual improvement in overall nuclear plant maintenance.

Analyses Of Non-Reactor Operational Experience

In addition to the screening and analysis of reactor operating experience, AEOD reviews the non-reactor operational experience associated with the activities and facilities licensed by the NRC and by the Agreement States. AEOD also conducts studies from a human factors perspective of non-reactor and medical misadministrations data files.

During fiscal year 1989, AEOD issued two survey reports which contain a review of 1988 non-reactor and misadministration reports. These reports were published in the *1988 AEOD Annual Report*, NUREG-1272, Vol. 3, No. 2. The staff also issued one engineering evaluation and one technical review. The non-reactor reports issued in fiscal year 1989, are listed in Table 2.

Medical Examination Report, Medical Misadministrations Reported to NRC for the Period January 1988 Through December 1988. A total of 12 therapy and 393 diagnostic misadministrations were reported to NRC in 1988. Five of the therapy misadministrations involved teletherapy, five involved brachytherapy and two involved radiopharmaceuticals. Of the 393 diagnostic misadministrations, seven involved the misadministration of iodine to patients. None of these iodine misadministrations resulted in a thyroid dose that was near a therapy equivalent dose. The report indicates that:

(1) Both the teletherapy and brachytherapy misadministrations reported in 1988 might have been prevented by quality assurance procedures demanding verification of dose calculations, the type of treatment, and patient identification.

(2) Essentially all of the diagnostic misadministrations involved either the administration of the wrong radiopharmaceutical or the administration of a radiopharmaceutical to the wrong patient.

Table 2. Non-Reactor Reports Issued During FY 1989

Designation	*Subject*	*Issued*
N901	Use of Radioactive Iodine for Infrequent Medical Studies and Those Performed Under an FDA Investigational Exemption for a New Drug (IND)	6/89
T908	Review of Therapy Misadministrations That Involved Multiple Patients and the Use of Computer Treatment Planning Programs	5/89
	Report on 1988 Nonreactor Events, NUREG-1272, Vol. 3, No. 2, Appendix A	6/89
	Medical Misadministration Report, Medical Misadministrations Reported to NRC from January 1988 through December 1988, NUREG-1272, Vol. 3, No. 2, Appendix B	6/89

The number, type and cause of the diagnostic misadministrations are about the same as reported in previous years. The primary cause of misadministrations involving the administration of millicurie amounts of iodine to patients was the failure of licensees to exercise adequate control.

Report on 1988 Non-reactor Events. The survey of 1988 non-reactor events shows that, as in previous years, most 1988 events concerned incidents of modest overexposure, lost or abandoned sources, or leaking sources.

One report of a leaking polonium-210 source manufactured by Minnesota Mining and Manufacturing Company (3M) led to a recall of substantially all of the static elimination devices manufactured by 3M. The investigation of the leaking static eliminator device showed that there was substantial under-reporting of leaks and losses of that device.

In addition, a capsule of cesium-137 leased from the Department of Energy (DOE) and used in an irradiator in Georgia was found to be leaking. As a result of this event, the NRC ordered Radiation Sterilizers, Inc., an NRC licensee, to suspend operations and to place similar kinds of capsules in storage.

None of the events reported to the NRC in 1988 had a significant impact on public health and safety.

Use of Radioactive Iodine for Infrequent Medical Studies and Those Performed Under an FDA Investigational Exemption for a New Drug (IND) (N901). AEOD reviewed five misadministrations that involved the use of iodine for infrequently performed studies or "new drug" studies. In all of the events, an organ other than the one being treated received a relatively high dose, or too much iodine was administered.

It was concluded that, because of the infrequent or developmental nature of the studies, that there is probably less management oversight than for the more common studies. Also, as is usual with these types of studies, the personnel did not have the opportunity to familiarize themselves with the appropriate quality assurance steps.

As a result of the evaluation, the Office of Nuclear Material Safety and Safeguards is planning to revise the NRC's inspection procedures to incorporate a routine review of licensee personnel training requirements for those performing these infrequent studies and the implementation of the FDA requirements for radiation safety quality assurance for a new drug.

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 Services, 5285 Port Royal Road, Springfield, Va. 22161. Copies are also available for public inspection and/or copying at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C., or at Local Public Document Rooms throughout the country (see Appendix 3 for list).)

There were four abnormal occurrence (AO) reports issued in fiscal year 1989— NUREG-0090, Vol. 11, No. 2 (April-June 1988); Vol. 11, No. 3 (July-September 1988); Vol. 11, No. 4 (October-December 1988); and Vol. 12, No. 1 (January-March 1989). The four reports cover two AOs at nuclear power plants, eight AOs at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and five AOs reported by the Agreement States. The reports also update the status of certain AOs previously reported.

A list of the AOs reported in the reports cited above is given in Table 3, and each one is described below. One of the events (AO 88-10, in the list) resulted in an escalated enforcement action, including civil penalty by the NRC (See Appendix 6 for a list of all civil penalties imposed by the Office of Enforcement during the report period, with capsule descriptions of the reasons therefor.)

Nuclear Power Reactors

Plug Failure Resulting in Steam Generator Tube Leak at North Anna Unit 1. On February 25, 1989, North Anna Unit 1 (Va.) automatically shut down, from 76 percent power, when a main feedwater regulatory valve closed because of a problem in the instrument air supply line. During recovery operations, the licensee identified a primary-to-secondary leak of about 74 gallons-per-minute in the "C" steam generator (S/G) and declared a Station Alert. The licensee continued cooldown and depressurization operations, and the Station Alert was terminated when the plant entered cold shutdown. At the time of the event, North Anna Unit 2 was in a refueling shutdown.

Investigation showed that the leak was caused by hot leg mechanical plug failure in a S/G tube which had been plugged by Westinghouse in 1985. The top portion of the plug was severed from the body of the plug, was propelled up the inner diameter of the tube by the primary system pressure, and punctured the tube just above the U-bend transition. The puncture was approximately 2a,a-inches long and 3/4-inch wide. The plug also dented an adjacent tube. Failure of the plug was attributed to stress corrosion cracking of heat-treated plug material (Inconel 600), the result of low anneal temperatures which rendered the material highly susceptible to stress corrosion cracking. The primary-to-secondary leak was well within the normal primary system makeup capability; in addition, the radiological releases were well below the technical specification limits. However, the event identified major safety concerns because: (1) this was an unexpected failure mechanism for S/G tubes; (2) it was a potential common mode failure mechanism with the possibility of multiple S/G tubes failing; and (3) there were generic implications for other plants using such susceptible tube plugs.

The licensee's recovery plan was to investigate, with Westinghouse, the cause of the failure, determine corrective actions, and place Unit 1 into a refueling/maintenance outage so that S/G repairs could be done concurrently. (The susceptible plugs were identified in the S/Gs for both Unit 1 and Unit 2.) Repairs consisted either of removing susceptible plugs and replacing them, or inserting a different type of plug into susceptible plugs. Repairs were completed for the S/Gs of both Units, and the NRC agreed that the plants could be restarted. Unit 2 returned to power operation by the end of April 1989. Unit 1 restarted in July 1989. The NRC was investigating the potential generic implications of heat treated mechanical plugs used by Westinghouse, Combustion Engineering, and Babcock & Wilcox designed plants, at the close of the report period.

Steam Generator Tube Rupture at McGuire Unit 1. On March 7, 1989, McGuire Unit 1 (N.C.) underwent a S/G tube rupture while operating at 100-percent power. The licensee declared a Station Alert, manually shut the reactor down, and initiated plant cooldown. The Station Alert was terminated when the plant entered "cold shutdown."

In February 1989, the reactor at North Anna Unit 1 automatically shut down at 76 percent power. The problem was traced to a leak caused by a plug failure in a steam generator tube. The plant is visible just around the bend to the right on the North Anna River in north central Virginia, between Richmond and Fredericksburg.

The maximum tube leak rate was estimated to be between 540 and 600 gpm. This leak rate considerably exceeds the normal primary system makeup capability, i.e., with the centrifugal charging pumps (CCPs) operating as part of the chemical and volume control system. To keep up with the leak, the operators switched the pumps to their safety injection lineup. (Steam generator tube rupture is one of the design basis accidents considered in theNRC safety review of nuclear power plants. Significant S/G tube ruptures, where the leak rate considerably exceeds the normal primary system makeup capability, occasionally occur, as, for example, events at Ginna (N.Y.) and North Anna Unit 1.)

The NRC sent an Augmented Inspection Team (AIT) to investigate the event. The team concluded that the operating crew performed competently, but weaknesses in both normal and emergency operating procedures were uncovered. The tube failure did not result in a radiological release to the environment in excess of regulatory limits, and the event did not exceed a technical specification (TS) safety limit. The whole body and thyroid doses from this event were a small fraction of the plant TS limits.

Investigation by the licensee established that the rupture was about 3-3/4-inches long, and was caused by intergranular stress corrosion cracking. The licensee's recovery actions included inspection of all tubes in all four S/Gs, metallurgical analysis of the ruptured tube, removal or plugging of tubes as necessary, and revision of the procedures which the AIT found deficient. The licensee committed itself to conducting a 100-percent inspection of the S/G tubes of all S/Gs at both McGuire Units 1 and 2 at their next refueling outages. The NRC staff concurred with the corrective actions taken, and the commitments made by the licensee. Permission to restart Unit 1 was given on May 5, 1989. The plant attained criticality on May 9, 1989, and reached full power operation on May 13, 1989.

Other NRC Licensees

Breakdown in Procedural Controls at Medical Facility. On March 24, 1988, the NRC conducted a special unannounced inspection at Riverton Memorial Hospital Health Trust, Inc., in Riverton, Wyo. The inspection was made to assess the licensee's corrective action following identification of nine violations in an inspection performed on September 30 and October 1, 1986. The violations resulted in a $2,500 civil penalty. The March 24, 1988 inspection identified eight violations, four of them the same as violations from the previous inspection and two related to previous findings. The causes were attributed to significant deficiencies in management oversight and control of the licensed program. On June 3, 1988, the NRC issued a proposed civil penalty of $5,000. The licensee took corrective action, including hiring a consultant to audit the hospital's radiation protection program for one year at quarterly intervals.

Diagnostic Overdose at VA Medical Center. On June 9, 1988, a patient at the Veterans Administration Medical Center, Los Angeles, Cal., was administered a dose of 15 millicuries of technetium-99m

Table 3. Abnormal Occurrence Reports Issued During During FY 1989

OCCURRENCES AT NUCLEAR POWER PLANTS

AO number	*Subject*	*NUREG-0900 Issue*
89-1	Plug Failure Resulting in Steam Generator Tube Leak at North Anna Unit 1	Vol. 12, No. 1 August 1989
89-2	Steam Generator Tube Rupture at McGuire Unit 1	Vol. 12, No. 1 August 1989

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

AO number	*Subject*	*NUREG-0900 Issue*
88-10	Significant Breakdown in Management and Procedural Controls at a Medical Facility	Vol. 11, No. 2 December 1988
88-11	Medical Diagnostic Misadministration	Vol. 11, No. 2 December 1988
88-12	Multiple Medical Therapy Misadministrations	Vol. 11, No. 3 January 1989
88-13	Medical Diagnostic Misadministration	Vol. 11, No. 3 January 1989
88-14	Medical Therapy Misadministration	Vol. 11, No. 4 April 1989
89-3	Medical Therapy Misadministration	Vol. 12, No. 1 August 1989
89-4	Medical Therapy Misadministration	Vol. 12, No. 1 August 1989
89-5	Medical Diagnostic Misadministration	Vol. 12, No. 1 August 1989

OCCURRENCES AT AGREEMENT STATE LICENSEES

AO number	*Subject*	*NUREG-0900 Issue*
AS88-2	Radioactive Material Released during a Transportation Accident	Vol. 11, No. 2 December 1988
AS88-3	Medical Diagnostic Misadministration	Vol. 11, No.3 January 1989
AS88-4	Multiple Medical Therapy Misadministrations	Vol.11, No. 4 April 1989
AS88-5	Medical Therapy Misadministration	Vol. 11, No. 4 April 1989
AS88-6	Multiple Medical Therapy Misadministrations	Vol. 11, No. 4 April 1989

diethylenatriamine-pentaacetic acid (DTPA) in a diagnostic procedure; the amount of DTPA exceeded the prescribed dose by a factor of 1,000. The licensee stated that no untoward effects on the patient were anticipated. The cause of the overdose was the failure of both a technician and a resident physician to follow protocols for radiopharmaceutical injections. As corrective action, the Chief of Service conducted a review and explanation of injection procedures for all nuclear medicine staff members.

Breast Irradiations at Less Than Prescribed Doses. On April 6, 1988, Marquette General Hospital, Marquette, Mich., reported to the NRC that a medical physicist, conducting a quality assurance review of patient treatment records, discovered that the doses given to two patients undergoing breast irradiation in November 1985 and March 1986, were about 85 percent of the prescribed doses. On May 5, 1988, the licensee reported that 19 similar misadministrations had been discovered during 1985, and until October of 1986 (when the procedure was discontinued). No medical damage to the patients was expected. The cause was an error in the manual calculations performed on the output of the treatment-planning computer. The licensee submitted a quality assurance program to prevent recurrence of the event. The program was incorporated into the licensee's license.

Iodine Overdose Deemed Negligible. On June 27, 1988, a patient at the Fairfax Hospital, Falls Church, Va., was administered 2.7 millicuries (thousandths of a curie) of iodine-131 meta-iodobenzylguanidine (MIBG), rather than the intended dose of 500 microcuries (millionths of a curie) of iodine-131 MIBG. The result was an estimated adrenal medullary dose of 268 rads. The thyroid burden should be negligible because the thyroid had been blocked with Lugols, as prescribed in the protocol. The patient exhibited no adverse health effects. The cause was a technologist's error in overlooking the proper dosage listed in the department's procedure manual. The technologist was reprimanded and retrained.

Misreading of Manual Causes Overdose. On November 17, 1988, a patient at Wilkes-Barre General Hospital, Wilkes-Barre, Pa., being treated for an endobronchial tumor, received a dose of 1,800 rads, rather than the prescribed dose of 750 rads, to the right bronchus at a distance of 0.5 centimeters, from an iridium-192 source. The licensee stated that no adverse health effects were anticipated and that the dose to the tumor was within standard treatment protocols for that type of tumor. The cause was human error; the staff radiotherapy physicist used the wrong table of a manual in developing a treatment plan. As corrective action, the licensee established a double independent verification of treatment calculations, provided additional training, and provided an additional chart for determining maximum treatment times for each treatment plan.

Irradiation of Wrong Thigh. On January 23, 1989, a patient at Abbott Northwestern Hospital, Minneapolis, Minn., suffering from a malignant tumor on his right femur (thigh-bone), received a 250-rad radiation dose to his left femur by mistake. The patient was scheduled for 12 treatments of 250 rads each to the right thigh, using a cobalt-60 teletherapy device. The misadministration was discovered after the first treatment. Treatment was subsequently performed correctly and the treatment schedule was continued. The licensee determined that the misadministration could cause increased fatigue and possible bone marrow suppression in the left femur of the patient.

The event involved a number of personnel errors. The simulator technologist, in turning the table on which the patient was lying, apparently disoriented herself and marked the wrong thigh. The therapy physician checked and approved the incorrect marking and treatment. The therapy technologist should have waited until the patient's simulator check list was available in the teletherapy unit before commencing treatment. As corrective action, the licensee provided additional guidance to personnel involved and upgraded its quality assurance/quality control program.

Patient Receives Look-alike's Brain Irradiation. On March 9,1989, a patient at Kennebec Valley Medical Center, Augusta, Me., received a therapeutic treatment which was intended for another patient. A radiotherapy physician had prescribed therapeutic treatments in fractionated doses to two elderly patients from a Veteran's Administration facility. One patient was to be treated for a brain tumor, while the second patient was to be treated for a lesion near the lower palate. Both patients were brought to the hospital at the same time. Because of an identification error, the second (lower palate) patient was brought into the treatment room and the procedure for the brain tumor treatment was begun. When the error was discovered, the procedure was stopped. A total of 100 rads had been delivered to the brain of the patient. In previous treatments, the patient had correctly received 2,400 rads to the lower palate. The licensee stated that no adverse effects are anticipated from the misadministration. The cause was human error on the part of the staff of the radiotherapy department at the medical center. It was noted that the names, physical appearances, and treatment planning pictures of both patients were similar. The licensee's planned corrective action included a strengthening of its patient identification policies, along with second-person confirmation of patient identity and treatment parameters.

Iodine Overdose: Patient Dies of Other Causes. On March 14, 1989, a patient at New England Medical Center Hospital, Boston, Mass., was intended to receive an administration of one millicurie of iodine-123 for a diagnostic scan. This would result in an exposure to the thyroid of about seven rads. However, a staff endocrinologist mistakenly requested an iodine-131 uptake and scan. A floor administrator, transcribing the request to a computer, selected an iodine-131 whole body scan as the intended request. The dosage for this incorrect procedure was prepared and administered to the patient by nuclear medicine department personnel, resulting in the patient's receiving five millicuries of iodine-131. The misadministration resulted in a therapeutic dose to the thyroid of approximately 4,000 to 5,000 rads, with a possible range between 1,200 and 9,000 rads. This dosage could affect the function of the thyroid. The licensee stated that the patient, a cardiac patient under the care of an endocrinologist, might later have been administered a similar dosage of iodine-131 for thyroid ablation as treatment for his cardiac condition. However, the incident should not have occurred with proper controls in place.

The licensee stated that the misadministration was caused by human error on the part of the staff endocrinologist and lack of training of involved personnel. The root cause was inadequate supervision of activities. An NRC inspection identified a violation. The licensee's corrective action includes a change in the radiopharmaceutical requisition forms to include the patient's name, type of study, and isotope; approval of all iodine-131 use by the Chief Nuclear Medicine Technologist before administration of doses to patients; and additional training to all radiology residents, endocrinologists, and technologists during regularly scheduled quality assurance meetings.

This incident was also reviewed by an NRC medical consultant. Among the consultant's recommendations was a follow-up study of the patient yearly, with thyroid function and imaging studies and palpation, to reduce the risk of thyroid cancer and hypothyroidism. The hospital committed itself to follow this course of action; however, prior to the July 10, 1989 enforcement conference, the patient died from a long-standing cardiac condition.

Agreement State Licensees

Radioactive Material Released in Truck Spill. On January 27, 1988, a Model SPEC 2T radiographic exposure device, which contains a 48-curie iridium-192 radioactive source, fell from the back of a Texas licensee's truck. A vehicle following the truck struck the device and dragged it for a considerable distance. At some point, the source became separated from the device. The licensee, Houston Inspection Laboratories, Inc., found the source along the roadway the following day. The causes of the occurrence were a failure to properly secure the device for transportation and failure to follow procedures. Corrective action included an upgrading of procedures for handling and securing the exposure device. The State agency issued a proposed administration penalty of $10,000 to the licensee.

Mistaken Dose of Over 30,000 Rads to Thyroid. On May 17, 1988, a patient at West Houston Medical Center, Houston, Tex., scheduled for a diagnostic scan of the thyroid, was mistakenly administered 30 millicuries of iodine-131, rather than the prescribed dose of 30 microcuries. The result was an estimated dose to the thyroid of over 30,000 rads, which would be expected to destroy the thyroid's function. The event was attributed to human error: the technologist placing the order for the radiopharmaceutical mistakenly asked for millicuries (thousandths of a curie) rather than the intended microcuries (millionths of a curie). For corrective action, the licensee is rewriting its protocols for each procedure to maintain stricter controls in the ordering and administering of radiopharmaceuticals.

Wrong Computer Data Entered: Nineteen Patients Overdosed. Between January and August 1988, a total of 19 patients at Rochester Hospital, Monroe County, N.Y., received cobalt teletherapy misadministrations. Fourteen received doses exceeding the prescribed amount by more than 10 percent, the largest being a total overdose of 81 percent. In addition, five patients received fractional doses that exceeded the prescribed dose-per-fraction by more than 50 percent. These treatments were terminated before the total error exceeded 10 percent, with the largest fractional overdose being 119 percent. An outside radiation oncologist was brought in to evaluate the possible effects on the patients. The cause of the overdoses was a mistaken alteration in data factors used in computer calculations of treatment dosages. The inadvertent change of data factors was due to lack of supervision, inadequate quality assurance, and an inadequate program to identify and eliminate errors. These deficiencies are to be rectified by the licensee.

Misalignment of Cobalt Unit: Brain Irradiated. Between August 8 and August 26, 1988, an 81-year-old patient at Sacred Heart Hospital, Cumberland, Md., received a total therapeutic dose of 1,400 rads to the wrong part of the body. The patient was scheduled to receive a total dose of 3,000 rads to the right maxillary sinus from two ports of a cobalt-60 teletherapy unit. However, the hospital oncologist misaligned one of the ports, resulting in a dose to the base of the brain of 1,400 rads. The oncologist maintained that the dose to the brain would not result in any medical side effects. Later,treatment to the right maxillary sinus was resumed to deliver the originally prescribed total dose of 3,000 rads. The oncologist affirmed that she would exercise greater vigilance and alertness in the future. The State agency discussed possible procedural changes with the oncologist to help prevent recurrence.

Computer Error Undetected for 13 Months. Between September 1987 and October 1988, 33 terminal patients, receiving palliative brain tumor treatments at Sacred Heart Hospital, Cumberland, Md., were given 75 percent in excess of the prescribed doses. The excessive dose was attributed to the hospital oncologist's use of a computer program file that had not been upgraded when a source change was made in March 1987. During a 13-month period, the therapy staff had noted severe skin erythemas (reddening) in several of the patients and had expressed their concerns to the hospital oncologist. The latter, however, decided that the erythemas were normal. Finally, the staff notified the hospital's consulting physicist who found the computer error and confirmed that there had been misadministrations. During the investigation, the hospital suspended the oncologist, removed her as Radiation Safety Officer, and removed her as Chairman of the Medical Isotopes Committee. Both the licensee and the State agency hired consultants to evaluate the occurrences and their effects on the patients. The State agency was awaiting written reports from the independent consultants at the close of the report period.

PERFORMANCE INDICATOR PROGRAM

During 1989, the Performance Indicator Program of AEOD saw the implementation of new performance indicators (PIs), a significant revision to the format of the report, and continued work on developing new indicators of plant maintenance effectiveness; the program also encompassed efforts by the Office of Research to develop indicators of safety system availability, training, and management effectiveness, and it included issuance of periodic reports on plant performance.

AEOD issued quarterly Performance Indicator Reports for each operating commercial nuclear plant in the United States. The reports provided graphical and tabular data regarding six indicators: (1) automatic scrams while critical; (2) safety system actuations; (3) significant events; (4) safety system failures; (5) forced outage rate; and (6) equipment forced outages-per-1,000 critical hours. The Performance Indicator Report presents a plant comparison of indicator data with an average of all other plants, as well as plant-specific trends in each indicator, over an 18-month period. Supplemental data are also provided on automatic scrams at low power levels and high power scrams-per-1,000 critical hours.

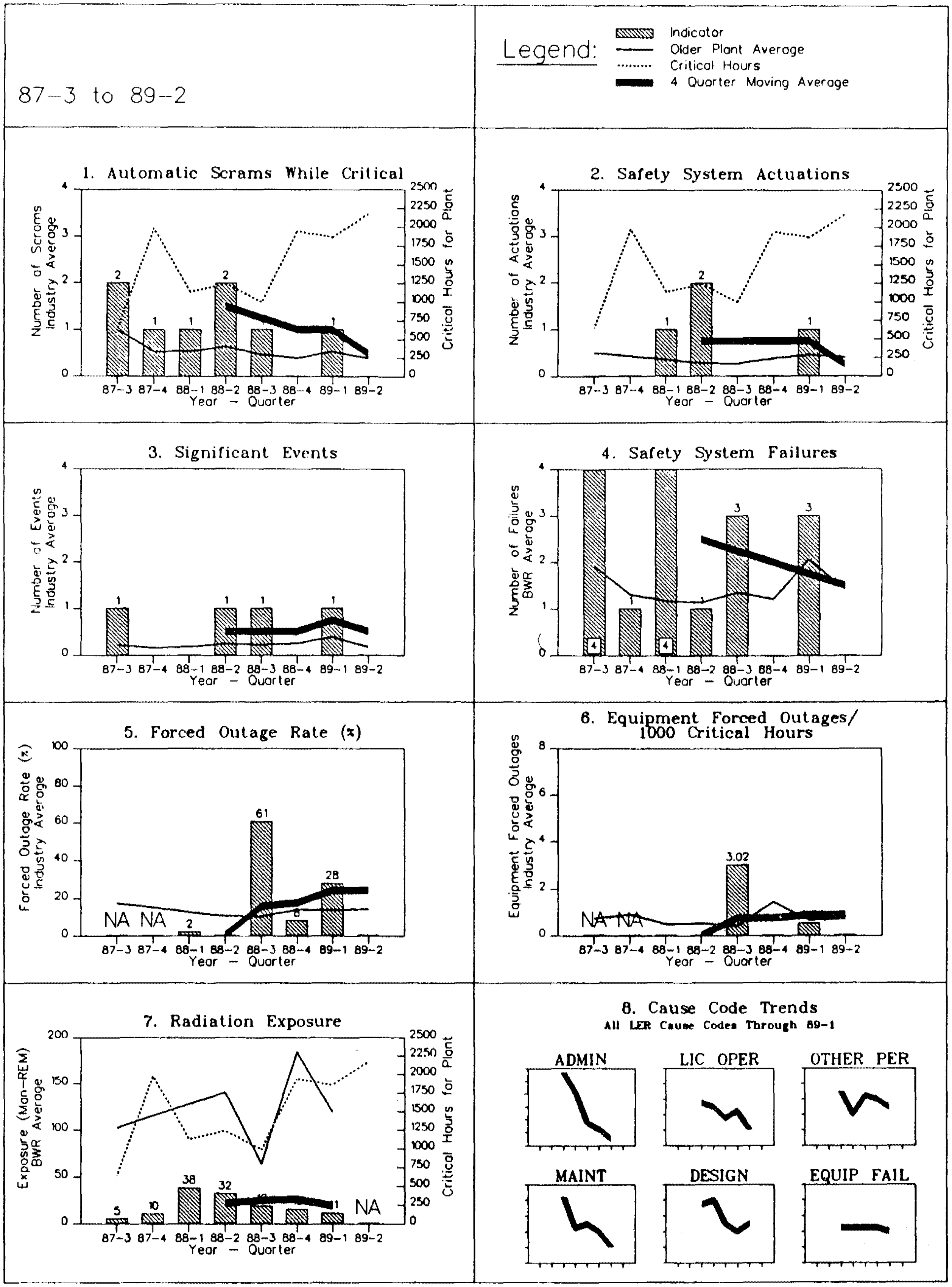
During fiscal year 1989, AEOD completed negotiations with the Institute for Nuclear Power Operations (INPO) to obtain quarterly collective radiation exposure data at each operating site (formerly, the data were provided annually). Beginning with the PI report for the first calendar quarter of 1989, these data were to be added to the Performance Indicator Report as a seventh indicator.

The Performance Indicator Report for the second calendar quarter of 1989 provided a major expansion of the indicators to include operating event "cause code" trends, as well as a reformatting of the report presentation. The cause codes provide a classification of all LERs by programmatic causes, for each operating unit. Quarterly trends are presented in these six cause code areas: (1) administrative control; (2) licensed operator; (3) other personnel; (4) maintenance; (5) design/installation/fabrication; and (6) equipment deficiencies. Supplemental cause code data is also tabulated in four maintenance areas: maintenance personnel; test and surveillance; maintenance equipment; and potential maintenance problems.

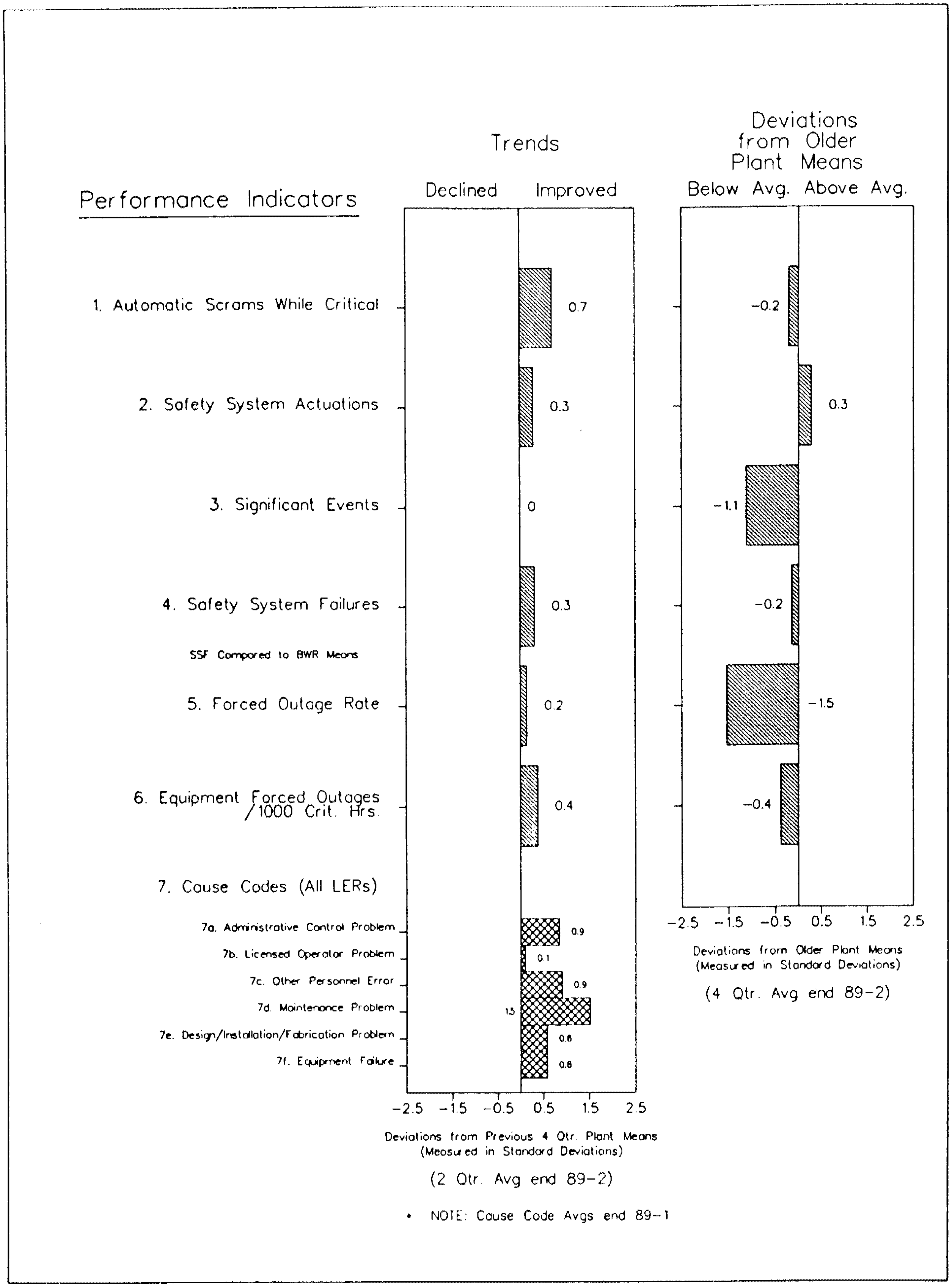
Cause code development represents the product of a multi-year joint effort of the Office of Research and AEOD, assisted by contributions from three national laboratories: Pacific Northwest Laboratory (PNL); Oak Ridge National Laboratory (ORNL); and Idaho National Engineering Laboratory (INEL). During fiscal year 1989, AEOD completed the developmental work, conducted a trial program, and validated the cause code concept. The results were submitted to the Commission in the SECY-89-046 report, "Results from Trial Program for Use of Cause and Corrective Action Data as Programmatic Indicators" (ORNL/NOAC-244).

Incorporation of cause codes into the Performance Indicators Report involved a significant modification of the report format, in order to remain within prescribed size constraints. AEOD developed format options which were presented the NRC senior managers at their May 1989 semiannual meeting. The options selected by the senior managers were then presented to the Commission in SECY-89-280. The Commission-approved format and cause trends were incorporated into the Performance Indicator Report, beginning with the second calendar quarter of 1989.

INCIDENCE OF PERFORMANCE INDICATORS



PERFORMANCE INDICATOR TRENDS



During fiscal year 1989, AEOD continued efforts to develop indicators of maintenance effectiveness. During the period, the staff kept the Commission apprised of such efforts by way of SECY-88-289 (October 7, 1988), and SECY 89-044 (February 6, 1989). The AEOD staff concluded that maintenance performance indicators based upon actual component reliability and failure data could provide a useful measure of maintenance effectiveness. The conclusions were documented in two reports, "Preliminary Results of the Trial Program on Maintenance Performance Indicators" (AEOD/S804A) and "Application of the NPRDS for Maintenance Effectiveness Monitoring" (AEOD/S804B). AEOD has completed an application of component failure data from all commercial nuclear power plants in a two-year period and concluded that the maintenance effectiveness indicator is valid for monitoring maintenance at both PWR and BWR plants and that successful validation has been achieved to support application of this indicator. That study was documented in the report "Maintenance Effectiveness Indicator," NUREG/CR-5442, dated November 1989.

In response to Commission direction, AEOD participated with industry in the framing of a demonstration project, in September 1989. The objective of the project is the sharing and comparing of developmental work on maintenance effectiveness indicators. Taking part in the demonstration project are six utilities with operating plants, the Nuclear Utility Management Resources Committee (NUMARC), the Institute of Nuclear Power Operations (INPO), and AEOD.

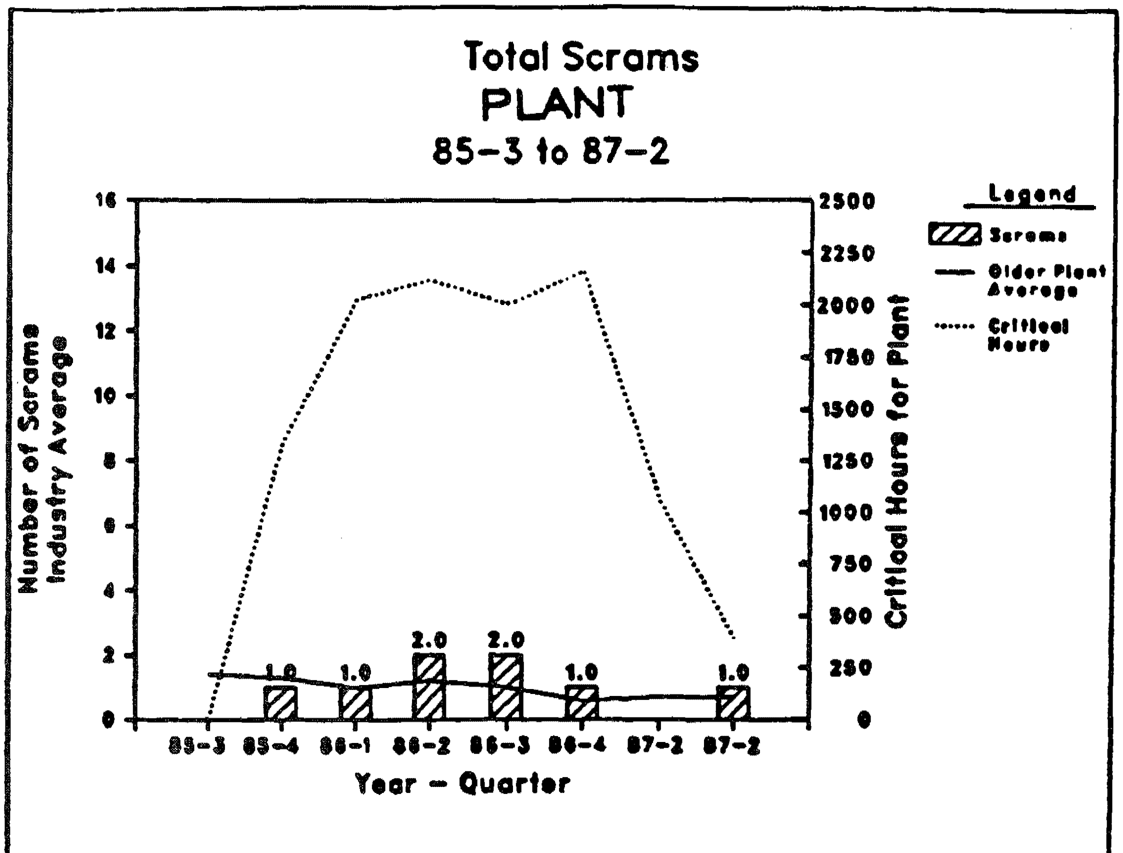
INCIDENT INVESTIGATION PROGRAM

The Incident Investigation Program (IIP) was established by the Executive Director for Operations (EDO) and approved by the Commission to assure that the NRC's investigation of significant events would be timely, thorough, well coordinated, and formally administered. The scope of the IIP covers the investigation of significant operational events involving both reactors and non-reactor activities 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, specifically, to gather all available information pertaining to the causes of the events—including those involving the NRC's activities—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 actions, the IIP process contributes to nuclear safety by providing 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 an 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. Of the approximately 4,000 reportable events which occurred during fiscal year 1989, no event was judged to have a sufficiently high level of safety significance to warrant an IIT investigation. AITs dispatched during fiscal year 1989 are shown in Table 4.

IIT Training Program. The purpose of this program is to provide prospective members of an IIT with comprehensive guidance and methodology in conducting systematic and technically sound investigations. The training program was developed by AEOD following discussion with representatives of the National Transportation Safety Board, Federal Aviation Administration, and National Aeronautics and Space Administration. A third IIT training course was completed in August 1989. A total of 24 NRC staff members and one industry representative from the Institute of Nuclear Power Operations attended the course.

The first week of the intensive training course constituted a detailed presentation and discussion of the procedures and techniques to be used during investigations. During the second week, participants learned and applied the techniques of accident investigation using the Management Oversight and Risk Tree (MORT) methodology. Five "IIT teams," made up of five members each, individually conducted simulated investigations. Each team reviewed known data, obtained information through mock interviews, and conducted a systematic analysis of the causes and implication of the "event." Each team exercise culminated in a "Commission briefing" or "Congressional inquiry," in which the team presented and defended its findings before a panel of NRC senior managers. Also, in a one-day media presentation, participants had an opportunity for mock interviews to prepare for exchanges with the media.


Total Scrams
PLANT
85-3 to 87-2
Number of Scrams
Industry Average
Critical Hours for Plant
Year - Quarter
85-3 85-4 86-1 86-2 86-3 86-4 87-2 87-2
1.0 1.0 2.0 2.0 1.0 1.0
Legend
Scrams
Older Plant Average
Critical Hours

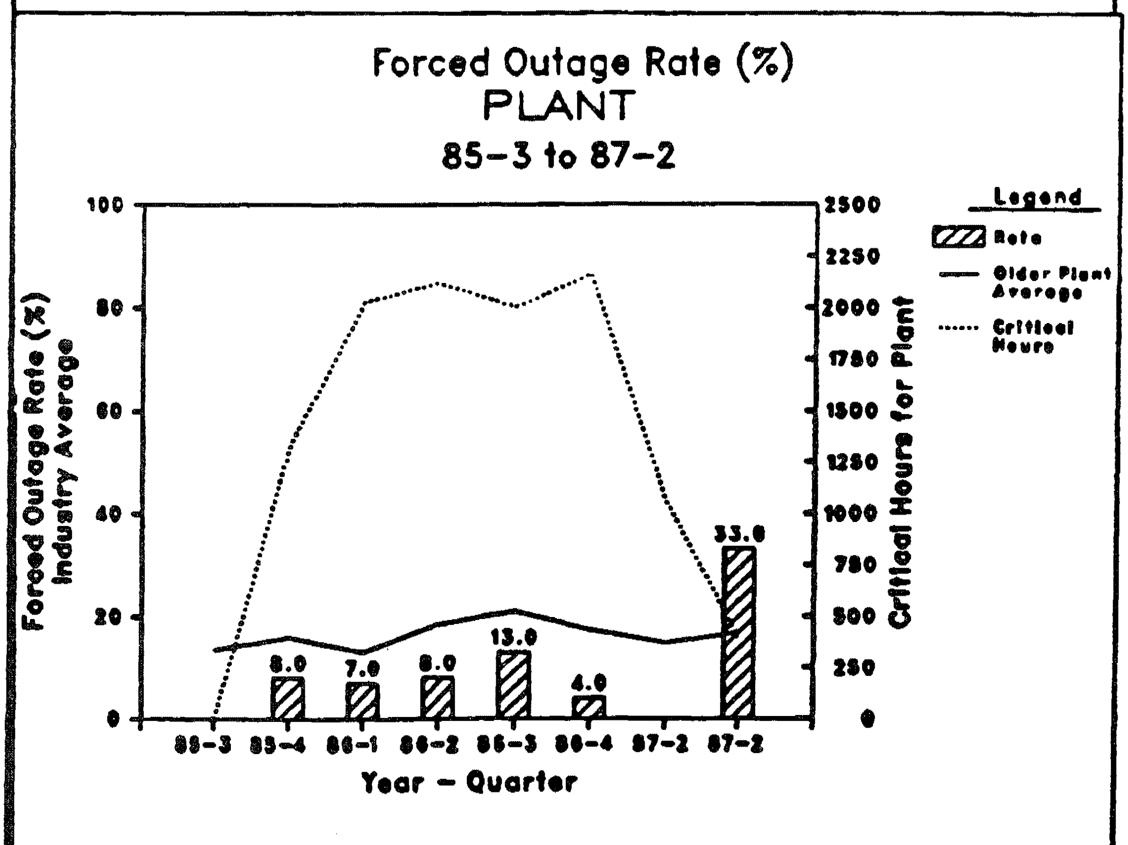

Forced Outage Rate (%)
PLANT
85-3 to 87-2
Forced Outage Rate (%)
Industry Average
Critical Hours for Plant
Year - Quarter
85-3 85-4 86-1 86-2 86-3 86-4 87-2 87-2
8.0 7.0 8.0 13.0 4.0 33.0
Legend
Rate
Older Plant Average
Critical Hours

Table 4. Augmented Inspection Teams (AITs) Dispatched in FY 1989

Event Date	*Plant & Unit*	*Event*	*AIT Criteria*	*Report Date*	*Report*
09/29/88 10/02/88	Oyster Creek	Loss of Shutdown Cooling Events	"A" & "B" Isolation Condensers Declared Out-of-Service	10/31/88	50-219/88-90
01/03/89	Oconee 1	Electrical Fire & Loss of Forced Coolant Flow	Fire Caused Trip of 1A1 & 1B1 Reactor Coolant Pump	02/10/89	50-269/89-03 50-270-89-03 50-287-89-03
01/20/89	Arkansas 1	Coolant Loss Outside of Containment	Check Valve Leakage Causes Coolant Loss Outside Containment	02/17/89	50-313/89-03 50-368/89-03
03/02/89	LaSalle 1 & 2	Unit 2 SAT Fault Causes Trip of Unit 1 Main XFMR/ Reactor Trip with Subsequent Malfunction	Unit 2 Transformer Fault Results in Unit 1 Trip Trip and Equipment Malfunction	03/21/89 50-374/89-007	50-373/89-007
03/03/89	Palo Verde 3	Multiple Equipment Failures Following Load Reject	Leak to Containment Sump Loss of RCPs	04/19/89	50-528/89-13 50-529/89-13 50-530/89-13
03/07/89	McGuire 1	Steam Generator Tube Rupture	Steam Generator Tube Rupture	04/10/89	50-369/89-06
04/12/89	Pilgrim 1	Unexpected RCIC Valve Opening	Similar to a Prior Event Involving the HPCI System	05/08/89	50-293/89-80
04/18/89 04/19/89	Braidwood 1 & 2	Inattentive Licensee Employees	Inattentive Licensee Employees	05/26/89	50-456/89-14 50-457/89-14
04/19/89	River Bend	Freeze Plug Failure In-Service Water System	Shutdown Cooling, Spend Fuel Cooling Were Lost	05/16/89	50-458/89-20
04/23/89	Comanche Peak 1	Hot Water Intrusion Into Auxiliary Feedwater System	Operator Error and Mechanical Failure	07/10/89	50-445/89-30 50-446/89-30
06/22/89	Seabrook 1	SDV to Main Condenser Failed to Close on Demand	Primary System Perturbation	08/17/89	50-443/89-82
08/14/89	Cook 2	Loss of Safety System Redundancy Resulted in Loss of Control Room Instrumentation	Loss of one 120 VAC0 Safety-Related Control Instrumentation Distribution (CRID) Panel	9/15/89	50-315/89-25 50-316/89-25
08/16/89	Robinson 2	Inadequate NPSH Feedwater Pumps	Design Deficiency Possible Loss of Auxiliary Feedwater Pump Due to Pump Cavitation	09/15/89	50-261/89-20
08/21/89	Nine Mile Point 1	Contamination of Sub-Basement by Leaking Drums	Possibility for Release of Radioactive Waste Products to the Environment Worker Safety Could be Jeopardized		
09/05/89	McGuire 2	Reactor Water Spill	RWST Water Into Auxiliary Building		

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 considerations informing 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 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. The evaluation process involves observation of plant activities, 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.

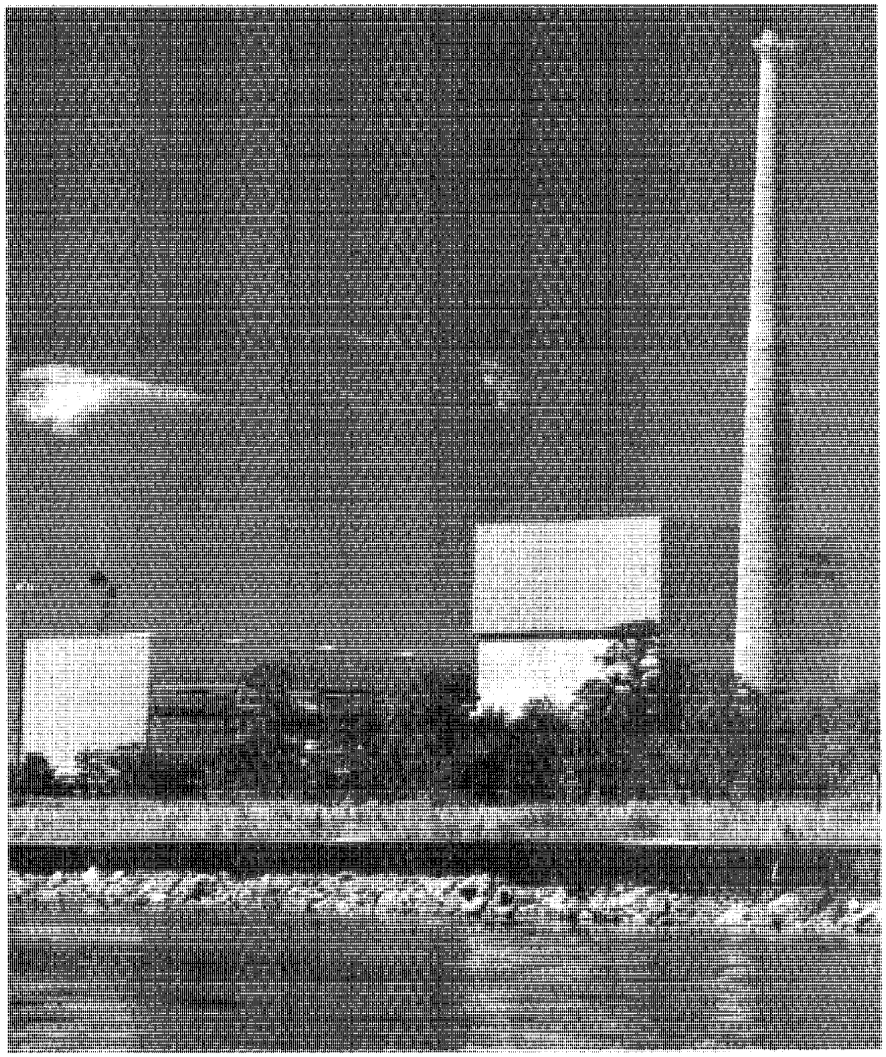
Diagnostic Evaluation of the Enrico Fermi Atomic Power Plant (Fermi 2). In June 1988, NRC senior managers undertook a detailed review of the regulatory and operational history of Fermi Unit 2 (Mich.) and decided that additional information was needed to make a genuinely informed assessment of overall plant performance there, and of the effects of recent corrective actions taken by the Detroit Edison Company (DECo) to improve operational safety at the plant.

A 19-member team subsequently spent a total of three weeks at the Fermi site, during August and September 1988. The team concluded that the root causes of Fermi's continued poor performance and apparent inability to sustain improvements were the protracted design and construction period, the failure of management to adequately and effectively plan for the transition from a design and construction project to an operating plant, lack of BWR operating experience throughout the organization, and management's slowness in determining and implementing effective solutions.

Overall, the team noted recent improvements in Fermi's performance and capabilities, but also identified a number of weaknesses that required greater attention and involvement by DECo management. Initially, DECo management was slow in taking aggressive and effective action to fill key positions with professionals having extensive nuclear plant operations experience, and to implement site-specific policies and programs to improve performance and accountability. However, actions taken in this regard during the last two years represented significant accomplishment and provided the foundation needed to achieve future progress. Further, the team concluded that the actions taken at Fermi generally address the causes of the performance problems, while maintaining an acceptable level of operational safety. Notwithstanding these judgments, the team determined that some areas needed additional management attention to increase the rate of progress and assure continued success. These concerns included the need to achieve organizational stability as soon as possible, improve effectiveness of first and second line supervisors, improve organizational climate, remedy and clarify fragmented and overlapping engineering support responsibilities, fix known equipment problems, set priorities according to plant needs, allocate resources to selected areas and better utilize existing resources, and improve effectiveness of operator training programs.

Diagnostic Evaluation of the Perry Nuclear Power Plant. In December 1988, NRC Senior Management decided to conduct a diagnostic evaluation at the Perry (Ohio) nuclear power plant. The decision was primarily based on uncertainty with the underlying causes and likely future direction in Perry's performance trends. Although performance indicators started to show improving trends, they were still below the industry average. Questions also remained concerning the plant's material condition and the perception that plant management accepted operational problems until violation of technical specification limits was closely approached.

A 16-member team spent a total of three weeks at the Perry site during February and March 1989. The DET concluded that the root causes of Perry's performance problems during the first operating cycle were the pressure to place and maintain Perry in productive operation; the emergence of a significant number of equipment design problems during startup testing andlater in the operating cycle; management's acceptance of a continuation of operations, despite an elevated number of equipment problems, while per-



The first dispatch of an Augmented Inspection Team (AIT) during fiscal year 1989 was in response to "loss of shutdown cooling events" at the Oyster Creek plant, on the Toms River in New Jersey. See Table 4 for a list of all AIT dispatches during FY 1989.

manent engineering resolutions were pursued; delayed management attention to procedural and personnel performance problems, caused by priority attention to technical/hardware problems and extraneous issues; and human relations weaknesses.

The team concluded that Cleveland Electric Illuminating Company management had effectively developed and implemented the management, organizational, and programmatic plans for the transition from construction to operations. A capable management team, a strong technical staff, and relatively comprehensive program documents had been put in place at the start of the first operating cycle. And yet a significant number of equipment problems emerged during the cycle with significant impact on plant performance, management attention, and technical resources. Many of these problems involved original design deficiencies, affecting both safety-related and non-safety-related equipment, which were not corrected in the pre-operational and startup test programs. The consequences included unplanned reactor shutdown, overburdened or distracted operators, and failed safety-related equipment.

The team concluded, however, that overall progress had been made during the cycle by Perry management in addressing the causes of the problems which adversely affected performance, as evidenced by the improving trends in most performance indicators over the last two quarters. Except for the human relations area, plant management was deemed to have effective self-analysis and performance monitoring programs, a fact which enabled the plant staff to develop an early and detailed awareness of technical, programmatic and performance weaknesses. In addition, Perry management had demonstrated the capability to effectively solve difficult problems once they became a management priority. Improvements involving station teamwork, personal accountability, equipment design and maintenance, and increased staff experience with equipment operations, testing, and procedures were also having a positive effect toward the end of the cycle.

Notwithstanding the improvements, the team determined that some areas needed additional management attention to increase the rate of progress and assure continued success. These included the need to expeditiously resolve continuing equipment problems affecting plant and personnel performance; to take more timely action to address human relations weaknesses, in order to ensure long term improvements in personnel performance; and to take prompt action to establish an effective Independent Safety Engineering Group to facilitate improved problem resolution. Because of the licensee's heavy reliance on contractor support, the team was concerned that an excessive reduction in contractors in certain areas, after the first refueling outage, could impair the licensee's ability to accomplish engineering support tasks.

Diagnostic Evaluation at Brunswick Steam Electric Plant. The NRC decided in December 1988 to conduct a Diagnostic Evaluation at the two-unit Brunswick (N.C.) nuclear power plant. The decision was based primarily upon the overall plant performance, which was poor or declining in several areas; those conclusions were based on the most recent Systematic Assessment of Licensee Performance (SALP) report, numerous equipment failures, and repetitive safety system failures that had resulted in the dispatch of two NRC AITs.

An 18-member team spent three weeks at the Brunswick facility, during April and May 1989. The DET concluded that the root causes of poor performance during the past few years had been inadequate corporate management, coincident with a period of past site management weakness; the failure to clearly define and communicate site goals, priorities, and expectations; cultural issues, including the failure of Carolina Power & Light Company (CP&L) management to adequately review and understand Brunswick's declining level of performance, plus a lack of individual accountability and teamwork; an ineffective root cause determination and corrective action program; and an ineffective engineering design and technical support program.

In general, the team concluded that recent management changes and initiatives were having a positive effect. Of particular note was the involvement and presence of senior site managers in the plant. The new management team was judged competent and capable of making the changes necessary to develop a safety culture and improve Brunswick's overall performance.

The team also determined that several areas needed additional management attention. These included implementation of an effective corporate oversight program, both to provide leadership and direction, and to accurately monitor and assess Brunswick performance; definition of site safety goals, priorities, and expectations to be effectively communicated to and understood by personnel at all levels; implementation and monitoring of the effectiveness of measures adopted to establish the desirable working culture at Brunswick; implementation of an effective corrective action program, one with a lower threshold for problem identification and effective steps toward root cause determination; and implementation of an integrated program to correct engineering design and technical support weaknesses involving both equipment failures and support activity weaknesses, such as configuration control and safety evaluations.

Diagnostic Evaluation of Arkansas Nuclear One. In June 1989, the EDO directed that a diagnostic evaluation of Arkansas Nuclear One be conducted. The evaluation was initiated in August 1989 and issuance of the final report was expected after the close of the report period.

TECHNICAL TRAINING PROGRAM

The NRC Technical Training Center (TTC), was established to develop and implement policy and programs for the technical training of the NRC staff. The TTC provides a variety of technical training for resident inspectors, region-based inspectors, operator license examiners, headquarters operations officers, project managers, technical managers, and other NRC technical staff. The TTC is located in Chattanooga, Tenn., but is structurally part of the NRC headquarters organization, in the Office for Analysis and Evaluation of Operational Data (AEOD).

The TTC provides technical training in broad areas of reactor technology and specialized technical training. The reactor technology curriculum comprises a full spectrum of courses, involving both classroom and full-scope reactor simulator training, and covering all of the major U.S. reactor vendor designs (Westinghouse, General Electric (GE), Babcock & Wilcox (B&W), and Combustion Engineering (CE)). The specialized technical training curriculum consists of a number of courses in engineering support, health physics, safeguards, and inspection or examination techniques. This specialized technical training is carried out through two basic processes. One involves making a few slots to regularly scheduled courses available to NRC employees. Courses such as these typically bring in students from a variety of organizations and are, therefore, not dedicated to specific NRC needs. The other process typically involves courses which are attended only by NRC employees or selected contractors. Courses of this type are normally tailored to meet specific NRC needs.

During fiscal year 1989, the TTC conducted or coordinated a total of 115 courses in the reactor technology areas and 85 more in the specialized technical training area. A total of 2,078 students attended TTC courses during the last fiscal year, although a number of students in qualification programs attended multiple courses. These courses represent a total of 249 course-weeks, of which 145 were associated with reactor technology training and 104 were associated with specialized technical training.

The TTC staff also accommodated a number of requests for special courses in reactor technology during the year. Courses were provided for State of Illinois Department of Nuclear Safety personnel in both Westinghouse and GE reactor technologies and for Atomic Safety and Licensing Board Panel (ASLBP) personnel in Westinghouse technology. Two national news media seminars were conducted. These included presentations on BWR and PWR technologies; plant operations and transient demonstrations, on NRC controlled simulators at the TTC; and question and answer sessions on health physics issues.

The phased plan for development of qualification and technical training requirements for headquarters personnel began in fiscal year 1988 and continued through fiscal year 1989. Those phases associated with grouping of positions with similar job tasks, identification of draft training requirements, integration and reconciliation of the individual training needs, and formalization of training requirements have been completed. The phase associated with development of priorities among initiatives started during fiscal year 1988 and will continue through fiscal year 1990.

A full course series was developed and implemented for CE technology. The full series of courses—attended by resident inspectors, operator licensing examiners, headquarters operations officers, and persons in other NRC technical positions—is now available in each of the four light water reactor vendor designs. Modifications were made to the reactor technology and simulator courses specifically developed for large parts of the NRC technical staff. These courses were presented several times in both the Westinghouse and GE technology areas. Extensions of the full course series were offered in all technologies, in support of operator licensing examiner training. The extensions involve additional reactor simulator time, to allow for more hands-on training in normal conditions, and also more exposure to symptom-based emergency operating procedures. A cross training series was developed and implemented. This series allows NRC technical personnel already formally qualified in one PWR technology to qualify in B&W or CE technology in less time than that required for the first qualification. Inspector refresher-course training was altered to emphasize emergency operating procedure simulator training, at the request of senior regional management.

Significant progress was made in the improvement and consolidation of health physics training. Plans were made to open enrollment in certain TTC courses to State personnel and, conversely, to open enrollment in certain State Progams courses to other NRC personnel. Revised curricula for NRC reactor and materials personnel have been defined. New courses associated with the revised curricula are scheduled for development and presentation over a two-year period. New courses associated with these curricula include Teletherapy and Brachytherapy, Health Physics Technology, Occupational Safety and Health Act (OSHA) orientation, Whole Body Counting/Internal Dosimetry, Panoramic Pool-Type Irradiators, Radwaste Management, and Advanced Health Physics.

Besides the efforts specifically geared to examination techniques training and health physics training, a number of other specialized technical training courses were initiated during the year—either through presentation by the TTC staff or through the use of TTC managed training contracts. These courses include Inspecting for Performance, Incident Investigation Team Training, Accident/Incident Workshops, Non-Power Reactor Technology, Cold Chemistry, Emergency Diesel Generator, and Power Plant Engineering. They were furnished through the use of contracts and Site Access Training, and Site Access Refresher Training by the TTC staff.

Long term, cost-effective solutions have been developed for NRC reactor simulator usage for the GE, Westinghouse, and B&W reactor vendor designs. The amount of simulator time required to meet NRC needs is estimated at about 1,500 hours each for the GE and Westinghouse vendor designs, and 600 hours each for the B&W and CE vendor designs. The CE simulator training time during the past year was made available by a contract with CE. The best options for securing a long term solution for CE simulator training were being explored as the fiscal year ended.

During the year, a major plan for upgrading the thermal-hydraulic modeling of the NRC training simulators was developed. These simulators currently have modeling which is typical of early generation simulators with two-phase flow capabilities. But NRC use of simulator time typically involves more extended scenarios, and exercises in emergency operating procedures (EOPs) and some research projects have been constricted by existing simulator modeling. For these reasons, a decision was made to upgrade NRC controlled simulators, within budget, by taking advantage of expertise within the AEOD staff. The upgrading involves simulator computer replacement (to establish a platform capable of supporting better codes), simulator instructor station replacement, addition of input/output override capability to the simulators, high fidelity thermal-hydraulic code procurement and conversion, and high fidelity code addition and integration with other simulator models. This project is scheduled to take place over the next three years.

TTC staff members provided technical consultation in a number of areas throughout the year. And the TTC staff supplied an operations team member for every Diagnostic Evaluation Team and support for a number of special inspections at different facilities. Significant support was also given the RES-sponsored human factors research projects involving team skills and behavior and simulator fidelity.

A significant new initiative over the next couple of years will be the introduction of expanded risk-based perspectives into TTC programs. The goal is to bring a risk-based culture to TTC courses, to complement the existing operationally oriented approach. The initiative is intended to generate an increased awareness of major risk contributors and more emphasis on risk-dominant sequences. The new emphasis, once realized, will manifest itself in instructor lesson plans, course manuals, and routine and special presentations.

INCIDENT RESPONSE

Events Analysis. The NRC maintains a 24-hour-a-day, 365-day-a-year Operations Center in Bethesda, Md. The Operations Center is the NRC's center for direct communications, through dedicated telephone connections, with licensed nuclear power plants and certain fuel cycle facilities, providing the capacity to receive reports of, and to deal with, significant events at these facilities. The center receives about 4,000 notifications each year from its licensees, primarily nuclear power plant operators. During fiscal year 1989, there were 230 incidents—10 alerts and 220 unusual events—reported to the Operations Center under the NRC emergency classification system.

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 simply recording the circumstances of the event for later evaluation to immediately activating response organizations within Headquarters and 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 those of the licensee and off-site organizations, whose immediate responses are defined in their own emergency plans.

Each of the 4,000 events reported each year to the Operations Center by licensees is evaluated to determine whether it bears any generic implications for other nuclear facilities. Event reports are screened for this purpose early during the first working day after receipt. Follow-up of plant-specific events is accomplished by the appropriate Region. Where an event indicates significant systems interaction and raises questions as to plant safety, an Augmented Inspection Team (AIT) or an Incident Investigation Team (IIT) may be formed. Events that may be significant from a generic standpoint receive additional in-depth evaluation and, if 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 appropriate facilities and tools to receive information, assess that information and communicate with other involved parties. During fiscal year 1989, the Operations Center was involved in several actual events which, while not requiring complete activation, necessitated the use of the Operations Center's capabilities. The Operations Center was staffed to monitor steam generator tube leaks at the North Anna Unit 1 (Va.) nuclear power plant and the McGuire Unit 1 (N.C.) nuclear power plant, losses of off-site power at the Crystal River (Fla.) nuclear power plant and the Brunswick (N.C.) nuclear power plant, and Hurricane Hugo's path near nuclear power plants in North and South Carolina. The telecommunications capability of the Operations Center was used by NRC management in teleconference discussions of a number of events that were potentially significant but not enough to warrant staffing of the Operations Center.

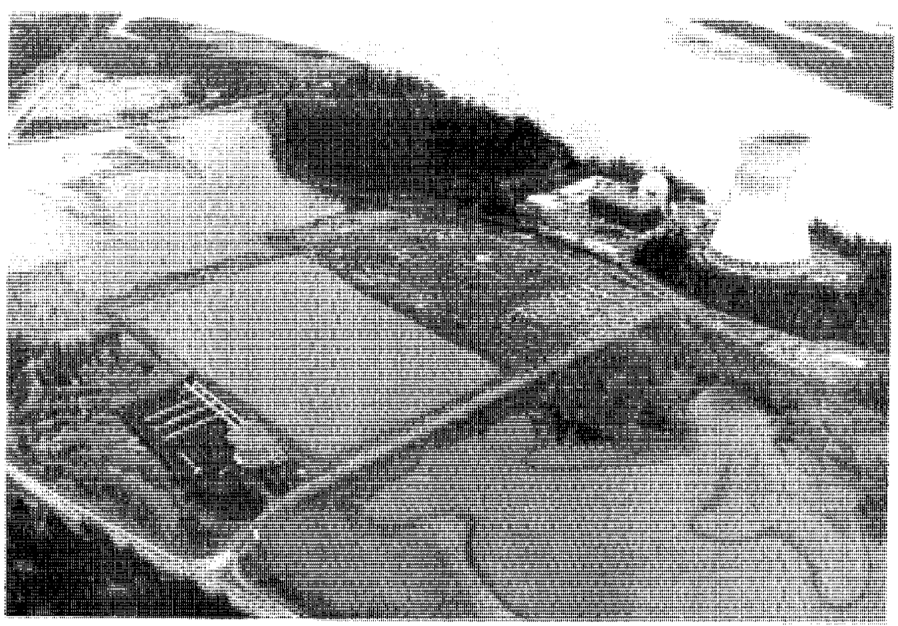
During fiscal year 1989, a number of exercises dealing with various accident scenarios and involving the Operations Center were conducted, in order to confirm and maintain the capabilities of the agency response personnel. Most of the scenarios were concerned with reactor plant incidents. The exercises took place at the Trojan (Ore.) nuclear power plant, the South Texas nuclear power plant, and the Crystal River (Fla.) nuclear power plant; and there were computer-generated reactor accident simulations in Regions II and IV. All of these exercises were supported through the Operations Center. The NRC has begun to plan for a post-emergency "tabletop" exercise to be held at the Riverbend nuclear power plant near Baton Rouge, La., in September 1990, and for the next Federal Field Exercise, planned for 1992. 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 activities within the Operations Center during an exercise or event.

Regional Response Capability. Each Regional Office also maintains its own incident response capability and an incident response center that is designed to support the agency response during a licensee Alert or in 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 normally arrives at the site some two-to-eight hours after being dispatched. Once the site team is fully briefed by licensee management and 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 become the NRC Director of Site Operations (DSO).

Each Region also has its own supplement, including specific implementation details, to the NRC Incident Response Plan. During fiscal year 1989, Headquarters and the Regions developed standardized regional supplements to reinforce the agency-wide response capability. Regional response capabilities are assessed annually, and the Regions participate in several exercises each year, at least one of which includes headquarters participation. In the event of an extended NRC response, the initial site team would be augmented by a number of team members from Headquarters and the other Regions.

Coordination with Other Federal Agencies. The NRC staff participated actively in many Federal emergency response planning and response activities directly involving other Federal agencies, in 1989. These activities included participation in the National Response Team (NRT), sponsored by the Environmental Protection Agency (EPA), which is responsible for national planning and coordination of Federal preparedness and response activities for hazardous materials incidents. NRC was particularly active in the NRT's preparedness activities and in making revisions to the National Contingency Plan. The NRC has also worked actively with EPA in connection with its proposed publication of the draft EPA Protective Action Guidance Manual in the *Federal Register.*

NRC representatives have also taken an active part in the Federal Radiological Preparedness Coordinating



Exercises to confirm and maintain the readiness of NRC response personnel and involving the Operations Center were carried out at several nuclear facilities during the report period. The exercises, which are based on varying accident scenarios, were performed during the report period at the Trojan (Ore.) plant, shown here, and the South Texas and Crystal River (Fla.) plants.

Committee (FRPCC), and its Subcommittee on Federal Response, with respect to implementation of Executive Order 12656, to revision of the Federal Radiological Emergency Response Plan (FRERP), to NUMARC's proposed revision of Emergency Action Level Guidance, and to the development and use of accident severity scales in other countries. NRC has also participated in meetings of the Subcommittee for Federal Earthquake Response Planning, sponsored by the Federal Emergency Management Agency (FEMA), to discuss the use of the FRERP during a catastrophic earthquake.

Emergency Response Data System (ERDS). The ERDS concept provides for licensee activated transmission of pre-selected plant data from the licensee to a computer at the NRC Operations Center during emergencies at commercial nuclear power plants. The implementation of ERDS was initiated in 1988. In 1989, the system hardware and software designs were completed, hardware was procured, and system integration and testing were begun. Currently, the ERDS program is moving forward with voluntary licensee participation. Efforts to arrange for licensee participation have entailed briefings for the Nuclear Utility Management and Resources Committee (NUMARC), the Edison Electric Institute (EEI), and individual utilities. At the end of 1989, eight utilities, representing 27 nuclear power units, had volunteered to participate in the ERDS program. Regulatory rulemaking to require the implementation of ERDS at all commercial nuclear power plants was begun in 1989. Following system testing with the initial individual utilities, delivery of the mainframe system to the Operations Center will be made in early 1990. It is expected that the remaining plant connections will be completed over the next two-to-three years.

Continuity of Government (COG) Program. During fiscal year 1989, the NRC continued to participate actively with the Department of Energy (DOE), FEMA and other governmental agencies in support of the COG program. The NRC has revised its Manual Chapter 0601, "Continuity of Government Program," to delegate authority to individuals in certain key positions to exercise responsibility during a national security emergency, and has issued a final rule, 10 CFR 50.54(dd), that allows licensees to take certain actions that depart from license conditions or technical specifications during a national security emergency.

Emergency Response Training. During fiscal year 1989, training sessions were conducted for more than 400 staff members, including five General Response Training sessions for all members of the headquarters response organization. These sessions discussed the NRC's response role and the work environment of the Operations Center. In addition, a course covering the technical training requirements for reactor accident protective measures assessment (Protective Measures Manual, NUREG/BR-0132) was presented to Headquarters and the Regions. The course included standardized procedures and computer codes for assessing public protective actions, projecting consequences, accessing weather information, and interacting with other Federal response organizations. The result is that all NRC response personnel with responsibilities for assessing protective measures during a reactor accident have a common basis for their assessments, using the same tools and procedures.

A similar training and procedure development program is under way for the reactor safety personnel responsible for assessing reactor conditions and for accident mitigation. A pilot course was conducted by the NRC's Office of Regulatory Research and the Technical Training Center which is designed to assure that the response staff is kept abreast of ongoing severe accident research and is prepared to perform an independent assessment of operator actions. The course included core-damage sequences, severe accident phenomenology, severe accident insights, event classification, and Emergency Operating Procedures (EOPs). Training development programs for response management, fuel cycle accidents, and materials accidents are planned for 1990.

Emergency Response Technical Tool Development. Work continued on various kinds of tools to assist in assessing the severity and possible consequences of reactor accidents. During 1989, the Radiological Assessment System for Consequence Analysis (RASCAL) dose model was published in NUREG/CR-5247. Protective Action Manuals were revised and distributed to all of the Regions. Development of Graphic Image Systems and of improved electronic mail capabilities is expected to be completed in 1990.

A Reactor Safety Assessment System (RSAS) is under development for use in assessing core status, in developing actions to restore plant stability, and in verifying the success of mitigative actions during emergencies at nuclear power plants. The RSAS system will provide the capability to represent, collect, store, and process the knowledge and plant-specific information required for the assessments.

The RSAS concept consists of generic models containing core protection knowledge for PWR and BWR types and will have the capability to extend the models to plant-specific versions. The knowledge base in RSAS includes critical safety function success criteria, available success path options, operational considerations from each of the Nuclear Steam Supply System (NSSS) vendors' emergency procedure guidelines, and severe accident insights.

RSAS will interface with ERDS to receive real time plant parameter data. The status of plant equipment will be obtained by voice transmission over the Emergency Notification System (ENS) and manually entered into RSAS. RSAS may prompt the user to provide additional information required for the analysis of the current data set.

RSAS work in 1989 consisted of the development of tools to allow a non-programmer to add new rules to the knowledge base, modify the generic knowledge structure, and extend the generic vendor models to plant-specific versions. The man-machine interface with RSAS has been modified to a more simplified system that can be operated with minimal training and experience. The ERDS-RSAS interface and the acquisition and input of the plant-specific information for PWRs are planned for 1990.

OFFICE OF INVESTIGATIONS

The Office of Investigations (OI) carries out investigations of alleged wrongdoing by individuals or organizations other than Nuclear Regulatory Commission (NRC) employees or NRC contractors, that is, by licensees, applicants and vendors, or their contractors.

In fiscal year 1989, OI opened 82 new cases and closed 88 cases. Fourteen of the closed cases were closed for administrative purposes; 28 closed cases were referred to the Department of Justice for consideration and possible prosecution.

During fiscal year 1989, OI again focused much of its attention on the sale of counterfeit and substandard parts, such as fasteners, flanges, valves, fuses, piping, and circuit breakers to nuclear power plants. Those investigations involving the execution of criminal search warrants in 1988 continue. OI efforts also resulted in the return of multiple-count Federal indictments in the Northern District of Texas and Southern District of Illinois, while other reports of investigation concerning product fraud/substitution have been referred to the Department of Justice for evaluation of prosecutorial potential. OI continues to work closely with the NRC Vendor Inspection Branch, NRR, as well as with the investigative units of the Department of Defense, the National Aeronautics and Space Administration, and the U.S. Customs Service.

Convictions/Guilty Pleas

On October 21, 1988, the radiation safety officer at Wright-Patterson Air Force Base pled guilty to charges of knowingly and willfully making a false statement to NRC Region III radiation specialists regarding the unlawful possession of an amount of unencapsulated Am-241 in excess of that allowed by the license. He was sentenced on December 13, 1988, to two years' probation and 200 hours of community service, and received a special assessment of $50.

The OI:Region IV investigation of Midwest Wireline Logging & Perforating, Inc., Seminole, Okla., resulted in the indictment of the firm's owner and the owner of a formerly NRC-licensed firm, Midwestern Wireline Corporation, for false statements to the NRC (18 U.S.C. 1001). Both negotiated guilty pleas and were sentenced on September 13, 1989, to 2½ years' probation and fined $2,500 each.

On May 25, 1989, the former radiation safety officer (RSO) of Met-Chem Engineering Laboratories, Salt Lake City, Utah, was sentenced to pay a $100 fine for the forgery of a letter to the firm's home office in which he denied that an employee overexposure had occurred. The OI:Region IV investigation of this matter reported the admission by the former RSO.

A nuclear medical technologist pled guilty to one count of an eight-count Federal indictment on August 21, 1989. The defendant had not been sentenced as of the close of the report period. The conviction came as a result of information that the defendant intentionally overdosed four hospital patients with diagnostic radiopharmaceuticals. The defendant was also charged with making material false statements in hospital records to conceal the overdoses.

The radiation safety officer at Bloomington Hospital, Bloomington, Ind., pled guilty on October 2, 1989, to covering up evidence of misadministrations and making false statements to Region III radiation specialists. Sentencing was pending at the close of the report period.

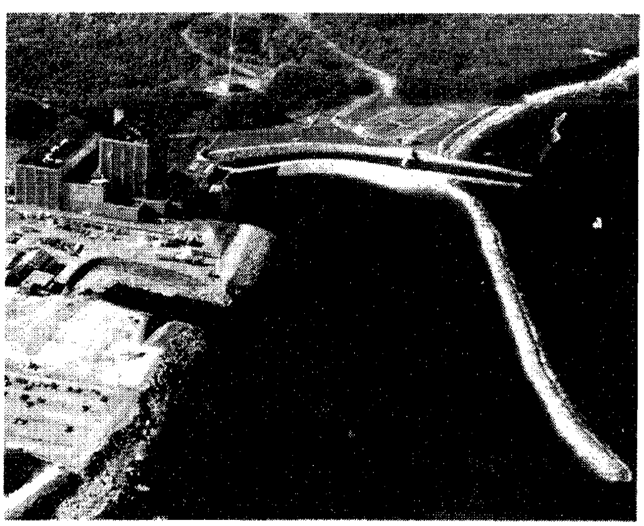
The former chief of radiology, Humana Hospital, Greenbrier Valley, West Virginia, who had been indicted for submitting false and altered preceptor statements pertaining to his application to become an authorized user of nuclear medicine, agreed on October 18, 1989, to the provisions of a pre-trial diversion agreement with the U.S. District Court, Southern District of West Virginia. This agreement, in lieu of Federal prosecution for violations of Title 18, U.S. Code, Sections 1341 (mail fraud) and 1001 (false statements), placed him under the supervision of a Federal probation officer for one year. Other terms of the agreement include stipulations that he refrain from employment in any activity licensed by the NRC or any of its Agreement States and from the practice of any form of nuclear medicine.

Enforcement Actions/Civil Penalties

In part as a result of an OI investigation, on November 1, 1988, the NRC sent a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $50,000 to Boston Edison Company for violations at their Pilgrim (Mass.) nuclear power plant. The violations involved three instances of degradation of vital area barriers, the failure of security personnel to report or correct one identified degraded condition, and a deliberate material false statement regarding the degraded condition made to an NRC inspector by an employee.

On January 18, 1989, in part as a result of an OI investigation into the destruction of records concerning a safety limit violation, the NRC sent a Notice of Violation to a former licensed operator at General Public Utilities Nuclear Corporation, Oyster Creek (N.J.) nuclear power plant, and a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $50,000 to the utility.

During fiscal year 1989, the following NRC enforcement actions were also initiated based on OI investigative reports:



Activities of the NRC Office of Investigation led to the discovery of violations by and levying of civil penalties on a number of utilities during the report period. The licensee for the Pilgrim facility near Plymouth, Mass., above, was among them.

(1) Midwest Wireline Logging & Perforating, Inc.: False Statement to NRC—License revocation pending.
(2) Waterford Unit 3 (La.): False Statements to NRC—Enforcement action pending.
(3) Rancho Seco (Cal.): Violation by Utility Managers in the Control of Radioactive Effluents—Civil fine of $100,000 paid.
(4) United States Testing Co.: Utilization of Non-Safety Certified Radiographers/Overexposure of Radiographers—Proposed civil fine of $280,000.
(5) Finlay Testing Laboratories: Illegal Transportation of Radioactive Materials on Commercial Passenger Aircraft and Employee Discrimination—License revoked.
(6) VA Medical Center, Loma Linda, Cal.: NRC Regulations Violated by RSO per Directions of Management and Radiation Safety Committee—Civil fine of $6,500.

OFFICE OF ENFORCEMENT

The NRC's 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, through the Office of Enforcement, under a revised enforcement policy (10 CFR Part 2, Appendix C (1989)) 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 Enforcement Policy was revised on October 13, 1988 to (1) provide greater discretion in determining the appropriate enforcement action in each case, (2) provide for higher civil penalties under certain specified conditions, (3) clarify the assessment factors, (4) modify certain severity level examples, (5) revise the Transportation and Safeguards supplements, and (6) make minor deletions and language changes.

The severity of NRC enforcement actions varies with the seriousness of the matter and the licensee's previous compliance record. Several levels of NRC actions are available:

- Written Notices of Violation are used in all 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. Civil penalties may also be imposed for particularly significant first-of-a-kind violations.

- Orders to "cease and desist" operations, or for modification, suspension, or revocation of licenses are used to dealwith 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.

The Regional Administrators have the authority to issue Notices of Violation not involving civil penalties and Notices of Violation proposing civil penalties with the concurrence of the Director of Enforcement and the Deputy Executive Director for Nuclear Materials Safety, Safeguards and Operations Support (DEDS). The DEDS is responsible for all enforcement decisions and issues all Orders, including those imposing civil penalties. The Director of the Office of Enforcement acts on behalf of the DEDS in his absence or as otherwise directed.

Appendix 6 provides a listing and brief summary of the 141 enforcement cases which resulted in civil penalty actions during fiscal year 1989, and also a brief description of the 13 enforcement Orders issued during fiscal year 1989. Recognizing that enforcement actions can sometimes span fiscal years, of the total cases for which there were actions in fiscal year 1989, 115 cases were proposed during this period for a total of $5,160,575; 31 were imposed during the period for a total of $1,690,333; and 118 were paid during the period for a total of $5,968,508.

Nuclear Materials Regulation

Chapter 4

The Office of Nuclear Material Safety and Safeguards (NMSS) of the U.S. Nuclear Regulatory Commission (NRC) and the NRC's five Regional Offices administer the regulation of nuclear materials, as distinct from nuclear reactor facilities (discussed principally in Chapter 2). The NRC conducts this 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) conversion of enriched uranium hexafluoride to ceramic uranium dioxide pellets and their subsequent fabrication into light water reactor fuel; (3) production of naval reactor fuel; (4) storage of spent reactor fuel; and (5) production and use of reactor-produced radioisotopes, or "byproduct material."

Highlights of actions taken during fiscal year 1989 include:

- More than 120 licensing activities dealing with fuel cycle plants and facilities.
- Approximately 2,800 fuel facility and material licensee inspections.
- Team assessments at three major licensee facilities.
- More than 5,500 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 this fiscal year, the NRC had completed 122 fuel cycle licensing actions. Table 1 shows the number of licensing actions by category.

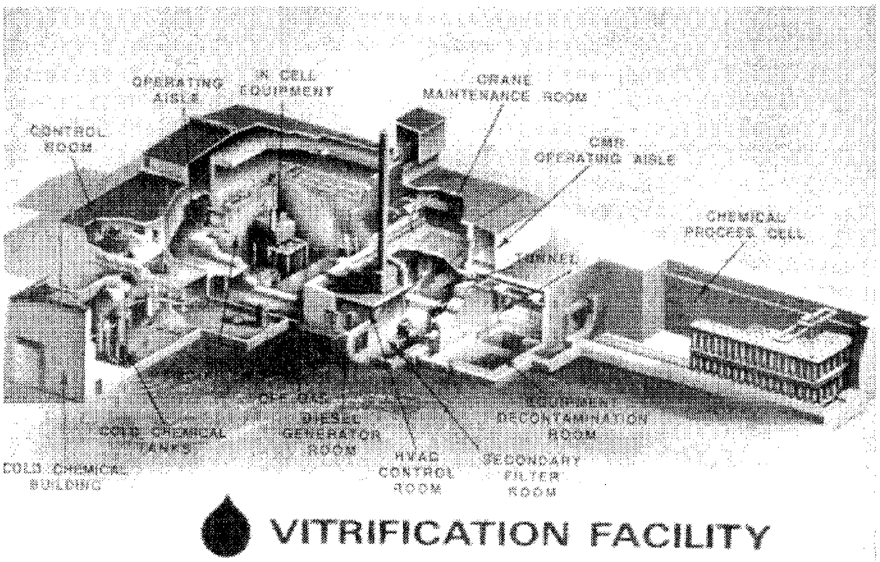
Uranium Enrichment

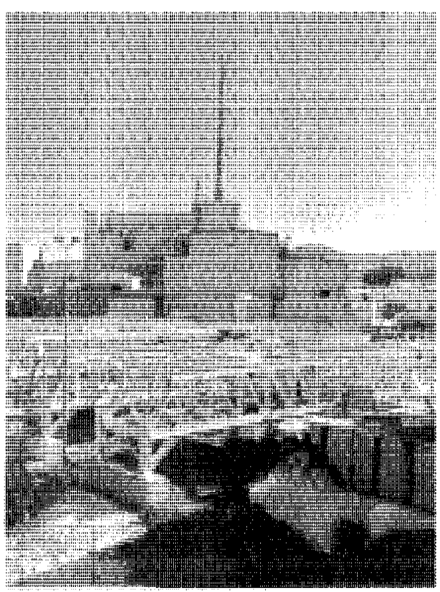
In fiscal year 1989, the Commission published an Advance Notice of Proposed Rulemaking on the regulation of uranium enrichment. After consideration of the comments received on the *Federal Register* Notice and uncertainties about potential applicants and legislative changes, the Commission, in April 1989, decided not to proceed with the proposed rulemaking. Nevertheless, the Commission stated that the general design criteria and other guidance in the Advance Notice would be used in licensing under the Commission's regulations in 10 CFR Part 50, which pertains to all types of production and utilization facilities—including uranium enrichment plants. During this fiscal year, Congress was considering, but did not pass, legislation which would change the definition of production facilities in the Atomic Energy Act, so that uranium enrichment plants would be licensed pursuant to 10 CFR Parts 40 and 70.

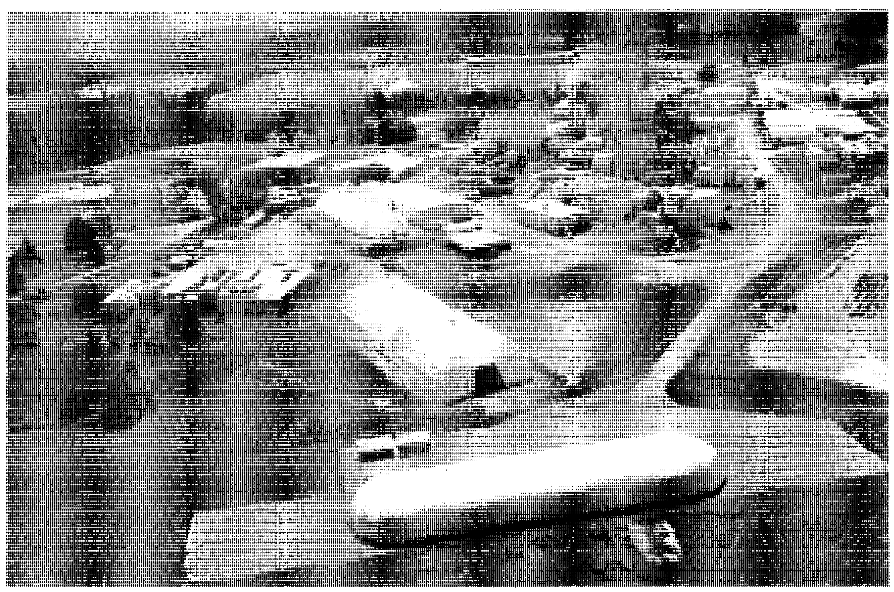
In June 1989, announcements were made of the formation of a partnership, Louisiana Energy Services, to build a gas centrifuge uranium enrichment plant in northern Louisiana, with an annual production capacity of 1.5 million "separative work units." The partners are Duke Power Company, Fluor Daniel, Inc., Louisiana Power and Light Company, a subsidiary of Northern States Power Company, and Urenco, Inc., which would supply the gas centrifuge technology based on its operating facilities in Europe. The partnership expects to submit an application for licenses in late calendar year 1990, and discussions on licensing issues with the NRC staff had already begun at the end of fiscal year 1989.

West Chicago: Kerr-McGee Rare Earths Facility

At the direction of the Atomic Safety and Licensing Board, the staff issued the Supplement to the Final Environmental Statement on the West Chicago, Ill., facility in May 1989. At issue are decommissioning and the on-site stabilization of thorium-bearing wastes. With the issuance of the staff's documents, the board has pursued resolution of the proceeding with the parties (Kerr-McGee, the State of Illinois, and the staff)







The NRC is responsible for safety oversight at the Department of Energy's (DOE) West Valley Demonstration Project, near Buffalo, N.Y. The project seeks to demonstrate the feasibility of solidifying high-level radioactive waste for disposal in a Federal repository. The wastes at the project site were generated from the reprocessing of commercial and Federal spent fuel between 1966 and 1972. DOE took possession of the site to conduct the project in February 1982.

At top is a drawing of the high-level waste vitrification facility, in which the waste will be solidified in borosilicate glass; the facility is still under construction. To the right above is a general view of the project site, with temporary radioactive waste storage structures in the foreground. Above is the reprocessing plant, also visible in the background of the photo above right, which is undergoing decommissioning; in the foreground above are newly built superstructures in which high-level waste from underground tanks is pumped out and processed. At right, a crew excavates a leaking tank from the old reprocessing plant burial ground on the site.

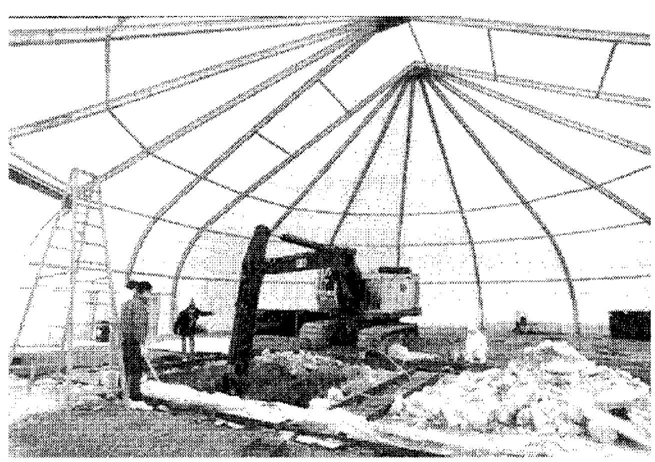


Table 1. Fuel Cycle Licensing Actions Completed in FY 1989

Category	*No. of Actions*
Uranium Fuel Fabrication	47
Uranium Hexafluoride Production	22
Fresh Fuel Storage at Reactor Sites	13
Critical Mass Materials	14
Interim Spent Fuel Storage	9
Uranium Fuel Research & Development	4
Advanced Fuel Research & Development	5
Other Source Material	6
Enrichment	2
Total	122

through a series of orders and required submittals. Concurrent with these actions, the NRC received a proposed amendment to the Agreement between the State of Illinois and the NRC which would extend the jurisdiction of the State over radioactive materials to include the type of waste at the West Chicago site. If the NRC approves the amendment to the State Agreement, Illinois would assume jurisdiction over the site and proposed disposition of the waste. (See the *1986 NRC Annual Report*, p. 88, for background.)

West Valley Demonstration Project Oversight

In 1989, the Commission staff continued its safety oversight activities at the Department of Energy (DOE) West Valley Demonstration Project (WVDP) near Buffalo, N.Y. The WVDP's purpose is to demonstrate the solidification and preparation of high-level radioactive waste from reprocessing for disposal in a Federal repository. Removal of dissolved cesium from the supernatant (liquid) portion of the waste began in early 1988. The cesium will be combined with the sludge (solid) portion of the high-level waste, which contains most of the other radionuclides. Beginning in 1992, the combined wastes will be solidified in borosilicate glass.

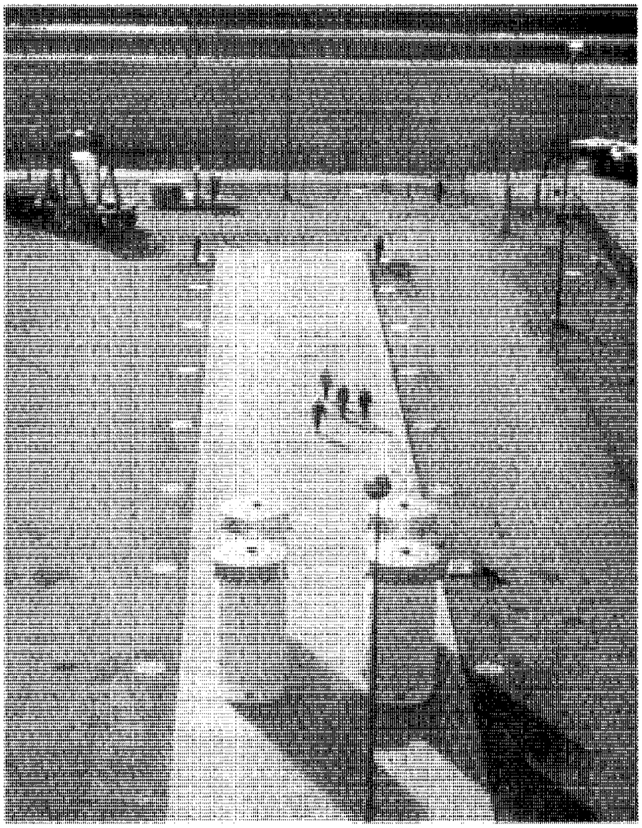
The staff monitors public health and safety aspects of the WVDP by inspecting the West Valley site and reviewing Safety Analysis Reports submitted by DOE. DOE normally submits a separate Safety Analysis Report for each segment of the waste process, including solidification in glass-making. The staff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

In 1989, the staff issued a Safety Evaluation Report on the Drum Cell, which is a storage facility for drums of cemented low-level waste. When the project is finished, the Drum Cell will contain 15,000, 71-gallon drums of solidified waste. A DOE Environmental Impact Statement is to consider options in which these drums may be transported elsewhere for disposal, or the Drum Cell converted into a permanent disposal tumulus. The staff monitored the operation of the cesium-removal process that started in 1988. By the end of 1989, about half of the cesium originally contained in the high-level waste will have been removed and immobilized on an ion-exchange medium. The staff will continue to monitor the safety of this process for its duration, and has begun reviewing safety documentation for the vitrification facility, the structure that will house the glass-making equipment.

Interim Spent Fuel Storage

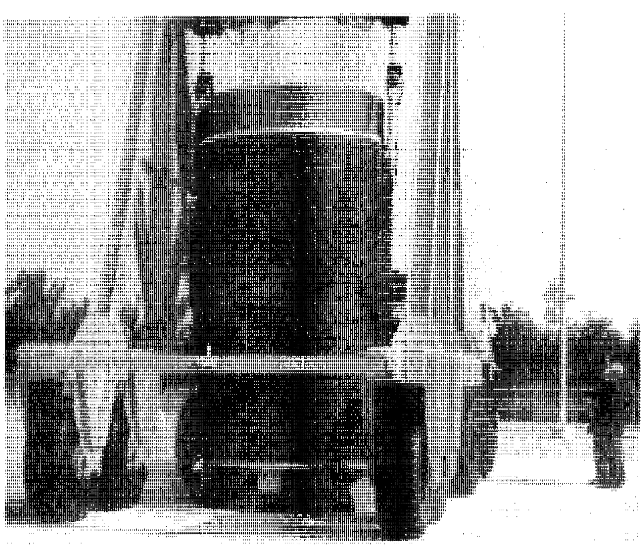
Under the Nuclear Waste Policy Act of 1982 (NWPA), utilities are responsible for interim storage of their spent fuel until a Federal repository or monitored retrievable storage installation is available. Utilities are continuing to develop plans for providing additional storage capacity as they approach the current storage limits of their reactor pools.

Where possible, utilities continue to re-rack spent fuel pools, a measure that has extended storage capacity for most reactors into the 1990s. Besides re-racking, some utilities are considering rod consolidation as a means of increasing pool capacity. On-site dry storage of aged spent fuel in modular units is also a means being used by utilities to meet storage needs.



Until an off-site repository is available, utilities are responsible for the interim storage of the highly radioactive spent fuel resulting from reactor operations. Spent fuel pools on the plant sites are the customary storage place for spent fuel, such as that shown below at the right (at the Perry (Ohio) facility), but dry spent fuel storage is an important option for utilties whose storage pool capacity is running out.

Above are four casks, each of them able to accommodate 21 spent fuel assemblies, at the Surry nuclear power plant of the Virginia Electric Power Company. The picture below shows the installation of the first cask at Surry, licensed by the NRC in 1986. The casks at Surry, made by General Nuclear Systems, Inc., of West Germany, weigh about 100 tons empty and 125 tons loaded.

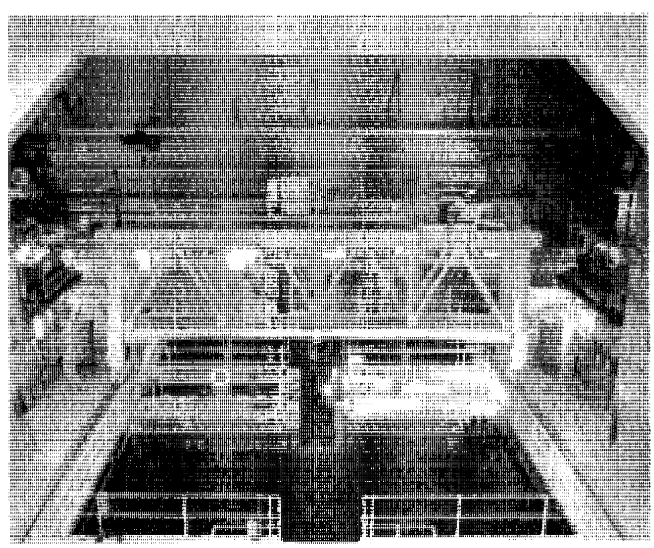


In 1986, the NRC issued the first two licenses for dry spent fuel storage to the Virginia Electric Power Company (VEPCO) for its Surry nuclear power plant and to the Carolina Power & Light Company (CP&L) for its H.B. Robinson (S.C.) nuclear power plant. The NRC staff continued to monitor development as the facilities were constructed and storage casks and canisters were fabricated.

In March 1988, the Duke Power Company applied for dry spent fuel storage at its Oconee (S.C.) nuclear power plant site. An environmental assessment and finding of no significant impact were published by NRC staff in November 1988. At the end of fiscal year 1989, the NRC staff essentially completed its safety review. In May 1989, NRC staff received an application for dry spent fuel storage from CP&L for its Brunswick (N.C.) nuclear power plant site. Staff had had this application under reveiw at the close of the report period.

In April and July 1989, the NRC staff issued letters of approval with related safety evaluations for two topical reports. The Nutech Engineers, Inc., report was approved for the Nutech concrete module and stainless steel canister system design, the NUHOMS-24P. The canister has a capacity of 24 pressurized water reactor (PWR) spent fuel assemblies, and one canister isemplaced in each module. This dry storage system is being installed at the Duke Power's Oconee plant. The Transnuclear, Inc. (TN) report on a TN-24 ferritic steel dry storage cask design, which has a capacity of 24 PWR assemblies, was also approved.

The NRC staff is reviewing four other topical reports on metal dry storage cask designs submitted by Nuclear Assurance Corporation, Combustion Engineering, and General Nuclear Systems, Inc., and



one topical report on a ventilated concrete dry storage cask design submitted by Pacific Sierra Nuclear Associates. Once the NRC approves such topical reports, a utility selecting a dry storage cask design may reference them in a license application or in an amendment to an existing 10 CFR Part 72 license, to expedite the review of a dry storage system or a proposed modification to an existing system.

To further streamline the licensing process for use of spent fuel dry storage casks at reactor sites, the NRC staff initiated amendments to 10 CFR Part 72. The rulemaking is consistent with that contemplated by Congress in the NWPA for "use at the sites of civilian nuclear power reactors without, to the extent practicable, the need for additional site-specific approvals by the Commission." Draft criteria and standards have been prepared to provide for formal certification of dry spent fuel storage cask designs and for the use of certified casks by reactor operators under a general license. The Commission issued a proposed rule for public comment on May 5, 1989. The NRC staff is reviewing the comments received to prepare a final rule for Commission consideration in early calendar year 1990.

Operational Safety Team Assessments

The NMSS staff continues to conduct operational safety team assessments at major fuel cycle and materials facilities. These assessments are expanded inspections with emphasis on all relevant aspects of safety management at the facilities. The assessments evaluate the areas of management organization and controls, chemical process safety, environmental protection, operations, transportation activities, fire protection, radiation safety, emergency preparedness, safety-related instrumentation and maintenance, and criticality safety. The assessment teams include representatives from the Regions, NRC Headquarters, and other Federal agencies, such as the Occupational Safety and Health Administration (OSHA) and the Environmental Protection Agency (EPA). Since the effort began in 1986, the staff has performed 25 such assessments. Recent assessments were at the two uranium hexafluoride production plants—Allied-Signal in Metropolis, Ill. (May 15-19, 1989) and Sequoyah Fuels, Gore, Okla. (July 24-28, 1989)—and at one fuel fabrication facility—Westinghouse, Columbia, S.C. (July 31-August 4, 1989).

Using the experience gained from these operational safety team assessments to improve the regulatory program for fuel cycle facilities, the NRC published four Branch Technical Positions (BTPs) in the March 21, 1989 *Federal Register* (54 FR 11590). The BTPs have been prepared to provide guidance in the areas of management controls/quality assurance, requirements for operation, chemical safety, and fire protection. The NRC intends to use the BTPs as the framework for developing Standard Format and Content Guides for applications and Standard Review Plans for the NRC staff, and to integrate the BTPs into the inspection programs through revised inspection modules.

Fuel Cycle Workshop

In an effort to improve communication with its licensees on regulatory matters, the NRC held its second Fuel Cycle Workshop in May 1989 in Bethesda, Md. The first workshop was conducted in October 1987 in Atlanta, Ga. The more recent two-day workshop stressed the need for licensees and the NRC to maintain safety awareness and avoid complacency. NRC staff made presentations on new rulemakings and Branch Technical Positions that have been published for comment. Staff also discussed several ongoing policy matters. The presentations were followed by open discussion and questions from the licensees. The workshop provided a forum for licensees to exchange views among themselves, learn from each other, and discuss the different means by which they are achieving mutual safety objectives. Both the NRC staff and the licensees agreed that similar workshops should be held at about 12-to-18 month intervals.

Interaction with OSHA

The NRC and the Occupational Safety and Health Administration (OSHA) have agreed on a coordinated effort toward worker safety in nuclear facilities that the NRC licenses. The agreement is contained in a Memorandum of Understanding (MOU) published in the October 21, 1988 *Federal Register*. The MOU with OSHA delineates the general areas of responsibility of each agency and describes the efforts of OSHA and the NRC to protect workers at facilities that NRC licenses. As part of the MOU, OSHA is participating in operational safety team assessments at certain NRC-licensed facilities, and is training designated NRC staff in non-radiological safety areas. Reactor resident inspectors at nuclear plants are also being trained. On April 3-7, 1989, 15 fuel facility and materials staff from the Regions and Headquarters took a one-week course on hazardous materials. On May 8-12, 1989, a second group of staff members took a one-week course on fire protection. OSHA and the NRC Technical Training Center are working together to develop a single, one-week course tailored to NRC needs. The MOU also stipulates that if, in the course of its radiological and nuclear safety inspections, NRC personnel identify safety concerns within areas of OSHA responsibility, they will notify OSHA and also bring the matters to the attention of licensee management.

MATERIALS LICENSING AND INSPECTION

The NRC currently administers approximately 8,200 licenses for the possession and use of nuclear materials in medical and industrial applications. Table 2 shows the distribution of these licensees by Region. The 29 Agreement States administer about 16,000 additional licenses. The program is designed to ensure adequate protection of the public health and safety in the conduct of activities involving these radionuclides. NRC regional staff completed over 2,700 inspections of materials facilities in 1989. The NRC Regional Offices administer all materials licenses, with the exception of exempt distribution licenses and sealed-source and device design reviews, which are done at NRC Headquarters.

The NRC completed nearly 5,600 licensing actions during this fiscal year. Of this total, about 500 were new licenses issued, 3,600 were amendments, 1,300 were license renewals, and 200 were sealed-source and device reviews.

Reducing the Licensing Backlog. During fiscal year 1989, NRC staff made a concerted effort to reduce the backlog of pending license applications. The number of "backlogged" cases was reduced from about 275 to 224 pending actions in September 1989. As more resources are applied to materials licensing casework, the staff expects a similar reduction in 1990. The ultimate goal is to reach licensing decisions on routine new license applications and amendments within 90 days, and on renewals within 180 days.

Human Factors. The NRC has recognized that human error associated with non-reactor uses of byproduct material is a significant factor in incidents that result in unnecessary or excessive public and occupational exposures. Therefore, in the last year, the NRC has undertaken to systematically apply human factors technology to the materials program.

One such initiative entailed support to NMSS by the Office of Nuclear Regulatory Research (RES) in the development of the medical quality assurance rule and Regulatory Guide. Regional inspections of medical facilities were conducted, and the results of a nationwide nuclear pharmacy licensee's evaluation of its own operations were reviewed. Information from these efforts has been disseminated throughout the NRC and to appropriate licensees to stimulate the application of human factors technology. Research projects in areas such as teletherapy, treatment planning computers, brachytherapy, nuclear pharmacies and nuclear medicine have also been identified.

The staff has supported efforts to reduce human error in industrial radiography. This includes participating in review of the radiography equipment rulemaking and review of the training and qualifications requirements for radiographers.

Industrial Uses

Industrial Radiography. This form of nondestructive testing uses radiation from byproduct material sources to examine the internal structure of materials. The NRC's radiography licensees perform testing within fixed radiography facilities or at temporary job sites. Portable devices can contain radiation sources of up to 200 curies of iridium-192 or up to 100 curies of cobalt-60. Devices at fixed facilities can contain sources of up to several hundred curies. At the end of this fiscal year, the NRC had issued a total of 261 radiography licenses; of these, 64 were for operations in fixed facilities and 197 for use at temporary job sites.

During the report period, the American Society of Nondestructive Testing (ASNT) Special Task Group continued its efforts to develop a national radiographer certification program. The NRC expended considerable effort working with the Task Group to further develop an acceptable program and to develop procedures for implementing the program. In early 1989, the Task Group presented the program to the ASNT Board of Directors and, in March 1989, the board approved the document describing its intended certification program. NRC staff arranged an April 5, 1989 Commission briefing by ASNT, the State of Texas, the Conference of Radiation Control Program Directors (CRCPD), and staff from NMSS and the Office of Governmental and Public Affairs (GPA). During the briefing, participants explained that under the ASNT Certification Program, either ASNT or CRCPD may administer the written examination (developed by the State of Texas). Therefore, in some States, CRCPD would administer the written examination and, in the remaining States, ASNT would administer the written examination.

By letter dated April 12, 1989, ASNT submitted Revision 0 of its "Certification Program for Industrial Radiography Radiation Safety Personnel" (IRRSP) for formal NRC review. In a letter dated July 20, 1989, Chairman Carr informed the ASNT Board Chairman that the NRC had completed its formal review of Revision 0 and that the Commission supports the ASNT initiative, since the certification program will help to assure the participants (NRC and the States) that only properly qualified individuals will conduct industrial radiography in the United States. ASNT has indicated that it intends to conduct a trial run of the testing portion of its program in late 1989.

Table 2. Regional Distribution of Nuclear Materials Licenses

(as of September 1989)

Region I	2,993
Region II	1,015
Region III	2,852
Region IV	810
Region V	285
Headquarters	259
Total	8,214

The NRC is drafting a proposed rule which would amend 10 CFR Part 34 to allow a radiography license applicant to affirm that individuals acting as radiographers are certified in radiation safety through ASNT's IRRSP program. This affirmation would be in lieu of describing an initial radiation safety training program for radiographers in the subjects outlined in Appendix A of Part 34, and the means used to determine a radiographer's knowledge and understanding of these subjects. The staff expects to publish the proposed rule for public comment in late 1989, and a final rule in early 1990.

On September 11, 1989, the Commission unanimously approved final amendments to 10 CFR Part 34 for radiographic equipment, which are intended to improve equipment reliability and safety. The amendments would also require that radiographic personnel use audible alarm rate-meters and report equipment failures to the NRC.

General License Effectiveness. General licensees include individuals or organizations that become licensees (without contacting the NRC) when they receive a byproduct source or a device containing a byproduct source from a specific licensee. These include certain measuring, gauging, illuminating, and controlling devices containing byproduct material with radioactivity ranging from microcuries to several curies. There are approximately 30,000 general licensees in non-Agreement States, using about 400,000 devices, and about twice this many in Agreement States. General licensees are expected to be able to use the devices safely by following simple instructions—without having radiological safety training or experience—because safety is built into the devices. The staff has reviewed the broad issues associated with the general license program and has worked with the Agreement States, as well as with the Regional Offices, with the objective of achieving better regulation and performance.

Source/Device Registration. The NRC and the Agreement States maintain a sealed source/device registration program to expedite the licensing review process when new requests for sources or devices are received. During the fiscal year, the staff completed 170 safety evaluations for radioactive sources and devices. The computerized registry system for approved sealed sources and devices produced about 260 reports for NRC Regional Offices, Agreement States, the Center for Devices and Radiological Health (CDRH), and the Atomic Energy Control Board of Canada. Two comprehensive Regulatory Guides (10.10 and 10.11) remain in wide use, to augment the registration program.

Irradiator Rule. Irradiators use gamma radiation from cobalt-60 or cesium-137 to irradiate products and change their characteristics in some way. Large irradiators are those delivering a dose exceeding 500 rems in one hour at a distance of one meter, not including irradiator devices in which the area being irradiated is totally enclosed within the device. About 85 percent of irradiator capacity is now used to sterilize disposable medical products. Most of the remaining capacity is used in chemical processing, primarily to induce polymerization in plastics. In the United States, there is very little irradiation of food (for the purpose of destroying pests or prolonging shelf life) at present, and the prospects for such irradiation are unclear.

There are currently about 70-to-80 large irradiators in the United States. About one-third are licensed by the NRC and two-thirds by Agreement States. Five additional irradiators are either under construction or proposed for construction in Agreement States. Almost all the irradiators use cobalt-60; four use cesium-137.

In addition to these, Congress has appropriated money to DOE in support of the construction of six

The NRC is responsible for the regulation and inspection of medical uses of radioactive materials, a steadily widening area of concern. Above, NRC Region I Health Physicist Marlene Taylor (right) goes over the records with a hospital technician.

irradiators to be used in food processing. These irradiators would be licensed by the NRC or by Agreement States, depending on their locations.

The staff worked on a proposed rule, to be published for public comment in early 1990, that specifies radiation safety and license requirements for the use of licensed radioactive materials in large irradiators. These safety requirements would apply to large panoramic irradiators and certain underwater irradiators. The proposed rule would make the NRC's licensing reviews and inspections more efficient, since licenses could be issued with fewer license conditions, and there would be a uniform set of requirements for inspections.

Sealed Sources Exceeding Part 61 Class C. Several licensees have experienced problems disposing of sealed sources they no longer need in their businesses. Certain well-logging sources, neutron sources, gauges, and irradiator and teletherapy sources are not accepted for disposal at commercial burial sites because, when concentrated for disposal, the radioactive material exceeds the limits for Class C low-level waste, as defined in 10 CFR Part 61.

Under Federal law, ultimate disposal of these wastes is the responsibility of the Federal Government, however, licensees must pay the disposal costs. The Deparment of Energy (DOE) is developing a program to establish a disposal facility for such waste, but this DOE disposal facility may not be available for 10 years or more. The NRC and DOE staff have discussed the need for DOE to accept and store these wastes at DOE sites, pending the availability of a disposal facility. In the interim, several hundred NRC and Agreement State licensees continue to possess sealed sources which will have to be stored indefinitely when they are no longer needed.

New Uses of Byproduct Material. The NRC has issued a license to the Federal Aviation Administration (FAA) for a new explosive detection system (EDS), which uses thermal neutron activation technology. The EDS device uses a moderated californium-252 (Cf-252) source to activate nitrogen atoms in plastic explosives. When radiation from the decay of activated nitrogen atoms is detected, a computer attached to the system gives a warning signal that explosives may be contained in luggage and further inspection efforts are initiated. In February 1989, the NRC issued a license for the first device, EDS-2, which was a prototype design. Since then, FAA's contractor has developed and built a more advanced version (EDS-3) which uses less Cf-252. In August 1989, the NRC staff completed the environmental assessment of the EDS-3, published a finding of no significant impact in the *Federal Register*, and issued a license to FAA for use and operation of up to six EDS-3 devices. The FAA is currently operating the first EDS-3 device at John F. Kennedy Airport in New York, and has plans for future use at other international airports in the United States and abroad.

Medical Uses

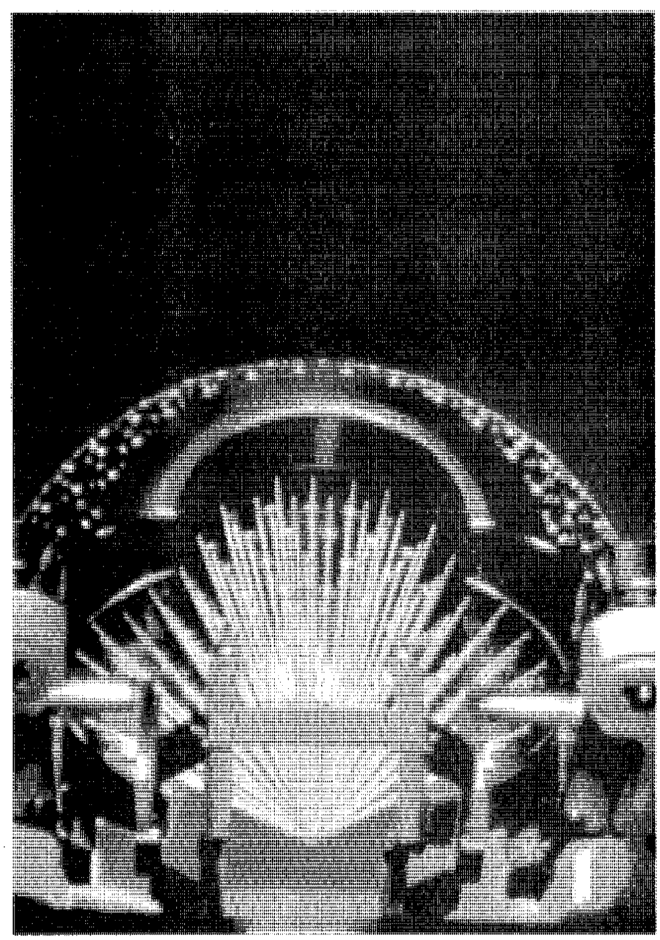
Medical Program Improvements. Increased emphasis on the safe use of byproduct material in human applications has continued this year. The headquarters and regional medical program staffs now include individuals with training and experience in nuclear medicine, radiation therapy and radiopharmacy. The NRC has improved communication with government agencies and professional organizations. The NRC's Advisory Committee on the Medical Uses of Isotopes (ACMUI) is being enlarged to include representatives from hospital management and technologists, in addition to the medical and scientific specialties presently represented on the Committee (see Appendix 2 for list of current members). The NRC has also increased communications with the Food and Drug Administration (FDA) and the Department of Veterans Affairs (VA). Of special interest is the initiative to improve radiation safety programs in the VA. The VA Central Office has, with NRC assistance, developed a program to provide additional audits of VA nuclear medicine programs to self-identify and correct items of non-compliance. NRC personnel provided familiarization training to 40 VA industrial hygienists, radiation safety officers, and administrative personnel. The training program was conducted at VA Medical Centers located near each of NRC's five Regional Offices.

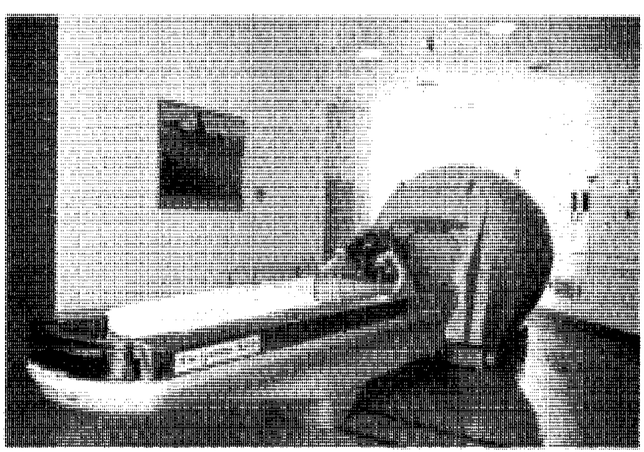
Training Standards. The NRC has requested public comments on the appropriate training and experience criteria for all individuals who participate in the medical use of byproduct material. The NRC has also hired a contractor to study training programs, accreditation and certification programs, and State requirements for accrediting such training. The NRC may revise its current training and experience criteria after analyzing the public comments and the contractor's report.

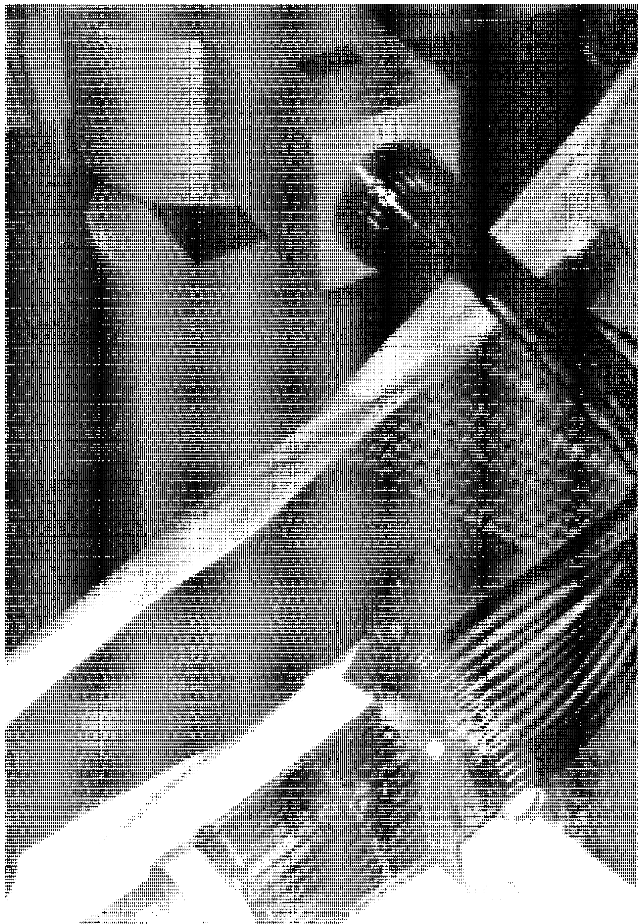
Petition for Rulemaking: Nuclear Medical and Pharmacological Issues. On June 8, 1989, the NRC received a Petition for Rulemaking submitted by the American College of Nuclear Physicians and the Society of Nuclear Medicine. The petitioners requested the following: (1) physicians be permitted to deviate from the package insert instructions of indications and method of administration of radiopharmaceuticals or biologics for therapy use; (2) physicians and pharmacists be permitted to deviate from the manufacturer's instructions for preparation of all biologics, radiopharmaceuticals, kits and generators; and (3) physicians and pharmacists be permitted to prepare and compound biologics, radiopharmaceuticals, kits and generators that are not the subject of an FDA-approved "Product Licensing Application" (PLA), or "New Drug Application" (NDA), or FDA-accepted "Notice of Claimed Investigational Exemption for a New Drug" (IND). The request involves changes to appropriate sections of NRC's regulations in 10 CFR Parts 30, 32, 33, and 35.

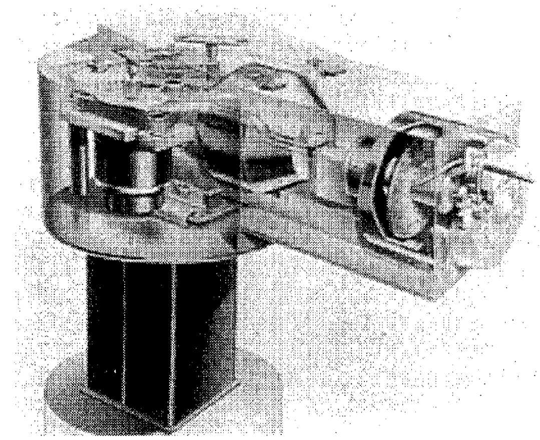
The proposed changes would affect NRC's medical-use licensees' receipt and use of byproduct radioactive drugs that are normally regulated by the FDA's Center for Drug Evaluation and Research and Center for Biologics Evaluation and Research. The "notice of receipt" of the petition was published in the *Federal Register* on September 15, 1989. The NRC is working closely with the FDA, the nuclear medicine community, and the radiopharmacy community to resolve the issues raised by this petition.

The Gamma Knife is a new surgical device employing radiation in the treatment of deep seated clots or tumors which are inaccessible or otherwise ill suited to conventional surgical procedures. The entire apparatus is shown below. The picture at the right is a schematic showing the radiation distribution within the device, calibrated for highly precise applications.









The medical device shown above, the microSelectron-HDR delivers high dose rate therapy using a computerized system incorporating special safety checks. The device, which contains a radiation source and calibration system, is shown at left hooked up to a patient taking treatment.

Quality Assurance in Medical Uses. In 1989, the NRC developed proposed amendments to 10 CFR Part 35 that would require medical use licensees to establish and implement a basic quality assurance (QA) program. The objective of this basic QA rule is to provide high confidence that errors in medical use of.byproduct material will be prevented. The proposed amendments would assure protection of the patients while allowing medical-use licensees the flexibility necessary to provide optimal medical care. The NRC is also proposing certain modifications to the definition of the term "misadministration" and to the related reporting and record-keeping requirements. In addition, the NRC has developed a draft Regulatory Guide that will assist licensees in the implementation of the basic QA rule. The NRC also intends to conduct a pilot program to determine the impact and efficacy of the proposed basic QA rule.

Advisory Committee on Medical Uses. The Advisory Committee on Medical Uses of Isotopes (ACMUI) was established in June 1958. The ACMUI is comprised of qualified physicians and scientists who consider medical questions referred to them by the NRC staff and provide expert technical advice on the medical uses of byproduct material. The ACMUI also advises the NRC staff, as required, on matters of policy. In March 1989, the staff published a call for nominations to the committee, in the *Federal Register*. A total of 21 nominations were received. From these, the staff recommended six individuals to the Commission for selection to four open positions on the committee.

By filling the open positions, the ACMUI will be able to place more emphasis on QA concerns. Current membership of the Committee is shown in the Appendix 2.

EVENT EVALUATION AND RESPONSE

The NRC continued to review and analyze operational safety data from nuclear fuel facilities and materials licensees, and maintained its ability to respond to events at these facilities. The NRC participated in an emergency response exercise at the Mallinckrodt materials facility in St. Louis, Mo. The event simulated an industrial fire that would have resulted in a large release of iodine. This exercise allowed NRC to evaluate new emergency response procedures related to events at materials licensees, and it gave the licensee a better perspective of the response it might expect from the NRC in such an event.

Event Evaluation Activities

Polonium-210 Contamination from Static Eliminators. In an event covered in the *1988 NRC Annual Report*, pp. 62, 63, contamination resulted from the failure of polonium-210 (Po-210) static eliminators that had been manufactured and distributed by Minnesota Mining and Manufacturing Company (3M). The NRC prohibited 3M from distributing the Po-210 devices to any customers, but permitted the company to continue manufacturing the devices for research and development purposes.

The 3M company continued the recall and testing of returned Po-210 static eliminators. By the end of fiscal year 1989, 3M reported that its customers had returned all but about 2,200 Po-210 static eliminators. These remaining devices had been disposed of by the customers or were considered lost. In view of the extensive radioactive decay of the Po-210 used in 3M's devices, there is no danger to the public health and safety from these unreturned devices.

Cesium-137 Contamination from Waste Encapsulation and Storage Facility (WESF) Sources. The staff continues to provide technical assistance to the Agreement State of Georgia and to monitor DOE's follow-up activities on the cesium-137 (Cs-137) contamination incident at the irradiator operated by Radiation Sterilizers, Inc. (RSI) in Decatur, Ga. (See the *1988 NRC Annual Report*, p. 84). During fiscal year 1989, DOE and its contractors identified one leaking WESF source and shipped it, along with several others, to the Oak Ridge National aboratory for analysis. Radioactivity levels in the pool water at RSI (Decatur) decreased with removal of the leaking WESF source and have remained constant since then. At the end of fiscal year 1989, the cause of the WESF source failure was still under investigation.

RSI operates a similar irradiator in Westerville, Ohio, that also used WESF sources. DOE and its contractors have tested all the WESF sources at both RSI facilities and have found no other leaking source. They are presently shipping all the WESF sources to DOE facilities, for storage and additional testing.

During fiscal year 1989, the NRC received a petition from the National Coalition to Stop Food Irradiation (NCSFI), asking that the NRC suspend the license issued to the Applied Radiant Energy Corporation, another commercial irradiator that uses the WESF sources, because of the incident at RSI (Decatur). NRC denied NCSFI's request. However, the NRC indicated that it would use the results of DOE's investigation into the WESF source failure to determine whether continued use of WESF sources will provide reasonable assurance that the public health and safety will be protected.

Emergency Preparedness

The final rule for ''Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees'' was published in the *Federal Register* on April 7, 1989 (54 FR 14051). The rule becomes effective on April 7, 1990, and will require approximately 30 major fuel cycle and materials licensees to maintain emergency plans to cope with serious accidents for which off-site response organizations might be needed.

This rule requires radioactive material licensees who are subject to radiological emergency planning requirements to demonstrate compliance with the Emergency Planning and Community Right-to-Know Act, for any hazardous chemicals they may possess. Affected licensees must make provision for biennial on-site emergency drills. Off-site response organizations must be invited to participate in these drills, but their participation is not required.

Safeguards and Transportation

Chapter 5

Pursuant to provisions of the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, the U.S. Nuclear Regulatory Commission (NRC) regulates safeguards for licensed nuclear materials, facilities, and activities 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 (NPRs) emphasize protection against theft or diversion of SNM, whereas safeguards for 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 1989, NRC safeguards requirements were applied to 110 power reactors, 51 NPRs, and 15 active non-reactor nuclear facilities. They were also applied to 42 shipments of spent fuel, 26 shipments of SNM involving more than one but less than five kilograms of HEU, and one shipment of SNM involving five or more kilograms of HEU.

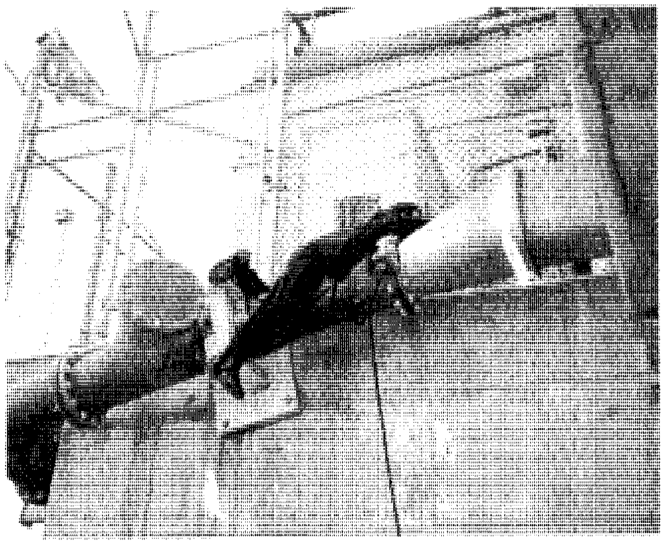
The Federal Government regulates safety in the transportation of radioactive materials primarily through the NRC and the Department of Transportation (DOT). These two agencies have delineated their respective regulatory responsibilities in this area through a Memorandum of Understanding (MOU). For international shipments, DOT is the designated U.S.authority and is responsible for implementing International Atomic Energy Agency (IAEA) standards. The NRC advises DOT on technical matters.

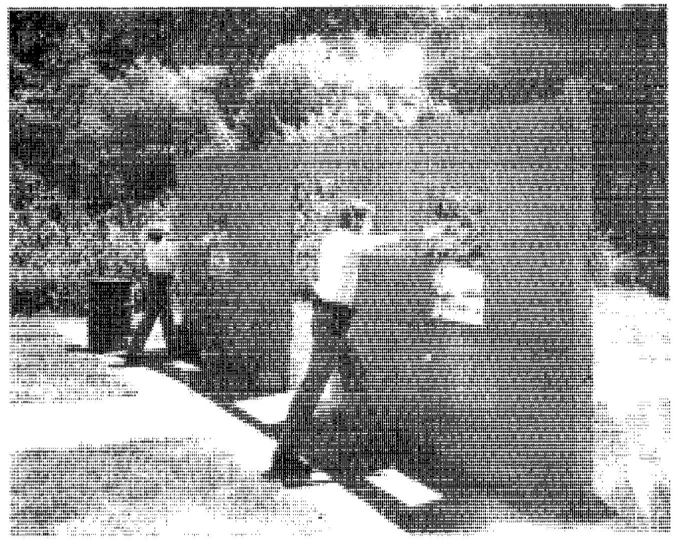
STATUS OF SAFEGUARDS AND TRANSPORTATION IN 1989

Reactor Safeguards

Power Reactors. Power Reactors. NRC safeguards regulations were implemented at 112 licensed power reactors. Although the ''design-basis'' threat remains unchanged, the Commission decided that, as a matter of prudence, power reactor licensees should develop contingency plans which could be put into effect within a short time after receiving notice from the NRC that the threat level had changed. A generic letter to this effect was issued to licensees. Also in the area of power reactor licensing, safeguards reviews of standard and advanced reactor designs focused on their consistency with the Severe Accident Policy Statement provision that ''issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants.''

Non-power Reactors (NPRs). Fifty-one operating NPRs were subject to the NRC's safeguards regulations during fiscal year 1989. Efforts have continued toward converting 25 NPRs from the use of high-enriched uranium to low-enriched uranium. The NRC regulation associated with this effort states that implementation, if required, would be deferred until Department of Energy (DOE) funding is available, and that a licensee can be exempted from conversion if the Commission finds that the reactor has a ''unique purpose'' requiring the use of HEU. At the end of the fiscal year, three NPR licensees had completed the conversion program, nine NPR licensees had been funded and were in the process of being converted, and four university NPR licensees were awaiting funding. Three commercial NPR licensees were not scheduled to receive DOE funding, and there was no suitable fuel developed for one university NPR. Two ''unique purpose'' applications were being considered by the Commission, two NPR licensees were in the process of decommissioning, and one NPR license was terminated.





Under inter-agency agreements with, among others, the U.S. Army, NRC personnel regularly take part in exercises designed to challenge safeguards protections against unauthorized entry. In the photo above left, a would-be intruder attempts to defeat an alarm system by means of various climbing aids. The weapons training provided to licensee security staff is also evaluated, in photo at right.

Regulatory Effectiveness Reviews (RERs) at Power Reactors. The NRC staff, assisted by U.S. Army Special Forces personnel, continued the RER program which is designed to evaluate the practical effectiveness of safeguards for vital equipment at licensed power reactors. RERs are conducted to assure that safeguards programs, as implemented by licensees, would be effective against the design-basis threat for radiological sabotage defined in 10 CFR 73.1. During the fiscal year, RERs were conducted at 14 power reactor units. Partly as a result of NRC Information Notice No. 88-41, which described certain common weaknesses found in previous RERs, the RER teams noted that licensees visited this year had increased their efforts to correct these weaknesses before the arrival of the RER team at their sites. Although RERs continued to identify some weaknesses, as well as strengths, in licensees' programs, upgrades have significantly improved safeguards at many sites. Problems and issues raised in RER reports are resolved through voluntary actions of the licensees or through licensing or enforcement actions, as appropriate.

Reactor Safeguards Inspections. Inspectors from the NRC's five Regional Offices conducted 353 inspections at operating power reactors, 6 pre-operating inspections at facilities where an application for license has been submitted to the NRC, and 17 inspections at NPRs (i.e., test, research, and training reactors). In addition, resident inspectors at operating power reactors continued to augment the safeguards inspection program at their respective sites. Enforcement actions resulting from NRC inspections are treated in Chapter 2 and listed in Appendix 6.

Fuel Cycle Facilities

The number of active licensed non-reactor nuclear facilities subject to NRC safeguards requirements was 15, of which 10 are major fuel fabrication facilities. Of these 15 facilities, four had holdings of formula quantities of SSNM, requiring the implementation of extensive physical security and material control and accountability measures. In June 1989, one of the four, which was previously engaged in the production of fuel for the Fort St. Vrain High Temperature Gas Reactor, reduced possession to less than strategic quantities. Accordingly, the only remaining licensed facilities possessing formula quantities of unirradiated SSNM are the three commercial plants associated with the Naval Reactor Program.

To assure that the physical protection of the SSNM at these facilities is essentially equivalent to the protection afforded like materials in the government sector, the NRC recently upgraded its protection requirements in several areas. Licensees for the three facilities that possess formula quantities have prepared and submitted, for NRC review and approval, plans for installing additional barriers at the site perimeter (including substantial vehicle barriers), conducting periodic tactical drills and exercises (including force-on-force techniques) and arming Tactical Response Teams with heavier firearms. Additional improvements under development include physical fitness training programs for facility guards and expanded firearms qualification requirements. In addition to physical protection regulations, the NRC requires licensees who possess SNM to implement systems for control and accounting of nuclear materials in process and in storage. In fiscal year 1989, licensees who produced low-enriched fuel for power reactors completed implementation of new requirements to, among other things, establish capability for timely detection of a loss of substantial quantities of material. In a corollary action, the NRC also upgraded control and accounting regulations for HEU. Licensees possessing formula quantities of SSNM submitted plans to the NRC for approval of proposed control and accounting systems designed to assure rapid detection of an abrupt loss of five or more kilograms of SSNM. These plans were approved in fiscal year 1989, and implementation is expected by the second quarter of fiscal year 1990.

Inspections at Fuel Cycle Facilities

Comprehensive physical security and material control and accounting inspections were conducted at 10 major fuel fabrication facilities. Special teams also completed performance-oriented physical security inspections at the three remaining facilities possessing formula quantities. Although no items of noncompliance were identified, the teams proposed various ways that security could be improved, both in the short and the long term.

Transportation

Spent-Fuel Shipments. The NRC approved 36 transportation routes as acceptable for protection against radiological sabotage. Forty-two spent-fuel shipments were made over these routes.

Spent-fuel shipping projects included two series of shipments by rail. The first began in fiscal year 1984 and was completed in fiscal year 1989. It entailed the movement of approximately 1,000 fuel assemblies from the Cooper nuclear power plant in Nebraska to the General Electric spent-fuel storage operation near Morris, Ill. Agreement by General Electric to receive and store this material was the confirmed in a fuel supply contract between General Electric and the utility, which has been in existence since the beginning of reactor operations. Receipt of this fuel will essentially fill the Morris pool under its present storage configuration. The second series of shipments began in fiscal year 1989, with the completion of three of 35 programmed shipments from the Brunswick nuclear power plant, Southport, N.C. to the Shearon Harris nuclear power plant near New Hill, N.C. Because the spent-fuel pool at the Shearon Harris plant is configured to store a larger number of spent-fuel assemblies than the Brunswick plant, Carolina Power & Light plans to move approximately 1,170 fuel assemblies to the Shearon Harris pool for storage over a five-year period.

Shipment Route Surveys. NRC regional personnel continued to work with local law enforcement agencies to conduct field surveys of routes proposed for shipments of spent fuel or SSNM. The NRC brochure entitled "Information Package on Spent Nuclear Fuel Shipments for Law Enforcement Agencies" (NUREG/BR-0020) was distributed to local officials and agencies during these surveys.

SSNM Shipments. There were 13 export shipments, 4 foreign shipments which transited the United States,

Drivers and armed guards awaiting the off-loading of a shipment of special nuclear material confirm their roles in the safeguarding of the procedure. The NRC supervises guard and driver training as part of its regulation of each phase in the transport of such materials.

and 9 domestic shipments—each involving less than five but more than one kilogram of HEU. One export shipment involving five or more kilograms of HEU was also made during fiscal year 1989.

Tracking International Shipments of SNM. NRC regulations require licensees to notify the NRC concerning international shipments. During fiscal year 1989, the NRC received approximately 380 such notifications, of which about half were forwarded to the Department of State for appropriate international notification. The number of these actions probably will continue to increase over the next several years.

Transport Inspection and Enforcement. In the safeguards area, the NRC continued to inspect selected domestic segments of import and export shipments of spent fuel. No significant problems were identified from the inspections carried out during this reporting period.

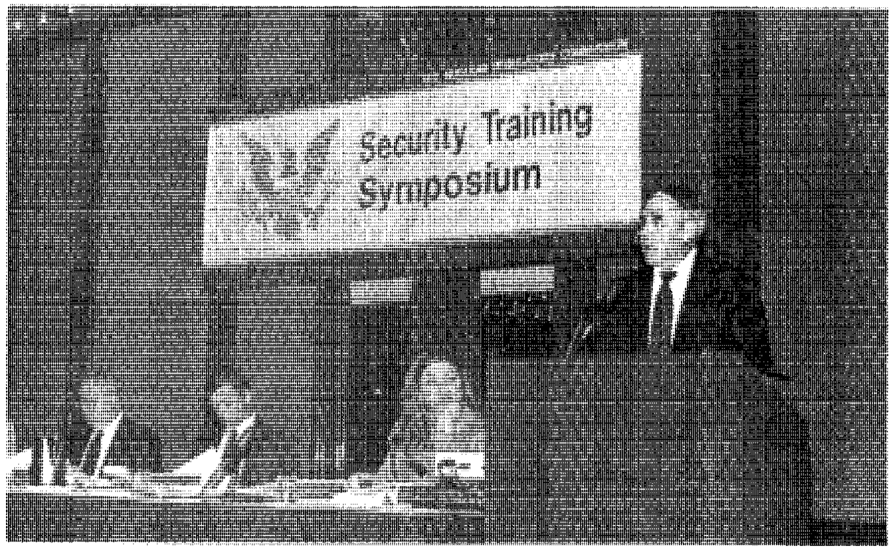
The NRC also continued its transportation-related safety inspection program. The total effort involved more than 1,000 individual inspections covering byproduct, source and SNM licensees; fuel cycle facilities; and shippers of spent reactor fuel.

An inspection program was initiated to assure that transportation containers certified by the NRC are fabricated in accordance with the NRC-approved design and quality assurance program of the container suppliers. Pilot inspections were conducted at six facilities that represented a broad spectrum of the industry. The container-supplier inspection program includes the designers, fabricators, and distributors that have NRC-approved quality assurance programs and certificates of compliance for transportation packages. The inspection program was designed to provide information on whether transportation packages are designed, fabricated, and procured in conformance with 10 CFR Part 71 requirements. The pilot inspections were conducted on a consistent basis using uniform inspection techniques in a comprehensive and systematic manner. The inspection results identified a number of items of nonconformance with the regulatory requirements, and pointed out the need for more inspections in this area.

Plutonium Air Shipment Criteria Development. Section 5062 of Public Law 100-203 imposes requirements on plutonium air transport packages to be used to ship plutonium from one foreign country to another, through U.S. air space. The law requires the NRC to certify the safety of plutonium air-transport-package designs to Congress. During fiscal year 1989, the NRC developed draft criteria for package drop and aircraft-crash testing of a candidate package design for certification by testing. Development of the draft test criteria was requested and funded by the Power Reactor and Nuclear Fuel Development Corporation (PNC), on behalf of the Japanese government. Contract support for this effort was provided by the Lawrence Livermore National Laboratory. A final report on the development of the draft criteria is scheduled for September 1990.

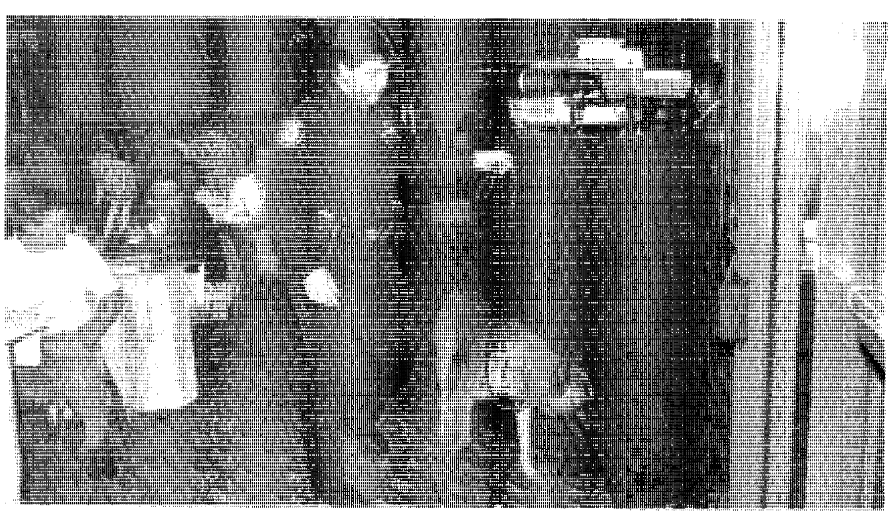
Trupact-II Shipping Container. In August 1989, the NRC approved the TRUPACT-II shipping-container design. This container will be used to transport contact-handled transuranic wastes from DOE facilities to the Waste Isolation Pilot Plant (WIPP) in Carlsbad, N.M. As part of its review, the NRC staff visited several DOE waste-generation and storage sites (Rocky Flats, Idaho National Engineering Laboratory, and Richland Hanford) to observe DOE procedures for waste handling and waste characterization. The staff also observed container testing at the Sandia National Laboratories. Four full-scale containers were subjected to impact, puncture, and fire tests, as specified in NRC regulations.

Incident Response Planning and Threat Assessment. The NRC staff assesses threats to NRC-licensed facilities, materials, and activities, and prepares the NRC's incident response plans for responding to actual thefts of nuclear material or radiological sabotage of nuclear facilities or activities. The staff maintains close and continuing contact with the intelligence community, including participating in regular interagency meetings of Federal agencies concerned with terrorism. As part of these liaison activities, the NRC and the Federal Bureau of Investigation (FBI) initiated a review and revision to their 1979 Memorandum of Understanding (MOU). The MOU is intended to provide guidance and procedures for matters regarding threat information exchange, incident response, and related mutual support. However, since 1979, new legal authorities have been enacted that bear on the roles and responsibilities of the agencies, and operational experience has suggested areas in the MOU which could be clarified or simplified. The revised MOU will be formally promulgated when signed by the NRC Chairman and the FBI Director. Also, on a daily basis, the staff reviews and evaluates intelligence reports on terrorist activities and incidents, and assesses any reported threats against licensees. Particular attention is paid to foreign terrorist groups, their activities, and their possible relationship to activities sponsored by foreign states. On the basis of the NRC's review and interaction with other agencies, the staff formally documents its analysis of the foreign and domestic threat environment every six months, to assure the adequacy of the NRC's current design-basis threat statements. The staff discerned no significant change in the threat environment that would impact the NRC's current safeguards regulations.



Welcoming participants to the Security Training Symposium is Robert F. Burnett, Director of the Division of Safeguards and Transportation in NRC's Office of Nuclear Material Safety and Safeguards. The symposium was held in Bethesda, Md., in November 1989.

At the head of the table (second from left), examining a display of weapons confiscated by the Bureau of Alcohol, Tobacco, and Firearms is Robert M. Bernero, Director of the NRC's Office of Nuclear Material Safety and Safeguards.



Among the on-site demonstrations at the Security Training Symposium was a display of canine olfactory acumen in finding hidden explosives. The participants are members of a U.S. Secret Service Canine Explosives Detector Team.

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. Additionally, the Communicated Threat Credibility Assessment Team, which is jointly funded by the NRC and the DOE, conducts analyses of written or recorded threats.

The staff continued its analysis of safeguards events, to identify trends, patterns, and anomalies. The "Safeguards Summary Event List" (NUREG-0525), a compilation of safeguards events, was revised in July 1989 (Rev. 15) to include events occurring through December 1988. In a related program, commencing in October 1987, licensees began a quarterly submission of Safeguards Event Logs to the NRC. In the two years that these submittals have been made, the staff has been developing a program of trend and pattern analysis to identify safeguards areas needing improvement, help focus regional inspection efforts, and provide feedback to licensees, to stimulate their self-improvement efforts. Since April 1988, the staff has presented briefings to licensees at their regional nuclear security associations, to acquaint them with the analysis program. Regional staffs have also been familiarized with the program. In July and September 1989, licensees and Regional Offices personnel were sent Safeguards Events Logs Facility Analysis Reports covering events reported through the end of June 1989. Similar reports are planned to be sent to licensees and Regional Offices on a periodic basis.

NRC Security Training Symposium. In November 1989, the NRC sponsored a three-day security training symposium on firearms and explosives recognition and detection, attended by approximately 270 persons. The purpose of the symposium was to facilitate a technology exchange between and among NRC staff, those NRC licensees required to perform firearms and explosives searches, and the Federal security community. Guest speakers included firearms and explosives experts from the Federal Bureau of Investigation, Departments of State and Defense, Secret Service, Federal Aviation Agency, Bureau of Alcohol, Tobacco and Firearms, and Sandia National Laboratories. This was a new initiative on the part of the NRC to improve communications among the entire commercial nuclear security community on technical issues and to assist in maintaining the highest standards of excellence with respect to security practice.

NRC/IAEA Interaction. The principal interaction between the International Atomic Energy Agency (IAEA) and the NRC during fiscal year 1989 involved the application of international safeguards to the General Electric Fuel Fabrication Plant in Wilmington, N.C. and the Babcock & Wilcox Nuclear Fuel Company in Lynchburg, Va. At General Electric, the IAEA continued the policy of unannounced random inspections initiated in late 1988 and performed a comprehensive physical inventory verification in August 1989. The Babcock & Wilcox facility was selected for the application of safeguards under the US/IAEA Agreement, during the first quarter of 1989. Routine inspections are being performed at this facility, and a physical inventory verification took place in June 1989. With respect to other areas of NRC/IAEA interaction, the NRC continued to coordinate reporting of accounting data to the IAEA on a monthly basis for Westinghouse (Columbia, S.C.), Advanced Nuclear Fuels (Richland, Wash.), and Combustion Engineering (Windsor, Conn.). An additional activity during 1989 involved NRC participation on a U.S. Government interagency team that provided support to the IAEA in upgrading INFCIRC-225, Rev. 1, "The Physical Protection of Nuclear Material." The NRC provided significant input to the U.S. position paper that served as the basis for many of the upgrades proposed for Revision 2 to INFCIRC-225.

In May 1989, representatives of the IAEA, the NRC and other U.S. agencies met in Washington, D.C., to discuss safeguards issues related to U.S. facilities.

REGULATORY ACTIVITIES AND ISSUES

NRC/DOE Physical Protection Comparability of SSNM

In fiscal year 1989, the NRC completed a review and analysis of NRC requirements for the physical protection of SSNM in transport, assessing them against the protections provided by the DOE Safe Secure Trailer system. (All shipments of formula quantities of SSNM were and are currently being made under the DOE protection system). The NRC concluded that its regulations in this area need to be significantly upgraded to match DOE standards, and the staff was preparing a rulemaking action at the close of the report period. In the meantime, the NRC has asked the DOE to continue making commercial shipments of formula quantities of SSNM, until NRC regulations have been upgraded and a commercial carrier is available and approved for this activity.

Tritium Shipper-Receiver Differences

At the request of the DOE, the NRC provided technical assistance to a DOE investigation of significant shipper-receiver discrepancies of tritium shipped by DOE, under NRC export licenses, to foreign customers. This investigation, includes a review of the alleged discrepancies reported by foreign facilities, as well as the examination of measurement, handling and shipping procedures and operations by both the Oak Ridge National Laboratory and domestic licensed facilities. At the conclusion of the DOE investigation, the NRC will re-examine its regulatory posture on tritium.

IAEA Regulations on Transportation of Radioactive Materials

The NRC has developed a proposed rule modification to 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." The changes, in combination with a parallel revision of the hazardous materials transportation regulations of DOT, will bring U.S. domestic transport safety regulations at the Federal level into conformance with relevant portions of the IAEA design and performance requirements, to the extent considered feasible. This consistency not only facilitates the free movement of radioactive materials between countries for medical, research, industrial, and nuclear fuel cycle purposes, but also contributes to safety by concentrating the efforts of the world's experts on a single set of safety standards and guidance (those of the IAEA) from which individual countries can develop their domestic regulations. The experience of every country that bases its domestic regulations on those of the IAEA can be applied by every other country with consistent regulations, to improve individual safety programs. Since DOT and NRC regulations supplement each other, DOT and the NRC are coordinating their rulemaking activities.

Site Storage of Spent Fuel

As the result of nuclear sites requiring additional storage facilities for spent fuel, the NRC published for public comment a proposed rule allowing a new method of storage of spent fuel at nuclear power plant sites. This method of storage is covered in the Nuclear Waste Policy Act of 1982 and provides for a dry spent-fuel storage program using NRC-approved storage casks.

Access Authorization at Nuclear Power Plants

In April 1989, the Commission directed the staff to prepare a general rule, rather than a policy statement, requiring all power reactor licensees to have an access authorization program. The NRC would also issue a Regulatory Guide which would endorse, with appropriate exceptions, the industry guidelines developed by the Nuclear Management and Resources Council as one acceptable means of complying with the rule. The Regulatory Guide should be ready for issuance at the same time the Commission acts on the final rule. Major elements of the access authorization program include background investigations, psychological evaluations, and the development and implementation of a behavioral observation program.

Fitness-for-Duty at Power Reactors

The Commission published, in the Federal Register (54 FR 24468) of June 7, 1989, a final rule that requires licensees authorized to construct or operate nuclear power reactors to implement a fitness-for-duty program by January 3, 1990. The general objective of the program is to provide reasonable assurance that nuclear power plant personnel are reliable and trustworthy, and not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause that in any way adversely affects their ability to safely and competently perform their duties.

Waste Management

Chapter 6

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

- Developing the criteria and the framework for regulating high-level waste (HLW), including determinination of the technical bases for licensing 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 the States and State compacts pursuant to the safety goals of the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA).
- Providing national program management for licensing, inspecting and regulating uranium recovery facilities and associated mill tailings.
- Reviewing and concurring in significant Department of Energy (DOE) decisions regarding inactive mill tailings sites and the licensing of stabilized tailings piles.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development Activities

A major aspect of the HLW program has been to re-examine and clarify selected areas of the NRC's regulations. Completing this effort prior to receipt of a license application for a repository for disposal of HLW will facilitate the licensing process for all parties involved, including the licensing staff, the DOE, the State, affected units of local government and Indian Tribes, and the adjudicatory boards.

Four rulemaking actions were completed during this reporting period. First, the NRC concluded its negotiated rulemaking to amend the Commission's Rules of Practice in 10 CFR Part 2 on procedures for the submission and management of records and documents related to repository licensing. This final rule was published in April 1989. After publication of this rule, the Commission appointed an Administrator for the Licensing Support System—the electronic information management system established as part of this rulemaking action for the licensing proceeding.

In September 1989, the NRC published new proposed amendments to 10 CFR Part 2 facilitating NRC compliance with the schedule for issuing a decision on construction authorization while allowing for a thorough technical review of the license application and equitable treatment of parties to the hearing.

In July 1989, the NRC published a final rule amending 10 CFR Parts 51 and 60. The purpose of this rulemaking is to set the standards and procedures that will be used by the NRC in determining whether adoption of the DOE's environmental impact statement (EIS) is practicable, as provided under the Nuclear Waste Policy Amendments Act (NWPAA). Under the new rule the NRC will find it practicable to adopt DOE's EIS unless the action that the NRC proposes to take differs in an environmentally significant way from the action described in DOE's license application, or significant and substantial new information or new considerations render the DOE EIS inadequate.

Finally, in September 1989, the NRC issued a proposed amendment to 10 CFR Part 51 on the timing of availability of a repository and the environmental impacts of storage of spent fuel at reactor sites after the expiration of reactor operating licenses. This amendment would conform Part 51 to proposed revised findings in the NRC's 1989 review of its "Waste Confidence" decision.

The staff continued to follow developments on Environmental Protection Agency (EPA) revision of its standards on the manangement and disposal of high level radioactive waste. The NRC will conduct its rulemakings on the conformance of Part 60 and the implementation of EPA standards after they are promulgated.

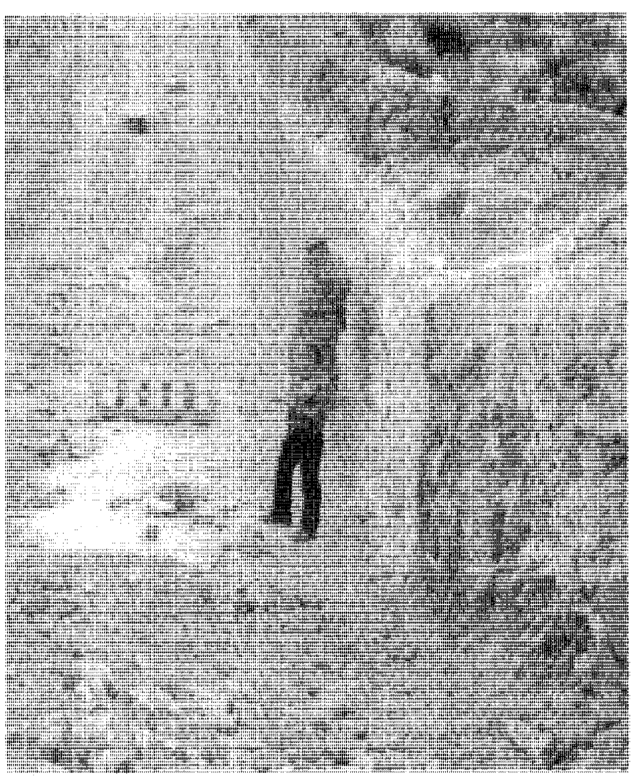
Regulatory Guidance Activities

The staff is continuing to conduct an active program to identify uncertainties in the regulatory framework and to develop regulatory requirements and guidance to resolve these uncertainties. Technical Positions (TP) are key mechanisms for providing guidance to DOE and are focused on staff criteria for acceptable methods of demonstrating compliance with 10 CFR Part 60.

The following TPs were published in final form or in draft form for public comment during fiscal year 1989:

- Final TP—"Postclosure Seals, Barriers, and Drainage Systems in an Unsaturated Medium" (NUREG-1373).
- Draft TP—"Tectonic Models under 10 CFR Part 60."
- Draft TP—"Methods of Evaluating Seismic Hazard at a Geologic Repository."

In addition to the staff effort to re-examine and clarify regulatory requirements and identify areas where guidance may be needed, the NRC's Center for Nuclear Waste Regulatory Analyses (CNWRA) identified and made recommendations on regulatory and institutional uncertainties with respect to specific DOE activities and the importance and timing of resolution. The staff will use the CNWRA's recommendations on ways to resolve uncertainties in deciding whether to pursue further rulemakings or to develop regulatory guidance documents.



The photo is of a trench at the proposed Yucca Mountain (Nev.) site for a permanent high-level nuclear waste repository. The repository would comprise a 1,500-acre grid of tunnels deep inside the mountain. Examining the calcite/silica deposits of the trench are an NRC-representative and a consultant from the DOE's Lawrence Livermore Laboratory.

Yucca Mountain Site Characterization Analysis

Under the NWPAA, DOE is required to submit to the NRC for review and comment a general plan for site characterization activities to be conducted at the candidate site before sinking a shaft. In January 1988, DOE issued the Consultation Draft Site Characterization Plan (CDSCP) for the Yucca Mountain, Nevada Site. The NRC provided objections, comments, and questions on the CDSCP in its final point papers in May 1988.

The DOE provided the Yucca Mountain, Nevada Site Characterization Plan (SCP) to the NRC on December 28, 1988. In its Site Characterization Analysis (SCA) of the SCP for the Yucca Mountain Site, issued on July 31, 1989, the NRC staff found that two of its five objections on the CDSCP remained unresolved. These related to DOE's not having a baselined quality assurance (QA) program in place and to the adequacy of both the exploratory shaft facility (ESF) Title I design control process and the design.

Of the 162 comments and questions that the NRC staff raised about the CDSCP, 103 were satisfactorily resolved. Of the remaining 59, many were partially resolved. These 59 have been incorporated into the 196 SCP comments and questions, all of which will be tracked as open items until they are resolved by means of information in SCP progress reports, other DOE documents, or by interactions between DOE and NRC staff.

Finally, the staff restated a programmatic concern raised by the Commission on the Draft 1988 DOE Mission Plan Amendment, that pressure to meet unrealistic schedules may leave DOE insufficient time for site characterization and for developing a complete and high-quality license application.

State Interactions

The NRC continues to include the State of Nevada and the three counties designated as affected units of local government as participants in the high-level waste program. The Commission held a public meeting with the State in December 1988. State representatives participated in numerous NRC-DOE technical interactions and in DOE QA audits observed by the NRC staff during fiscal year 1989. Items of interest to the State and local governments are included on the agenda for all NRC-DOE meetings, and the NRC routinely involves the State in all other interactions. In addition, the NRC informs the State, local governments, and potentially affected Indian Tribes of all Commission meetings and meetings of the Advisory Committee on Nuclear Waste (ACNW) on the high-level waste program.

As requested by the State of Nevada, NRC staff reviewed the "QA Manual for the Nevada Agency for Nuclear Projects/Nuclear Waste Project Office" and found it acceptable and consistent with NRC regulations. Unlike staff review of DOE QA programs, the review of the Nevada QA manual was not a requirement, but was carried out under a policy of cooperation with the State to help guide it in developing high-quality data which potentially may be used during the licensing hearings on the repository. The NRC is not required by the NWPAA to review technical activities carried out by the State in connection with DOE's characterization of the Yucca Mountain site.

On July 6, 1989, Acting Governor Robert Miller of Nevada signed legislation making it illegal to store nuclear waste anywhere in the State of Nevada. It is not clear what effect this will have on the repository program.

Quality Assurance Activities

The NRC staff's objective in its review of the DOE QA program is to establish confidence that work performed during site characterization is appropriately controlled and defensible in licensing before site characterization begins. The staff's QA review is divided into a review of DOE QA plans and procedures (document reviews) and evaluations of DOE's effectiveness in auditing its program to identify and correct problems in program implementation.

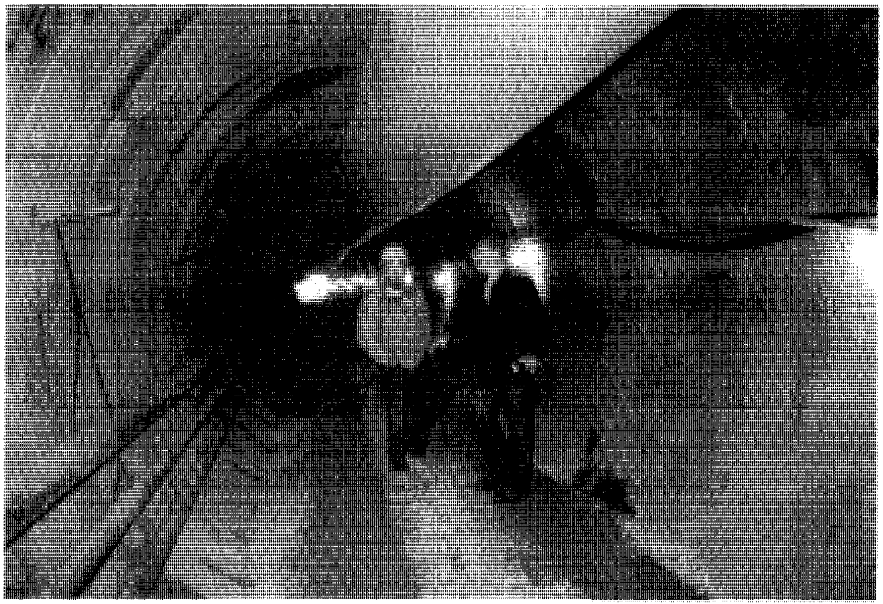
As stated earlier, in December 1988, DOE submitted the SCP for NRC staff review. The SCP contains general information on DOE's QA program including QA organizations, regulations, activities covered by the QA program and references to more detailed QA plans and procedures. In fiscal year 1989, the staff not only reviewed the QA information provided in the SCP but also conducted reviews of the detailed QA plans for all of the Yucca Mountain program contractors and provided formal comments to DOE.

To conduct its evaluation of DOE's effectiveness in auditing, the NRC staff conducted eight observation

NRC staff members are shown observing the taking of core samples of volcanic tuff—the same material widely found at the proposed high-level waste repository site at Yucca Mountain (Nev.)—derived from drilling at the Apache Leap site near Superior, Ariz.

audits, using teams composed of technical and QA staff from the NRC and QA staff from the Center for Nuclear Waste Regulatory Analyses (CNWRA). The DOE audits were conducted at all major contractor organizations participating in the site characterization program for the Yucca Mountain Project. Formal staff reports were issued for all of the audit observations, and DOE will have to respond to those where improvements are needed in the audit process.

The staff also reviewed and commented on the QA requirements document for glass waste producers (the West Valley Demonstration Project and Defense Waste Processing Facility at Savannah River). Each glass-waste-form producer and the major participating organizations must have a QA program that meets the applicable requirements in the QA requirements document.



An NRC geologist visits a water tunnel in Colorado Springs, Colo., to observe underground excavation with a Tunnel Boring Machine. The proposed high-level waste repository calls for a grid of tunnels deep below ground and sealed off from the environment.

Waste Confidence

In August 1984, the NRC issued its "Waste Confidence" decision. In its decision, the Commission made five findings on: (1) the technical feasibility of disposal; (2) the timing of repository availability and sufficient disposal capacity; (3) safe management of wastes until a repository is available; (4) duration of safe storage; and (5) ability to provide additional storage capacity, if needed. The Commission committed itself to reviewing its findings at least every five years, until a repository for high-level waste is available.

In September 1988, the Waste Confidence Review Group was established to carry out the first five-year review of the original decision. The Review Group provided the Commission its Proposed 1989 Waste Confidence Decision and Conforming Amendments to 10 CFR Part 51, in June 1989. In September 1989, the Proposed Decision and Proposed Rulemaking were published for comment.

The proposed decision would revise two of the findings so that the timing of repository availability would be extended to the first quarter of the 21st century, and the duration of safe storage would be revised to cover 30 years beyond the licensed life for operation of a reactor (which may include the term of a renewed or extended operating license). The proposed amendment to 10 CFR Part 51 essentially conforms the regulation to these two revised findings. A final decision and rulemaking are planned for mid-1990.

Center for Nuclear Waste Regulatory Analyses

The Center for Nuclear Waste Regulatory Analyses (CNWRA) completed its second year of operation on October 14, 1989. The NRC originally envisioned a three-year "phase-in" plan for the establishment of the Center and the transfer of essentially all NRC technical assistance work from existing contractors. However, the NRC accelerated the plan so that, by the end of the second year, nearly all of its technical assistance work had been transferred to the CNWRA.

The level of support that the Center provided to the NRC increased throughout the second year. The CNWRA continued to develop its technical and analytical capabilities, taking the following steps: the hiring of additional technical staff; continuing work on four research projects and the three-year transportation risk study begun in its first year; and advancing the ongoing systems engineering program to assure that all NRC high-level waste activities required under the NWPA, as amended, are optimally planned integrated, implemented, documented and managed. The CNWRA provided technical support to the NRC staff by recommending regulatory requirments that should receive priority attention during the NRC's review of DOE's SCP; assisting the NRC in resolving technical concerns raised in the NRC's comments on DOE's CDSCP (such as those raised regarding the exploratory shaft); assisting in the NRC review of DOE's SCP, including the description of the exploratory shaft facility; assisting in QA observation audits; providing technical support in developing NRC technical positions and rulemakings, and initiating two new research projects.

Revised Schedule for the Repository Program

The Department of Energy (DOE) issued the "Report to Congress on Reassessment of the Civilian Radioactive Waste Management Program" in November 1989. In the report, DOE announced several extensions in the schedule for major repository program milestones. For example, the earliest date for a repository to be available is 2010, 12 years later than the 1998 date established in the Nuclear Waste Policy Act. The NRC will be conducting reviews and interacting with DOE to support DOE activities planned for its restructured program. The NRC's goal will continue to be the fulfillment of its responsibilities under the NWPAA in a manner that will not unnecessarily delay implementation of DOE's restructured program.

LOW-LEVEL WASTE MANAGEMENT

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

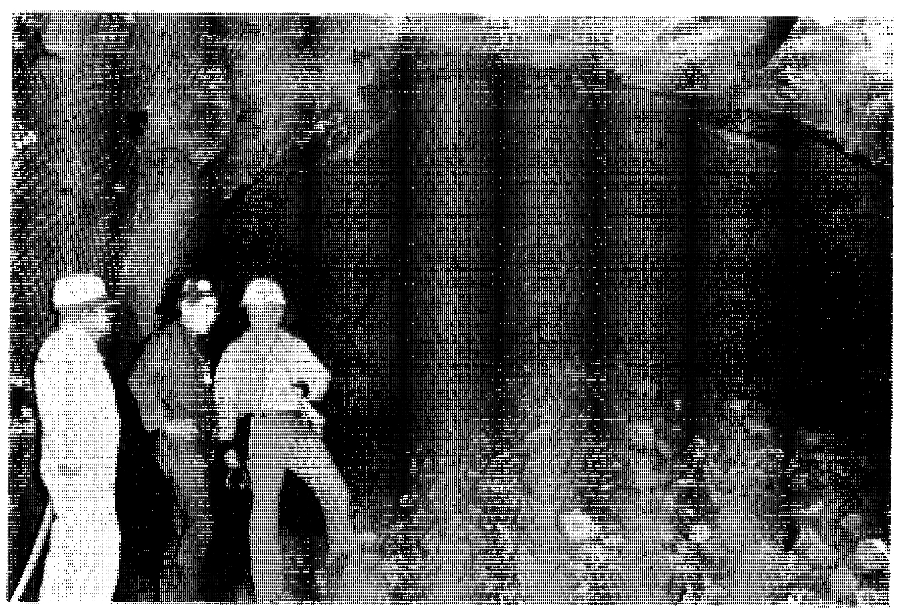
The NRC staff has developed the rules and guidance documents required under the LLRWPAA, including a rule on emergency access and a proposed rule on data base management for low-level waste (LLW) disposal activities. In addition, the Act calls for the NRC to coordinate with other Federal agencies and to provide technical assistance to the States on issues related to development of LLW disposal capacity. For example, the staff provided a review of a prototype LLW disposal facility and reviewed Department of Energy (DOE) plans for stabilizing LLW from the West Valley Demonstration Facility.

Regulation and Guidance

The NRC staff continued its efforts to develop regulations and to provide guidance that will assist the States and State compacts in developing the LLW disposal capacity that the LLRWPAA requires.

Section 6 of the LLRWPAA authorizes the NRC to grant emergency access to any non-Federal LLW disposal facility, when necessary to eliminate an immediate and serious threat to the public health and safety or to the common defense and security. On February 3, 1989, the NRC issued a final rule (10CFR Part 62) establishing criteria for making determinations on requests for emergency access. This regulation became effective on March 6, 1989.

The Commission initiated steps to strengthen its oversight of import and export of low-level radioactive wastes. An Advance Notice of Proposed Rulemaking, to solicit public comments on several options the Com-



NRC engineers appraise the effects of an underground excavation test blast, using the "smooth wall" blasting technique, at the Nevada Test Site. Such tests are important to work on the proposed high-level waste repository.

mission is considering to amend existing regulations governing the import and export of radioactive wastes, has been prepared and will be issued early in fiscal year 1990.

In 1989, the NRC initiated a rulemaking to amend Parts 20 and 61 to: (1) augment and improve the quality of information contained in manifests accompanying shipments of LLW to disposal facilities; (2) require operators of licensed LLW disposal facilities to store this manifest information in computerized record-keeping systems; and (3) require operators to routinely submit to the NRC, in an electronic format, reports of manifest information. These amendments will ensure that the chain of custody for LLW can be tracked—from generation, through processing and disposal. The amendments also will ensure the availability of information needed for safe operation of LLW disposal facilities, and will remove roadblocks to complete development of a national computer data base containing waste manifest information from all disposal facilities.

A final rule was published in May 1989, amending 10 CFR Part 61 to require disposal of "Greater than Class C" wastes in the deep geologic repository for high-level waste, unless the Commission has approved disposal elsewhere. The final rule obviates the need for altering existing classifications of radioactive wastes as high-level or low-level.

Technical Assistance to States

The NRC staff provided assistance to a number of State programs related to LW management. This assistance included: (1) participation in Agreement State program reviews; (2) response to specific inquiries related to Governors' Certifications; (3) development of guidance designed to facilitate State regulation of LLW disposal; and (4) the NRC staff sponsorship of a workshop for State regulators of LLW disposal.

Governors' Certifications Update. Section 5(e) of the LLRWPAA sets forth milestone requirements for States and State compacts to ensure continued access to operating LLW disposal facilities. To meet the January 1, 1990 milestone, Governors may submit to the NRC a written certification that theState will provide for storage, disposal, or management of LLW generated in that State after 1992. The LLRWPAA requires NRC to transmit the certifications to Congress and publish them in the Federal Register. NRC staff estimated that as many as 33 States would submit Governor's Certifications. States or State compacts that do not meet the January 1, 1990 milestone may lose surcharge rebates from the DOE, as well as access to existing LLW disposal facilities.

On February 10, 1989, the Commission sent identical letters to the Governors of the 33 States subject to the 1990 milestone. The letter provided technical and administrative guidance to the States and State compacts, suggesting that Governors' Certifications include the following types of information:

(1) Estimates of the volume and type of waste and who will generate it after December 31, 1992.

(2) A description of proposed storage, disposal or management actions for LLW, including mixed LLW, generated within the State or compact.

(3) A statement that proposed actions are within existing legal authorities and are consistent with the NRC or Agreement State regulations and guidance.

(4) Implementation provisions for the proposed action including organizational responsibilities, timing and schedules.

This administrative guidance to States was designed to ensure that the certifications can be processed promptly upon receipt by the NRC. The staff has encouraged Governors to file their certifications as early as possible; has offered opportunities for consultation on the guidance; and has offered to review programs, if requested by the Governor. The guidance was published in the Federal Register on February 22, 1989 (54 FR 7616).

After issuing the February 1989 guidance, the NRC received several requests for further clarification. In response, on August 9, 1989, letters were sent to all State liaison officers and designated representatives of LLW compact officials and Agreement State regulatory agencies. These letters clarified the NRC's position on: (1) the extent of State responsibilities, where States are planning to rely on generators to store wastes after 1992; (2) States' abilities to file complete license applications for disposal facilities in 1992, even if a State may not yet have an authorized State agency for the regulation of mixed wastes; and (3) the need for States submitting joint certifications to define clearly their separate and discrete plans for managing wastes after 1992.

LLW Disposal Regulators' Workshop. On September 7 and 8, 1989, the NRC's Office of Governmental and Public Affairs (GPA) hosted a workshop with Agreement State regulatory staff who will be involved in licensing LLW disposal facilities. NMSS staff had lead responsibility for conducting the technical discussions. The workshop provided the NRC and Agreement State staffs an opportunity to exchange information and to improve their technical licensing reviews. Regulatory staff attended from 15 States which either currently regulate the disposal of LLW or are developing licensing programs pursuant to the LLRWPAA.

The workshop was organized to provide an opportunity for candid dialogue among Agreement States and NRC staff on technical, licensing and regulatory issues. Participants were especially interested in recent NRC staff experience in reviewing the earth mounded concrete bunker (EMCB) Prototypex License Application Safety Analysis Report (PLASAR), and also in having the benefit of the NRC's technical expertise in evaluating adequacy of wasteforms. NRC staff in turn benefitted from the State staffs' perspective on a number of topics, including mixed waste and performance assessment issues. This was the second NRC-sponsored LLW regulatory workshop. In both workshops reactions from all participants have been very positive.

Work with Other Federal Agencies

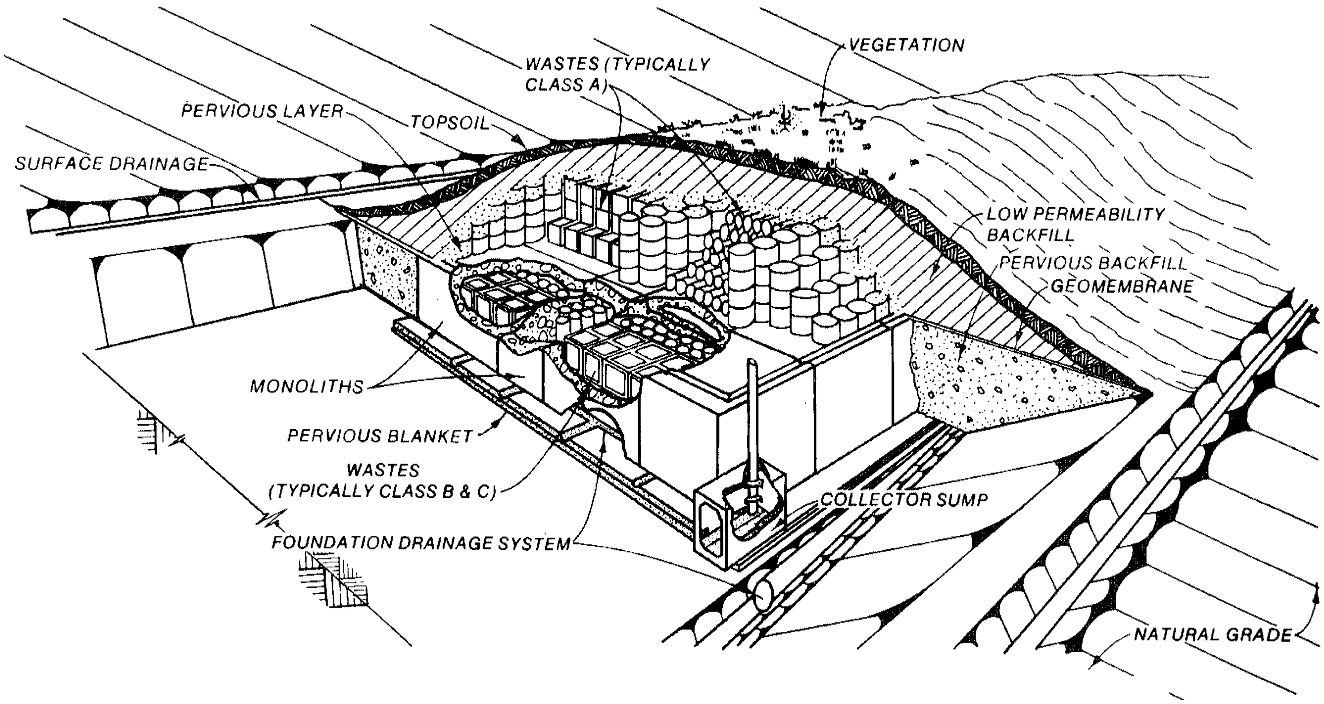
The NRC staff continued to work with the DOE and the Environmental Protection Agency (EPA) in resolving LLW management issues. Interaction with DOE has focused on providing guidance to States on meeting the requirements of the LLRWPAA. In addition, the NRC has provided reviews of DOE's waste stabilization efforts at the West Valley Demonstration Project. The NRC and EPA staffs continued to work on resolving the mixed low-level radioactive and hazardous waste ("mixed waste") issue. The joint guidance on the definition of mixed waste was formulated and issued in October 1989. The NRC also provided comments to EPA on a number of proposed standards, such as the Clean Air Act and the standard on inactive mill tailings.

NMSS Review of Prototype License Application Safety Analysis Report (PLASAR). In response to requests from Agreement States for additional licensing and technical guidance on engineered alternatives, the NRC agreed to perform a "prototype licensing review" of two soil-covered designs that the NRC considered to be most viable from a licensing standpoint. These include: (1) the earth mounded concrete bunker (EMCB) and (2) the below-ground vault (BGV). The NRC staff performed a technical review of the EMCB Prototype License Application Safety Analysis Report (PLASAR), which was prepared for DOE under the Department's technical assistance program to States and compact regions (established in response to the LLRWPAA). The EMCB PLASAR was prepared by Ebasco Services Inc. (Ebasco), under contract to EG&G Idaho Inc., DOE's lead contractor for its LLW management program. The NRC staff review of the EMCB PLASAR was published as NUREG-1375 in July 1989. The NRC staff was reviewing the BGV PLASAR, prepared by Rogers & Associates Engineering Corporation, at the close of the report period.

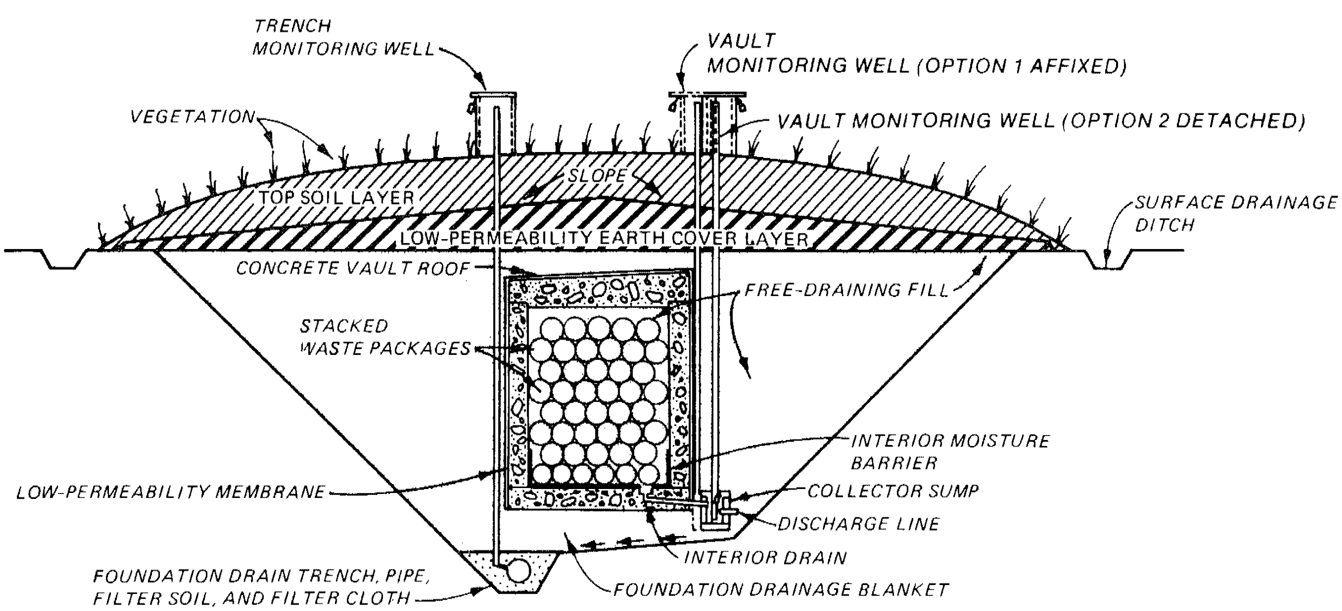
The primary objective of the EMCB PLASAR review was to provide assistance to the States and regional compacts by: (1) identifying acceptable and unacceptable alternative design features and concepts, and (2)demonstrating, by example, how to use the NRC's Standard Review Plan (NUREG-1200). It was also recognized that the prototype licensing review would provide the NRC staff with valuable practical experience in using the NUREG-1200 Standard Review Plan to conduct a licensing review of a low-level radioactive waste facility. Because the emphasis of the review, from the start, was focused on the need to provide assistance on the specific technological aspects of engineered alternative methods of disposal, not on near-surface disposal in general, the review concentrated on the design and operations-related portions of the EMCB study.

NRC staff expended approximately two staff years of efforton the review and contractor staff performed another half staff year of work. A summary report of the review effort was prepared and is entitled "Safety Evaluation Status Report" (SESR). The SESR follows, to the extent practicable, the format and content of a final safety evaluation report (SER) that would be developed for an actual application review. The SESR describes the adequate and inadequate aspects of the information provided in the PLASAR and the basis for conclusions. The NRC staff considers the completed sections in the SESR to be good examples of the safety assessments that are a necessary part of a licensing review, in support of the technical bases for regulatory acceptance or rejection and defensible before a licensing review board.

West Valley Demonstration Project. The NRC staff and its contractors, Brookhaven National Laboratory and the National Institute of Standards and Technology, reviewed several documents describing the cement solidification of a decontaminated supernatant (liquid) waste that resulted from the treatment of HLW at the West Valley Demonstration Project, near Buffalo, N.Y. The DOE had provided the documents for the NRC's review, in an attempt to demonstrate that the cement solidification processing at the West Valley site would meet the long-term (300 years) structural stability provisions of 10 CFR 61.56 and the relevant portions of the May 1983 NRC Staff Technical Position on Waste Form. The NRC's role, as provided for in the September 1982 Memorandum of Understanding between the NRC and DOE, is to review and provide consultation to DOE on any potential radiological danger to the public health and safety which may be presented by the West Valley Demonstration Project.



The NRC has agreed to perform "prototype licensing reviews" of two soil-covered designs for low-level waste repositories. One is the earth-mounded concrete bunker (EMCB), above, in which wastes would be separated according to the level of radioactivity. The other, below, is the below-ground vault (BGV), with stacked waste packages covered over by a concrete roof and layers of earth.



The supernatant is a waste product from the spent-fuel reprocessing plant at West Valley and is decontaminated by an ion-exchange process using zeolite to remove most of the dissolved cesium from the liquid waste. The majority of the reprocessed liquid waste, approximately 600,000 gallons consists of water and dissolved solids—predominantly sodium nitrate and sodium nitrite. The retention tank also contains a hard bottom sludge layer, about 21 inches in depth, that is highly radioactive. Removal, treatment and processing of the bottom sludge layer into borosilicate glass is not anticipated to begin until 1992, following completion of the solidification of the decontaminated supernatant waste. The supernatant is chilled from 180°F to 62°F and diluted before passing it through the zeolite beds in the supernatant treatment system. After the process of dilution and decontamination, the supernatant is sent to the liquid waste treatment system, where it is concentrated to 39 weight-percent in an evaporator. The 39 weight-percent decontaminated supernatant is then sent to the cement solidification system to be mixed with Portland Type I cement and selected additives in a high shear mixer. The resulting waste cement mixture is then pumped into square carbon steel 71-gallon drums that are transferred into a shielded truck for transportation and storage in the constructed drum cell facility. The number of drums of solidified waste to be produced is not expected to exceed 15,000. The ultimate permanent disposal of the solidified decontaminated supernatant waste is to be decided after completion of the DOE Environmental Impact Statement for the West Valley Demonstration Project.

NRC staff completed a Technical Evaluation Report on the proposed cement solidification process in November 1988. The staff's report describes the DOE and NRC efforts that culminated in the NRC's conclusion that there is reasonable assurance that cement solidification of the decontaminated supernatant waste will meet the waste form stability requirements of 10 CFR Part 61, and fulfill provisions of the NRC's technical position on waste form. The conclusion on reasonable assurance is predicated on the expectation that the ongoing short- and long-term testing programs will continue to show favorable and acceptable test results.

URANIUM RECOVERY AND MILL TAILINGS

Under this program area, the NRC licenses and regulates uranium mills, commercial in-situ solution mining operations, and uranium extraction research and development projects. The NRC also evaluates and concurs in DOE's remedial action plans for inactive uranium mill tailings sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act of 1978.

Regulatory Development

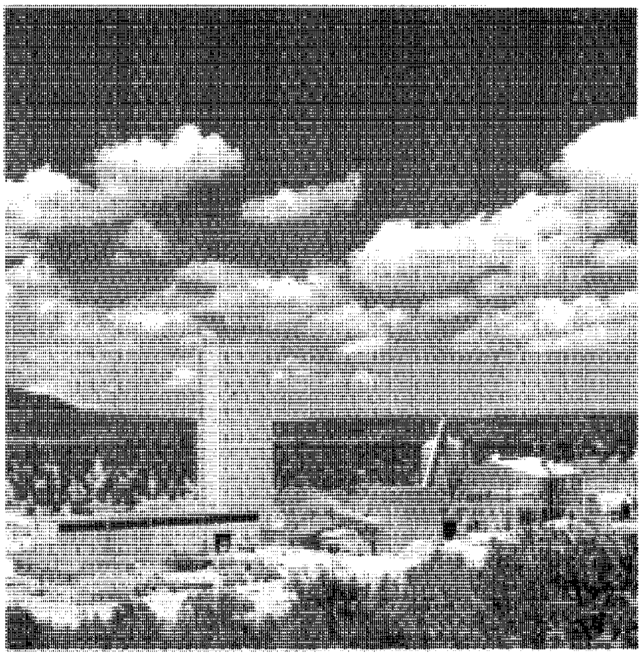
The Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), which was enacted to prevent or minimize environmental hazards from active or inactive mill operations, requires the EPA to develop radiation standards for mill tailings sites and the NRC to develop regulations for uranium recovery operations. The final EPA standards for inactive sites were issued in October 1983. The NRC then embarked on a two-step process to conform its regulations (10 CFR Part 40) to these standards. The first step, completed in October 1985, was modification of NRC regulations on radiological protection and long-term stabilization of mill tailings sites, to bring them into congruence with the EPA standards. The second step incorporated the EPA groundwater protection standards. The NRC's final rule addressing groundwater protection was published November 13, 1987. The NRC also developed a proposed rule for licensing the custody and long-term care of uranium mill tailings sites covering commercially licensed as well as Uranium Mill Tailings Remedial Action Program (UMTRAP) sites. The draft rule should be issued early in fiscal year 1990.

In 1989, NRC staff worked with DOE and EPA in implementing EPA's proposed groundwater protection standards at inactive uranium mill tailings sites. Section 84a(3) of the Atomic Energy Act requires that the NRC's regulations for mill tailings be comparable to the EPA's requirements that are applicable to similar wastes under the Solid Waste Disposal Act, as amended. The NRC completed an initial evaluation of the two regulatory frameworks as a first step in determining whether additional rulemaking is needed to achieve comparability. The EPA has this assessment under review.

The NRC staff continued its work on regulatory guidance for uranium recovery operations by issuing a final Regulatory Guide on calculation of radon flux attenuation by earthen covers, and by issuing for comment a draft technical position on design of erosion-protection covers.

Licensing and Inspection Activities

The NRC's Uranium Recovery Field Office (URFO) performed 37 inspections of uranium recovery facilities. In other regulatory actions, the URFO staff completed 2 license renewals, 26 major license amendments, and 49 minor license amendments.



The facility pictured is a uranium mining and milling operation, owned by the Rio Algom Corporation, in Utah. The milling of uranium is subject to NRC regulation, with particular regulatory concern for the radioactive "mill tailings" or waste products of the milling process.

Of the 29 NRC-licensed uranium recovery facilities, 19 are uranium mills, 3 are either heap leach or other byproduct recovery operations, 4 are research and development solution mining operations, and 3 are commercial in-situ facilities. Only eight of the licensed facilities were in operation at the end of fiscal year 1989: three uranium mills, two research and development solution mining operations, two commercial-scale solution mining facilities, and one secondary recovery operation. Given the economic condition of the uranium industry, few new facilities are expected to be licensed in the near term, except for solution mining operations. The NRC has five new commercial-scale solution mining applications under review, and two more are expected in fiscal year 1990. Over the next few years, much of the casework confronting the uranium recovery program will be in the areas of remedial activity and decommissioning, including remedies for groundwater contamination.

Remedial Action at Inactive Sites

The NRC continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive mill tailings sites, as required by Title I of the UMTRCA. The NRC is a cooperating agency and is required by UMTRCA to concur in DOE's selection and performance of remedial actions at 24 inactive mill tailings sites. During fiscal year 1989, pursuant to this responsibility, the NRC staff completed 57 review actions. These included 2 Remedial Action Plan (RAP) reviews, 3 design reviews, 9 inspection plan reviews, 2 RAP modification reviews, 20 other site-specific reviews, and 18 reviews of generic items related to the program. In addition, the NRC staff prepared three Technical Evaluation Reports documenting its review of DOE's remedial action selection for the Riverton, Wyo.; Tuba City, Ariz.; and Spook, Wyo. sites. Inspections of remedial action construction activities were performed at the Lakeview, Ore.; Green River, Utah; and Spook, Wyo., sites, and NRC technical staff conducted additional site visits at the Durango, Rifle, and Grand Junction, Colo., and Green River, Utah, sites.

During the past year, the NRC staff examined ways to streamline the UMTRA Program review and concurrence process. The NRC and DOE reached several distinct agreements that will increase efficiency for future UMTRA Program documentation and review. In support of the agreements, the NRC prepared and issued a Staff Technical Position on Standard Format and Content for Documentation of Remedial Action Selection at Title I Uranium Mill Tailings Sites.

DECOMMISSIONING OF NUCLEAR FACILITIES

NRC staff activities have continued to focus on developing the guidance that licensing staff and licensees need to implement amendments to Commission regulations for decommissioning nuclear facilities. These amendments pertain to planning, financial assurance and record-keeping for decommissioning, and procedures for terminating licenses. Although the requirements became effective on July 27, 1988, holders of existing licenses subject to the financial assurance requirements have until July 27, 1990 to provide the necessary financial assurance certifications.

Guidance Documents

The guidance documents that NMSS developed in fiscal year 1989 include Revision 1 of the Standard Format and Content Guide (NUREG-1336), and Standard Review Plan (NUREG-1337) for document preparation and review of financial assurance arrangements. NUREG-1336 is being developed as a draft Regulatory Guide to be issued for public comment in fiscal year 1990. The NRC staff is preparing a Standard Review Plan (SRP) for the review of preliminary decommissioning plans for reactors (which must be submitted five years before projected end of operations) and for the review of decommissioning plans for reactors (submitted at the time of termination of operations). These SRPs will provide information to licensees and NRC staff on methods for review of the licensee submittals. They are based on the transfer of responsibility for decommissioning from the Office of Nuclear Reactor Regulation (NRR) to NMSS.

Reactor Decommissioning

NMSS staff has continued to assist NRR licensing staff in reviewing decommissioning plans for power reactors that have been shut down. The staff developed and has implemented a protocol for the transfer from NRR to NMSS of licensing responsibility for power reactors, after approval of a decommissioning plan and issuance of a possession-only license.

Under the new protocol, licensing responsibility for the Humboldt Bay Unit 3, Vallecitos, and Fermi Unit 1 inactive nuclear facilities was transferred from NRR to NMSS in fiscal year 1989. A dismantlement plan submitted for the Pathfinder facility, which had been partially decommissioned in 1970, was under NMSS review at the close of the report period. The Public Service Company of Colorado has informed the NRC that it plans to decommission the Fort St. Vrain plant, and the NRC staff was reviewing its Preliminary Decommissioning Plan. In addition, Sacramento Municipal Utility District notified the NRC of its intent to decommission the Rancho Seco facility, and Long Island Lighting Company was working out plans with the State of New York for the decommissioning of the Shoreham nuclear power plant (see Chapter 9), at the end of the fiscal year.

ADVISORY COMMITTEE ON NUCLEAR WASTE

In May 1988, the Commission approved the establishment of an Advisory Committee on Nuclear Waste (ACNW). By its charter, the ACNW is to "report to and advise the Nuclear Regulatory Commission (NRC) on all aspects of nuclear waste management, as appropriate, within the purview of NRC's regulatory responsibilities. The primary emphasis will be on disposal of high-level nuclear waste but will also include other aspects 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 specific areas of concern referred to it by the Commission or designated representatives of the Commission, and it is authorized to undertake other studies and activities on its own initiative, as appropriate to carry out its responsibilities."

ACNW reports (except for any 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, which is drawn from scientific and engineering disciplines includes individuals experienced in geosciences, radiation protection radioactive waste treatment, environmental engineering, nuclear engineering, and chemistry. (See Appendix 2 for a list of current members.)

During fiscal year 1989, the ACNW reported to the Commission and others on a variety of issues, including:

- The Proposed Deletion of Section 20.205 from the Proposed Revision of 10 CFR Part 20, "Standards for Protection Against Radiation" (SECY-88-315).
- Draft Generic Technical Position: Guidance for Determination of Anticipated Processes and Events and Unanticipated Processes and Events.
- Advance Notice of the Development of a Commission Policy on Exemptions from Regulatory Control for Practices Whose Public Health and Safety Impacts are Below Regulatory Concern.
- West Valley Demonstration Project.
- Final Rulemaking on 10 CFR Part 61 Relative to Disposal of Greater-Than-Class C Low-Level Radioactive Wastes.
- Proposed Waste Confidence Decision by the Waste Confidence Review Group.
- Draft Technical Position on Postclosure Seals in an Unsaturated Medium.
- Proposed Commission Policy on Exemptions from Regulatory Control.
- Management of Mixed Hazardous and Low-Level Radioactive Wastes (Mixed Wastes).
- NRC Comments on DOE Site Characterization Plan for the High-Level Waste Repository.
- Reporting Incidents Involving the Management and Disposal of Low-Level Radioactive Wastes.
- NRC Analysis of the DOE Site Characterization Plan for the High-Level Waste Repository.
- Technical Position Paper on Environmental Monitoring of Low-Level Radioactive Waste Disposal Facilities.

In performing the reviews and preparing the reports cited above, the ACNW held nine full committee meetings and one working group session.

Communicating With Government And The Public

Chapter 7

The NRC communicates in a variety of ways with a broad spectrum of governmental entities, domestic and international, and with the general public. Several NRC Headquarters Offices and the Regional Offices participate in the dissemination of information on NRC activities, by various means and at numerous locations throughout the country. The Commissioners and senior management frequently take part in Congressional hearings (see Table 1, below) and appropriate Congressional Committees are kept informed of NRC actions and decisions on a regular basis. Liaison with Federal and State agencies, Indian Tribes and local community organizations, the news media, Congress and the international community is provided by the NRC Office of Governmental and Public Affairs (GPA).

PUBLIC COMMUNICATION

Commission Meetings

The five 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 rare occasions when the Commission decides that a meeting should be closed. In general, that decision will be taken only if the meeting deals with one or more of the following subjects: classified documents, internal personnel matters, information that is confidential by statute, trade secrets, personal privacy, investigatory records, or adjudicatory matters, as specified under the Government in the Sunshine Act. Members of the public are not allowed to participate in Commission deliberations unless specifically requested to do so by the Commission.

Transcripts of open 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 posted in the lobby of NRC's Headquarters and in the Public Document Room disclosing the time, place, subject matter of the meeting, and whether it is an open or closed meeting, and giving the name and telephone number of the official designated to respond to requests for information about the meeting. Information regarding scheduled Commission meetings for the upcoming four-week period is also available by a recorded telephone message, on (301) 492-0292.

Advisory Committee Meetings. Under provisions of the Federal Advisory Committee Act, the Nuclear Regulatory Commission seeks and receives advice and recommendations from a number of standing advisory committees, such as the Advisory Committee on Reactor Safeguards and the Advisory Committee on Nuclear Waste (see Chapter 2), as well as certain *ad hoc* committees. These meetings are held at NRC Headquarters locations or in other venues throughout the United States, as appropriate to the issues and/or projects under review. Appendix 2 gives a listing of the membership of the standing committees. 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) 492-1968.

Public Information

As NRC decisions, plans and actions unfolded throughout fiscal year 1989, information on them was provided to the news media and the general public by means of news releases, fact sheets, formal orders, official decisions and similar informational material. News conferences and interviews were arranged with Commissioners and senior staff when major actions were taken. In the Regional Offices, the NRC Public Affairs staff assisted news media at public meetings, emergency exercises, Licensing Board hearings, and other activities. In addition, the staff responded to thousands of requests from the news media and the public for information on various NRC programs and activities.

During the report period, more than 400 public announcements were issued by Headquarters and Regional Offices to news media, interest groups, industry and the general public, covering numerous NRC activities. And new fact sheets were issued on Nuclear Material Safety and Safeguards, Research Programs at the Nuclear Regulatory Commission, and Regulating the Disposal of Low-Level Radioactive Waste.

News reporters get "hands-on" instruction in the workings of nuclear power plant during a media seminar at the NRC Training Center.

Media Seminars. Two national media seminars were conducted at the NRC Training Center at Chattanooga, Tenn., to provide news reporters with basic "hands-on" instruction about how nuclear power plants work. The programs took place January 17-18, 1989, and July 6-7, 1989, attracting reporters from major publications throughout the nation.

School Volunteers. The 152 NRC employees who served as volunteers during the school year in programs conducted at more than 50 schools in Montgomery County and Prince George's County, both in Maryland, the District of Columbia, and Northern Virginia received certificates and letters of appreciation. At a special event honoring the volunteers, on April 6, 1989, Chairman Lando W. Zech, Jr., and Montgomery County (Md.) Schools Superintendent Harry Pitts renewed an agreement to continue the highly successful program. The NRC as an agency received a 1988-89 Outstanding Service Award from the Montgomery County Public Schools system for its work with school children. NRC staff assisted in the schools by serving as science fair judges, tutors, mentors, classroom lecturers, career awareness counselors, faculty advisors, and field trip hosts, as well as by their participation in the "English for Speakers of Other Languages Program."

Consumer Affairs. The NRC's fourth annual observance of National Consumers Week, coordinated by the Public Affairs staff, was held between April 24-30, and featured an address by Linda F. Golodner, Executive Director of the National Consumers League, Washington, D.C.

Headquarters Public Document Room

Persons interested in detailed information about commercial nuclear facilities have found the NRC's principal Public Document Room (PDR) a voluminous source of useful materials. The PDR is located at 2120 L Street, N.W., Washington, D.C. This specialized research center houses extensive documentation, available to the public, of significant nuclear regulatory decisions and actions. Users of the center can have documents reproduced for a nominal fee.

Researchers in the PDR can examine copies of a wide variety of materials, such as NRC reports; transcripts and summaries of meetings; licenses and amendments; existing and proposed regulations; and correspondence on technical, legal and administrative matters. Most of these documents are related specifically to nuclear power plants—their design, construction, operation, and inspection—and to nuclear materials, including the use, transport, and disposal of radioactive wastes. The PDR features extensive indexes and an on-line bibliographic data base available for staff and public use.

The Headquarters PDR contains about 1.64 million documents, and the collection is enlarged by an average of 265 new items every day. During an average month, the PDR serves about 1,200 users. Professionally trained reference librarians are available to assist with data base searches and to provide printouts on demand. These tailored bibliographies comprise the

NRC Chairman Lando W. Zech, Jr., at left, and Harold R. Denton, Director of the NRC Office of Governmental and Public Affairs, chat with reporter Dave Airozo during a tour of the Headquarters Public Document Room.

publicly available legal, technical and administrative documents which the NRC generates or receives. For those interested in automatically receiving selected serially published documents, the PDR provides a Standing Order subscription service. Certain documents of perennially high interest, such as Press Releases and Meeting Notices, are also available on an expedited basis.

Persons wishing to use or obtain additional information regarding the holdings, file organization, reference, reproduction services, and procedures of the PDR may call (202) 634-3273 or write to the U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. Requests may also be transmitted by facsimile to (202) 634-3343. A "Public Document Room Users' Guide" and "Public Document Room File Classification System" guide are available upon request. In addition, orientation sessions are provided for individuals or groups interested in using the facility, and training sessions are scheduled regularly in how to use the PDR automated bibliographic retrieval system, which can be accessed from public terminals in the Reading Room and or by dial-in access using personal computers.

Local Public Document Rooms

As of the close of the report period, the NRC was maintaining 97 Local Public Document Rooms (LPDRs) throughout the country. Of these, 78 are related to power reactor sites (eight of the 78 also maintain records on fuel or waste disposal facilities); six are dedicated to fuel cycle or radioactive waste facilities; 11 are "mini-LPDRs," housing selected data collections for a limited time, usually in support of an NRC hearing procedure; and two are related to the prospective high-level waste disposal facility. (LPDRs for this facility near Yucca Mountain in Nevada have been established in as Vegas and in Reno, Nev.)

LPDR document collections are usually located in university or public libraries. The NRC now provides financial assistance to 70 LPDR libraries, of which 68 are associated with power reactor sites and two with the prospective high-level waste disposal site. Under the LPDR program, NRC staff perform periodic audits of the collections and conduct workshops for the public at the PDR sites. (See Appendix 3 for a complete listing of LPDRs.)

During the report period, a pilot project providing on-line access to the NRC's computerized Nuclear Documents System (NUDOCS, see Chapter 10) continued at six LPDRs. (See *1988 NRC Annual Report*, p. 107, for background.)

The NRC has amended its Rules of Practice to improve the hearing process and widen the availability of information and documentation. Under the new procedure, a temporary LPDR was established in Los Angeles County in connection with a Rockwell International Corporation facility; hearings on a license renewal application for the facility were to take place in the vicinity. Temporary LPDRs have since been installed in a number of venues (as noted in Appendix 3).

Renewing the Volunteer Program Agreement between the Montgomery County, Md., Public Schools and the NRC are, at left seated, Schools Superintendent Dr. Harry Pitt and, at right seated, NRC Chairman Lando W. Zech, Jr. Standing is Johnnie A. Moore, NRC coordinator for the program.

Local librarians and their patrons may use a toll-free telephone number to request assistance and advice from LPDR staff on collection content, search strategies, and the use of reference tools and indices. The number is 1-800-638-8081.

Commission History Program

The Commission History Program studies the origins and evolution of regulatory policies in their historical context. The History Office is currently preparing 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 will cover the period from 1963 to 1971 and focus on reactor safety and siting, radiation protection, and environmental issues. Like the first volume, it is intended to serve as a reference for general readers, as well as for agency staff.

CONGRESSIONAL OVERSIGHT

NRC witnesses participated in 11 hearings before subcommittees of the Congress during the period from October 1, 1988, through September 30, 1989. The Commission and the NRC staff testified on a wide variety of topics, as shown in Table 1, below.

NRC Legistative Proposals

The Commission submitted two legislative packages to Congress on February 2, 1989. The first consisted of nine proposals, aggregated as an omnibus bill, which were aimed at enhancing nuclear safety, physical security and the agency's enforcement authority. The second package related to the expenditure of funds received from a financial surety arrangement established to accomplish decommissioning, decontamination and reclamation. The proposed legislation was referred to the House Committee on Interior and Insular Affairs and to the Senate Committee on Environment and Public works.

COOPERATION WITH THE STATES

The NRC's contacts with regional, State, and local agencies, and with Indian Tribes—for purposes other than inspection, enforcement or emergency planning—are administered through State Programs (SP). These include the State Agreements Program and various liaison and cooperative programs administered in accordance with policies and procedures established by Headquarters and implemented primarily by the Regional Offices. A more detailed discussion of NRC and States' interface is contained in NRC's report, "The U.S. Nuclear Regulatory Commission Program With State and Local Governments and Indian Tribes" (NUREG-1309).

Conference of Radiation Control Program Directors. NRC Chairman Kenneth M. Carr (as of July 1989) and William H. Spell, Chairman of the Conference of Radiation Control Program Directors, Inc. (CRCPD), met at NRC Headquarters in September 1989. The CRCPD—which serves as a forum for addressing radiation protection issues at the Federal, State and local levels of government—explored a number of items of potential interest and importance to the NRC, such as their plan of action regarding the regulation and control of "Naturally Occurring and Accelerator Produced Material" (NARM); the role of CRCPD in offering the Texas Industrial Radiographer Test to interested States; and CRCPD's efforts in developing an accreditation process for State Radiation Control Environmental Laboratories.

The CRCPD also discussed several items of interest to the States, including NRC training of State personnel, NRC policy on general licensees, and the role of the NRC and the Environmental Protection Agency (EPA) in radiation protection matters at the Federal level.

National Governor's Association. The National Governors' Association (NGA) held its winter meetings in Washington, D.C., from February 26 through 28, 1989. The NGA adopted a Comprehensive National Energy Policy which

> "encourages early resolution of nuclear power issues, consistent with safety and environmental requirements. These issues include plant standardization and timely permitting, consistent regulatory oversight of operations, plant life extension, decommissioning, and waste disposal. States should continue to have the right to monitor operating conditions at nuclear power plants."

The policy also recognizes State responsibility to ensure timely decisions on permit issuance, siting and licensing of energy facilities, consistent with State and Federal land and health and safety requirements. The NGA policy calls for research and development of advanced reactor designs, waste management technology, nuclear fusion, plant retrofit and life extension.

Table 1. Congressional Hearings at Which NRC Witnesses Testifed—FY 1988

Date	*Committee*	*Subject*
10/6/88	Committee on Interior and Insular Affairs Subcommittee on Oversight & Investigations (House)	Fitness for Duty
2/23/89	Committee on Interior and Insular Affairs Subcommittee on Energy and Environment (House)	NRC FY 1990 Budget
2/28/89	Committee on Energy and Commerce Sub-committee on Energy and Power (House)	NRC FY 1990 Budget
3/1/89	Committee on Science, Space & Technology Subcommittee on Energy Research and Development (House)	Advanced Reactors
3/15/89	Committee on Government Operations Sub-committee on Environment, Energy and Natural Resources (House)	GAO Report on NRC Personnel Security
3/21/89	Committee on Appropriations Subcommittee on Energy and Water (House)	NRC FY 1990 Budget
5/4/89	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	NRC Actions at Comanche Peak and Rancho Seco
5/11/89	Committee on Energy and Commerce Sub-committee on Energy and Power (House)	NRC Rule on Standardization and Licensing Reform
7/13/89	Committee on Public Works & Transportation Subcommittee on Surface Transportation (House)	Hazardous Materials Transportation Act
7/19/89	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Single Administrator
8/3/89	Committee on Government Operations Sub-committee on Environment, Energy and Natural Resources (House) and Natural Resources (House)	Decommissioning of Fuel Facilities

State Agreements Program

A total of 29 States, under formal agreements with the NRC, have assumed regulatory responsibility over certain source and by-product nuclear materials and small quantities of special nuclear material. Negotiations for an agreement with the State of Maine were under way at the close of the report period. At that time, there were about 16,200 radioactive material licenses in the Agreement States, representing about 65 percent of all the radioactive materials licensees in the United States.

Review of State Regulatory Programs. The NRC is required by the Atomic Energy Act of 1954 to review Agreement State radiation control programs periodically to confirm they are adequate to protect public health and safety and are compatible with NRC programs. The "routine review"—a complete, in-depth examination of a State's radiation control program—is conducted every 18-to-24 months for each Agreement State. To maintain continuity during the time between routine reviews, intermittent "program visits" are conducted. These visits provide an opportunity to discuss areas of concern on an informal basis and to verify the satisfactory status of the State's program. Follow-up and special reviews are conducted as needed to confirm the adequacy of State actions in specific areas. Nineteen routine reviews, nine interim program visits and one follow-up review were conducted in fiscal year 1989. The NRC technical staff also accompanied State inspectors to State-licensed facilities to evaluate inspector performance.

NRC Technical Assistance to States. The NRC continued to provide technical assistance to Agreement States with respect to licensing, inspection, and enforcement activity, and also regarding proposed statutes and regulations. Assistance covered matters ranging from responding to telephone requests for information to conducting State reviews of license applications and State inspections. Agreement States are expected to maintain a core staff knowledgeable about materials radiation safety and to use in-State technical resources, such as advisory committees and consultants. Unusual radiation applications may involve radiation safety programs that needspecialized expertise or knowledge, in which case NRC experts can be a valuable resource. An example of such NRC technical assistance was the aid provided to the State of Maryland in which NRC staff assisted the State in the inspection of a facility which the State had ordered closed because of an uncontrolled release of contamination.

Training Offered by NRC. State radiation control personnel regularly attend NRC-sponsored courses to improve their ability to maintain high quality regulatory programs. The NRC sponsored 17 short term training courses and meetings during the report period, attended by about 350 people, including State personnel, NRC staff and military personnel. Three of these training courses were hosted by Agreement States. Courses included such subjects as health physics; industrial radiography safety; nuclear medicine procedures; inspection procedures; well logging; radiation protection engineering; and the transportation of radioactive materials, nuclear materials, and low-level waste. The States provided 12 lecturers and 36 panelists. Other participants acquired on-the-job training in licensing and compliance either in the States or in visits to NRC Headquarters and Regional Offices.

Annual Agreement States Meeting. The annual meeting of Agreement State radiation control program directors took place in October 1989 at the Overland Park Marriott Hotel in Overland Park, Kans. The official welcome was delivered by Overland Park Mayor Ed Eilert and by Stanley Grant, Ph.D, Secretary of Health and Environment for the State of Kansas. Dr. John Montgomery, Region IV Deputy Regional Administrator, addressed the meeting, emphasizing the important partnership between NRC and the States in protecting the public health and safety. The keynote address was delivered by Carlton Kammerer, Director, State Programs, GPA, in which he addressed the need for Agreement States to meet NRC's guidelines for assuring the adequacy of their programs and their amenability to criteria employed in the NRC's periodic reviews. The meeting also included panel discussions on such topics as low-level waste, new radiation regulations, operational events, and materials licensing. Representatives of several key Federal agencies were invited to the meeting, and this year the Navy and Air Force were also represented. The non-Agreement States represented at the meeting were Maine, Michigan, Ohio, Pennsylvania and Oklahoma.

Regulation of Low-Level Waste. The NRC provided technical assistance to Utah, Texas, Nebraska, Michigan and New York in establishing their low-level waste regulatory programs and in meeting the requirements of the Low-Level Radioactive Waste Policy Act Amendments of 1985. Technical assistance was also provided to Pennsylvania, Nebraska and New York with their formulation of low-level waste regulations compatible with NRC regulations. Assistance on specific cases was furnished to Florida, Utah, Colorado, Georgia and Nevada. South Carolina, Washington and Nevada continue to participate in the NRC review of topical reports on high-integrity containers, waste solidification processes, and computer codes to be used in implementing 10 CFR Part 61.

Technical assistance in the area of low-level waste regulation was also given the States through the medium of two low-level waste meetings: a Low-Level Waste Regulatory meeting, held in November 1988, and a Low-Level Waste Workshop, in September 1989. These meetings provided an opportunity for States and the NRC to discuss certain current regulatory issues related to low-level waste disposal.

Regulation of Uranium Milling. The NRC assisted Agreement States in their programs for regulating uranium milling. The assistance included guidance on surety arrangements and on the Environmental Protection Agency's requirements. Direct technical assistance was provided to the State of Washington on specific cases. Representatives from Colorado, Texas, Washington, Illinois, New Mexico and Utah participated in a management meeting on groundwater requirements for uranium mill tailings licensees.

Special Projects. In September 1989, the State Agreements Program staff developed and sponsored a nationwide pilot videoconference on the NRC's revised 10 CFR Part 20 "Standards for Protection Against Radiation." The training was conducted in cooperation with the NRC's Offices of Nuclear Regulatory Research and Nuclear Reactor Regulation, the State of Texas, the U.S. Department of Agriculture and the U.S. Department of Health and Human Services—Center for Devices and Radiological Health. The videoconference, which supplied an overview of Part 20 revisions, allowed for viewers to ask questions during the program.

State, Local and Indian Tribe Liaison Activities

The NRC Five Year Plan calls for the agency to assume a more active role in fostering cooperation and communications between the NRC and State and local governments and Indian Tribe representatives, pursuant to the more general purpose of promoting broader and deeper understanding of issues and activities affecting nuclear safety.

In September 1989, the Agreement States Program staff developed and sponsored a nationwide pilot video teleconference—involving other NRC offices, the U.S. Departments of Agriculture and of Health and Human Services—to discuss the NRC's revised radiation protection standards (10 CFR Part 20). Participants in the pilot program included Seated, left to right, are Mark Bennett, Food and Drug Administration, moderator; Dr. Donald Cool, NRC Office of Nuclear Regulatory Research; Thomas Essig, NRC Office of Nuclear Reactor Regulation (NRR); and Richard Ratliff, Texas Bureau of Radiation Control. Standing, left to right, are John Buchanan, NRR; Carlton Kammerer, Director of NRC State Programs (SP); Vandy L. Miller, SP; and John Kendig, SP.

Final Policy on Cooperation with States. On February 22, 1989, NRC published a policy statement in the *Federal Register* entitled "Cooperation With States at Commercial Nuclear Power Plants and Other Nuclear Production or Utilization Facilities" (54 FR 7530). In developing the policy, the NRC considered comments from utility groups, State agencies and a public interest group. The policy is intended to introduce uniformity into the handling of State requests to monitor and/or participate in regulatory oversight of their nuclear plants and facilities. More specifically, it affirms that NRC will continue to cultivate close working relationships with the States and allow State representatives to observe and participate in NRC inspections and in entrance and exit meetings with licensees.

For more than 11 years, the NRC has been entering into various agreements—based on memoranda of understanding with the States—which deal with topics ranging from the stationing of State resident engineers at nuclear power plants to low-level waste package and transport activities at licensed facilities. In recent years, States have generally become more deeply involved in activities related to the operation of power plants within and adjacent to their borders. States also play an important role in the non-radiological aspects of plant safety, such as fire protection.

The NRC staff has developed implementing guidance, in the form of a model "subagreement," for State resident engineer inspection programs, in response to a request from the Illinois Department of Nuclear Safety. The guidance will be modified to encompass other agreements with States whose inspection programs may be less comprehensive and would not involve placing resident engineers at power plants. Other States which have expressed an interest either in observing or taking part in NRC inspections are Massachusetts,New Hampshire, Maine, New York, New Jersey, Pennsylvania and Vermont.

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 volume of waste and milestones for specific action by the States. The Act established a system of incentives and penalties to promote steady progress toward new facility development and granted Congressional consent to seven interstate low-level waste disposal compacts, covering 37 States. In 1988, Congress approved the Appalachian Compact and the Southwestern Compact. In February and March of 1989, South Dakota and North Dakota each enacted legislation to join the Southwestern Compact.

To meet the next Congressional milestone, States which are not in sited compacts must, by January 1, 1990, either submit an application for a low-level waste disposal site or a certification. The certification will describe how the State will be capable of providing for the storage, disposal or management of any low-level waste generated within the State which will require disposal after December 31, 1992. The NRC will receive the certifications, forward them to Congress and publish them in the *Federal Register*.

During 1989, the NRC published notices in the *Federal Register* which set forth policies and procedures for administering its responsibilities for the certifications and for coordinating its efforts with the Department of Energy (DOE) and the sited States of South Carolina, Washington and Nevada. The NRC is also continuing its program of assistance to States and compacts by reviewing enabling legislation and providing training and regulatory assistance.

State Liaison Officers. The NRC continues to use the State Liaison Officers (SLOs) appointed by Governors as its primary point of contact with States regarding NRC activities. The policy statement on cooperation with States identifies the SLO as the primary State contact for all requests involving observation of NRC inspections.

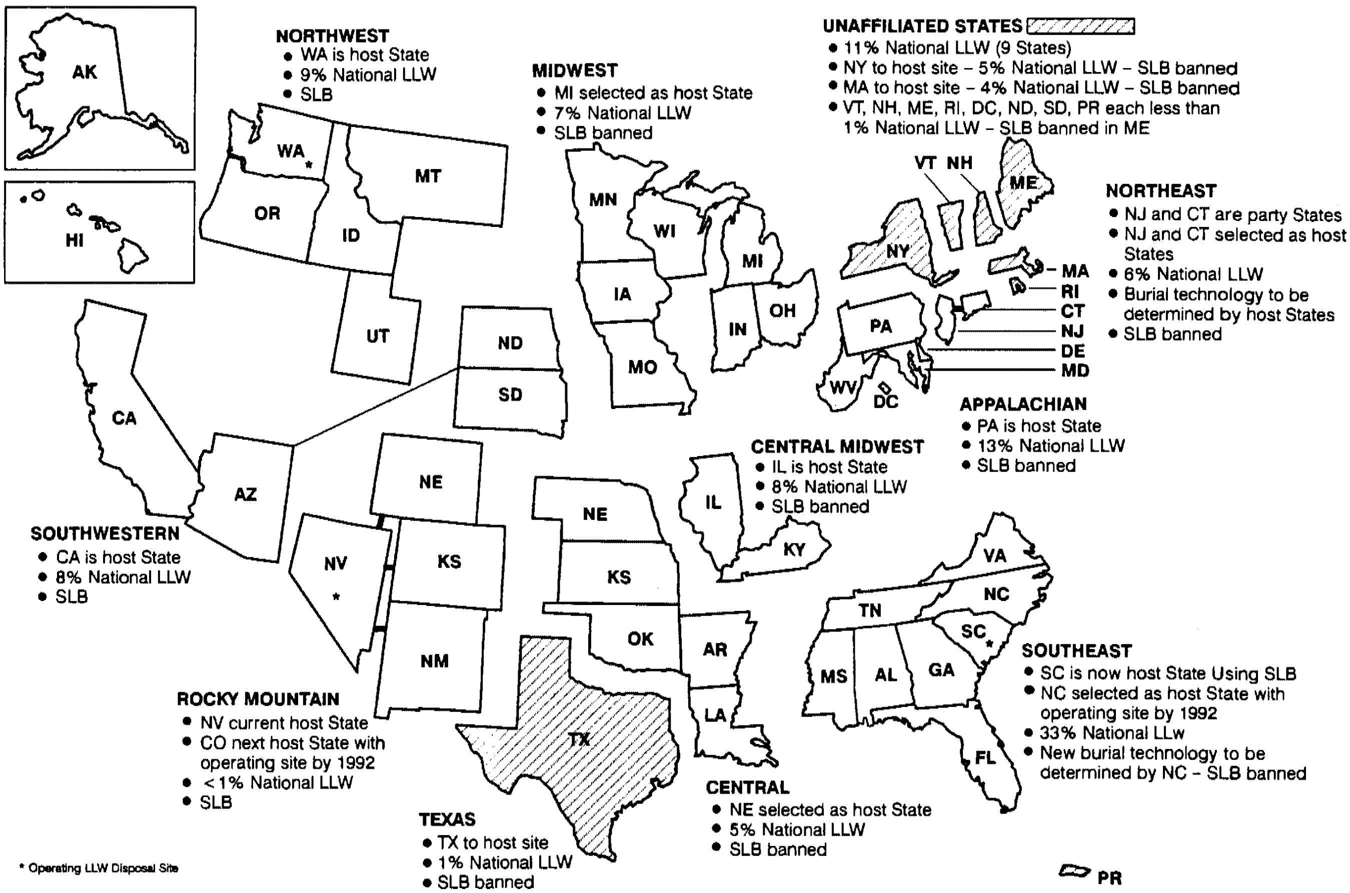
Region I SLOs met at the Regional Office in King of Prussia, Pa., on April 11 and 12, 1989. Their discussions focused on the States' concerns with "No Significant Hazards" license amendment issues, the NRC's increased use of utility groups for studies and research, the lack of timeliness in reporting of malfunctions by utilities, and the need for the NRC to publish national performance trends for operating reactors. The States also expressed a strong need to attend NRC training courses. The final policy statement on cooperation with States was also discussed, and each State reported on its activities with respect to power reactors within its borders.

An SLO meeting was also held in NRC's Region II office in Atlanta on September 19 and 20, 1989. Topics discussed there included State attendance at certain meetings between the NRC and its licensees, State accompaniment of NRC inspections of materials licensees, State concerns with naturally occurring radioactive materials, and the training of nuclear physicians. Presentations were made by State attendees and representatives from the Southern States Energy Board and the Federal Emergency Management Agency.

Regional State Liaison Officers. Each NRC Regional Office has a Regional State Liaison Officer (RSLO) who acts as that NRC Region's principal contact with SLOs

LOW-LEVEL RADIOACTIVE WASTE COMPACT STATES

OCTOBER 1989



Note: National LLW volume for 1988 = 1.4 million cubic feet.
SLB = shallow land burial

Source: State, Local and Indian Tribe Programs
Office of Government and
Public Affairs, NRC

and other State and local officials. RSLOs generally coordinate NRC activity involving State and local governments and Indian Tribes. RSLOs often attend and participate in local meetings when local issues under the NRC's purview are involved. And RSLOs often address State legislative committees and meet with State and local officials to address concerns and respond to questions. The RSLOs routinely respond to requests for information from SLOs and other State officials concerning nuclear power facilities or other areas under NRC's jurisdiction. RSLOs attend regional low-level radioactive waste compact commission meetings and monitor State progress in developing additional disposal capacity for low-level waste. SP conducted a Regional State Liaison Officers counterpart meeting from January 31 through February 1, 1989, at the Region I office. The meeting provided an opportunity for SP staff and the RSLOs to review and discuss NRC policies and initiatives with regard to the States and to other governmental entities.

Outreach Activities. In keeping with the mandates of its Five Year Plan, the NRC has continued to broaden its cooperation with the States and their organizations. In addition to routine interaction with State and local government and Indian Tribe officials, NRC representatives have taken part in a number of special State-related activities. For example, Commissioner Kenneth C. Rogers was active in the proceedings of the National Association of Regulatory Utility Commissioners (NARUC) and Commissioner James R. Curtiss addressed a meeting of the National Conference of State Legislators (NCSL) on August 9, 1989.

Liaison with American Indian Tribes. The NRC continues to maintain a government-to-government relationship with those American Indian Tribes involved and/or interested in NRC's programs. While no Tribes have as yet been formally accorded "affected status" under the 1987 Nuclear Waste Policy Amendments

Act, Indian Tribes are kept apprised of NRC's activities in connection with the high-level waste program. Those Tribes potentially affected by the Department of Energy's siting of a high-level waste repository at Yucca Mountain in Nevada 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 activities of the NRC's Advisory Committee on Nuclear Waste.

The NRC also involves Native American organizations—such as the National Congress of American Indians (NCAI)—in pertinent NRC activities. For example, the NCAI assisted the NRC and Indiana University staff in revising a 1980 Survey of State Radiological Emergency Response Capabilities for Transportation Related Incidents (NUREG/CR-1620). The NCAI helped develop a questionnaire to determine tribal emergency response capabilities and the level of interaction with the States; it also provided a letter of introduction endorsing the study and encouraging cooperation by the 15 selected Indian Tribes, whose reservation boundaries are crossed by spent fuel shipment routes. The updated report (NUREG/CR-5399) was expected to be publicly available in January 1990.

The NCAI and other tribal representatives met with Commissioner Curtiss and Commission staff in March 1989 to discuss concerns related to high-level waste transportation, including the Tribes' desire to be notified of planned high-level waste shipments through Indian reservations. The NCAI continues to represent tribal interests as a member of the high-level waste Licensing Support System Advisory Committee. This Committee negotiated a rule designed to streamline the high-level waste licensing proceeding by means of an electronic information management system.

The NRC also participates in biannual interagency meetings sponsored by the EPA. The meetings are geared toward sharing experiences and exploring new ground in the area of Federal and tribal government interaction.

INTERNATIONAL ACTIVITIES

The NRC's international activities are concerned with

- Contributing to the safe operation of licensed U.S. reactors and fuel cycle facilities and the safe use of nuclear materials.
- Improving worldwide cooperation in nuclear safety and radiation protection.
- Assisting U.S. efforts to restrict U.S. nuclear exports to peaceful use only.
- Supporting U.S. foreign policy and national security objectives.

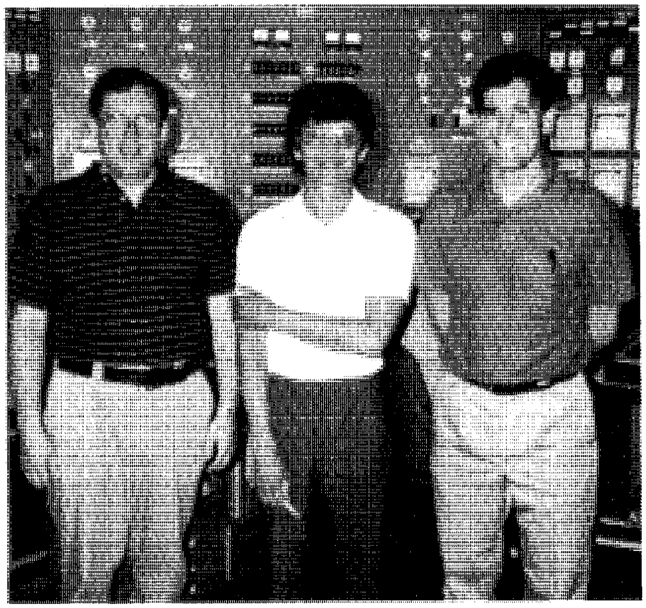
The Office of Governmental and Public Affairs is the Commission's primary medium for the coordination of international activities and policies. Other NRC offices contribute to international cooperation by taking part in international meetings, providing technical expertise, and conducting research, both in this country and abroad.

The NRC's international program in nuclear safety has traditionally included bilateral regulatory and research cooperation agreements and participation in multilateral research and other safety cooperation, through the International Atomic Energy Agency (IAEA) and the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA). While power reactor safety is the primary focus of these efforts, increased attention is also being given to broader radiation protection matters, to waste management issues, and to other areas of materials safety—including source and by-product material, fuel handling, and the international transport of radioactive waste.

Continued world-wide interest in the Chernobyl accident has amplified the importance of nuclear safety in U.S. foreign policy andexpanded the NRC's involvement in international nuclear safety cooperation. Through its international programs, the Commission is continuing bilateral U.S.-U.S.S.R. activities and is working with the IAEA in promoting broad international cooperation on nuclear safety and regulatory matters. Some of these efforts are described in greater detail below.

Highlights of Fiscal Year 1989

- Hosted meetings with Chairman Vadim Malyshev of the U.S.S.R. State Committee for the Supervision of Nuclear Power Safety.
- Organized and held meetings of seven of the 10 working groups under the Protocol of the U.S.-U.S.S.R. Joint Coordinating Committee for Civilian Nuclear Reactor Safety (JCCCNRS).
- Conducted a seven-week reactor inspector exchange with the Soviet Union, sending a U.S. team to the Zaporozhe nuclear power plant in the U.S.S.R. and receiving a U.S.S.R. team at the Catawba nuclear power plant in South Carolina.



Members of a U.S. Inspection Team are shown with the local Chief Inspector in the control room of the Zaporozhe nuclear power plant in U.S.S.R. They are, left to right, Philip Brochman, NRC Region III (Chicago) Resident Inspector; Victor Koltunov, Chief Inspector at the site; and Joe Callan, Director, Division of Reactor Safety in NRC Region IV (Dallas). Zaporozhe is in the Ukraine, 550 miles south of Moscow.

- Hosted the fifth annual regulatory meeting with Japan's Ministry of International Trade and Industry (MITI), and sent a nine-member team to Japan for an in-depth look at Japanese maintenance techniques.

- Initiated review of policy on the export and import of radioactive wastes.

- Formalized nuclear ties between the NRC and Canada's Atomic Energy Control Board (AECB) by signing a five-year renewable information exchange and nuclear safety cooperation arrangement.

- Hosted a visit by a delegation from the Federal Republic of Germany (FRG), led by Environment and Nuclear Safety Minister Toepfer; the delegation was in the U.S. for discussions with EPA, NRC and DOE on waste management and nuclear safety issues, and to visit Three Mile Island and Yucca Mountain.

- Coordinated a visit to the FRG and the United Kingdom (U.K.) for detailed discussions in both bilaterals and in a multilateral symposium on Regulatory Practices and Safety Standards.

- Concluded an arrangement for cooperation in nuclear safety and exchange of information with Czechoslovakia, NRC's first such arrangement with an Eastern European country.

- Renewed NRC's bilateral information exchange and nuclear safety cooperation arrangements with Brazil, Spain, Sweden, and Mexico.

- Worked closely with the Executive Branch and the IAEA in strengthening international safeguards and physical security. Sent experts to Japan, France, the FRG, U.K., European Community, Belgium, Switzerland, Greece, Romania, Yugoslavia and Austria for discussions and site visits.

- Participated in an IAEA-sponsored Technical Committee Meeting to review IAEA's Guidelines on physical protection of nuclear materials against theft or sabotage.

- Participated in the 1989 IAEA General Conference held in Vienna from September 25 through 29 and presented papers and chaired sessions at the Scientific Program for Nuclear Safety held during the conference.

- Sponsored an IAEA Operational Safety Review Team (OSART) mission to the Byron (Ill.) nuclear power plant from May 15 to June 2. Team members visiting the plant came from the FRG, Sweden, the U.S.S.R., Belgium, Finland, the German Democratic Republic, Argentina, Japan and Canada; and there were four members from the IAEA Secretariat. Observers from Mexico, Czechoslovakia and Brazil also attended the two-week review.

- Hosted an informal OSART review meeting with U.S. experts who had been participants in OSART missions during the previous three years, discussing the effectiveness of U.S. involvement in the program. Suggestions for improving the program, which was considered successful by the meeting participants, were provided to the IAEA.

- Sent 14 U.S. experts to participate in 11 IAEA OSART missions to Japan, France, Hungary, the U.S.S.R. (two missions), Brazil, China, the U.K., Korea, Czechoslovakia, and Poland.

- Participated in IAEA's Nuclear Safety Advisory Group (NUSSAG) meeting in Vienna in April to review reactor safety standards-related activities and to set priorities in IAEA's nuclear safety program.

- Sent an NRC expert on a Radiation Protection Advisory Team mission to Ghana and Zimbabwe.

International Cooperation

U.S.-Soviet Civilian Nuclear Safety Cooperation. The Soviet Union and the United States continued to develop potentially fruitful cooperation in nuclear reactor safety, through meetings of working groups of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) and continuing interactive meetings and exchanges. Cooperative activities included a visit to the United States in May by Vadim Malyshev, Chairman of the U.S.S.R. State Committee for the Supervision of Nuclear Power Safety.

The largest bilateral technical meeting ever hosted by NRC was held June 5 through June 9, 1989, when 32 Soviet scientists came to Rockville, Md., for discussions in seven of the 10 working groups set up under JCCCNRS. Discussion topics included Safety Approaches and Regulatory Practices, Analysis of the Safety of Nuclear Power Plants in the U.S.S.R. and U.S., Radiation Embrittlement and Annealing, Fire Safety, Severe Accidents, Exchange of Operational Experience, and Erosion/Corrosion of Piping and Components.

A successful seven-week inspector exchange was concluded during the summer at reactor sites in the United States and in the Soviet Union. The U.S. team was assigned to the Zaporozhe plant in the U.S.S.R. and the Soviet team to the Catawba plant in South Carolina.

The NRC participated in working group meetings in Moscow and Kiev in September dealing with the environmental and health effects of the Chernobyl accident. The potential for joint effort in this area appears far greater than previously thought. Areas of intergovernmental and interagency cooperation were being developed at the close of the report period. The NRC is urging private industry to join in these activities, in order to broaden their scope and to help accommodate the large flow of information and attendant activity.

Bilateral Information Exchange Arrangements. The NRC participates in a wide-ranging, mutually useful program of information exchange and safety and research cooperation with its counterparts in the international community. The NRC has conducted much of its technical information exchange through a series of general safety cooperation arrangements formally concluded with the regulatory authorities of Belgium, Brazil, Canada, China, Czechoslovakia, Denmark, Egypt, the Federal Republic of Germany, Finland, France, Greece, Israel, Italy, Japan, South Korea, Mexico, the Netherlands, the Philippines, Spain, Sweden, Switzerland, Taiwan, the Union of Soviet Socialist Republics, the United Kingdom and Yugoslavia.

These arrangements establish formal channels with foreign nuclear regulatory organizations to ensure prompt and reciprocal notification of reactor safety problems that could affect both U.S. and foreign nuclear facilities and to facilitate identification of possible "precursor events" that warrant further investigation. These arrangements also provide a framework for bilateral cooperation on nuclear safety, safeguards, waste management, and environmental protection. The bilateral arrangements are normally effective for five years but contain provisions for renewal by mutual agreement.

Canada. The NRC and the Atomic Energy Control Board of Canada (AECB) formalized their nuclear ties by signing a five-year renewable information exchange agreement at the Canadian Embassy in Washington, D.C., in June 1989. Under the arrangement, both parties will exchange safety-related information related to the regulation of activities at nuclear facilities for which each agency is responsible—including siting, constructing, commissioning, operating and decommissioning. The agreement covers a broad range of topics, such as regulatory standards and procedures for nuclear facilities; technical reports and safety assessments; safety research programs; possible exchanges of personnel; and reports of radiological events, accidents or emergencies.

The Federal Republic of Germany (FRG). In November 1988, the FRG sponsored and co-hosted with the IAEA and the NEA an "International Symposium on Regulatory Practices and Safety Standards for Nuclear Power Plants," in Munich, Germany. Commissioner Rogers and several senior NRC managers participated, a number presenting papers or chairing panels. An IAEA survey of individual country practices revealed wide differences among countries' practices with respect to IAEA codes, principles and standards. Most participants supported further meetings to explore the implications of these differences and ways to minimize them.

In August 1989, FRG Minister of Environmental Affairs, Klaus Toepfer, led a delegation to the U.S. for discussions with NRC, EPA and DOE. He met with Commissioners Rogers and Curtiss, and his delegation exchanged information with officials from a number of NRC offices on a variety of subjects—including high-level waste disposal, regulatory approaches to spent fuel management, accident management, and risk assessment. Minister Toepfer also visited Three Mile Island, and members of his delegation visited the Yucca Mountain Test Site, the Waste Isolation Pilot Project storage facility, and the Diablo Canyon (Cal.) nuclear power plant.

Participants in the bilateral discussions between nuclear officials of the U.S. and the Federal Republic of Germany (FRG), in August 1989, included, left to right: NRC Chairman Lando W. Zech, Jr.; Dr. Toepfer, FRG Minister of Environment and Nuclear Safety; Dr. Birkhofer, General Manager of the FRG Nuclear Safety Co.; Dr. Hohlefelder, Assistant Secretary to the Ministry of Environment and Nuclear Safety; and James Shea, NRC Director of International Programs.

A direct line of communication was opened between senior NRC and West German nuclear safety officials on such time-sensitive matters as unusual occurrences at operating nuclear power plants. The first visit of a technical team to the FRG took place in November of 1989.

The United Kingdom (U.K.). Commissioner Rogers visited the United Kingdom (U.K.) in May to learn more about ongoing nuclear safety programs and regulatory practices and to appraise the effects of privatization on nuclear programs and institutions in the U.K. The Commissioner visited the URENCO enrichment plant, the Heysham 1 and 2 nuclear power plants, the THORP reprocessing facility at Sellafield, and a low-level waste disposal site. Earlier, in March, officials of the U.K. Nuclear Installations had visited the NRC for discussions of emergency planning, and of studies they had sponsored comparing U.S. and U.K. requirements.

In July, the U.K. Health and Safety Executive's Director General, John Rimmington, and his Deputy met in Washington with Commissioners Rogers and Curtiss to discuss recently revised licensing rules for standardized plants in the U.S., as well as the new role that the Health and Safety Executive now assumes in the area of safety research. Mr. Rimmington reported on progress in the U.K.'s transition from government to private ownership of electric power production, and indicated that privatization is proceeding without significant impact on electric energy supplies.

Japan. It was a year of active cooperation between Japan's nuclear safety program and the NRC. Senior officials and technical personnel from Japan and the U.S. visited each other's facilities and held meetings on current issues and joint programs. The sharing of useful information included significant data on Japan's experience with thermal stress in piping and weld cracking. The NRC receives, on a regular basis, MITI's press announcements of operational events at Japan's 37 power reactors.

The NRC hosted the fifth regular NRC-MITI meeting on nuclear regulatory matters on October 2-3, 1988, in Washington, D.C. A 20-member MITI delegation of government and utility representatives met with the NRC to discuss such topics as severe accident issues, reactor plant life extension, and advanced light water reactors. Following the discussions, several members of the MITI delegation visited the Three Mile Island and Limerick nuclear power plants in Pennsylvania and the Brookhaven National Laboratory on Long Island, N.Y., where NRC safety research is conducted. Also in October, a nine-member team of regional and resident inspectors, led by one Headquarters representative, visited Japan for 10 days to take an in-depth look at Japan's approach to surveillance testing and maintenance procedures. Following discussions with government representatives, the team spent a week visiting various reactor plant sites to meet with utility management and plant personnel.

Taiwan. In May, several members from NRC participated as delegates to the American Institute in Taiwan (AIT) and Coordination Council for North American Affairs (CCNAA) Joint Standing Committee Meeting on Civil Nuclear Cooperation, which was held in Taipei. Presentations were made by both sides on the status of their nuclear activities, followed by a discussion of current and future items of mutual interest. Proposed cooperation includes visits to NRC by Taiwan's nuclear specialists, short term assignments of Taiwan nuclear safety experts at NRC, visits to Taiwan by NRC experts to present information on current safety topics, and joint cooperation on safety research projects.

Korea. The NRC has a long history of close cooperation with Korea in nuclear safety and regulation, both bilaterally and through the IAEA. The U.S.-Korea Joint Standing Committee on Nuclear and Other Energy Technologies (JSCNOET) met at the Department of State in early October 1989. In these discussions, Korea's Ministry of Science and Technology (MOST) identified operational safety, radiation protection, public acceptance, and confirmatory research as areas in which Korea hopes to work closely with NRC in the future.

Czechoslovakia. In April, former NRC Chairman Lando W. Zech, Jr., and Dr. Stanislav Havel, Chairman of the Czechoslovak Atomic Energy Commission, signed an agreement to exchange nuclear safety-related technical information and to cooperate in civilian nuclear safety matters. This is the first NRC nuclear safety agreement with an Eastern European country. Commissioner Rogers led the first NRC nuclear safety delegation to Czechoslovakia in October 1989 for technical discussions with that nation's nuclear officials and tours of the facilities there. The discussions led to identification of several potential areas for the exchange of information under the agreement. The Czech nuclear program is considerably smaller than that of the Soviet Union, but the former has achieved several independent advances in research of interest to the NRC.

Hungary. On July 12, during a visit to Hungary, President Bush announced that the U.S. had proposed a U.S.-Hungary agreement on scientific and technical cooperation in various areas, including nuclear safety. Following the President's initiative, a delegation of representatives from involved U.S. agencies met with their Hungarian counterparts in Budapest, from July 31 through August 2. NRC officials, as part of the U.S. delegation, sought to assess Hungarian interests and expectations in the nuclear safety area.

The People's Republic of China. For most of fiscal year 1989, NRC carried out an active program of bilateral cooperation with its counterpart, the Chinese National Nuclear Safety Administration (NNSA), assisting China in building its nuclear safety program preparatory to commissioning its first nuclear power reactor, at Qinshan. Several NRC staff members visited China to give technical assistance, and several Chinese worked at the NRC for on-the-job training. Following the Chinese government's suppression of the pro-democracy movement in June, the NRC and other U.S. government agencies put cooperative programs with China on hold while the impact of the Chinese actions on our policies toward China was assessed. As part of the same policy decision, the Department of State chose to withhold action on all export applications licensed by the Department of Commerce—and concurred in by the NRC—involving transfers of commodities to the Chinese government for nuclear-related uses and controlled for nuclear non-proliferation reasons.

Foreign Assignees to the NRC Staff. The NRC work/training assignee program continues to be of strong interest to foreign regulatory organizations and the Commission. Six countries sent 12 staff members to participate in the program during the report period. While licensing activities related to engineering and system technology have continued to attract a number of participants, an increasing number of foreign visitors have been accommodated in activities related to the analysis and evaluation of operational data, safety programs and waste management.

Participation in International Organizations and Conferences

Meeting of the IAEA Board of Governors and General Conference. The NRC was represented at both the February and June Board of Governors meetings where IAEA policy decisions on program, budget and staffing are taken. NRC Chairman Kenneth M. Carr participated in the 1989 IAEA General Conference, held in Vienna from September 25 through 29. As in the past two years, special scientific meetings took place in conjunction with the conference, and, in one such session, chaired by Chairman Carr, the NRC's Individual Plant Safety Examination program was discussed, as part of the Scientific Program for Nuclear Safety.

OSARTs and Other IAEA Activities. Four NRC staff members participated in separate IAEA Operational Safety Review Team (OSART) missions to the PAKS nuclear power plant in Hungary, the Rovno plant in the U.S.S.R., the Qinshan plant in China, and the Dukovany nuclear power plant in Czechoslovakia. The NRC arranged to have U.S. utility experts take part in OSARTs in Japan, France, Brazil, the UK, Korea, Poland, and again in the U.S.S.R. The NRC also was represented on a Radiation Protection Advisory Team mission in Ghana and Zimbabwe, in June, to assess the infrastructure and training of personnel for control of radiation sources in those countries. The NRC also took part in several IAEA meetings, some of which were joint NEA/IAEA sessions, to discuss experience and potential benefits in the use of severity scales to rank the significance of events at nuclear power plants. The NRC continued its practice of providing nuclear safety advice and assistance through the IAEA's technical assistance program and through its bilateral contacts with countries developing their own nuclear power programs.

Activities in the OECD/NEA. The NRC continued its involvement in the reactor safety, radiation protection and waste management programs of the Nuclear Energy Agency (NEA) in Paris. Agency representatives served on the key standing committees of the NEA, as well as with the governing body, the Steering Committee for Nuclear Energy. Under the sponsorship of the Committee on Safety of Nuclear Installations (CSNI), an NRC-proposed multi-national research program to study the Three Mile Island Unit 2 reactor vessel lower head is being implemented. Substantial funds are being provided by other member countries to conduct the research. A major development during the year was the restructuring of the standing committee organization, with the assignment of certain functions of the CSNI Subcommittee on Licensing to a new committee dealing with regulatory issues. The mandate of the new Committee on Nuclear Regulatory Activities (CNRA) was approved by the Steering Committee in early October 1989, for an interim term. The first meeting of the committee was to take place in early November. The Director of NRC's Office of Nuclear Reactor Regulation (NRR) is the U.S. delegate to the committee.

During the fiscal year, steps were taken to transfer the NEA Incident Reporting System (IRS) data bank operations to the Oak Ridge National aboratory as part of the U.S. contribution to the NEA program and budget. An Agreement between the OECD and the U.S. government formalizing this transfer was to be signed before the end of 1989.

Signing a renewal of the U.S.-Spain Information Exchange agreement are, seated at left, Commissioner Donato Fuejo, President of Spain's Nuclear Safety Council (CSN) and, seated at right, NRC Chairman (as of July 1989) Kenneth M. Carr. The event took place in Vienna, at the IAEA General Conference in September 1989. Observers are, left to right, CSN Commissioners Echavarri and Gonzalez; U.S. Ambassador Kennedy; Theodore Sherr, Senior Science Attache; James Shea, NRC Director of International Programs; Frank Kinnely, U.S. Department of State; and U.S. Ambassador Newlin.

A U.S. delegation meets with the U.S.S.R. Plant Director of the Rovno nuclear power plant, Vladimir Korovkin (at head of table), and his staff. NRC delegate Ashok Thadani (in striped sweater) poses a question to the group. Rovno is in the western U.S.S.R., about 80 miles from the Polish border.

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, and to certain nuclear-related components and 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.

The NRC issued 119 new export licenses and 51 minor amendments to existing licenses. Of these cases, 43 involved routine exports of low-enriched uranium fuel for various power reactors around the world using uranium of U.S. origin or purchasing Department of Energy (DOE) uranium enrichment services. Discussions were held with Japanese officials concerning the proposed issuance of multi-year export licenses for routine fuel reloads for all Japanese light-water-moderated power reactors. This step would reduce considerably the administrative burden of processing export license requests and would also conform with NRC's existing practice concerning exporting uranium fuel to other countries. The NRC also issued seven licenses authorizing the export of more than 334 kilograms of high-enriched uranium (HEU) for use in research and test reactors in the Euratom countries and in Canada and Japan. DOE transport of commercial HEU shipments by its safe, secure transport (SST) vehicle continued without problems in 1989. The NRC continues to cooperate with the DOE Office of Security and Safeguards in a detailed review of special nuclear material protection during the transportation phase.

NRC Consultations with the Executive Branch on Nuclear-Related Export Matters. The NRC consults with the Executive Branch on other nuclear-related exports involving "dual-use" items licensed by the Department of Commerce, retransfer requests of U.S.-origin nuclear material, and nuclear technology transfers. Cooperation with the Soviet Union and Eastern European countries has continued to increase, resulting in more transfers of nuclear technology assistance in the safety area.

The NRC continues to participate in the interagency committee that oversees the U.S. nuclear export control system. The committee reviews primarily Department of Commerce-licensed export requests for commodities controlled for nuclear non-proliferation reasons. An important initiative of the committee has been the update of the Department of Commerce's Nuclear Referral List, which is in the final stages of completion. The NRC also participated in consultations with other countries on upgrading the related international "trigger list" of nuclear commodities, as part of the U.S.'s concerted effort to obtain multilateral controls on these items.

U.S.-Japan Agreement for Cooperation. In August 1989, a five-member team consisting of representatives from the Departments of State and Energy and the Nuclear Regulatory Commission were in Tokyo to discuss administrative arrangements for the implementation of the recently concluded U.S.-Japan agreement for cooperation with Japanese authorities from the Ministries of Foreign Affairs and International Trade and Industry and the Science and Technology Agency. Discussions included the tracking of produced special nuclear material, inventory reconciliations, retransfers, and proposals for multiple "reload" licensing of future exports of special nuclear material from the U.S. to Japan.

Nuclear Materials Safety. To support increasing interest by the Commission in nuclear materials safety, the NRC has expanded its international involvement in the areas of import and export, radioactive waste, radiation protection and fuel cycle activities. In early 1989, the NRC began a broad review of its international policy and import/export regulations for radioactive waste and identified a number of changes for consideration. An advance notice of proposed rulemaking was scheduled for release in late 1989.

In May, the NRC attended an IAEA meeting in Vienna to begin developing an international voluntary "Code of Practice" for transfers of radioactive waste. The voluntary code is sought in an effort to help assure that no incidents of illicit radioactive waste dumping will occur across national borders. The key areas of discussion concerned appropriate waste elements to be included. Another meeting was scheduled for early 1990 to conclude the agreement, subject to final approval by the IAEA Member States at the September 1990 General Conference.

During 1989, several NRC Commissioners and staff visited radioactive waste facilities in the Federal Republic of Germany, Sweden and France to improve U.S. understanding and bilateral cooperation. The NRC has also increased its scope of involvement in multi-national radiation protection and radioactive waste activities with the OECD/NEA and IAEA.

Following the General Conference, the Chairman visited the FRG, Sweden, and Norway for discussions and site tours of proposed underground radioactive waste repositories and facilities where research on radioactive waste management is performed, as well as for briefings and other site visits related to reactor safety. In Stockholm, the Chairman signed the renewal of NRC's information exchange and cooperation arrangement with Sweden.

International Safeguards and Physical Security. In all pending export cases to be reviewed by the NRC, the staff reviews the effectiveness of the implementation of IAEA safeguards and the physical security arrangements to be applied to the exported materials in the receiving country. These reviews are performed in compliance with U.S. non-proliferation laws to ensure that U.S. exports will be protected during transit and use in the importing country and that the exports will be used for peaceful purposes only.

The NRC participates in U.S. government efforts to improve and strengthen the IAEA safeguards system through the U.S. Program of Technical Assistance to IAEA Safeguards (POTAS) and the U.S. Action Plan Working Group (APWG), providing direct assistance to the IAEA and participating in international projects in support of the international safeguards regime. Under the auspices of the APWG, the NRC participated in bilateral and multilateral discussions on IAEA safeguards with Japan, France, the United Kingdom, the FRG, and the European Community. The POTAS program funded the assignment of two NRC experts to work in the IAEA Division of Safeguards in 1989.

In support of its review of physical security arrangementsof U.S.-controlled materials in other countries, the NRC participates in Department of Energy-sponsored trips to importing countries to study and discuss their physical security programs. During the report period, U.S. delegations visited Belgium, Switzerland, Greece, Romania, Yugoslavia and Austria for this purpose.

The NRC was part of a U.S. delegation to the IAEA-sponsored Technical Committee on Physical Protection (TCPP) meeting to review IAEA Information Circular (INFCIRC) No. 225/Revision 1, which provides guidelines applicable to the protection of fuel facilities, reactors and transportation systems against theft or sabotage of nuclear material. The TCPP identified and recommended revision of 25 specific points in a proposed INFCIRC No. 225/Revision 2 text that was to be submitted to the IAEA Board of Governors for review and approval in late 1989.

Nuclear Regulatory Research

Chapter 8

Activities of the Office of Nuclear Regulatory Research (RES) provide an essential contribution 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 rulemaking, including the issuance of regulatory guides and rules that govern NRC licensed activities. (See "Regulations and Guides," below.) Regulations issued by NRC in 1989 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which includes a listing of those guides issued, revised, or withdrawn during fiscal year 1989.

This chapter summarizes RES activities during fiscal year 1989 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 addressing the prevention of damage to reactor cores and mitigation of severe accident consequences encompasses the operations of the reactor as a system and consideration of the operator as an integral part of the reactor system. Also included are the establishment and maintenance of accident management programs designed to minimize the risk to the public in the event of severe accidents.

ACCIDENT MANAGEMENT

The accident-management program plan developed in 1988 defined two major areas of research support: (1) assessment of severe accident-management strategies, and (2) identification of the attributes of a functioning utility accident-management plan.

As part of the activity to define the attributes of an accident-management plan, it was recognized that the ability of plant instrumentation systems to present appropriate and accurate information to the plant staff was a critical element in the successful management of severe accidents. Thus research was initiated to develop and test a methodology to be used to systematically evaluate the capabilities of plant instrumentation. The approach chosen was to develop "tree structures" for each of the critical safety objectives in the plant: to prevent core dispersal from vessel, to prevent containment failure, and to mitigate fission product release from the containment. A tree was then constructed defining (1) safety functions that must be maintained, (2) challenges to the safety functions, (3) the challenge mechanisms, (4) appropriate strategies to prevent or mitigate these mechanisms, and (5) the availability and environmental qualification of the instrumentation, precursor indicators, etc., that are essential to evaluating safety parameter availability to the staff. Work in 1989 was completed for evaluation of a representative pressurized water reactor (PWR).

As a part of the strategy assessment activity, the NRC staff surveyed NUREG-1150, probabilistic risk assesssment (PRAs), and other documents and identified a list of candidate strategies to prevent or mitigate certain severe accident conditions, with the maximum use of existing plant facilities and available resources. These accident-management strategies are those considered relatively mature and well understood. The NRC, with contractor support, assessed the general applicability of each strategy and investigated to ensure that no potential adverse effects had been overlooked.

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 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 describe techniques used by the staff to evaluate specific situations. Others provide guidance to applicants concerning the information needed by the staff in its review of applications for permits and licenses. Many NRC guides refer to or endorse national standards (also called "consensus standards" or voluntary standards) that are developed by recognized organizations, often with NRC participation. The NRC makes use of a national standard in the regulatory process only after an independent review by the NRC staff and after review of public comment on the NRC's planned use of the standard.

The NRC encourages comments and suggestions for improvements in Regulatory Guides and, before staff review is completed, issues them for comment to many individuals and organizations, along with the value/impact statements that set forth the objectives of each guide and both its expected effectiveness and its likely impact, in terms of resources and effort involved.

The strategies and their assessment are being presented for the licensees' consideration during their conduct of individual plant examinations (IPEs).

The intentional depressurization of a PWR reactor coolant system (RCS) was examined analytically as an accident-management strategy for mitigating the possibility of direct containment heating (DCH) during a station blackout transient. A sequence involving the loss of all a.c. power and immediate loss of auxiliary feedwater, with depressurization, was simulated from transient initiation to a point after the relocation of molten material to the lower plenum was predicted to occur. Two strategies to mitigate DCH by depressurization of the RCS were considered. One strategy, called early depressurization, assumed that the reactor head vents and pressurizer power-operated relief valves (PORVs) were latched open at steam generator dryout. The second strategy, called late depressurization, assumed that the heat vents and PORVs were latched open at a core exit temperature of 922 K (1200°F). Depressurization of the RCS to a low value that might mitigate DCH is predicted prior to reactor pressure vessel breach with both early and late depressurization. Based on current analyses, late depressurization is preferable to early depressurization or to no action.

PLANT PERFORMANCE

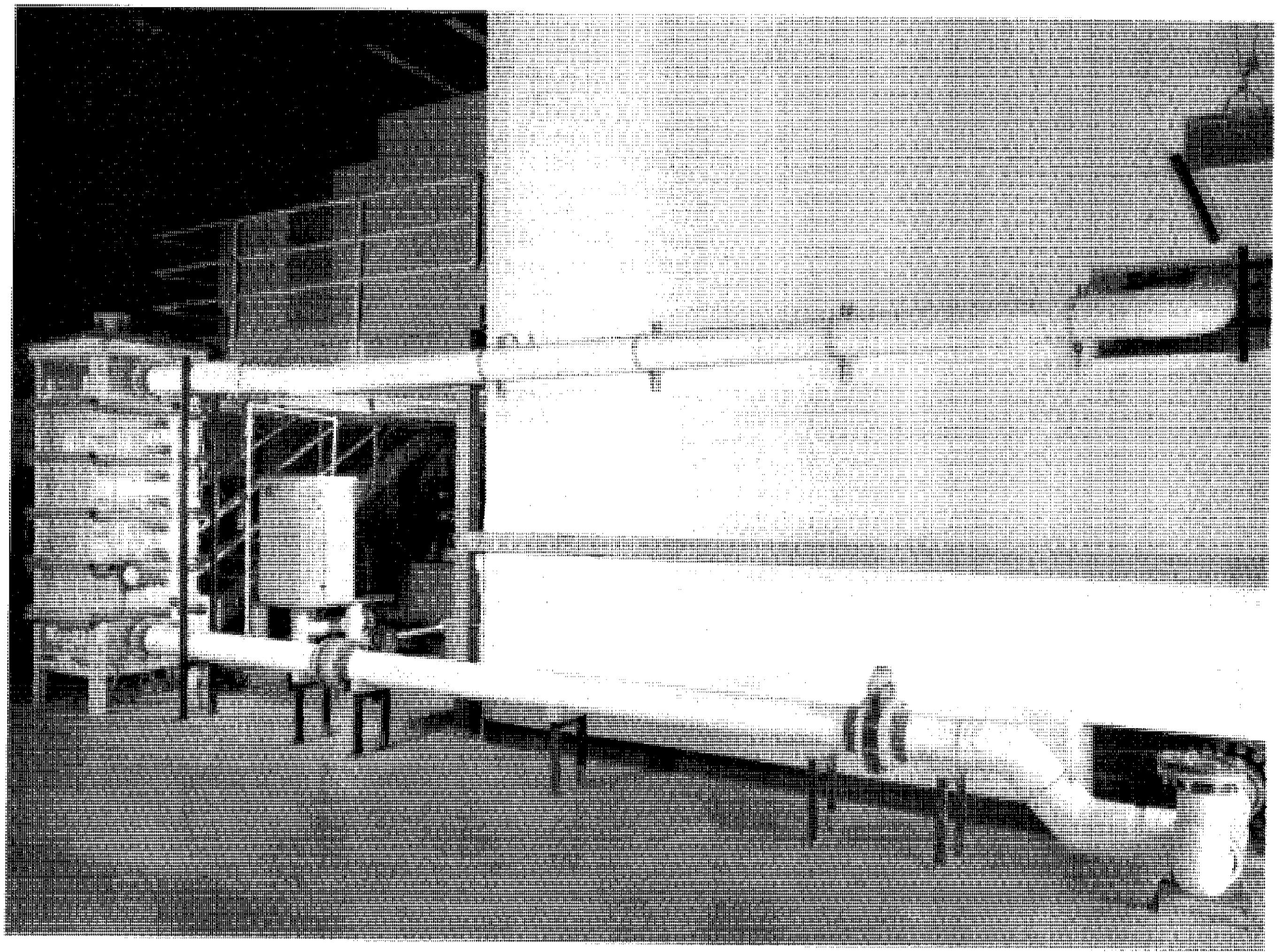
Modeling

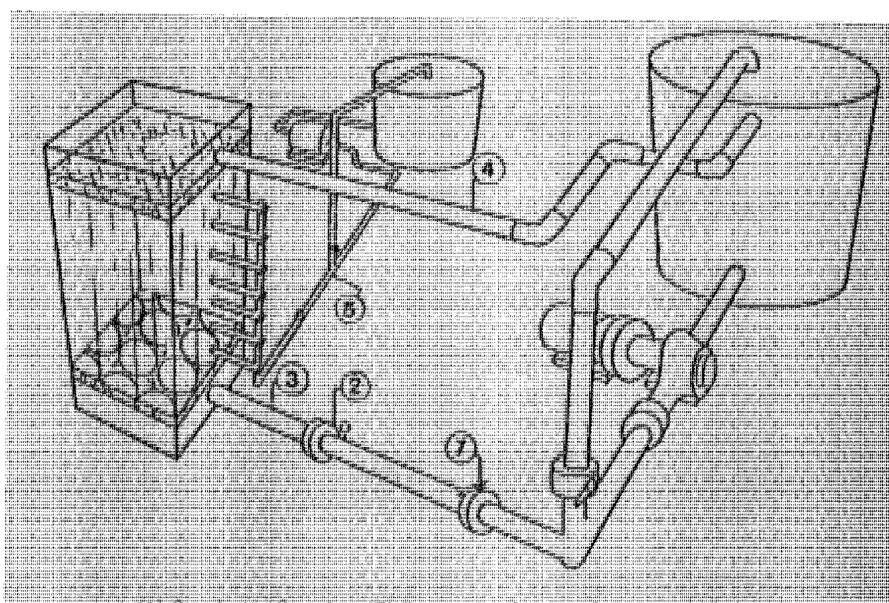
General Design Criterion 12 (in Appendix A to 10 CFR Part 50) states that the reactor core and associated coolant, control and protection systems shall be designed to ensure that power oscillations of a kind that could produce conditions exceeding the specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The power oscillation event at the LaSalle (Ill.) reactor in March 1988 raised two basic questions: (1) what is the potential extent of fuel damage resulting from power oscillations if they are not detected and suppressed, and (2) what are the potential implications of instability with respect to "Anticipated Transient Without Scram" (ATWS) events (where the oscillations might complicate the recovery).

To address these questions and provide audit capability for analysis of industry solutions, four computer codes are being used: RAMONA-3B, TRAC-BF1, APUR, and HIPA. During 1989, the four codes were appraised for their usefulness in analyzing instability, the code validation requirements were determined, and the code assessment calculations were completed. The physical processes identified as causes of the LaSalle event were confirmed. The power increase that occurs during an oscillatory event was quantified, and the conditions leading to asymmetrical oscillations were identified.

B&W Testing

Multiloop Integral System Test (MIST) Program. The MIST program is a joint government/industry experimental attempt to obtain information on the thermal-hydraulic behavior of Babcock and Wilcox





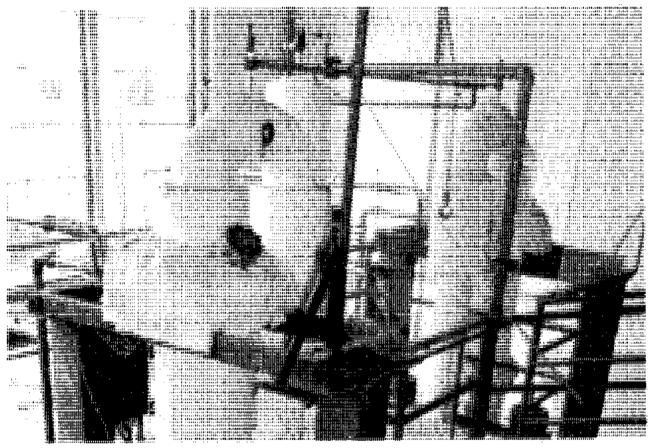
(1) flow butterfly valve; (2) flow measuring nozzle; (3) inlet probe; (4) outlet probe; (5) injection system.

Boiling water reactors rely on the injection of a soluble neutron absorber (sodium pentaborate) to control power in the event of a complete failure of the control rod scram system. Achieving reactor shutdown calls for an effective mixing of the injected liquid with the coolant that is recirculating through the core and other parts of the reactor. Because the flows are rather low and the density of the injected solution is much higher than that of the hot reactor water, concerns have been raised about the mixing process. Experiments at the University of California at Santa Barbara support the conclusion that complete mixing will occur with recirculation flow rates down to 8.2 percent of the normal full flow rate. The experimental apparatus is shown above, with schematic at right.

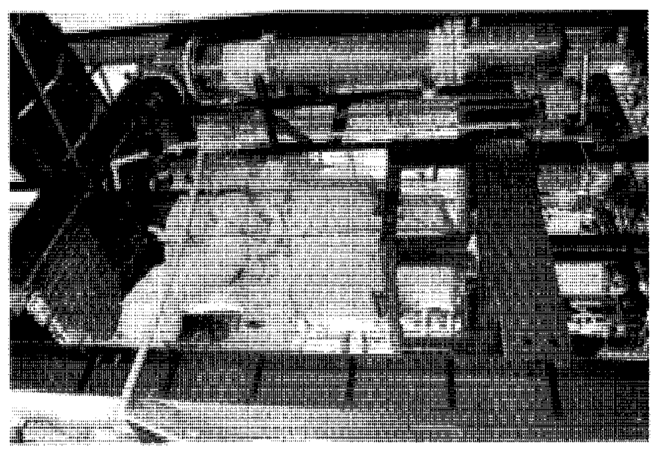
(B&W) reactors. The participants in the program are the NRC, B&W, the Electric Power Research Institute, and the B&W Owners Group. The experimental program involves conducting tests in the MIST facility, which is designed to operate at typical plant pressure and temperature. The MIST facility, located in Alliance, Ohio, is a scaled model of a B&W lowered-loop nuclear steam supply system. The experimental data for MIST have proved sufficient to validate calculations from B&W small-break loss-of-coolant accident (LOCA) models. (See the *1988 NRC Annual Report*, p. 126, for background.)

Since December of 1985, 62 tests have been conducted in the MIST facility. The tests investigated the thermal-hydraulic behavior in MIST for small-break OCA transients, steam generator tube rupture transients, "feed-and-bleed" recovery procedures, the effects of non-condensible gas and reactor coolant pump operation on transient progression, and strategies to cope with station blackout. In fiscal year 1989, the analyses of the first 58 tests were completed, and four natural circulation tests were successfully conducted. Various tests based on their similarity are grouped together and published in 11 NUREG/CR reports. These reports include a summary of the findings from all the testings, data and analyses of all the tests, and code comparisons against selected tests. Seven of 11 volumes were published in fiscal year 1989; the remainder are to be published in fiscal year 1990. All testing and essentially all documentation are now completed. The MIST project is scheduled for official closeout near the middle of fiscal year 1990.

B&W Once-Through Steam Generator (OTSG) Data Needs. In fiscal year 1989, the NRC and the Babcock & Wilcox Owners Group (B&WOG) concluded their joint effort, initiated in 1988, to investigate thermal-hydraulic issues and data needed to further understand the Babcock & Wilcox once-through steam generator (OTSG) behavior under transient or accident conditions. A report describing all the work completed in this effort was published in March 1989. As to whether additional experimental data are needed, the report concluded that (1) although there is little prototypic data related to OTSG auxiliary feedwater (AFW) behavior under transient and accident conditions, best-estimate code sensitivity studies showed that detailed knowledge of AFW behavior in an OTSG is not required for transient predictions; (2) certain OTSG geometry-dependent thermal-hydraulic phenomena, most of which are associated with OTSG depressurization resulting in excess steam flow (e.g., entrainment, de-entrainment, liquid carryover) were found to have data deficiencies. The deficiencies identified only affect data needed for assessment of best-estimate thermal-hydraulic code predictions of some B&W transients, and not data required for licensing purposes. The B&WOG agreed to go forward in a joint effort with NRC to obtain the necessary experimental data. However, because of NRC budget constraints and licensing fee questions raised by the industry, both parties agreed not to pursue experimental testing.



A scale model of the primary and secondary loops of a Babcock & Wilcox pressurized water reactor, designed and constructed at the University of Maryland, operates at a maximum pressure of 300 psi and has a maximum heat input of 180 kilowatts. The model is one-fourth the height of an actual B&W reactor, with a nominal volume scale of 1-to-500. Test performed in the model complement the MIST program (see "B&W Testing," under "Plant Performance," in the text).



Experiments and Analyses

2D/3D Program. Under the 2D/3D International Loss-of-Coolant Accident (LOCA) Research Program, the final test series in the full-scale Upper Plenum Test Facility (UPTF) has been completed. This now completes all testing in the German as well as the Japanese facilities used in this program. The UPTF tests have provided answers to major issues involved in large-break LOCAs and in transients that could lead to pressurized thermal shock (PTS).

The UPTF data analyses show that emergency core coolant reaches the lower plenum much more quickly than previous small-scale tests had indicated. While most of the emergency core coolant injected in the cold leg near the break bypasses the core by flowing out the break, most of the emergency core coolant injected in the cold legs farther away from the broken cold leg does not bypass the core.

The data obtained served to resolve a number of other technical issues. For example, the UPTF fluid mixing tests showed that emergency core coolant and the primary coolant mix very well, and thus the PTS concern is greatly alleviated. It was also found that countercurrent flow limit (CCFL) in the hot leg was observed for steam flows that are much greater than those expected under PWR small-break LOCA conditions. That finding relieves concern that decay heat removal might be impeded or prevented under such conditions.

The Federal Republic of Germany (FRG) plans to extend the use of the UPTF to the accident-management area. Possible U.S. participation in this extended use was being negotiated at the close of the report period.

RELAP5 Code. The RELAP5 code is used to analyze thermal-hydraulic phenomena occurring in the reactor system for a variety of transients. The Idaho National Engineering Laboratory (INEL) is the principal laboratory responsible for the maintenance and development of the code. For the past few years, the code was evaluated extensively by various organizations, under the International Code Assessment Program (ICAP). At the NRC's direction, INEL developed a code improvement plan to correct deficienciesidentified by ICAP. The NRC and Siemens KWU from the FRG (an ICAP member) jointly funded this developmental effort in fiscal years 1988 and 1989. In addition, many ICAP members participated actively in the effort by contributing new models, as well as by providing personnel at INEL to implement these models. As a result, a MOD3 interim version of the code was developed and distributed to various organizations for assessment. These organizations—which include INEL, Yankee Atomic Electric Company, Northeast Utilities, Siemens KWU, AEC-Winfrith/United Kingdom, and Imatran Voima Oy/Finland—completed their assessments of the code employing separate-effect and integral experiments. INEL served as the coordinator in compiling the results from all these developmental assessments. The final version of the RELAP5 code (RELAP5/MOD3) was scheduled for release during the first quarter of fiscal year 1990. Further assessments of the code will be performed under the ICAP through 1991.

HUMAN PERFORMANCE

In close coordination with the NRC Office of Nuclear Reactor Regulation, the Office of Nuclear Material Safety and Safeguards, and the Office for Analysis and Evaluation of Operational Data, RES revised the Human Factors Regulatory Research Program Plan and, on June 16, 1989, issued the Commission Information Paper, SECY-89-183, "NRC Human Factors Programs and Initiatives."

This paper updated the research program and provided the regulatory context for the ongoing research. Both the Commission Paper and the human factors research program plan were subsequently issued as NUREG-1384. The human factors research program is conducting research on personnel performance measurement, the personnel sub-system, human-systems interfaces, organizational factors, probabilistic data acquisition and quantification, data management systems, human reliability analysis (HRA) and PRA integration, HRA and PRA results applications, and human factors generic issues. These activities, grouped into (1) human factors research, and (2) reliability assessment research, are discussed below.

Human Factors Research

The personnel performance measurement program has provided the means for improving the collection, screening, storage, retrieval and analysis of data relevant to other activities as well. Two new initiatives under the program are the development of a human factors investigation method—to be used for identifying the causes of human error involved in a reportable event—and the establishment of a comprehensive data management system organizing personnel performance information.

Personnel sub-system research on the effects of overtime continued during the report period, focusing on the correlation between working hours and incidents at nuclear power plants, and on the effects of 12-hour shifts on nuclear power plant operations. Work has begun on the development of training effectiveness methods, starting with a workshop involving experts in the field.

Human systems interface research continued with NRC participation in the "Halden Project," including a workshop of experts convened in January 1989 as a first step toward developing effective tools to measure and evaluate computer-driven interfaces. The results of the workshop were published in NUREG/CR-5348. A survey of the commercial nuclear power industry's current and planned use of artificial intelligence, expert systems, and computers has been completed. The survey identified existing and proposed uses of these high technology systems and schedules for their implementation. In follow-up research related to procedural aspects of the Chernobyl accident in the Soviet Union, a study was undertaken to determine the nature and extent of procedure violations in the U.S. and their consequences. Assessment of the costs and benefits of expanded regulatory guidance on other than emergency operating procedures (EOPs) is continuing and will include judgments as to whether and where upgrades are necessary for improving the use of normal and abnormal operating procedures.

In organizational factors research, a two-day workshop was conducted, with a diverse group of participants, resulting in recommendations of new research into (1) the organizational factors influencing human performance, and (2) the technology for translating measurable organizational factors into reliability and risk assessments. A conceptual model was developed for characterizing and measuring organizational factors, and a field characterization process for selected organizational factors was tested for further development.

Reliability Assessment Research

This continuing RES program provides the tools and data necessary for assessing human performance in ways adaptable to plant probabilistic risk assessment (PRA) studies, and systematically applying the results of those studies to the resolution of generic issues and consequent regulatory decision-making. Major activities included: (1) development of a direct link between the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) automated data management system and other computational codes, such as Integrated Reliability and Risk Analysis System (IRRAS), and expansion of both human and equipment failure rate categories to accommodate common-cause failure and team performance data; (2) field testing of an artificial intelligence-based Cognitive Environment Simulation (CES) and its reliability assessment application by means of a Cognitive Reliability Evaluation Technique (CREATE); (3) field evaluation of the user friendliness and utility of the Maintenance Personnel Performance Simulation (MAPPS) at licensed facilities; (4) development of criteria for equating human actions in nuclear facilities with human actions in military, aviation, and surface transportation facilities, in order to use the data to support reliability assessments of nuclear facilities; (5) development of a technique to manage the integration of human and hardware expertise into the reliability assessment process; (6) field implementation and testing of the safety-system-function-trend indicator involving auxiliary feedwater, diesel generators, and heat removal systems; (7) programmatic work focused on development of leading indicators of safety related to maintenance, training, and organization problem-solving; (8) resolution of TMI Action Item II.C.4, regarding the application of reliability engineering to maintain operational safety throughout the plant life cycle; and(9) resolution of the Chernobyl follow-up item on NRC Testing Requirements by drawing upon ongoing work. In addition, guidance was provided for the human reliability analyses that are to be performed with the individual plant examinations.

During fiscal year 1989, cooperative research with other governmental and private organizations was initiated through joint funding of a core of research tasks directed by the Human Factors Committee, National Academy of Sciences, National Research Council. The core research will focus on human error, organizational effectiveness, the augmentation of intelligence functions by artificial intelligence, and the acquisition of human factors performance information.

Reactor Containment Performance

To ensure that existing regulations related to severe accidents adequately protect the public, research is needed to confirm the technical bases upon which the regulations are founded. These bases include 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 environment is also an important consideration in protecting the public. With these kinds of data, 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 terms" are identifications of the quantity, timing, and energy of radioactive materials released to the environment following a postulated severe reactor accident. The NRC conducts research in this area to help define and focus accident-management concerns, containment performance improvements, and individual plant examinations to seek out potential vulnerabilities previously undetected.

At present, research is under way to develop theoretically based fission product behavior models by which to predict fission product release and transport in the reactor coolant system and the containment. As described in the *1988 NRC Annual Report*, p. 130, 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 debris is expelled from the reactor vessel, and the extent of fission product revaporization from the reactor coolant system. The code called TRENDS is being developed to estimate the partition of iodine between the aqueous phase and the gas phase in the containment, the production of organic iodide species, boiling-water reactor (BWR) suppression pool chemistry, and the extent of iodine revaporization and resuspension from containment surfaces and sumps.

Besides the fission product research cited above, the NRC is participating in an internationally sponsored project called Advanced Containment Experiments (ACE). The project comprises three 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; and phase C deals with molten core-concrete-interaction research. Three integral core-concrete-interaction tests have been conducted addressing the effects of various corium compositions on typical concrete substrates. Phase C of the program was expanded to address melt cooling issues, seeking a determination as to what debris configurations (power level and depth) can be cooled by an overlying water pool. A scoping test was performed to demonstrate the viability of such experiments.

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 counter-current 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 period of certain severe accidents in a pressurized water reactor (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) 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 description of calculation analyses, see the *1987 NRC Annual Report*, pp. 134 and 135.) However, uncertainties in these calculations are yet to be estimated or bounded, and more work is needed to validate the results.

REACTOR CONTAINMENT SAFETY

Core-Melt Progression and Hydrogen Generation

In-vessel core-melt progression is concerned with the state of the reactor core in a severe reactor accident from the initiation of core uncovery up to reactor vessel melt-through, including the mode of reactor vessel failure. Sensitivity studies to date suggest that uncertainties in the state of the core at vessel failure (the melt mass, composition, and temperature) generate the greatest uncertainties in assessing the core-melt threat to the integrity of the containment. The details of core melting are also primary determinants of in-vessel hydrogen and fission product generation.

Current knowledge of in-vessel severe accident behavior has come from experiments such as the series of severe fuel damage tests performed in the Power Burst Facility test reactor that included extensive post-irradiation examination (PIE) and from the extensive core examination of the Three Mile Island Unit 2 (TMI-2) reactor performed by the Department of Energy (DOE). During fiscal year 1989, the NRC undertook to procure and examine test specimens from the lower head (bottom) of the TMI-2 reactor vessel (which did not fail), including the head penetrations, in order to obtain information on the melt attack on the lower head and on the margin-to-failure of the lower head during the accident. New research on the mode of vessel failure under melt attack has been started that goes beyond current experiments on vessel penetration failure and the current theoretical analysis of vessel failure modes.

During the report period, there was a particular emphasis on severe accident behavior in boiling water reactors (BWRs), which are significantly different from the PWRs to which most existing severe accident data relate. This BWR research is relevant to the BWR Mark I liner melt-through investigation, suggesting the initial conditions for the core-melt attack on the liner. Most of the current information on severe accident behavior in a BWR core has come from the DF-4 (damaged fuel) experiment in the Annular Core Research Reactor (ACRR). DF-4 results suggest that the boron-carbide control blades and the zircaloy channel boxes that isolate the fuel assemblies in a BWR are highly significant. In 1989, results from the extensive PIE of the DF-4 test assemblies and analysis of the experimental results have contributed considerably to an understanding of core-melt progression. Information from DF-4 results is being augmented by a series of out-of-reactor experiments in the West German CORA fuel damage test facility, in which the NRC participates closely, and a full-length BWR geometry test in the Canadian National Reactor Universal (NRU) reactor was in preparation at the close of the report period.

Except for the results of the TMI-2 core examination, there are no available data on the late phase of core-melt progression, namely, significant melting and relocation of the ceramic fuel. Preparations have been made for a new program of relatively small separate-effect experiments in the ACRR on the mechanisms of failure (thermal, chemical, and mechanical) of a metallo-ceramic lower crust across the fuel-rod stubs, such as that which supported the growing molten fuel pool in the core at TMI-2 during the accident. The failure threshold and location determine the mass of fuel melt that drains from the core, as TMI-2 demonstrated. The initial experiment in this series, MP-1 (melt progression), was performed during fiscal year 1989.

Core-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 core-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.

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. CORCON MOD2 (released August 1984) includes the effects of head and mass transfer, attack on structural concrete in the reactor cavity, and the influence of an overlying water layer. CORCON has now been integrated into the ONTAIN and MELCOR codes. Most recent code modeling improvements include experimentally based models for interphase heat and mass transfer between oxide and metallic components in the core debris. Additional efforts have addressed condensed-phase chemical reactions of zirconium, transient heat conduction into concrete, and improved axial heat transfer models. Considerable effort was expended in fiscal year 1989 in the validation of the CORCON code by comparison with integral and separate-effect core-concrete-interaction tests. The code is used in research institutions throughout the world. Large-scale integral experiments with sustained induction heating were continued, in order to study the effect of core debris mixtures of various compositions interacting with limestone and siliceous 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 initiated, and data are used to assess the VANESA code. The degree to which refractory radionuclides are thrown off from molten debris depends in part upon the relative vapor pressures of the pool constituents. A refined model, based on recent high-temperature measurements of chemical activity coefficients, is in preparation for incorporation in VANESA. Consolidation of the VANESA and CORCON codes into a single code, CORCON MOD3, was initiated during the report period 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 have been initiated to determine the manner in which it may spread and relocate within the reactor cavity. The ability of the BWR Mark I steel drywell shell to survive a core-melt accident may depend upon such debris behavior. With respect to that same Mark I safety issue, studies of heat transfer from high-temperature melts to non-horizontal steel barriers have also begun.

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. 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).

A research program was developed at the Sandia National Laboratories to explore the phenomenon of core debris dispersal at various scales. The 1/20th linear scale system pressure injection tests (SPIT) and the 1/10th linear scale high-pressure screening tests (HIPS) have been completed. Four 1/10th-scale tests have been conducted in the Surtsey facility at Sandia to investigate energy exchange and the generation of aerosols, for the purpose of identifying and quantifying the phenomena important to DCH accidents. In fiscal year 1989, three experimental programs were under way to generate information from which to develop a data base to estimate the risk associated with high-pressure core-melt accidents. The three programs are: (1) the low-pressure cutoff melt dispersal 1/42th-scale experiments, at Brookhaven National Laboratory; (2) the 1/10th-scale high-pressure melt ejection scale tests, at Sandia; and (3) the companion 1/30th-scale tests, at Argonne National Laboratory. Details of these programs are set out in the *1987 NRC Annual Report*, p. 137.

Hydrogen Combustion

The hydrogen combustion program assesses both the consequences and methods used to control or mitigate deflagrations, diffusion flames, accelerated flames, transition from deflagration to detonations (DDT), and detonations that might be caused by hydrogen burns in a severe reactor accident. The HECTR lumped-parameter computer code was developed at the Sandia National Laboratories and is used in the analysis of nuclear reactor accidents involving the transport and combustion of hydrogen. A flame propagation model was incorporated into HECTR. The HMS-BURN code, a three-dimensional finite-element analysis tool developed at Los Alamos, is also employed to provide more detailed hydrogen transport and mixing calculations. The assessment of HECTR and HMS-BURN codes continues, employing data generated from the large-scale hydrogen transport experiments performed at the HDR reactor in the Federal Republic of Germany.

Flame acceleration, deflagration-to-detonation transition, and detonation experiments have been analyzed and documented. A review of the effect of elevated temperature and high steam concentration on the various modes of combustion is now complete. The

The Sandia National Laboratories, shown at left, is the site of NRC-sponsored research to explore core debris dispersal. The Main Technical Area of the Sandia complex is shown, comprising scientific and technical laboratories, administrative buildings, specialty shops and environmental test facilities.

combustion of hydrogen mixtures at elevated temperatures and steam concentrations typical of severe accidents represents one of the important remaining uncertainties. The ZND detonation model predicts increased likelihood of detonations at elevated temperatures. In 1989, an experimental plan was developed to address the data needs for detonations and other combustion modes.

CONTAINMENT STRUCTURAL INTEGRITY

Structural Tests

The NRC participated in a test of a model of the Sizewell "B" containment performed by the Central Electricity Generating Board in the United Kingdom. Sizewell "B" is a PWR housed in a prestressed-concrete containment. The design pressure used for the containment and the model is 0.345 MPa. The containment structure is based on a Bechtel design—making it very similar to some of the prestressed containments in the United States. The containment model was tested to structural failure to demonstrate its pressure reserve and provide data for computer analyses. One major difference between the containment model and full-size prototype is the lack of a steel liner in the model, which used a rubber bladder as the pressure boundary during the hydrostatic tests.

The 1/10-scale model was pressurized several times to 1.15 times its design pressure, during the week of July 24, 1989. Testing culminated in an over-pressurization test to failure that was completed on July 31, 1989. The containment model carried over 500 sensors embedded in or attached to it which monitored its behavior during testing. Intensive study of the model and the data was under way at the close of the report period. Japanese and French authorities also participated in the experiment, which was intended to check the fidelity of predictions of behavior up to the point of failure. An assessment by the participants will be completed in early 1990.

Equipment Hatch Tests

These tests were part of the effort to develop a more complete understanding of research results after a 1/6-scale model of a reinforced concrete containment was tested to failure in July 1987. (For a description of the model, see the 1987 NRC Annual Report, p. 135.) A 40-inch (1.0-m) diameter equipment hatch, typical of equipment hatches found in U.S. containment buildings, was incorporated into the model.

The first test on the scaled pressure unseating equipment hatch was with an aged gasket with total bolt (10 bolts) pre-load of 57.2 kips (5.72k/bolt; a kip is 1,000 lbs. deadweight). The resultant finding was that gross leakage depends on average response of the hatch (around the circumference) and not on local deviations. Three other tests performed to date on the scaled equipment hatch include variations in gasket material, gasket aging, bolt stiffness, and bolt pre-load.

Seven more tests are planned on the scaled equipment hatch which will include high temperature, in addition to the test variations mentioned above. A final NUREG report on the hatch tests is expected by early 1990.

REACTOR ACCIDENT RISK ANALYSIS

Review of PRAs

Probabilistic risk analysis (PRA) is used by the NRC staff to support the resolution of a wide 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 1989 included the following:

Three Mile Island Unit 1 (Pa.). This PRA was submitted voluntarily by the licensee for NRC staff review. The review has uncovered several weaknesses in the licensee's analysis, which were under discussion with the licensee at the close of the report period.

Brunswick (N.C.). This PRA was also 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 and a draft report was under review at the close of the report period.

Crystal River (Fla.). This PRA was also a voluntary submittal by the licensee. The review was completed and a final report issued in fiscal year 1989.

Browns Ferry (Ala.). This PRA was done by the licensee, but the review was initiated by the NRC staff. The review has been completed, and the licensee is updating the PRA, partly in response to review results.

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 is performing a PRA. Because the seismic portion of the work involves the development of some new PRA methods, the staff review is proceeding as the various stages of the PRA are being performed. The review was nearly completed, and a draft report in preparation, at the close of the report period.

CESSAR System 80 Plus. A review of a PRA for the CESSAR System 80 (based on the Palo Verde (Ariz.) facility) was completed. The PRA and its review are used by the applicant in the design of the CESSAR System 80 Plus. Thus, even though this was a PRA on the existing standard System 80 design, the work is being performed as part of the System 80 Plus application.

GE Advanced BWR. A PRA has been submitted as part of the licensing application for this advanced BWR. Review of this PRA, which is being submitted in several modules, was under way at the close of the report period.

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 risk analyses on regulatory issues such as the technical bases for present emergency planning regulations and 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. In addition, the American Nuclear Society sponsored and published a review of the draft report.

The NRC staff and supporting contractors have updated the five risk analyses. The updates, which are quite extensive, are 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 completed new 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, has begun a formal review of the document. The review is now expected to be completed by the summer of 1990.

New 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-technology methods for performing and reviewing PRA and to develop, maintain and provide quality assurance for such methods.

In support of NRC staff performance and review of PRAs, a new, fast-running computer model for in-plant severe accident analysis has been developed. The model, MELCOR, analyzes such accidents from the initiating event, such as a pipe break, through core degradation and vessel and containment failure (i.e., when all core and containment protection systems have failed). Version 1.8 of MELCOR has been completed, with subsequent delivery of the code and draft code documentation to users. This version and previous versions have seen significant use in staff severe accident analyses and in the analyses of DOE facilities (under contract to DOE).

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 in time to permit its use in the final consequence calculations in the second draft of NUREG-1150. Final quality assurance and benchmarking of the code with international standard problems was in progress at the close of the report period.

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 in which 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 and unresolved safety issues. Other uses, such as targeting inspection activities, 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 needs described above and also to provide the NRC with 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 1989, feedback from a limited trial of the two codes was incorporated and "production" versions made final. Courses have been conducted to train staff personnel in the use of the codes.

By means of these codes and other methods, risk analysis support was provided to the staff in support of the resolution of a number of issues, including:

- An evaluation of a diesel generator technical specification change at Grand Gulf.
- An analysis of the impact of degradations in the reliability of reactor protection system trip breakers.
- A review of the most important accident sequences potentially leading to core damage for the Browns Ferry plant, for an NRC Senior Management Meeting.
- An analysis of the impact of degradations in the reliability of the high-pressure injection system and auxiliary feedwater system in Surry, for an NRC Senior Management Meeting.
- An analysis of the impact of degradations in the reliability of the high-pressure coolant injection and reactor core isolation cooling systems in the Brunswick plant, for an NRC Senior Management Meeting.
- An analysis of alternative accident-management strategies.
- A research information letter on the probabilistic evaluation of allowable outage times and surveillance intervals.
- An analysis of the impact of check valve failure probabilities on the estimated core damage frequency at the Surry plant.
- A report on insights into plant safety emerging from probabilistic analyses, presented to the Senior Management Meeting.

Integrity of Reactor Components

That sector of NRC research activity dedicated to the integrity of reactor components examines reactor plant systems and components to see that they perform as designed and that they continue to do so over the life of the plant. Reactor safety 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 operators' ability to cool the reactor core and could lead to a loss-of-coolant accident accompanied by release of hazardous fission products.

REACTOR VESSEL AND PIPING INTEGRITY

Pressure Vessel Safety

The reactor pressure vessel is the key element in the primary pressure boundary. It houses and supports the reactor core and provides channelling 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. This effort addresses the 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, there were three areas 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 1989, the research included work on developing and refining analysis methods and evaluation criteria for reactor pressure vessels fabricated with welds that could be susceptible to low-energy ductile fracture, developing crack arrest data and analyses, and designing pressurized thermal shock experiments by which to assess low-energy ductile fracture and stainless steel cladding effects.

The NRC's regulations require that precautions be taken to avoid non-ductile failure of the reactor pressure vessel. They also require that the ductile fracture resistance remain above a specific limit, as measured by the material's "Charpy V-notch upper-shelf" energy. If the upper-shelf energy falls below the

50 ft.-lb. regulatory limit, a detailed analysis must be performed to demonstrate that an adequate margin against failure is ensured, or the vessel must be thermally annealed. There are some vessels currently in service with welds in which the Charpy V-notch upper-shelf energy is projected to fall below the existing regulatory limit before the end of the vessel's design life. These welds are commonly called "low upper-shelf" welds. Research has begun to determine whether there is a firm technical basis justifying continued operation below the 50 ft.-lb. limit and to validate the salutary effects of thermal annealing.

During fiscal year 1989, NRC-funded research evaluated both the technical aspects of the low-upper shelf weld problem and the technical acceptability of proposed regulatory criteria for permitting operation below the 50 ft.-lb. limit. Significant progress was made during 1989 in understanding and resolving several of the technical aspects of the problem. The U.S. Navy's David Taylor Research Center organized an *ad hoc* group to resolve questions about the correct methodology for analyzing the fracture toughness data from low upper-shelf welds. Group participants included researchers from government, industry, and universities. The work was completed and final recommendations will be forwarded for consideration by the NRC and the ASME Section XI Code Committee.

Researchers at the Oak Ridge National Laboratory (ORNL) participated in a joint venture with the Electric Power Research Institute and the Babcock & Wilcox Owner's Group to remove the low upper-shelf welds from the Midland Unit 1 (Mich.) reactor pressure vessel (the Midland plant was cancelled before startup). The welds were inspected using state-of-the-art techniques to determine the number and size distribution of defects in these welds. This information will be used in improving the initial flaw data used in probabilistic analyses of reactor pressure vessel integrity. The welds removed from the Midland Unit 1 pressure vessel are being used in destructive testing to determine the fracture behavior of these welds in both the irradiated and unirradiated conditions.

The Heavy Section Steel Technology (HSST) program continues to perform most of the NRC's pressurized thermal shock (PTS) research. The research in recent years has been focused on crack arrest evaluations and on benchmark experiments to define specific details of postulated PTS accidents and the possible vessel fracture associated with them. During 1989, ORNL researchers used the analytical tools developed in past years to evaluate the relative importance of several variables, including crack arrest toughness, on the risk attributable to PTS. The earlier analyses had indicated that, in general, high values of

An *ad hoc* group of researchers from government, industry and universities was brought together at the U.S. Navy's David Taylor Research Center, above, to deal with methods for analyzing fracture toughness data, during fiscal year 1989. The Center is located at the mouth of the Severn River, at Annapolis, Md.

crack arrest were beneficial. The effort started in 1989 explored not only the impact of high crack arrest toughness levels, but the shape of the crack arrest toughness curve and the impact of ductile tearing that might follow arrest of a rapidly propagating crack. The results illustrated that the shape of the crack arrest curve and the possibility of ductile tearing following crack arrest are as important to vessel integrity as achieving high crack arrest toughness levels. In parallel with the crack arrest analyses, ORNL researchers were working with others in the national and international community to evaluate the effects of other factors, such as crack tip constraint. There is a general consensus that some of these other factors are as important to vessel integrity as crack arrest toughness. That conclusion prompted the NRC to reduce the emphasis on crack arrest testing and to increase attention to these other factors.

The 1989 efforts sharpened the NRC's focus on fracture evaluation research and has resulted in a significant restructuring of the overall effort. The two PTS experiments planned for 1990-1991 were dropped from the overall plan. Effort was directed instead toward detailed analyses of other large-scale benchmark experiments that have been performed in the international community, as part of international activity under the auspices of the Committee on the Safety of Nuclear Installations (CSNI) Principal Working Group 3. The results of this research are expected to point up the technical areas that warrant further work, as against those that are already adequately developed. As a result of these analyses, additional large-scale experiments may be warranted in the future.

Radiation Embrittlement. Neutron radiation embrittlement of reactor vessels has been found to be higher in many plants than previously thought. The NRC's regulatory documents are being updated to reflect this realization. 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 embrittlement of reactor vessel materials is characterized by changes in a "reference temperature for nil-ductility transition," which can be characterized as follows. For many reactors now in operation, the toughness of certain vessel materials at room temperature is too low to permit full pressurization of the vessel with adequate safety margins. As temperature is raised, toughness increases, slowly at first, but then, at the "reference temperature," much more rapidly. At normal operating temperatures, vessel materials are quite tough.

To monitor radiation embrittlement in reactor vessels, specimens of the most radiation-sensitive materials are exposed in surveillance capsules positioned inside the vessel near the wall. Destructive tests of these specimens, when the capsule has been withdrawn after several years of exposure, provide the data for thorough study of the relationship of embrittlement to neutron fluence and material composition.

In May 1988, the NRC published Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," containing a correlation of the shift in reference temperature to neutron fluence and copper and nickel contents. Analyses of the surveillance data base by two independent investigators furnished the technical basis for the guide. Public comment reflected general agreement with the findings, and there was further peer review by two national standards committees that were using the guide as a basis for their standards. The guide was also checked against the considerable body of surveillance data generated since the original correlations were made, and it was found to be satisfactory. The maintenance and analysis of these data are done by ORNL using a comprehensive computerized collection of data, called the Embrittlement Data Base (EDB), from pressure vessel surveillance reports of commercial reactors as well as irradiation heat data from research reactors.

Work has begun to amend the PTS rule, 10 CFR 50.61, making the formula for reference temperature in the rule consistent with that in the guide. Publication for public comment was pending at the close of the report period. When the amended rule becomes effective in 1990, a few nuclear power plants will have reached the screening criterion given in the rule sooner than had previously been projected. At the close of fiscal year 1989, licensees for these plants were taking flux reduction measures to slow the accumulation of neutron fluence in the reactor vessel wall.

In addition to analyzing the surveillance capsule specimen data, the NRC is evaluating radiation embrittlement in certain research programs. These research efforts use test reactors to provide accelerated embrittlement of various reactor pressure vessel materials so that many different variables can be evaluated in a relatively short period of time. In fiscal year 1989, results from the fifth series of test reactor irradiations performed in the HSST program were analyzed. They indicated that the method used by the American Society of Mechanical Engineers (ASME) for accounting for radiation embrittlement effects on fracture toughness slightly underestimates the actual loss in fracture toughness. The impact of these results is being assessed, and the possibility of a change in ASME procedures is being considered. Other studies are under way to evaluate the effects of neutron irradiation on crack arrest toughness, stainless steel cladding fracture toughness, and low upper-shelf weld fracture toughness. Actual material from the Midland Unit 1, cited above, will be used for the low upper-shelf experiments.

In 1986, ORNL discovered that surveillance specimens exposed to a low neutron flux for many years in their High Flux Isotope Reactor (HFIR) had suffered more embrittlement damage than had been projected on the basis of embrittlement data then available. The HFIR surveillance specimens and attendant data were scrutinized in great detail, leading to the conclusion that the greater-than-expected embrittlement may be due to a "flux effect" that had not been observed before. The data used in predicting the service life of the HFIR were derived from materials test reactor (MTR) data in a test where the specimens were irradiated at an accelerated rate of neutron bombardment, i.e., a high flux. The HFIR vessel was exposed to a flux five orders of magnitude lower than the MTR irradiation fluxes. Apparently, the lower rate of bombardment was more embrittling to the steel, producing the unexpected damage.

In order to examine the effects of actual power reactor operation on vessel supports, a program was initiated to study the results of low-temperature, low-flux irradiation on the mechanical properties of the neutron shield tank of the out-of-service Shippingport (Pa.) reactor. Availability of the Shippingport neutron shield tank (a vessel support structure) constituted an excellent opportunity to check such effects, because its material of construction is equivalent to the material used in present-day core support structures. (For further discussion, see below, under "Aging of Reactor Components.") In addition to the Shippingport neutron shield tank, efforts are under way to obtain samples from the decommissioned Belgian BR-3 neutron shield tank, and other sources of suitable materials are being sought.

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 to compute 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, in order to ensure that the plant is operating in conformance with NRC safety regulations.

A proposed Regulatory Guide identifying acceptable methods and assumptions for establishing pressure vessel fluence has been prepared for publication for public comment. The guide incorporates developments coming out of the surveillance dosimetry program.

Steam Generator Integrity

The Steam Generator Group Project at Battelle-Pacific Northwest aboratories (PNL) has been using an out-of-service steam generator from an actual PWR facility as a test bed for measuring the effectiveness of eddy current (EC) inspection techniques for detecting and measuring flaws in steam generator tubing. In addition, tube segments removed from the generator were burst-tested to validate empirical models of remaining tube integrity developed earlier. Testing of EC techniques prior to the current report period is described in the *1987 NRC Annual Report*, pp. 111 and 112.

In fiscal year 1988, based on results from this research, draft revisions of Regulatory Guides 1.83 and 1.121 for improved guidance on inservice inspection and plugging of steam generator tubes were prepared. In fiscal year 1989, initial value-impact analyses for implementation of the improved recommendations were completed and being refined.

Piping Integrity

Environmentally Assisted Cracking. A very significant problem encountered in BWRs has been the intergranular stress corrosion cracking of austenitic stainless steel piping at weldments. This condition has been responsible for hundreds of pipe-cracking incidents throughout the world over the last 10 years. Because these problems have resulted in extended and unscheduled outages—with extensive repairs and replacements, and significant occupational exposures—the NRC and the electric utility industry have devoted a good deal of research to their resolution. (For background on the issue, see the *1986 NRC Annual Report*, pp. 163 and 164, and the *1987 NRC Annual Report*, pp. 112 and 113.)

The use of alternative materials and other measures intended to mitigate intergranular stress corrosion cracking have been investigated. Three different grades of stainless steel—Type 316 NG, Type 347, and CF-3—have been evaluated under a variety of environmental and mechanical loading conditions and found to be significantly more resistant to cracking than the materials initially used in nuclear plant piping. However, tests have shown that, under certain water chemistry conditions, even these superior materials become susceptible to cracking. At normal reactor operating temperatures of approximately 290axaC, cooling water containing low levels of dissolved oxygen and sulfate was found to significantly increase the susceptibility of these materials to stress corrosion cracking.

Extensive research has been carried out to demonstrate the strong interactions among dissolved oxygen and various impurities, as well as the effects of individual impurities on stress corrosion of sensitized Type 304 SS in low-oxygen, high-temperature water. The data provide the basis for confirming the benefits of good water quality and the role of different impurities in stress corrosion cracking of sensitized austenitic stainless steels. The program is described in the *1988 NRC Annual Report*, p. 145.

The process of crack growth in weld-overlay repairs of cracked pipe has been studied in simulated BWR environments and at low strain rates. The test specimens were fabricated so that the crack would propagate through the original sensitized pipe material into the weld clad overlay. The results of the experiment indicate that cracks do not extend into the weld overlay, confirming the suitability of this type of repair. In fiscal year 1989, research effort in this area concentrated on an evaluation of irradiation-assisted stress corrosion cracking of reactor internals and on corrosion fatigue of pressure boundary materials. The significant effect of the BWR environment in reducing the fatigue life of Type 316 NG stainless steel was demonstrated.

A thermal aging program was initiated in 1982 to evaluate the long term effects on degradation of toughness in cast stainless steel as afunction of time of exposure, temperature, and material composition. Through 1988, results have been accumulating to allow a quantitative evaluation of the degree and significance of toughness loss at reactor operating temperatures and operational times. Also, the mechanisms responsible for the toughness loss are being identified by evaluating both laboratory-exposed specimens and specimens removed from actual components in nuclear power plants. By fiscal year 1989, enough data had been accumulated to evaluate the effects of material variables on embrittlement. Chemical composition and ferrite morphology of the casting have a strong effect on the extent and kinetics of embrittlement. Procedures and correlations for predicting the extent of thermal embrittlement (fracture toughness) of cast stainless steel components during reactor service were developed. A heat treatment has been devised for recovering toughness. However, re-embrittlement during subsequent exposure occurs at a much faster rate than the initial aging embrittlement.

Erosion/Corrosion. Very significant pipe wall thinning has occurred in a number of steel piping systems of nuclear plants because of "erosion/corrosion" of the material from high velocity single-phase coolant water. The problem was highlighted at the Surry Unit 2 (Va.) plant, where part of the feedwater piping had become so thin that the pipe failed catastrophically. A survey of 28 U.S. plants and two foreign plants was undertaken to ascertain the general experience with erosion/corrosion and to establish the significant variables that might be related to the problem. These variables included feedwater velocities, pressures, temperatures, water chemistry histories, and materials. A survey disclosed that the problem is significantly widespread.

A comprehensive review of the available data and current mechanistic understanding of erosion-corrosion revealed that susceptibility depends strongly on the interaction of flow and certain environmental and material variables. Thus, it is insufficient to identify the critical limit of just one variable, such as velocity or pH or geometry, for erosion-corrosion; all relevant factors must be taken into account together. A qualitative understanding has now been developed of the interaction of important variables, and quantitative predictive methods have also been developed. (The latter are subject to considerable uncertainty.) A workshop on the subject was conducted with scientists of the U.S.S.R., as part of the activities of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety.

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: would it leak or break? The matter of the leak-or-break alternatives had been addressed for years, without the emergence of a strong consensus. The NRC and the industry have pursued parallel research efforts in evaluating pipe fracture behavior. The industry's effort has focused on the behavior of stress corrosion-cracks, and the NRC has explored the broader questions regarding "leak-before-break" phenomena for all piping.

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 has been the Degraded Piping Program, conducted by Battelle's Columbus 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. Some of these are under investigation in other NRC-funded piping research programs, such as the International Piping Integrity Research Group (IPIRG) program. IPIRG is a consortium of government and industrial organizations formed to jointly fund research on the integrity of piping subjected to seismic and dynamic loadings, and other piping integrity issues. In 1989, pipe fracture experiments were completed on six-inch-diameter carbon and stainless steel pipe subjected to dynamic loadings. The work advanced to dynamic pipe fracture experiments on a typical piping loop configuration, using 16-inch-diameter pipe. The test pipe materials include carbon and stainless steel pipe, and the welds made on those materials are tested as well.

The NRC plans continued research into the causes of piping fracture, with new studies to begin in 1990.

Inspection Procedures and Technologies

This program includes studies of improved methods for the detection and sizing of flaws during inservice inspection of carbon steel (wrought) and cast stainless steel piping and pressure vessels. It also includes studies of online 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 light-water reactor primary circuit components is called the SAFT-UT (Synthetic Aperture Focusing Techniques for Ultrasonic Testing). The SAFT-UT technology is based on the 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 the flaw(s). (For background on the SAFT-UT technology, see the *1988 NRC Annual Report*, p. 147.) In 1989, extensive SAFT validation exercises were conducted at the Electric Power Research Institute's Nondestructive Evaluation Center in Charlotte, N.C., for the inspection of heavy-section pressure vessel steels and for the detection and sizing of intergranular stress corrosion cracks in stainless steel piping.

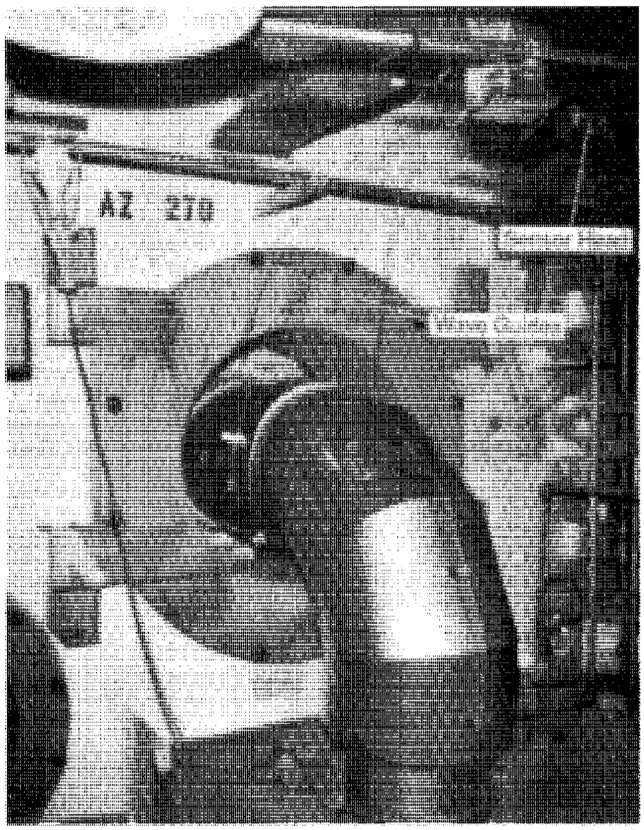
Inservice Inspection System Qualification. Research that included both national and international studies and field experience over the last several years have indicated that inservice inspection, as currently practiced, is not always reliable or effective. NRC research results have indicated a need for qualification of the entire inservice inspection (ISI) process, including the personnel, procedures, and equipment, as described in the *1987 NRC Annual Report*, pp. 115 and 116.

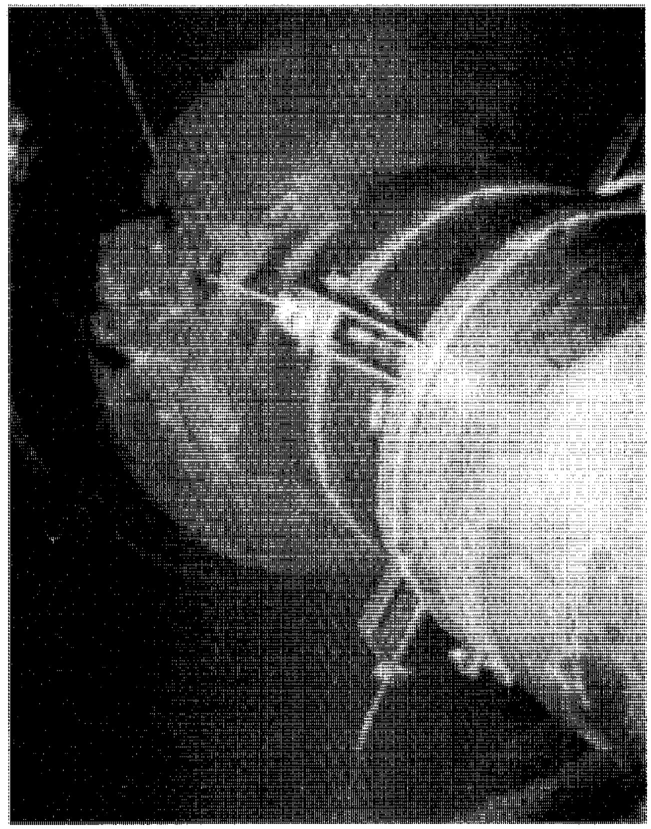
In 1987, two mandatory appendices to Section XI of the ASMECode were being assessed by the appropriate ASME committees, with NRC participation. In 1988, one of the appendices—on personnel training and qualification—was approved and incorporated into the code. The other appendix—on criteria for performance demonstrations—was approved in 1989. Also in 1989, the NRC research staff and contractor personnel began working with the ASME to incorporate in its code the qualification criteria for personnel and systems used for eddy current inservice inspection of steam generator tubing. 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. A technical basis is being elaborated for the new criteria for overcoming identified shortcomings.

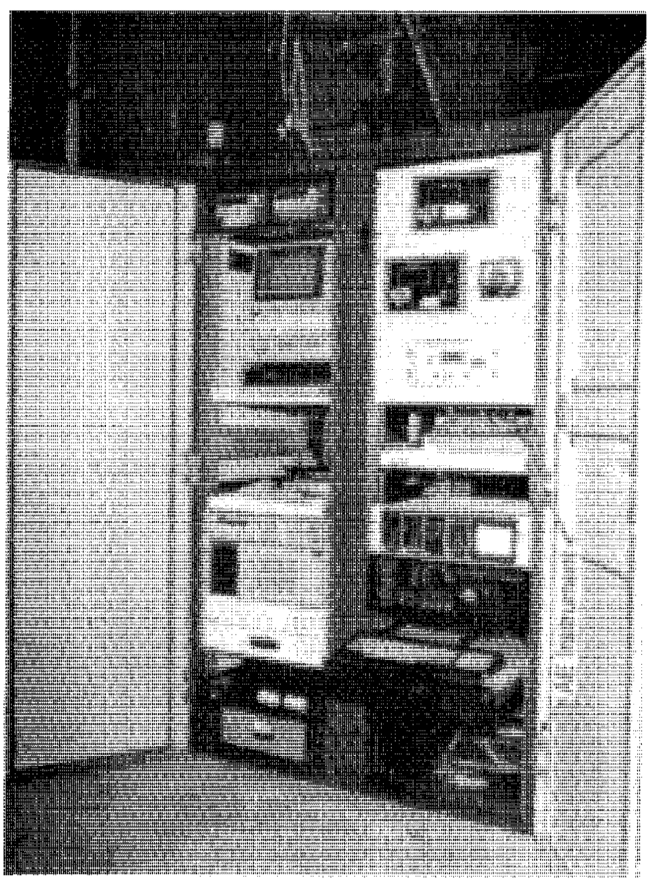
Continuous Monitoring for Crack Growth and Leak Detection. Research has been under way to develop the use of acoustic emission (AE) for the continuous online monitoring of reactors to detect and locate crack growth and to estimate the severity of the cracking from the AE signals. Up to 1986, a large body of laboratory and field data had been developed to establish feasibility and methodology for inservice monitoring of reactors and for evaluation of data. In 1985 and 1986, a great deal of data from an intermediate-scale vessel test was thoroughly evaluated to upgrade and validate existing models and technology, as described in the *1987 NRC Annual Report*, p. 116.

In 1987, activities focused on preparing a code case for the ASME Section XI Code for continuous monitoring of reactor pressure boundaries during operation. This code case was in the final stages of code committee approval at the close of the report period.

The 1989 inservice inspection of Philadelphia Electric (PECO) Company's Limerick Unit 1 (Pa.) reactor identified a flaw indication in a feedwater nozzle-to-safe-end weld. The flaw appears to be a shallow intergranular stress corrosion crack. Although fracture mechanics calculations indicated that the flaw would not compromise safety during the next operating cycle, the decision was made by PECO to apply AE monitoring and a crack-arrest-verification specimen technique to provide added assurance that the crack would not grow without detection. The AE monitoring equipment used for this purpose is the equipment used by PNL in an NRC research program. It was made available by NRC at no cost to PECO; in return, the NRC will have access to the AE data acquired during the monitoring. This application of AE constitutes a validation step for the technology developed under the NRC-sponsored AE research program. PNL performed the installation and will provide data analysis service under contract to PECO and with assistance from the PECO staff. In the course of this effort, PNL will work with PECO to transfer the technology and to provide the support for PNL to perform complete data analysis at their facility.







The use of "acoustic emission" (AE) in continuous monitoring to detect and locate crack growth and to estimate the severity of cracking is the subject of intensive NRC research. AE monitoring equipment was made available by the NRC to the Philadelphia Electric Co. (PECO) in 1989 to provide added assurance that an identified shallow intergranular stress corrosion crack would not grow without detection.

The monitoring equipment, installed at the Limerick Unit 1 (Pa.), facility is shown. At top left is an arrangement of AE waveguides and sensor head in the drywell; top right are sensor waveguides mounted on the pipe weld; and at left is the AE monitoring instrument cabinet.

Evaluation of a stand-alone, "Smart system" for AE leak monitoring was completed in 1988. The system is capable of accurate detection location and sizing of leaks in the pressure boundary. A detailed topical report was published giving details of equipment, calibration, and operation procedures and of data analysis and evaluation procedures. A draft revision of Regulatory Guide 1.45 on leak detection systems was developed in 1989 for staff review. The draft revision incorporated changes and improvements based on the findings from the research.

AGING OF REACTOR COMPONENTS

Aging Research

Aging is a key concern with currently operating plants and will clearly be crucial to any assessment of the safety implications of license renewal. Aging affects all reactor structures, systems, and components and has the potential to increase risks to public health and safety. There are significant uncertainties about aging-related degradation processes and about whether time-related degradation can be detected and managed before safety is impaired. Specifically, there is concern that multiple failures of age-related components could occur during transients or accidents and result in core damage of melting and a release of radiation. In the past, failures of safety 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 and civil engineering components.

The purpose of research into the aging of reactor components is primarily to establish the safety margins of operating plants as they progress through their design life, to define the aging mechanisms, to confirm existing and/or develop new detection and mitigation methods 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 required to understand the effects that aging has on the safety function of electrical and mechanical components of commercial nuclear plants. For the NPAR program, aging refers to the cumulative degradation of a system or component that occurs with time, which, if unmitigated, can lead to an impairment of continuing safe operation. The NPAR program provides systematic research effort to learn from operating experience and expert opinion, identify failures attributable to age degradation, predict safety problems resulting from age-related degradation, and develop recommendations for surveillance and maintenance procedures that will alleviate aging concerns. At the present time, NPAR consists of 15 separate, but related, projects concerned with the study of the effects of aging on 21 individual mechanical and electrical components and 15 systems composed of such components. An additional seven components and six systems have been targeted for study in the coming years. A phased approach to the research has been adopted to facilitate interim reviews and evaluations and to help arrange for availability of resources.

Based on the review of operating experience, including the available data base, expert opinions, and interactions with codes and standards committees, Phase 1 aging assessments were completed on the following special topics and safety-related components and systems:

(1) Solenoid-Operated Valves

(2) Risk Evaluation of Aging Phenomena

(3) RHR System in BWR Plants

(4) Auxiliary Feedwater Systems

(5) Cables, Connectors and Electrical Penetrations.

Reports were issued on the above-mentioned 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 Committee 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 in lieu of tests that cause excessive wear; (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 have been completed on the following components:

(1) Auxiliary Feedwater Pumps

(2) Battery Chargers and Inverters

(3) Batteries

(4) Emergency Diesel Generators.

A significant Phase II effort was completed in fiscal year 1989 with tests to determine whether naturally aged Class 1E batteries will function under earthquake loads. Specifically, the purpose of this part of the program was to determine if it is possible that naturally aged batteries, of adequate electrical capacity, might not be rugged enough to provide needed electrical power during and after large-scale earthquakes. The batteries used in these tests were obtained from an operating nuclear plant and had been in use for approximately 14 years.

The results of the tests indicate that the batteries did not suffer any degradation significant enough to reduce the required electrical capacity. Test results also showed that measurement of capacitance and internal resistance may give an indication of aging-related battery conditions.

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 capability to predict the residual operational lives of major light-water reactor (LWR) components and structures can be indispensable to resolving technical issues associated with plant aging and license renewal. The objective of the RLA, as an element ofthe NPAR program, is to develop technical bases and criteria for the NRC to assess methods of 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.

As of fiscal year 1989, the RLA of 11 components important to plant safety has been completed. The components are reactor coolant pumps, PWR pressurizers, PWR pressurizer surge and spray lines, PWR reactor coolant system charging and safety injection nozzles, PWR feedwater lines, PWR control rod drive mechanisms and reactor internals, BWR containments, BWR feedwater and main steam lines, BWR control rod drive mechanisms and reactor internals, electrical cables, and emergency diesel generators. In these assessments, the degradation sites, degradation mechanisms, stressors, and failure modes have been identified for each component. The assessments also include a review of the current methods for inspection and surveillance of these components. The results of this effort have been documented in NUREG/CR-4731, Volumes 1 and 2.

The two-volume NUREG report includes significant contributions from recognized industry experts on each of the components studied. The strategy of involving outside technical experts has resulted in a creditable up-to-date document. An overview of the Electric Power Research Institute programs related to the aging of major LWR components and structures is included. The NUREG report is structured to be a reference useful to the regulatory process and to the entire nuclear industry.

Priorities Among Safety-Related Components Based on Risk Significance. Time-dependent analyses and calculations that take into account the effects of aging are necessary means to identifying and setting priorities among risk-significant components, systems and structures. Building on those measures, program development is necessary to continually improve understanding of and to manage aging in those components, systems and structures.

A methodology has been developed to quantify core-melt frequency changes attributable to component aging and maintenance. The methodology is useful for evaluating maintenance and reliability programs and for assessing their effectiveness in controlling aging impacts on system unavailability, core-melt frequency, and risk. The application of the methodology has proved useful in performing regulatory analysis for license renewal, including justifying needs for aging management during a renewed license period. The methodology will be employed to identify where in plant safety systems aging is significant to risk. Based on risk significance of aging, priorities will be set for components for which aging management programs should be implemented.

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 mid-1991. 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. This guidance is expected to be completed by 1992.

The NPAR program anticipated these needs for a timely strategy and guidance by initiating, in 1988, a study aimed at developing regulatory guidance and review procedures for nuclear power plant license renewal. The overall goal of that effort was to provide the technical basis for detailed guidance and the requirements deriving from the rule to be developed in 1991-1992. This approach complements the rulemaking process and will allow the NRC to prepare for license renewal review in an orderly and timely way.

An assessment was undertaken to support one of the alternative positions on regulatory requirements for the license renewal rule. To that end, the aging effect on component failures was included in existing PRA models to quantify the changes in core-melt frequency ascribable to various aging failure rates of key components. In addition, the effects of incorporating maintenance programs and replacement schedules to control aging failure rates and the resulting impact on core-melt frequency were evaluated. Both a PWR and BWR plant were evaluated in this work.

The dominant component aging contributors identified for the PWR plant were the diesel generators, specific check valves and motor-operated valves in the emergency core cooling systems, and motor-driven and turbine-driven pumps in the auxiliary feedwater system. The dominant component aging contributors for the BWR plant were the diesel generators, motor-driven pumps in the service water system, and the turbine-driven pumps in the reactor core isolation cooling system.

It was shown that changes in core-melt frequency are highly dependent on maintenance and replacement intervals. Incorporation of short term maintenance and replacement intervals for either PWR or BWR plants can reduce the core-melt frequency for maintenance and replacement schedules by as much as three orders of magnitude.

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 five major LWR components: reactor pressure vessels, steam generators, primary piping, pressurizers, and emergency diesel generators. 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 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.

Maintenance to Manage Aging. Maintenance, in the broadest sense, is one of the keys for managing plant aging and will play a pivotal role in life extension/license renewal. The Surry (Va.) feedwater pipe break and the North Anna (Va.) steam generator tube rupture are examples of events that confirm the premise on which the NPAR program is based, stressing evaluation of component maintenance effectiveness to alleviate aging concerns. That premise is that component aging, if not adequately managed, will lead to component degradation and often to failure, which will result in (1) reduced component reliability, (2) increased system unavailability, and (3) a concomitant increase in overall plant risk.

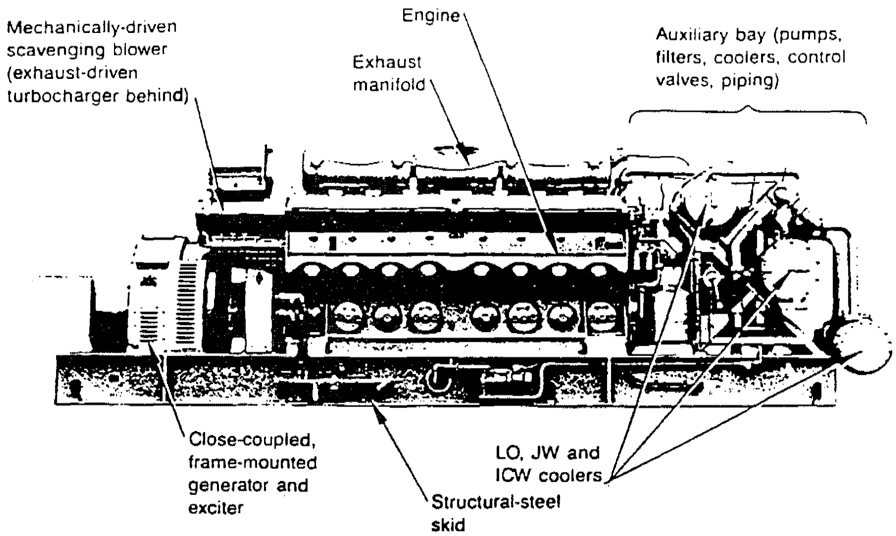
To identify the considerations that can contribute to effective management of component aging, the NPAR program has focused on resolving three major questions with respect to maintenance: (1) what components, systems, and structures to maintain, being susceptible to aging and thus risk-significant; (2) when to maintain them; and (3) how best to maintain them.

The NPAR program approach to dealing with maintenance issues comprised five major activities: (1) setting priorities for the selection of risk-significant components, systems, and structures; (2) understanding the aging degradation mechanisms in these selected components, systems, and structures; (3) identifying the degradation sites within the component boundary of interest; (4) evaluating the adequacy of current inspection strategies and detection and mitigation methods for these aging degradation mechanisms; and (5) developing recommendations for improved maintenance practices for these components. Recommendations were made, relevant to the maintenance rulemaking process, for achieving maintenance practices that will be effective in managing aging in components and structures.

The IEEE Working Group 3.3 prepared a draft guide on Maintenance Good Practices for Nuclear Power Plant Electrical Equipment. For three components included in the guide—motors, motor-operated valves, and solenoid-operated valves—plant aging research was used in addressing degradation processes and recommending guidelines for effective maintenance.

Components, Systems, and Facilities

Emergency Diesel Generators. Emergency diesel generators (EDGs) used in nuclear power plants are exposed to aging stresses from the environment and from operating and testing practices. Roughly half of the generator failures appear to be caused by some form of aging degradation. It has been concluded that the main purpose for the monthly testing of the EDGs



Shown is a typical skid-mounted engine generator package in which the skid length varies form 28-to-33 feet, the height is approximately 11 feet, and the weight varies from 65-to-90 tons for the engine alone.

should be reformulated from that of gaining statistical information to that of developing operational information on key engine performance parameters, in order to understand whether the trends in the engine operation are normal or possibly a sign of aging and wear problems. Since the testing requirements imposed on the EDGs constitute a major service condition and may cause the most severe aging degradation, it was recommended that (1) a more complete inspection and performance monitoring program be considered to help in mitigating certain aging-failure processes; (2) major engine overhauls not be based entirely on inspection considerations; and (3) preventive maintenance programs be improved to mitigate stresses that result in wear and vibration on such components as the engine governor.

Service Water System. The service water system is important to aging assessments because it is the final link in the transfer chain between the reactor core and the ultimate heat sink. Studies indicate that the accumulation of biological and inorganic matter, as well as corrosion, is the primary aging degradation mechanism in this system. However, the current level of surveillance and post-maintenance examination could be improved to more accurately track and detect the system aging degradation. It was recommended that an improvement in record-keeping of failures and aging degradation be implemented. In 1989, a method of analysis for root causes was developed and employed to help define the depth of knowledge required to accurately characterize the system's age-related degradation.

RHR Systems in BWR Plants. An analysis of past operating experience for residual heat removal (RHR) systems has shown that a significant number of failures are related to aging. The predominant cause of failure is normal service, which includes all the operating and environmental stresses to which the system is normally exposed. The predominant failure mechanisms are wear and calibration drift, which are associated with mechanical and electrical components, respectively. The components most frequently failing indude valves and instrumentation/controls.

An evaluation of the time-dependent effects of aging on component failure rates has shown that, while the mechanical components age moderately, the electrical components show little or no increase in cable component failure rate. This combination produces a moderate increase in the unavailability of the RHR systems. Comparison with previous results indicates that the potential for the unavailability of standby systems such as RHR is less severely affected by aging than is the unavailability of a continuously operating system. That finding can be attributed to the predominantly standby status of the RHR system, as well as to the stringent tests and inspections required for such a safety system.

Shippingport. The Shippingport (Pa.) nuclear power plant, out of service after 25 years of operation and undergoing decommissioning, is a valuable source of aged equipment for the NPAR program. As the first U.S. large-scale, central-station nuclear plant, the Shippingport reactor is similar to current commercial PWRs in design and operation. Its quarter-century of service exceeds the operating history of most currently active nuclear power plants. Also, because of substantial modifications during the mid-1960s and 1970s, Shippingport offers unique examples of identical or similar equipment which has been in side-by-side operation but which represent different vintages and exhibit different degrees of aging.

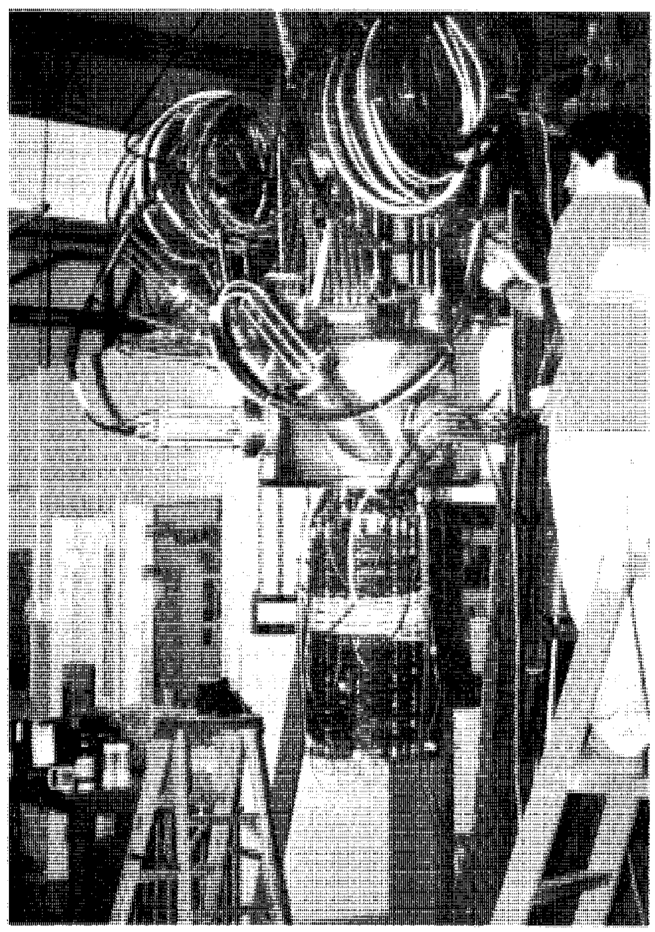
The decommissioning of Shippingport has been coordinated with activities of programmatic importance in the NPAR, e.g., data acquisition, including records and operating histories.

Electrical and Mechanical Components. In fiscal year 1988, the NRC entered into a six-year international agreement with the French Commissariat a l'Energie Atomique (CEA) for cooperative research on long term electrical cable aging degradation in nuclear power plants. Under the agreement, the CEA is to irradiate and thermally age both French and U.S. cables in their OSIRIS test reactor at Saclay, at a tenfold acceleration rate. The French are to irradiate the same cable materials in parallel at the Saclay POSEIDON cobalt-60 gamma irradiation facility. Periodically during the aging, cables and material specimens will be subjected to a LOCA (loss-of-coolant accident) qualification test in the Saclay CESAR steam chamber. The research results are expected to provide a realistic assessment of electrical cable degradation with age for in-containment safety-related service and to determine the effect of age on cable behavior during a design basis accident.

The French CEA have completed the first year of irradiation aging (in their OSIRIS test reactor) of two U.S. and two French commercial cables used extensively in both French and U.S. nuclear power plants for safety-related service. Physical property and electrical measurements were performed on segments of these cables removed after one year's accelerated exposure to determine the degree of degradation.

LOCA tests at the Sandia National Laboratories LICA test facilities have been completed for 12 of the most frequently used U.S. nuclear class 1E electric cable products, after 20, 40 and 60 years equivalent aging. A few motor-operated valves (MOVs) in cable samples aged an equivalent 60 years experienced a loss of insulation resistance during the LOCA test exposure, based on one-ampere fuse failure. The earliest such failure occurred at 80 hours in the LOCA test. However, most cables successfully passed the accident exposure following artificial aging to an extended lifetime of 60 years.

In response to the need for improved MOV condition monitoring methods, an assessment of five commercially available MOV monitoring systems and the use of motor current signature analysis was completed and reported in NUREG/CR-4234. The potential value of measuring valve stem position, valve stem veloc-



Final adjustments are made on the experimental cable test setup, used in the 60-year simulated service tests.

ity, valve stem strain, torque and limit switch actuation, torque switch angular position, motor current, etc., was tested in the laboratory and also in the field at a nuclear power plant. Most of the measurement systems available to the industry for detecting such parameters were shown to be effective in revealing changes in valve stem taper, stem nut wear, degraded valve stem lubrication, stem packing, torque switch setting, etc. Implementation of improved MOV condition monitoring practices and maintenance diagnostic systems at nuclear facilities in the future is expected to significantly reduce the incidence of MOV failures.

Snubbers are restraining devices used to restrict movement of pipes and equipment during such dynamic actions as might be induced by earthquakes, turbine trips, and safety valve venting. Snubber designs are either hydraulic or mechanical. A snubber will allow free thermal movement during normal operation but will restrict components in off-normal conditions. Environmentally induced degradation of snubbers is a principal concern of designers and users of these devices. In this program a comprehensive survey of snubber service operations, inspection, testing, surveillance, monitoring, and maintenance will be conducted with the cooperation of the utilities. Aging trends will be developed and used to improve methods of inspection, surveillance, and maintenance of snubbers.

Structural Components. Three planning and organizing reports were issued in fiscal year 1989: Structural Aging (SAG) Program Five-Year Plan; A Review and Assessment of Materials Property Databases with Particular Reference to Concrete Material Systems; and Plan for Use in Development of the Structural Materials Information Center. Extensive efforts are being made to obtain all currently available data on the aging characteristics of concrete structures from either national or international sources. Direct contact has been made with a large and diverse group of potential sources of such information, and the SAG program has been presented in a number of seminars, conferences, and similar forums. An electronically accessible data base is being created to contain the inventory of aging characteristics of structural materials (initially for concrete only). Collection of samples of aged and exposed concrete from some decommissioned nuclear facilities and the testing of these samples have begun. Work also began on a computer matrix identifying primary safety-related concrete structures in a nuclear power plant, their functions, and a rating of the safety significance of each structure. An identification of potential environmental stressors and aging factors that can affect the performance of these structures, as well as useful nondestructive concrete examination techniques, are also being sought. Plans have been drawn up for the development of condition assessment models and reliability-based prediction tools.

Decommissioning

Development of information on the safety, costs, and wastes related to the decommissioning of LWRs and other nuclear facilities has continued. Data are being for reports of the decommissioning of the La Crosse (Wis.) and the Rancho Seco (Cal.) nuclear power plants, and the Shippingport reactor, mentioned above. These data will cover costs, radiation doses, and low-level waste resulting from decommissioning activities. Identification of radionuclide source terms for decommissioning and low-level waste burial is continuing, using materials from the Shippingport reactor, a decommissioned West German reactor, and several U.S. operating reactors.

Three Regulatory Guides for LWRs were issued for public comment or as final guides related to the assurance of funds for decommissioning financing, the format and content of decommissioning plans, and record-keeping for decommissioning. Similar Regulatory Guides for other nuclear facilities are in preparation. Regulatory Guides for LWRs on methods for facilitating decommissioning and on the standard format and content of financial assurance mechanisms required for decommissioning under 10 CFR Parts 30, 40, 70 and 72 are also in preparation.

Chemical Decontamination

The NRC continued to develop an information base for reducing occupational doses in nuclear power plants and for assessing the effects of decontaminations on nuclear plant solidification systems. Measurements were made of recontamination rates following chemical decontaminations at operating nuclear power plants. A report analyzing these results was in preparation at the close of the report period.

Shippingport Shield Tank

A program has continued during the report period to determine the effect of low-temperature, low-flux irradiation on the mechanicalproperties of the neutron shield tank of the Shippingport reactor (see above). With the identification of embrittlement in the pressure vessel, an urgent need was recognized to assess possible embrittlement in present-day reactor supports. (The construction material in the Shippingport neutron shield tank (vessel support structure) is equivalent to that used in present-day support structures.)

Test results (Charpy impact testing) from both the inner wall and outer wall of the neutron shield tank base material show that the low-temperature, low-flux irradiation has significantly lowered the toughness of the shield tank (support) material. The irradiation reduced the upper-shelf energy and increased the transition temperature of the material. Annealing studies on material from the neutron shield tank showed a complete recovery of irradiation embrittlement after an anneal at 400°C for two hours.

Because of the value of these Shippingport data in evaluating the entire support structure integrity issue, considerable effort is being expended to verify the exact level of property changes for comparison to similar changes from other reactors and environments. Other experimental irradiations and analyses are planned to gain more information upon which to judge the Shippingport data.

REACTOR EQUIPMENT QUALIFICATION

Valve Operability

Experiments were conducted in 1988 and 1989 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.

Most valve actuators are sized by analytical methods and provide the force required to close the valves under conditions postulated for design. However, since the water in a high-energy pipe is subject to both high pressure and high temperature, some of the water will rapidly flash to steam at the break. If the rapid changes of water into steam extend to the inside of the valve, the effect of the flashing on valve hardware is very difficult to assess by analytical methods. It is also difficult to assess by analytical methods the effects of the large friction forces that develop between the internal parts of valves because of the high-velocity flows. Therefore, experiments were conducted to directly measure the magnitudes of the various forces that must be overcome during valve closure under pipe-break conditions.

Evaluation of the results of the first series of 14 experiments conducted on two gate valves manufactured by different companies showed that the valves were capable of stopping the flows in all the experiments. Some of the data indicated, however, that complete closure did not occur during one blowdown. (It should be noted that the valve actuators were set to deliver larger thrusts to ensure closure, so that the various forces could be quantified.) Other important findings were:

(1) The main friction force that must be overcome by the valve actuator is under-predicted when the typical friction factor is used.

(2) Neither valve would have completely stopped the flows for all the experiments if the recommended valve actuator settings had been used.

(3) One valve experienced damage during closure; therefore, the analytical methods are not applicable to this valve.

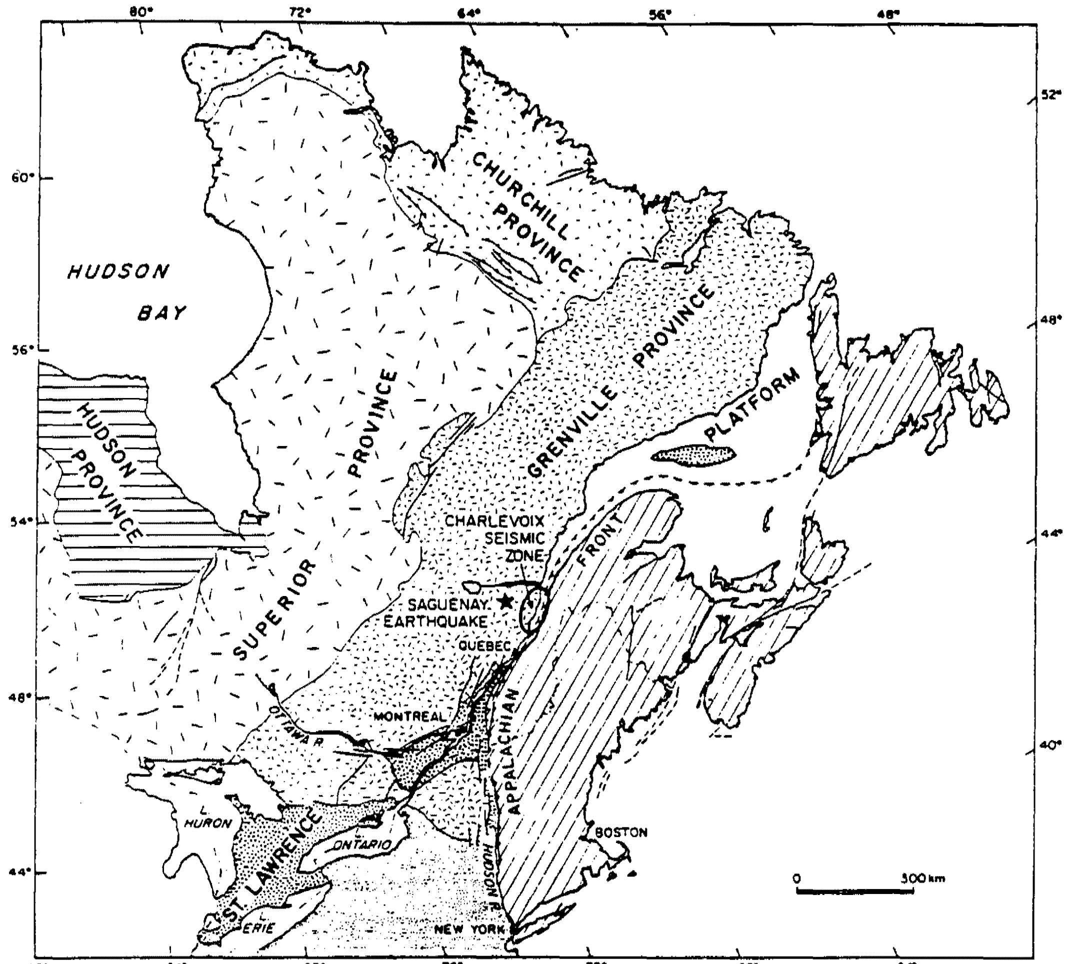
The results of the experiments were presented to representatives from valve and valve actuator manufacturers, from utilities, and from research laboratories. These expert representatives found the results generally acceptable and agreed that the data contribute to an understanding of valve behavior. However, the experts recommended that more experiments be conducted to obtain more information before establishing regulatory requirements. The NRC accepted this recommendation and is currently funding additional experimentation on valves. These latest experiments will be evaluated and the results presented to the panel of representatives cited above. It is intended that the results and data from all valve tests be made available to the utilities for their use in assessing valve closure capabilities during the accident conditions described above.

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

The publication, in 1989, of seismic hazard curves by both the NRC (NUREG/CR-5250) and EPRI (NP-6395) marks the end of major efforts to characterize the seismic hazard at U.S. nuclear reactor sites. While the best information and procedures have been employed, it is understood that large uncertain-



The November 25, 1988 Saguenay earthquake occurred west of the Appalachian Front within the Precambrian Grenville Province. The Saguenay region has undergone few historic earthquakes and has been considered relatively aseismic, in contrast to the active Charlevoix seismic zone located about 80 kilometers southeast of the November 25 epicenter.

ties still remain in seismic hazard estimates. Also, recent full-scope probabilistic risk assessments, performed as part of the NUREG-1150 effort, continue to show that seismic hazard uncertainties contribute significantly to overall uncertainty in nuclear reactor risk estimates. These large uncertainties make it difficult to judge the contribution of seismic risk in the development of individual plant examination guidelines.

Recent successes in geological, geophysical and seismological studies, sponsored by RES and discussed below, show that it is possible to answer basic scientific questions underlying seismic hazard uncertainties. It is the goal of the RES earth science program to significantly reduce the uncertainty in seismic hazard estimation in the next decade by pursuing this type of research.

Seismographic Networks. During fiscal year 1989, the NRC continued its support of regional seismographic networks. The networks provided essential information on the seismicity of the central and eastern United States. This includes not only seismicity recorded during the year and attendant parameters, but also earthquake epicenter trends that emerge only after long periods of earthquake monitoring. Such trends are important for deciphering tectonic structures that affect the seismicity.

A new National Seismographic Network is being established and should be fully in place by the end of fiscal year 1992. The central and eastern region are being developed under joint NRC and U.S. Geological Survey (USGS) funding. Under an interagency agreement between the NRC and the USGS, the USGS will have responsibility for monitoring earthquakes in the

United States east of the Rocky Mountains and for operating network stations as they are installed. The NRC provides funding for the permanent equipment and operating software. The national network will provide a more uniform station spacing and, hence, earthquake detection capability, although at the expense of more widely spaced stations.

In conjunction with the establishment of the new network, the NRC has acquired a Sun computer work-station to provide an in-house capability for analyzing important seismic events. Software for this purpose is being and tested on the work-station.

Northeastern Neotectonics. The Saguenay, Quebec, earthquake of November 25, 1988, with a magnitude of 5.9, was the largest earthquake to occur in eastern North America since the Temiskaming earthquake of 1935. The earthquake was strong enough to induce liquefaction in the meizoseismal area. The epicenter was located near the Charlevoix seismic zone, 80 kilometers away, in an area with little historic seismicity. This fact raises a question about estimation of seismic hazards based on historic and present, instrumentally determined, seismicity. Intra-plate regions are characterized by long recurrence intervals, and in some regions the historic record may be too short to adequately represent the seismic hazard. Thus, the Saguenay earthquake suggests that seismic hazards in intra-plate regions with little historic seismicity may be higher than expected. Another conclusion derived from the Saguenay event and other east coast events is that near-field intensities in this region appear to be low, compared to far-field intensities. The reason may be that a thick sedimentary cover can absorb high frequencies, thereby reducing shaking effects near the epicenter, while the low crustal attenuation at depth produces relatively large seismic motion at a distance.

Detailed investigations of the seismic role of pre-existing structures were conducted in the epicentral areas of the 1983 Goodnow earthquake in the Adirondack Mountains, and the 1985 Ardsley earthquake in Westchester County, N.Y. These areas are characterized by brittle, apparently seismogenic structures that form a high angle with the structure of the surrounding rocks. Results suggest that faults with little accumulated displacement can be the source of significant earthquakes, but that the segmented nature of many fault zones in this region may limit the size of earthquakes generated by such faults.

Paleoseismic studies in the epicentral areas of moderate historic earthquakes in the northeastern United States and adjacent Canada have found paleoliquefaction features induced by a prehistoric, Holocene earthquake and have provided data on seismically induced soil deformation structures that can be used throughout the northeastern United States to distinguish between seismically induced features and similar features induced by other phenomena.

Studies of the seismicity at Lancaster, Pa., have shown that the relatively high seismicity of this area is concentrated in an elongated, north-south trending zone of about 50-kilometer length. The zone coincides with the Fruitville fault, where it is mapped, and extends beyond it. Overall, a variety of evidence clearly shows an association of the seismicity with this fault. Among the lines of evidence available are remote sensing lineaments and travertine deposits that, as in other locations, are found to be forming downstream from the trace of the fault, indicating recent movement on the fault. Thus, the study has also served to confirm the value of travertine deposits as indicators of recent movement.

Charleston Studies. New evidence from soil liquefaction studies near Charleston, S.C., gathered during the report period, supports a previous conclusion that the seismicity in the Charleston area has unique characteristics and that the large earthquakes that have occurred here are not necessarily to be expected in other parts of the east coast. Investigations during the past five years have documented the occurrence of about five large prehistoric earthquakes of magnitudes similar to that of the 1886 Charleston event. Dating of these events that happened over a span of about 3,000 years suggests that the recurrence interval for large earthquakes in this area is on the order of 500-to-1,500 years.

The search for seismically induced liquefaction features was extended southward to Jacksonville, Fla. The farthest south that such features were found was near the South Carolina-Georgia State line. Two features were mapped in that location, and both were caused by the 1886 earthquake. Investigations in the north have still not identified any paleoliquefaction features farther north than Southport, N.C. It was found that, as the distance from Charleston increased, the liquefaction features became less frequent and smaller, indicating, because of the large distance, either a prehistoric event larger than the 1886 earthquake or smaller events with hypocenters located closer to the liquefaction features.

Relative Travel Time Residuals (RTTRs) of seismic waves were used in the study of the velocity structure of the earth's crust in South Carolina. The data show a systematic increase in RTTR values from the Piedmont toward the coast. The reasons for the increase are not completely understood, but it may be related to an increase in crustal thickness or a decrease in velocity toward the coast. The RTTRs also provide corroborative evidence of an Ashley River fault, previously postulated on the basis of tectonic and stratigraphic investigations. A significant increase in RTTRs occurs from west-to-east across this subsurface fault; azimuthal bias exists at stations west of the fault but not east of it. Both results indicate differences in crustalvelocity properties across the fault.

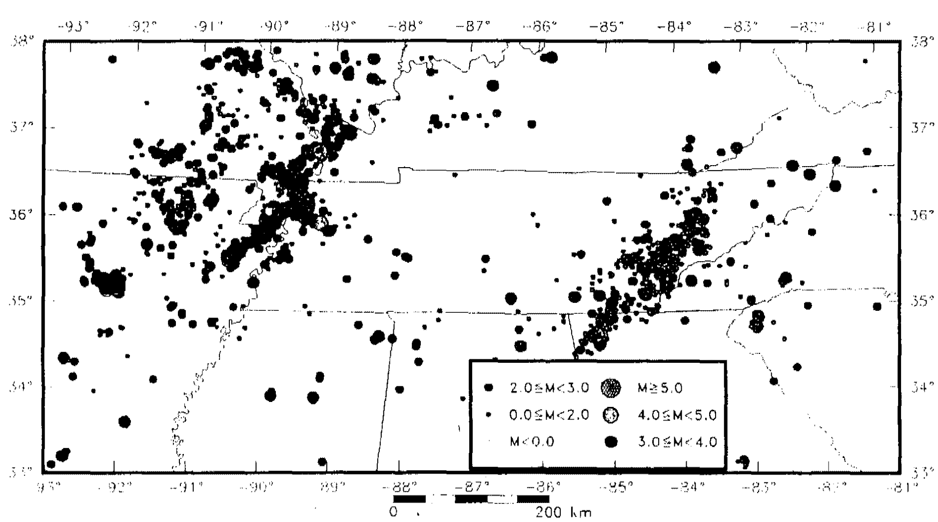
New Madrid and Eastern Tennessee. The New Madrid (Mo.) area experienced an earthquake sequence in 1811-1812 that included the most severe shocks ever generated east of the Rocky Mountains. Today the area is still the source of considerable microearthquake activity. The source of the seismicity has been identified as reactivated faults within a rift in the crystalline basement. The presence and extent of this ancient rift structure has been defined by geological and particularly by geophysical means. Plots of epicenters in this area also clearly reveal the trends of the underlying rift structure.

The southeast Tennessee/northwest Georgia area also shows considerable seismic activity. The seismicity both here and at New Madrid has been monitored for over a decade, and a comparison between the two areas reveals certain interesting parallels and also some differences. Both zones of seismicity cover about the same area, and both are oriented in a northwesterly direction. The New Madrid zone has more tightly clustered epicenters, and it also has more regional seismicity surrounding it. The tectonic settings are quite different for the two areas: a former intracontinental rift is the causative structure at New Madrid; eastern Tennessee's seismicity is related to a former continental suture zone that could also be called a continental margin/accreted terrain complex. Seismically, the New Madrid zone is somewhat more active than the eastern Tennessee area in the magnitude range from 1.5 to 4.2, generating more and larger earthquakes that are also shallower and less diffuse. The stress field in both zones is similar so that the difference must be related to the different tectonic regimes.

Meers Fault. Paleoseismological studies are being conducted along the Meers fault, which is part of the northwest-striking Wichita frontal fault system of southern Oklahoma. Although the Meers fault is currently aseismic, a segment of it, 26 kilometers in length, has been shown to have undergone substantial displacement during at least one event in the late Quaternary or Holocene. Detailed trenching and logging at selected sites indicate that the prominent scarps along this segment of the fault system were produced by two Holocene surface faulting events. Charcoal samples for age dating were obtained from faulted and unfaulted alluvium, and these suggested that the two events occurred about 1,200 and 2,000 years ago. An earlier event was identified during fiscal year 1989 but has not been dated because of the lack of geologic evidence. Evidence indicates that it could have occurred anywhere from several thousand years to several tens of thousands of years ago.

Research has been completed into dating the displacement along the Washita Valley fault zone of central Oklahoma. Results suggest that this fault zone has not experienced late Quaternary displacement. Regional reconnaissance is under way to identify other late Quaternary faults (e.g., the Criner fault in the Wichita frontal fault system) and to delineate the northwestern and southeastern extent of the Meers fault.

Nemaha Uplift and Oklahoma. The Nemaha uplift trends from southeastern Nebraska through Kansas into central Oklahoma and is an area of the mid-



The map, centered on Tennessee and including portions of 12 contiguous States, shows the frequency of the very considerable seismic activity in the New Madrid and eastern Tennessee seismic zones.

continent that shows elevated seismicity. The uplift is associated with subsurface faulting, particularly the Humboldt fault zone on its eastern flank. With 10 years of earthquake monitoring, the Kansas and Nebraska areas have revealed fairly definite fault-related epicenter patterns. In Oklahoma, however, the patterns of seismicity are much less clear. The State has higher seismicity than Kansas or Nebraska, but definite epicenter patterns pointing to causative structures have not so far emerged, even though there are epicenter clusters in south-central Oklahoma. This is undoubtedly because of the more complicated tectonic setting in Oklahoma, where the Nemaha uplift meets other more substantial tectonic units, such as the Anadarko and Ardmore basins and the Wichita uplift. Although seismic monitoring data in this area also span more than a decade, more time is needed to better resolve this question.

Pacific Northwest. The Pacific Northwest is underlain by 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 the geological and geophysical evidence indicates active subduction, but 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 is conducting a major study of the geology and tectonics of this region. The NRC is partially funding two neotectonic research projects of this program, one in southwestern Washington and the other in central Oregon. These projects are continuations of investigations that revealed geologic evidence suggesting the occurrence of several prehistoric and Holocene large/great 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.

Similar cycles of marsh deposits and shallow marine sediments have been identified along the relatively stable coast of northern Europe. These deposits are attributed to non-tectonic causes. The features mapped in Europe are being compared with those in the Pacific Northwest to confirm whether or not Washington and Oregon have experienced large subduction zone events in the past.

Subsidence in Maine. In addition to a 45-station crustal motion network spanning the eastern two-thirds of the nation, the NRC sponsors a second, smaller Global Positioning System (GPS) network of 21 stations in southeastern Maine. This network was measured for the first time in the fall of 1986 and was re-measured in January 1989. The purpose of the network is to establish absolute values for the subsidence that is taking place on the eastern coast of Maine. Subsidence rates as high as nine millimeters-per-year have been postulated for this area. Some investigators question these rates, and half of that amount may be a more reasonable assumption. The GPS measurements should eventually establish valid subsidence rates, because they will not be influenced by sea level changes. Considering that the vertical component of GPS is less accurate than horizontal positions, a second set of measurements will probably not reveal any motion but serve instead to provide a better baseline. For the same reason, these stations will be re-measured at five-year intervals in the future.

Strong-Motion Studies. The NRC supports several programs to analyze and disseminate information obtained from strong-motion accelerograms of portions of the eastern United States and California. As part of an interagency agreement with the USGS, the NRC is funding detailed analysis of strong-motion records from earthquakes in southern Illinois and in Palm Springs and Whittier, Cal. It is also funding detailed analyses of teleseismic records of large intra-plate earthquakes to estimate strong ground motions. Site effects of strong ground motions are identified by analyzing the effects on a site of several events at different locations, and propagation characteristics are investigated by studying the effects of a single event at several sites. In cooperation with Canada, the NRC has providing support for the installation of strong-motion instruments in eastern Canada.

Many nuclear power plants in the United States are built on relatively thin soil columns over competent bedrock. One of the most significant seismic uncertainties affecting these facilities is the behavior of shallow soils over rock when subjected to strong ground motions, and the way in which these soils propagate the seismic waves. To address these questions, instruments have been placed in boreholes at various levels at McGee Creek, Mammoth Lakes, Cal., an area of relatively high seismicity with a soil column of thin glacial till over hard bedrock. Analysis of micro-earthquakes and two earthquakes of magnitude 5.7 and 4.9, recorded by instruments at depths of 167, 35, and O meters in the till and bedrock, indicates that the glacial moraine generally increases the amplitude and duration of waves recorded at depth. However, corner frequencies of S-waves measured at depth and at the surface are very nearly the same. It can be inferred from this that the local site effect attributable to glacial till is not the cause of the nearly constant source radius observed for earthquakes in the region.

The McGee Creek experiment is producing significant data on the propagation of strong ground motion in a shallow soil column. Results have helped identify the strengths and weaknesses in the experimental plan, so that a significantly improved experiment can be carried out in a more typical sedimentary soil column, such as that at Anza, Cal., where five instruments were installed in boreholes near a splay of the San Jacinto fault zone. Seismometer depths vary from 0-to-300 meters, ranging from the surface through the soil column and into bedrock. This site was selected because it has a soil profile like that of many United States nuclear plant sites (20-to-30 meters of soil over competent basement rock), is in a highly active seismic area with epicentral distances ranging from two-to-three kilometers and magnitudes from 1.0 to 6.4, and includes excellent coverage by other seismometers that allow accurate determination of the source parameters of earthquakes.

Seismic Engineering Research

In addition to earth science research, the NRC seismic research program includes several engineering-oriented projects to determine the possible effects of earthquakes on nuclear plants and their safety systems.

Seismic Strength and Response of Nuclear Plant Concrete Shear Wall Buildings. Since 1980, in its Seismic Category I Structures Program, the NRC has conducted analytical and experimental research on low-rise reinforced concrete shear walls, buildings, and building segments. The objective has been to resolve certain structural concerns related to the adequacy of criteria currently used in design and analysis and to determine whether existing nuclear power plants can withstand earthquakes greater than those considered in the original design.

From early static and dynamic tests conducted under this program, it was concluded that shear wall buildings have significant design margin against gross failure because of earthquakes. However, it was found that the measured natural frequencies of dynamic response at design levels were significantly lower (up to 50 percent) than those computed analytically, and further reduction in the measured natural frequencies was observed as the shake table accelerations were increased. The major implication of this finding is that a reduction in building natural frequencies would tend to change the shape of the calculated in-structure floor response spectra used to specify dynamic loads for equipment mounted on the floor slabs. Depending on the frequency of the component, it is possible that the dynamic loading criteria specified may be either significantly increased or significantly decreased. Thus, the seismic capacities of some equipment and piping may be lower than expected. New tests completed in fiscal year 1989 indicate that the amount of stiffness reduction may not have been as severe as previously indicated.

Test data from the Seismic Category I Structures Program have prompted the American Society of Civil Engineers Dynamic Analysis Committee to establish a Working Group on Stiffness of Concrete Shear Wall Structures. The working group is developing new guidelines for the seismic design of nuclear power plants.

The impact of the "frequency reduction" issue on existing seismic probabilistic risk assessments (PRAs) is being assessed. Three PRAs are being re-evaluated to determine if the frequency reduction issue significantly changes estimates of seismically induced core damage frequency or priorities given various earthquake damage scenarios. The initial re-evaluation of the Peach Bottom plant was completed in fiscal year 1989.

Seismic Component Fragilities. Results of data collection and evaluation activities by Brookhaven National Laboratory (BNL) on both low and medium voltage switchgears have led to revisions in the "generic equipment ruggedness spectra," or GERS, used by industry for these equipment categories. GERS are important to the implementation of Unresolved Safety Issue A-46 (USI A-46: Seismic Equipment Qualification in Operating Plants) and will be used in the external events portion of the individual plant examination effort. Relay testing completed by BNL led to a General Electric Part 21 report identifying deficiencies in the seismic fragilities of Class IE IAV relays. This same test program revealed how relays from various manufacturers are sensitive to the direction of input motion, the frequency distribution of the input motion, the electrical state of the relay (energized or de-energized with contacts either normally open or normally closed), and the number of axes in which independent motions are applied (uniaxial, biaxial, or triaxial). Testing will continue in order to determine the impact of coil voltage and device current on relay fragility and the influence of multiple, closely spaced, short-duration (less than two milliseconds) "chatter intervals" on the performance of a remotely located device.

Seismic Design Margins Reviews. As developed and used by the NRC and EPRI in recent years, seismic margins reviews have been found to be effective and efficient in assessing the capability of nuclear power plants to safely withstand earthquakes larger than their design basis level. The results of seismic margins evaluations can be used to answer questions regarding the effects of higher seismic hazard at a site or to identify what systems and plant functions are most relied upon to lessen the probability of core damage resulting from earthquake events. Current decision-making regarding the implementation of the NRC's Severe Accident Policy Statement will take the use of seismic margins reviews into consideration.

EPRI, Georgia Power, and the NRC are completing the cooperative seismic margins review of Hatch Unit I (Ga.). While two other plants have undergone such reviews, Hatch is the first BWR and the first plant on a soil site to have a margins review. Liquefaction potential at high seismic ground motions and relay chatter evaluation are also new and important parts of this review. Georgia Power is performing its USI A-46 review of components and tanks in conjunction with the margins review.

An extensive group inspection of Hatch was performed in November of 1988, with participation by NRC staff members and the NRC-sponsored Hatch Peer Review Group. The EPRI-sponsored analysis was to be complete in the fall of 1989, as was a separate, NRC-sponsored fault-tree system analysis. The completion of the Hatch review in early 1990 will conclude the Seismic Design Margins Program.

Cooperative International Seismic Programs. The NRC's participation in international seismic test programs is beneficial both for the sharing of research resources and in bringing 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 NRC has cooperated in joint experiments in Taiwan, the Federal Republic of Germany, and Japan. Tests at the Heissdampfreaktor (HDR) facility in Kahl, Federal Republic of Germany, generated data on the response of an actual piping loop to large dynamic loads far exceeding those expected from a safe shutdown earthquake. An experiment at the Tadotsu shake table in Japan also demonstrated the ability of a model PWR piping loop to withstand multiple repetitions of loads far in excess of design and furnished data that will help in studies of the growth of cracks under extreme seismic loads. Analyses of these experimental data were completed in 1989.

Current activities include the planning of a soil-structure interaction experiment to be constructed in a seismically active location near Hualien, Taiwan. This experiment is being organized by EPRI, and interested parties in France and Japan will also participate.

Another undertaking attempts to benefit from the massive Japanese investment during the 1980s to ensure the reliability and safety of nuclear power plants located in densely populated, highly seismic regions. Hundreds of millions of dollars were invested in large-scale testing programs that led to the revision of Japanese seismic design standards for nuclear power plants. A research program was begun in 1989 to scrutinize just how Japanese practice may differ from that in the United States and to assess the safety significance of any differences.

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 1989, research program plans for both high-level waste (HLW) and low-level waste (LLW) were developed to help ensure the responsiveness of the program to the needs of the licensing staff.

High-Level Waste

The NRC maintains active research programs in 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 identify the physical processes that control and determine repository performance in the unsaturated volcanic tuff at the Yucca Mountain site (Nev.) currently under consideration by DOE as directed by the Congress in December 1987. In 1988 and 1989, NRC research focused on tuff, but the ultimate goal of the NRC's waste 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 with NRC requirements and with the Environmental Protection Agency's HLW standard, while DOE goes about the task of providing a permanent HLW repository. Key technical issues being addressed are unsaturated flow and transport mechanisms, fault delineation, and assessment of seismic potential and geochemical effects.

Geohydrology. Since transport by ground water is the most likely path by which most 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 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 characterize the hydrology of fractured media and how measurement data should be analyzed to model ground-water flow. This work was begun in 1987 and currently entails assessment of techniques and methods for fracture characterization, infiltration and percolation studies, and rock and matrix permeability testing. The project is using vapor-phase flow and transport assessment and numerical simulations of flow and transport in partially saturated media to assess the importance of large, natural, anomalous hydrologic features, appropriateness of continuum-versus-discrete fracture models, measurements of effective porosity, theories of spatially projecting dispersion measurements, and distinctions between and among matrix diffusion, dispersion and sorption.

The validity of the models used to describe ground-water flow and radionuclide transport is being appraised in an international project called INTRAVAL. The research at the University of Arizona has been formally accepted as part of the INTRAVAL project. The NRC staff and research contractors from the University of Arizona, Sandia National Laboratories, Massachusetts Institute of Technology, and Battelle Pacific NorthwestLaboratories 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, will augment the field testing program cited above.

Stability of Underground Openings. When specifying suitable site conditions for a repository, 10 CFR Part 60 specifically requires consideration of natural phenomena and site conditions that could adversely affect achievement of the prescribed performance objectives. An important phenomenon that could affect both the short and long term performance of a repository is ground motion resulting from seismic activity. 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, which could violate established repository performance objectives.

To investigate the effects of seismicity on the underground openings for an HLW repository, the NRC is sponsoring research at the Center for Nuclear Waste Regulatory Analyses in San Antonio, Tex. Initial results from the study indicate that structural damage at depth is incurred through accumulation of joint shear deformation resulting from repetitive loading.

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 migration of radionuclides to the accessible environment.

To evaluate the performance of seals in the unsaturated HLW tuff environment, the NRC is supporting research studies at the University of Arizona. Both laboratory and field performance of seals is being investigated.

Waste Package Performance. Understanding the performance that can be expected from the waste form and waste package is essential to the NRC's ability to independently evaluate DOE's eventual demonstration that both form and package comply with the containment and controlled radionuclide release requirements of 10 CFR Part 60. During 1989, NRC sponsored research on the testing of six candidate HLW container materials proposed for use in tuff environments by DOE. Research studies to establish the accuracy and precision of available measurement techniques for determining the corrosion environment (pH and Eh) at the surface of waste packages—under repository conditions and on time scales to support long term predictions of waste package performance—are under way at the National Institute of Standards and Technology.

The Japan Atomic Energy Research Institute (JAERI), under a cooperative research agreement with the NRC, continued a series of experiments on the stability of HLW when it is in the form of glass and on the durability of HLW containers in high-radiation environments. This work complements the laboratory research studies supported by the NRC of radioactive waste containers and of the various forms of radioactive waste.

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 as a result of mineral dissolution or precipitation following waste emplacement. The NRC has an active research program in geochemistry as it affects the management of HLW. Work is being completed at the University of California at Berkeley on the geochemistry of radioactive wastes in the repository environment. In 1989, the thermodynamic data base used to predict chemical reactions in tuff and ground water in the thermally affected area of an HLW repository was examined, and research into the modeling of the evolution of water as it moves toward the waste packages was started at the Center for Nuclear Waste Regulatory Analyses. The NRC is participating in an international field study at the Koongarra ore body in northern Australia, observing the actual movement of radionuclides. This study is providing a basis for validating performance assessment models to be used in HLW repository licensing. The second year of the study has seen the completion of the hydrologic and geochemistry modeling scenarios. A good deal of the field data on transport properties of the site have already been collected. At Oak Ridge National Laboratory, laboratory studies suggested that there may be a practical approach for simplifying coupled hydro-geochemical models of radionuclide transport at Yucca Mountain, Nev. A study was begun at Johns Hopkins University to develop coupled hydro-geochemical transport models and to test them against data from natural systems such as the Koongarra ore body.

Rulemaking. In May 1989, the NRC published the final amendments to a rule (10 CFR Part 61) that requires geologic repository disposal of above Class C waste unless an alternative has been approved by the NRC. In April 1989, the initiation of a rulemaking to clarify the terms "anticipated processes and events" and "unanticipated processes and events" was approved.

Low-Level Waste

NRC research in support of licensing activities 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, including environmental water entry into disposal units. This research is useful not only to the NRC licensing staff but also to the States regulating LW disposal (see Chapter 7). In order to make their research results available to the States, NRC research contractors, besides publishing their work, gave presentations at meetings well attended by State representatives—such as "Waste Management '89," the DOE Oak Ridge Model Conference, and the Annual DOE LW 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 nuclear waste. In 1988, the Idaho National Engineering Laboratory completed research on the reliability of "engineered components" for alternatives to shallow land burial of LLW. The research indicated that the cover component was most important for the reliability of the engineered alternatives designs. Concrete is expected to play an important role in engineered alternatives to shallow land burial. In 1989, the National Institute of Technology and Standards continued investigating, for the NRC, the durability of concrete in engineered alternatives to shallow land burial, and the Idaho National Engineering Laboratory is continues to develop a mathematical model describing concrete performance.

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, and the compressive strength of the solidified waste, are consistent with NRC technical positions and requirements of 10 CFR Part 61 for waste form stability. Under examination is the stability of decontamination waste obtained from operating nuclear reactors using commercial decontamination processes—such as LOMI, CANDECON, DOW NS-1, and CITROX—and solidified in cement. Field studies using lysimeters are being conducted at the Oak Ridge and Argonne National Laboratories to determine whether radionuclides on ion-exchange resins are released from solidified waste forms under environmental conditions involving natural precipitation. The effects of radiation on the stability of ion-exchange resins containing radioactive material are under study by the Idaho National Engineering Laboratory.

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 designed to inhibit water percolation into waste disposal units. Covers under investigation include types being considered for future LLW disposal sites, such as (1) a compacted clay cover, (2) a compacted clay layer beneath an erosion protection layer (rip-rap), and (3) a compacted clay layer above a conductive layer barrier (flow layer above a capillary break). The project is discussed in the 1988 NRC Annual Report, p. 161.

Table 1. Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved

Number	*Title*	*Report Number*	*Date*
A-1	Water Hammer	NUREG-0927, Rev. 1 NUREG-0933	March 1984
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	NUREG-0609	November 1980
A-3	Westinghouse Steam Generator Tube Integrity	NUREG-0844	September 1988
A-4	CE Steam Generator Tube Integrity	NUREG-0844	September 1988
A-5	B&W Steam Generator Tube Integrity	NUREG-0844	September 1988
A-6	Mark I Short-Term Program	NUREG-0408	December 1977
A-7	Mark I Long-Term Program NUREG-0661 Suppl.	NUREG-0661	July 1980
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808	August 1981
A-9	Anticipated Transients Without Scram	NUREG-0460, Vol. 4	September 1980
A-10	BWR Feedwater Nozzle Cracking	NUREG-0619	November 1980
A-11	Reactor Vessel Material Toughness	NUREG-0744, Rev. 1	October 1982
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	NUREG-0577, Rev. 1	September 1982
A-17	Systems Interactions	NUREG-1229 Generic Letter 89-18	August 1989
A-24	Qualification of Class IE Safety-Related Equipment	NUREG-0588, Rev. 1	July 1981
A-26	Reactor Vessel Pressure Transient Protection	NUREG-0224	September 1978
A-31	Residual Heat Removal Shutdown Requirements	SRP 5.4.7	1978
A-36	Control of Heavy Loads Near Spent Fuel	NUREG-0612	July 1980
A-39	Determination of SRV Pool Dynamic Loads and Pressure Transients	NUREG-0802	September 1982

Table 1. Unresolved Safety Issues for Which a Final Technical Resolution Has Been Achieved (continued)

Number	*Title*	*Report Number*	*Date*
A-40	Seismic Design Criteria	NUREG-1233	September 1989
A-42	Pipe Cracks in Boiling Water Reactors	NUREG-0313, Rev. 1	July 1980
A-43	Containment Emergency Sump Performance	NUREG-0897, Rev. 1	October 1985
A-44	Station Blackout	Regulatory Guide 1.155 NUREG-1032 NUREG-1109	August 1988 June 1988 June 1988
A-45	Shutdown Decay Heat Removal Requirements	NUREG-1289 NUREG/CR-5230	September 1988
A-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211	February 1987
A-47	Safety Implications of Control Systems	NUREG-1217 NUREG-1218 Generic Letter 89-19	September 1989
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	NUREG-1370	September 1989
A-49	Pressurized Thermal Shock	Regulatory Guide 1.154	February 1987

LLW Source Term Modeling. A computer code has been developed by the Brookhaven National Laboratory to provide an estimate of the radionuclide release from a shallow land low-level waste disposal facility. The source term code—named Breach, Leach and Transport (BLT)—consists of four submodels based on current shallow land burial practices and the four major processes expected to control the rate of radionuclide release from a disposal facility. The computer codes FEMWATER and FEMWASTE were selected to model the processes of ground-water flow and contaminant transport in unsaturated media. Models developed to predict container corrosion and waste form leaching were incorporated into the FEMWATER code. To provide confidence in the model predictions, the BLT code will be benchmarked against lysimeter experiments of saltstone waste forms at the Savannah River Laboratory and cement, bitumen, and polymer waste forms at PNL. Results of sensitivity analyses will 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 and the Massachusetts Institute of Technology, is intended to confirm the reliability of unsaturated flow and transport models ofLW disposal facilities. This work has been formally accepted in the INTRAVAL international study that deals with model validation of ground-water flow and transport models.

Rulemaking. A final rule on criteria and procedures for evaluating requests for emergency access to LLW disposal sites was issued in February 1989. A Regulatory Guide presenting a practical approach and a computer program for evaluating cover designs for uranium mill tailings was issued in June 1989.

Resolving Safety Issues and Developing Regulations

UNRESOLVED SAFETY ISSUES

The Energy Reorganization Act of 1974, as amended, requires that the annual report of the Commission to the President and the Congress include progress reports on those items previously identified as Unresolved Safety Issues (USIs). During fiscal year 1989, four more USIs were resolved. Table 1 is a listing of former USIs for which a technical resolution has been achieved. USIs resolved during fiscal year 1989 are discussed in the summary that follows. With the resolution of these issues, all USIs have now been completed.

SUMMARY OF STATUS

Systems Interactions
(USI A-17)

Adverse systems interactions are events that may jeopardize the independent functioning of nuclear plant systems. Because of the potentially broad bounds of this safety issue, the staff spent considerable effort in defining a workable scope. The final scope concentrated on identifying subtle, unintended dependencies (1) between redundant, safety-related structures, systems, and components, and (2) between safety-related and non-safety-related structures, systems, and components.

The staff considered a number of alternatives for the resolution of this USI, as was reported in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," dated August 1989. The analysis was largely based on the findings reported in NUREG-1174, "Evaluation of System Interactions in Nuclear Power Plants," dated May 1989.

Based on this work, the staff concluded that the best way to resolve the issue was to incorporate A-17 findings into a number of programs. Specifically, the A-17 insights developed for plant flooding and water intrusion from internal plant sources have been made part of the individual plant examination, or IPE, program. Potential adverse systems interactions during an earthquake are being addressed as part of the implementation of USI A-46, "Seismic Qualification of Equipment in Operating Plants."

The staff issued Generic Letter 89-18 setting out the bases for the resolution of USI A-17 and providing a summary of the lessons learned from operating events regarding adverse systems interactions. It is anticipated that this information will be factored into ongoing evaluations of operating experience.

The resolution of USI A-17 has been completed. A notification of actions taken was published in the *Federal Register* in August 1989.

Seismic Design Criteria
(USI A-40)

Rapid advancements in seismic design technology over the past decade have made it possible and necessary to update the NRC acceptance criteria for the seismic design of structures, systems, and components of nuclear plants. Under contract to the NRC, the Lawrence Livermore National Laboratory compared NRC seismic design criteria with current knowledge and published the results in "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria" (NUREG/CR-1161, dated May 1980). A cost-benefit analysis for the proposed changes in review criteria is reported in "Value/Impact Assessment for Seismic Design Criteria" (NUREG/CR-3480, dated August 1984). Based on these recommendations and the results of astaff-sponsored workshop for soil-structure interaction held in June 1986, the staff developed modifications to related review criteria.

These modifications consisted of proposed revisions to Standard Review Plan (SRP) Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3 and were published for public comment in June 1988. The public comments were reviewed in detail, and staff responses thereto are reported in "Recommendations for Resolution of Public Comments on USI A-40" (NUREG/CR-5347, dated June 1989). Subsequently, appropriate changes were made to the proposed SRP sections. A regulatory analysis was prepared and published as the "Regulatory Analysis for USI A-40, Seismic Design Criteria" (NUREG-1233, dated September 1989). The revised SRP sections have been completed, and a copy thereof placed in the Public Document Room. An announcement of their availability was published in the *Federal Register* in September 1989.

The revised review criteria will be used in review of new construction permits (CP), preliminary design approval (PDA), final design approval (FDA), and combined license (CP/OL) applications under 10 CFR Part 52. Incorporation of the proposed review criteria is expected to eliminate some potential sources of unwarranted conservatism and result in a seismic design that reflects an up-to-date understanding of this technology.

All actions on USI A-40 have now been completed, and this issue is considered to be resolved.

Safety Implications of Control Systems
(USI A-47)

The staff has evaluated the control systems provided by the four U.S. nuclear steam supply vendors; the systems are those typically in use during normal startup, shutdown, and online power operations of nuclear power plants. The purpose of the studies was to identify those control systems whose failure could (1) cause either transients or accidents to become more severe than was assumed at the licensing stage for each plant, (2) adversely affect any assumed or anticipated operator action during the course of an event, (3) cause technical specification limits to be exceeded, or (4) cause transients or accidents to occur at a frequency in excess of those established for abnormal operational transients and design basis accidents.

The NRC has completed its technical work on USI A-47 and has developed a resolution. The resolution is documented in the final reports of NUREG-1217, "Evaluation of Safety Duplications of Control Systems in LWR Nuclear Power Plants—Technical Findings Related to USI A-47"; NUREG-1218, "Regulatory Analysis for Resolution of USI A-47"; and Generic Letter 89-19.

The staff concluded from A-47 investigations that certain actions should be taken to enhance safety in LWR plants. The staff recommends that plants (1) provide systems to protect against reactor vessel/steam generator overfill events and to prevent steam generator dryout, (2) make provision in plant procedures and technical specifications for periodic verification of the operability of these systems, and (3) modify selected emergency procedures to ensure safe plant shutdown following a small-break loss-of-coolant accident. The recommendations apply to a limited number of plants for which additional protection is deemed warranted.

To implement this resolution, the NRC issued Generic Letter89-19 to all LWR plants recommending that these actions be effected under a prescribed schedule.

Hydrogen Control Measures
(USI A-48)

This issue arose out of the Three Mile Island (TMI) Unit 2 (Pa.) accident in 1979. Approximately 1,000 pounds of hydrogen burned up in the TMI containment when it was ignited. Depending on hydrogen concentrations, this combustible gas can deflagrate or detonate. Both occurrences can affect containment integrity and/or the operation of safety equipment within the containment. Following the TMI accident, extensive research programs were initiated by both the NRC and the nuclear power industry to control hydrogen produced by metal-water reactions in several types of containments and to study the effects of hydrogen combustion on safety-related equipment.

Based on this research, the Commission published hydrogen control standards in 10 CFR Part 50, addressing four of the five containment types in use. The standards are discussed in the 1987 NRC Annual Report, pp. 148 and 149.

In 1985, the National Research Council was requested to conduct a peer review of the hydrogen research programs as part of the NRC evaluation. Their report, "Technical Aspects of Hydrogen Control and Combustion in Severe ight-Water-Reactor Accidents," was published early in 1987. The nuclear industry research program of the BWR Hydrogen Control Owners Group has been appraised by the staff, and a report was issued in September 1988 on the evaluation of the adequacy of this program.

In September 1989, a generic summary report, NUREG-1370, was published as the resolution of this issue. A key conclusion from this report was as follows:

"On the basis of the extensive research effort conducted by the NRC and the nuclear industry, current regulatory requirements, including their implementation, and the independent review by the National Research Council Committee on Hydrogen Combustion, the staff concludes that no new regulatory guidance on hydrogen control for recoverable degraded-core accidents (like that which occurred at TMI-2) for these types of plants is necessary and that USI A-48 is resolved."

Based on this research, the Commission published hydrogen control standards in 10 CFR Part 50, addressing four of the five containment types in use. BWR Mark I and Mark II containments have been inerted. The owners of BWR Mark III and PWR ice condenser type plants have elected to use igniters as a hydrogen control system in compliance with the rule. Large, dry containment type reactors, because of their increased hydrogen dilution volumes, were not included in the rulemaking and are subject to further research; potential hydrogen requirements for these containments are being addressed in Generic Safety Issue (GSI) 121. (USI A-48 was originally developed to assess all reactor types, but with this rulemaking, the issue is more narrowly focused on BWRs with Mark III containments and PWRs with ice condenser containments.)

GENERIC SAFETY ISSUES

In December 1983, the Commission approved a priority listing, prepared by the staff at the behest of

Table 2. Issues Prioritized in FY 1989

Number	*Title*	*Priority*
15	Radiation Effects on Reactor Vessel Supports	HIGH
125.1.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	DROP
131	Potential Seismic Interaction Involving the Moveable In-Core Flux Mapping System Used in Westinghouse Plants	MEDIUM
139	Thinning of Carbon Steel Piping in LWRs	RESOLVED
B-31	Dam Failure Model	LI(DROP)
D-2	ECCS Capability for Future Plants	DROP

the Commission, of all generic safety issues, including TMI-related issues, based on the potential safety significance and cost of implementation of each issue. Information and guidance on generic safety issues are reflected in the NRC's Five Year Plan.

Priorities of Generic Safety Issues

The NRC has continued to use the methodology set out in the 1982 NRC Annual Report for determining the priority of Generic Safety Issues (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 (supplements in June and December). The list of issues includes TMI Action Plan (NUREG-0660) items and USIs (discussed in detail earlier in this chapter). 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 1989, the NRC identified six new generic issues, established priorities for six issues (Table 2), and resolved 14 GSIs (Table 3), other than USIs. One GSI scheduled for resolution was integrated into the action plan related to another ongoing NRC activity. Table 4 contains the schedules for resolution of all unresolved GSIs.

STANDARDIZED AND ADVANCED REACTORS

Advanced Reactor Concepts

The NRC staff has completed detailed technical reviews of three advanced reactor concepts submitted by the Department of Energy (DOE). The reviews sought to determine the acceptability and licensability of these unique advanced reactor designs. The conceptual designs comprise two advanced liquid metal reactors (PRISM and SAFR) and one advanced modular high-temperature gas-cooled reactor (MHTGR). Key policy issues associated with these designs were identified and submitted for Commission consideration. The staff will be updating recommendations to the Commission on the key policy issues, factoring in any insights gained from the evaluation of a report on the containment issue being prepared by DOE. Draft safety evaluation reports (SERs) for MHTGR, PRISM, and SAFR have been completed and ACRS comments received. The Commission decided to issue the draft SERs to DOE for their use in further evaluation of the designs. The HTGR draft SER, NUREG-1338, was issued in March 1989; the PRISM draft SER, NUREG-1368, was issued in September 1989; and the SAFR draft SER, NUREG-1369, will be issued in early fiscal year 1990 to document the work completed to date and to close out this review, since DOE has stopped work on the SAFR advanced reactor design.

Standardization

The NRC believes that standardization of nuclear power plant designs is an important initiative that can significantly enhance the safety, reliability, and availability of nuclear plants. The Commission intends to improve the licensing process for standardized nuclear power plants and to reduce complexity and uncertainty in the regulatory process. In this regard, the Commission issued a revised Standardization Policy Statement on September 15, 1987, affirming the

Table 3. Generic Safety Issues Resolved in FY 1989

Number	*Title*
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems
82	Beyond Design Bases Accidents in Spent Fuel Pools
99	RCS/RHR Suction Line Interlocks on PWRs
101	BWR Water Level Redundancy
115	Enhancement of Reliability of Westinghouse Solid State Protection System
122.2	Initiating Feed and Bleed
124	Auxiliary Feedwater System Reliability
125.I.3	SPDS Availability
134	Rule on Degree and Experience Requirements for Senior Operators
HF1.1	Shift Staffing
HF4.1	Inspection Procedures for Upgraded Emergency Operating Procedures
I.F.1	Expand QA List
II.C.4	Reliability Engineering
II.E.6.1	In-Situ Testing of Valves—Test Adequacy Study

Commission's intention to develop a rule codifying the process for approving standard plant designs.

In August 1988, the Commission issued for public comment proposed regulations (10 CFR Part 52) to implement the revised standardization policy. A public/industry seminar was conducted in October 1988 to facilitate the public comment process on the proposed rule. The final 10 CFR Part 52 rule on standardization was issued in April 1989. A regulatory framework is provided for certification of reference designs by means of rulemaking to obviate the need for the reconsideration of design issues in individual licensing proceedings on future license applications that reference the certified designs.

FUEL CYCLE, MATERIALS, TRANSPORTATION AND SAFEGUARDS

In fiscal year 1989, the NRC continued incipient rulemaking efforts and developed final rules on activities pertaining to the transportation of radioactive materials, the physical protection of special nuclear material, the use or disposal of material containing very small quantities or concentrations of radioactive material, and access authorization at nuclear power plants.

Specifically regarding nuclear material transportation, a proposed major revision of the NRC's regulations was issued in June 1988 for public comment. The

Table 4. Generic Safety Issues Scheduled for Resolution

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
15	Radiation Effects on Reactor Vessel Supports	HIGH	TBD
23	Reactor Coolant Pump Seal Failures	HIGH	07/91
29	Bolting Degradation or Failures in Nuclear Power Plants	HIGH	05/90
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	07/92
70	PORV and Block Valve Reliability	MEDIUM	11/89
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NEARLY RESOLVED	03/90
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	01/90
83	Control Room Habitability	NEARLY RESOLVED	03/90
84	CE PORVs	NEARLY RESOLVED	02/90
87	Failure of HPCI Steam Line Without Isolation	HIGH	11/89
94	Additional Low-Temperature Overpressure Protection for Light-Water Reactors	HIGH	11/89
103	Design for Probable Maximum Precipitation	NEARLY RESOLVED	10/89
105	Interfacing Systems LOCA at LWRs	HIGH	10/91
106	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	05/91
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	HIGH	01/92
121	Hydrogen Control for Large, Dry PWR Containments	HIGH	02/91
128	Electrical Power Reliability	HIGH	03/90
130	Essential Service Water Pump Failures at Multi-plant Sites	HIGH	03/90
135	Steam Generator and Steam Line Overfill	MEDIUM	11/90
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage	MEDIUM	10/89
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	01/92

Table 4. Generic Safety Issues Scheduled for Resolution
(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
B-55	Improved Reliability of Target Rock Safety Relief Valves	MEDIUM	05/90
B-56	Diesel Reliability	HIGH	11/89
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	04/90
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	03/90
C-8	Main Steam Line Isolation Valve Leakage Control Systems	HIGH	10/89
I.D.3	Safety System Status Monitoring	MEDIUM	TBD
I.D.5(3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	12/89
II.H.2	Obtain Technical Data on Conditions Inside TMI-2 Containment Structure	HIGH	12/91
II.J.4.1	Revise Deficiency Report Requirements	NEARLY RESOLVED	TBD
HF4.4	Guidelines for Upgrading Other Procedures	HIGH	11/89
HF5.1	Local Control Stations	HIGH	09/90
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	05/91

comment period for this proposed rule has been extended until the Department of Transportation issues a companion rule. The rule maximizes compatibility between U.S. and international regulations and imposes additional requirements on the transportation of low specific activity (LSA) material—typically material with such a limited concentration of radioactivity that packagings are not required to be accident resistant.

In the safeguards area, the NRC reviewed its safeguards regulations and guidance documents, identifying areas in the regulations that were out-of-date, susceptible to differing interpretations, or otherwise needed clarification. A proposed rule correcting these deficiencies was published for public comment in July 1989.

The most significant alterations deal with policy questions that apply to the use or disposal of radioactive materials containing quantities of radionuclides so small that they do not need to be further regulated. On March 14, 1988, a status report was issued on the subject. It was followed by Commission discussions of a proposed policy statement and a decision to release an advance notice of the development of a Commission policy statement. The advanced notice was published in the *Federal Register* in December 1988. An international workshop to promote cooperation and foster mutual understanding on the subject was held in October 1988. In January 1989, a public meeting was held to gather information and solicit comments on the Commission's draft policy. A revised draft policy statement on exemption from regulatory control was sent to the Commission in June 1989. In connection with this broader effort, a proposed rulemaking allowing on-site incineration of waste oilgenerated at nuclear power plants was issued for public comment in August 1988. The comments have been evaluated and incorporated into the final rulemaking package.

Finally, on the subject of unescorted access to nuclear power plants, a proposed Commission policy statement endorsing industry guidelines for an access authorization program was issued on March 9, 1988. Following evaluation of public comments, a decision paper on access authorization was forwarded to the Commission in March 1989. The Commission decided that the provisions of this policy should be incorporated into NRC regulations by way of a final rulemaking, rather than the issuance of a policy statement.

DEVELOPING AND IMPROVING REGULATIONS

In a program initiated in 1985 and continued through 1989, the NRC staff undertook to evaluate existing regulatory requirements in terms of their risk effectiveness and to eliminate or modify requirements with only a marginal safety importance. A three-volume research report (NUREG/CR-4330) provided detailed technical assessments of requirements associated with a number of topics. Based on these and continuing studies, the NRC staff, in March 1989 and again in August 1989, forwarded Commission papers that identified progress achieved to date and outlined the resources needed to conclude the program to eliminate or modify regulatory requirements of marginal safety importance.

It is clear that a major need exists for additional spent fuel storage space at commercial nuclear power reactor sites, to be available in the near future. In response to this need, the Nuclear Waste Policy Act of 1982 directed the Secretary of Energy to establish a dry spent fuel storage demonstration program, with the objective of coming up with one or more technologies that the NRC might approve for use at civilian nuclear power reactor sites without, to the extent practicable, requiring additional site-specific approvals. A proposed rule, 10 CFR Part 72, published for public comment in the *Federal Register* in May 1989, would allow holders of nuclear power reactor operating licenses to store spent fuel in NRC-approved casks at reactor sites under a general license.

The Commission issued an advanced notice of proposed rulemaking to inform the public that the NRC is considering a proposed amendment to its regulations regarding enhanced professional or educational credentials for senior nuclear power plant operating personnel. The proposed requirements are intended to further ensure the protection of the health and safety of the public by improving the capability of shift operating crews to effectively respond to off-normal situations; they could also add operating experience to plant management by opening a career path for senior operators into plant management. The proposed rule was published for public comment in the Federal Register in December 1988. In April 1989, the Commission decided to issue a policy statement, rather than a final rule, on the subject. The final policy statement was issued in August 1989.

A proposed rule has been developed to amend the 10 CFR Part 35 regulations that apply to the medical use of byproduct material. The amendments would require medical-use licensees to implement quality assurance (QA) programs and would revise misadministration reporting requirements. Implementation of the new requirements would be supported by issuance of a Regulatory Guide that would include specific criteria for medical QA programs. The feasibility of this approach will be evaluated during a pilot program involving several medical-use licensees. The NRC expects to issue the proposed rule for public comment in early fiscal year 1990.

In March 1988, the Commission issued a Policy Statement on the Maintenance of Nuclear Power Plants. In the statement, the Commission indicated its intention to pursue a rulemaking on maintenance. In developing this proposed rulemaking, the staff had extensive interactions with U.S. industry (airline and nuclear) and studied foreign nuclear maintenance programs and practices. In addition, a three-day public workshop was held in July 1988 to solicit comments on rulemaking options. Information gathered in these interactions and from the workshop was used in formulating the proposed rule and its supporting Regulatory Guide. The Commission issued the proposed rule for public comment in November 1988. In June 1989, the Commission decided to hold the final rule on maintenance in abeyance for 18 months and to issue instead a revised policy statement. The draft proposed Regulatory Guide was published in August 1989, and the revised final policy statement on maintenance will be published in early fiscal year 1990.

In April 1989, the NRC issued two final rules to amend its 10 CFR Part 73 and 10 CFR Part 50.62(c)(4) regulations. The Part 73 regulation relates to access to safeguards information and, as amended, requires licensees to conduct an FBI criminal history check for certain individuals who need access to safeguards information. The Part 50 regulation was amended to larify equivalent control capacity for BWR standby liquid control systems.

Table 5. Rulemaking Actions Processed During FY 1989

Rulemaking Activities		*Number*
Final Rulemakings Published		17
Rulemakings Terminated/Withdrawn		3
Ongoing Rulemaking Actions		42
Proposed Rulemakings	(20)	
Final Rulemakings	(11)	
Rulemakings on Hold	(11)	
	Total Rulemakings	62

Emergency Preparedness

On April 7, 1989, the Commission approved a final regulation on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees. The regulation requires approximately 30 major NRC fuel cycle and other radioactive material licensees to maintain emergency plans. The plans are for coping with serious accidents involving licensed radioactive materials for which responses by off-site response organizations (such as police, fire, and medical organizations) might be needed. This action is intended to ensure that these licensees are prepared to take action to protect public health and safety if an accident occurs.

In addition, on September 1, 1989, the staff forwarded to the Commission proposed resolutions to those petitions for rulemaking dealing with the size of the emergency planning zones. The petitions evaluated are: PRM-50-31 filed December 21, 1981, by the Citizens Task Force of Chapel Hill, N.C.; PRM 50-45 filed on August 6, 1986, by Mr. Kenneth G. Sexton; and PRM-50-46 filed on October 14, 1986, by the Attorney General, State of Maine. The Commission is currently evaluating the staff recommendations.

Summary of Rulemaking Actions

During fiscal year 1989, 62 rulemaking actions were processed, of which 17 rules were formally published, 3 were terminated/withdrawn, and 42 are ongoing. (See Table 5.) Besides the 42 ongoing rulemaking actions, it is estimated that in fiscal year 1990, there will be approximately 12-to-15 new rulemakings requiring RES review and approval by the Executive Director for Operations.

Regulatory Analysis

The NRC has, as one of its prime tasks, responsibility for the oversight of regulatory impact analyses (RIAs) of rulemakings, backfits, generic safety issues, or Regulatory Guides. Pursuant to this obligation, the NRC has published operating procedures for agency use for the support and/or review of regulatory impact analyses affecting all regulatory actions. The staff is also concerned with the development and implementation of systematic methods for performing RIAs. For example, the NRC has issued RES Office Letter Number 2, Regulatory Impact Analysis Guidelines, dated November 18, 1988. The office letter provides procedures for oversight support for initiating offices/divisions during their development of regulatory impact analyses. Also for agency use are improvements completed on "Forecast," a PC-based cost evaluation model. This software package presents a checklist of major elements for appraising the costs of nuclear plant physical modifications. And an update of the NRC's Generic Cost Estimates for construction-related activities at nuclear power plants has been completed. This work, published as NUREG/CR-4627, Revision 1, was developed to support NRC analysts in determining generic estimates for removal, installation, and total labor costs for construction-related activities at nuclear power stations.

Development of these types of methodologies, coupled with existing RIA methods, will continue in an effort to facilitate NRC decision-making in evaluating 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 this report period, approximately 17 safety-related regulatory impact analyses (both initiated and completed) have been processed.

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 twice 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 requested comments on NUREG-1317, "Regulatory Options for Nuclear Plant License Renewal," issued in August 1988. Comments were summarized and analyzed in NUREG/CR-5332, "Survey and Analysis of Public Comments on NUREG-1317: Regulatory Options for Nuclear Plant License Renewal," issued in March 1989. Before the close of the report period, the NRC staff developed a plan for the completion of the rulemaking; the draft and final rules are expected to be published in mid-1990 and in mid-1991, respectively.

SEVERE ACCIDENT IMPLEMENTATION

In the 10 years since the TMI-2 accident, the NRC has pursued an active program of research into severe nuclear power plant accidents. The research has been multi-faceted, encompassing such broad areas as 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 included the judgment that existing plants posed no undue risk to public health and safety. The Commission took note of the fact that systematic examinations of existing plants could identify plant-specific vulnerabilities to severe accidents for which further safety improvements could be justified. Modification of the Commission's rules or policies regarding siting, emergency planning, containment design, as well as by the resolution of specific severe accident issues are ways in which the results of severe accident research translate into regulatory action.

Individual Plant Examinations

Consistent with the Commission's Severe Accident Policy Statement, the staff has required individual plant examinations (IPE) of all existing plants to identify any plant-specific vulnerabilities to severe accidents. This effort has involved development of guidance for performance of the IPE, preparation of a generic letter to plant operators requesting the IPE, and the development of review plans and eventual review of IPE results in concert with NRR. Any requirement to correct identified plant-specific vulnerabilities not already corrected voluntarily will be subject to the backfit rule. IPE results will be applied to accident management planning and preparation.

On November 23, 1988, after the close of the report period, the NRC issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR 50.54(f)," to all licensees of nuclear power reactor facilities. This letter requested that licensees perform a plant examination in search of vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. The specific objectives for these IPEs were for each utility (1) to develop an overall appreciation of severe accident behavior; (2) to understand the most likely severe accident sequences that could occur at its plant; (3) to gain a more quantitative understanding of the overall probability of core damage and radioactive material releases; and (4) to reduce, if necessary, the overall probability of core damage and radioactive material release by appropriate modifications to procedures and installation of hardware that would help prevent or mitigate severe accidents. Upon completion of the examination, the utility is required to submit a report to the NRC describing the results and conclusions of the examination. That submittal will be reviewed and evaluated by the NRC.

The NRC had also issued NUREG-1335, "Individual Plant Examination: Submittal Guidance," as a draft for comment by January 1989 to provide guidance on the conduct of the IPEs. A workshop was held on February 28 and March 1 and 2, 1989, in Ft. Worth, Tex., for utilities and interested members of the public to frame comments and questions on the IPE process and the guidance document. NUREG-1335 was revised to take account of comments received and was issued in final form in August 1989. The issuance of NUREG-1335 formally started the IPE process. Utilities will have three years (until September 1, 1992) to complete and submit their IPEs to the NRC.

External Events

In December 1987, the NRC established an External Event Steering Group (EESG) to make recommendations concerning individual plant examinations for vulnerabilities to severe accidents initiated by external events (e.g., earthquakes, floods, fires). Recommendations are needed regarding (1) what external events need to be considered in the IPE, (2) what methods can be used in the examination, and (3) how the IPE for External Events (IPEEE) is to be coordinated with other ongoing regulatory activities involving external events, particularly in the seismic area.

In April 1988, three subcommittees were established to make recommendations in the areas of (1) seismic events, (2) fires, and (3) high winds, flood and others (e.g., man-made hazards such as those caused by nearby transportation and military and industrial facilities).

During 1989, the three subcommittees completed their studies and made recommendations on the IPEEE to the EESG. These recommendations will form the basis for a generic letter to licensees calling for consideration of external events in the IPEEE. The letter is expected to be issued during fiscal year 1990.

Containment Performance Improvements

Severe accident research has engendered a number of insights concerning containment performance during a severe accident. These insights encompass both strengths and weaknesses in 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 scrutinizes severe accident research insights, in order to identify containment vulnerabilities and potential improvements to correct vulnerabilities. Because of concerns about Mark I containments, the CPI program initially focused on this containment design. However, studies of all other types of containments are also in progress. If potential improvements that can provide significant enhancements to safety are identified—and are shown to be cost-effective pursuant to 10 CFR 50.109—this program will produce specific regulatory recommendations.

The CPI program is closely related and complementary to the IPE (see above) and accident-management program. Under the CPI program, containments are appraised for vulnerabilities on a generic basis, so that utilities do not have to deal with complex and highly uncertain severe accident phenomena on an individual basis. The IPE, on the other hand, deals with plant-specific containment vulnerabilities, those unique to a particular plant and are not treated under the generic CPI program. Utilities will not have to make containment changes related to uncertain phenomena disclosed by the IPE program until results of the CPI program have been considered and factored into their IPEs.

In January 1989, the NRC staff presented recommended improvements for BWR plants with Mark I containments to the Commission (SECY-89-017). The staff concluded that the best way to reduce overall risk in BWR Mark I plants was to pursue a balanced approach, using accident prevention and mitigation. Five specific improvements were recommended: (1) an improved hardened vent capability, (2) improved reactor vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, (4) extended emergency procedures and training, and (5) accelerated staff actions to implement the station blackout rule. These improvements, although not representing large changes to the plants, would substantially enhance safety of Mark I plants by reinforcing basic defense-in-depth provisions, including enhancement of containment performance. The staff evaluated the improvements and found them to be cost-effective.

Based on subsequent Commission guidance, the staff initiated, late in the report period, certain plant-specific backfits to require a hardened vent on all Mark I plants for which it could be shown to be cost-effective. This backfit process should be completed in fiscal year 1990, and the hardened vent should be implemented at plants for which it is cost-effective within three years. The other recommended improvements are to be considered by utilities with Mark I plants as part of the IPE process. A supplement to the IPE Generic Letter 88-20 was issued on August 29, 1989, forwarding information on the recommended improvements to utilities with Mark I plants.

RADIATION PROTECTION AND HEALTH EFFECTS

The NRC conducts research and standards development in radiation protection to ensure continued protection of workers and the public from radiation and radioactive materials in connection with licensed activities. The radiation protection program is currently focused on improvements in health physics measurements and the review and dissemination of dose reduction research performed by other Federal agencies and industry. One goal is to provide acceptable performance standards for the many measurements required of licensees. The program also contributes to monitoring licensee performance in areas such 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, in addition to conducting a limited number of 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. In fiscal year 1989, a feasibility study was initiated to determine whether the extensive data base on cellular and molecular effects can be used to reduce the uncertainties in health risk estimates for low dose and dose rates.

Radiation Protection Issues

Brookhaven National Laboratory ALARA Center. The Brookhaven National aboratory (BNL) ALARA Center, funded by the NRC, continued its work on surveillance of DOE and industry dose reduction and ALARA research. BNL has published a series of reports (NUREG/CR-3469) that abstracts 252 national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination. In 1989, BNL focused on 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 its licensees for information and advice on all aspects of radiation protection and dose reduction.

In 1989, BNL hosted a second International Workshop on New Developments in Occupational Dose Control and ALARA Implementation at Nuclear Power Plants. The workshop was attended by 125 representatives from U.S. utilities, the NRC, DOE, all NRC Regions, and technically advanced countries. Numerous new techniques and reports on demonstration of dose reduction efforts were discussed.

Low-Level Liquid Effluent Exposure Study. A study is in progress to investigate the potential reconcentration of radioactivity from nuclear power plant liquid effluents, for selected nuclear power plants having "holdup" systems. This study will characterize low-level liquid effluent systems, the accumulation of radioactive material from these systems, and resultant radiation doses. The information will be used by the NRC staff to identify systems likely to result in a significant buildup of radioactive materials and to determine whether further regulatory action is needed.

Accreditation and Testing of Personnel Dosimetry Processors. NRC requirements for the 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, 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. A draft Regulatory Guide, expected to be published in fiscal year 1990, will discuss the radiation categories required, mixed field requirements, dosimeter exchange rates, and other topics.

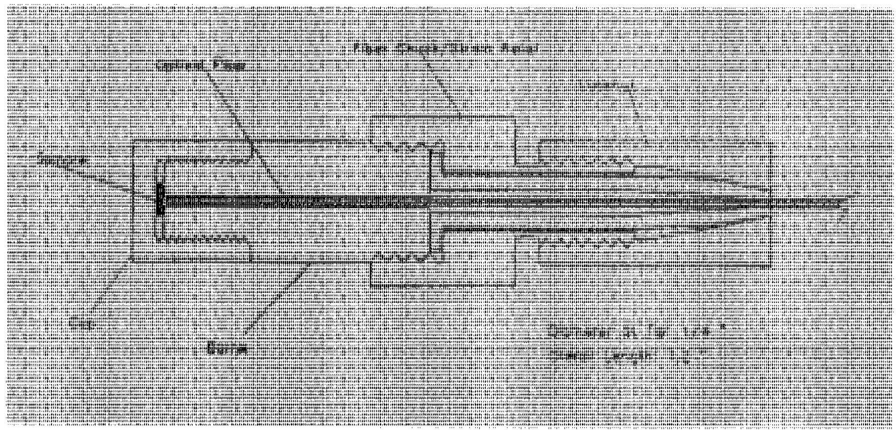
As of June 1989, 61 laboratories were accredited. These include commercial dosimetry processors, military establishments, commercial shipbuilders, nuclear power companies, and other commercial establishments that use radiation measurement techniques.

NRC continued a testing program for the personnel extremity dosimetry processors, undertaken to establish an accreditation under NVLAP for these kinds of processors similar to that for personnel whole body dosimetry processors discussed above. Initial testing of processors against the draft national consensus standard indicated that processors were having difficulty meeting the performance criteria for accuracy and precision specified in the standard. A second series of tests was instituted, and it is expected that a final set of tests will begin as soon as the draft standard is adopted in final form. Successful performance in the final set of tests is expected to lead to a requirement for NVLAP accreditation of processors to meet the performance requirements of the national consensus standard.

External Dosimetry. NUREG/CR-5100, "Integrating Fiber Optic Radiation Dosimeter," published in March 1989, describes the development and testing of a device for remote measurement of radiation fields. The research, carried out under a Phase II Small Business Innovative Research and Development (SBIR) contract, involves the use of electron-trapping phosphors coupled to long optical fibers of around one-half millimeter diameter, which in turn are coupled to a photomultiplier. Remote readout is accomplished by exciting the phosphor with infrared pulses, which then produces visible light pulses in the phosphor that travel through the optical fiber to the photomultiplier, where they are analyzed. The contractor is filing a patent application on the invention and intends to develop commercial applications for the process.

The computer code SADDE (Scaled Absorbed Dose Distribution Evaluator) was developed to supplement the previously developed VARSKIN code. VARSKIN is used for calculating radiation dose from radioactive contamination on the skin. The application of the VARSKIN code has been limited to radionuclides for which critical data had previously appeared in the scientific literature. SADDE allows the user to calculate necessary input data for VARSKIN for any radionuclide. The code and its application is described in NUREG/CR-5276 published in January 1989.

Besides development of general methods for calculating radiation doses to the skin, work is progressing



In the fiber optic dosimeter, radiation striking the phosphor sample at the left end of the probe excites electrons up to trapping levels in the phosphor. The phosphor is read by sending a beam of infrared light down the optical fiber, causing the trapped electrons to fall back to the ground state. In doing so, they emit visible light that travels up the optical fiber to a light-sensing detector. The amount of light emitted is proportional to the quantity of radiation that struck the phosphor sample.

on defining the effects of irradiation of the skin by very small radioactive particles ("hot particles"). This effort arises out of recent incidents of hot-particle exposure at NRC-licensed facilities and a current report by the National Council on Radiation Protection and Measurement (NCRP) emphasizing the need for a better understanding of these effects and their relative importance within the universe of known health effects resulting from radiation exposure. The results of this work, as well as other ongoing work by non-government groups, will be considered by the Commission in re-examining its regulatory requirements with respect to the radiation exposure of the skin.

Effects of Worker/Dosimeter Geometry on Dose Measurements. Work has begun under an SBIR contract to determine the quantitative effects of worker/dosimeter geometry on dose measurements, by simulation and experiment. A firm proposed to apply fractal mathematics in an innovative approach to the investigation of this generic problem. In the first phase of the project, the contractor will develop a computer simulation experiment to model the effects, and provide quantitative values of the effects, to be used in the validation and verification of experimental results.

Self-Powered Gamma-Ray Detector. Research under an SBIR contract is being carried out to develop a self-powered gamma-ray detector (SPGRD), employing 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). The contractor has fabricated and tested two basic designs and established the feasibility of the concept. Present work involves construction and testing of prototypes of a directional detector and an isotropic detector. If successful, the device is expected to be used by radiation workers to minimize exposures to radioactive gamma-ray sources.

Health Effects Research

Embryo/Fetal Dose from Maternal Intake. A study to improve understanding of the effects of radionuclide burdens on the mother on prenatal radiation exposure to the embryo or fetus was continued in fiscal year 1989, with significant progress. In fiscal year 1990, the NRC expects to issue a preliminary report on methodology for calculating embryo/fetus dose attributable to the maternal radionuclide burden. This information is needed to assess consequences of accidental releases of radionuclides, and also to ensure compliance with the proposed 10 CFR Part 20.

Improvement of Health Effects Models. Considerable progress has been made in the development of models for predicting early health effects resulting from combined internal and external radiation in case of severe accidents.

Two reports—NUREG/CR-5351, "Models for Pulmonary Lethality and Morbidity After Irradiation from Internal and External Sources," and NUREG/CR-5353, "Inhaled 147pm and/or Total Body Gamma Radiation: Early Mortality and Morbidity in Rats"—were published during the report period. These, together with three three NUREG/CR reports on the subject published in fiscal year 1988, culminate a 10-year multi-laboratory research effort that has provided the scientific basis for development of models to predict the early effects of external and internal irradiation from reactor accidents.

Revision 1 to NUREG/CR-4214, "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis, Part II, Scientific Bases for Health Effects Models," was also published. This document describes models reviewed and validated by U.S. and international experts that can be used to assess the radiological health effects, including potential morbidities and mortalities, from nuclear power plant accidents. Models of early and continuing effects, cancers and thyroid nodules, and genetic effects are provided. The models will be incorporated into computer codes applied in NUREG-1150, the Commission's risk re-baselining study, "Severe Accident Risks: An Assessment for Five Nuclear Power Plants." The new models have also been used in developing the staff-proposed Policy Statement on Exemption From Regulatory Control. Worldwide interest has been expressed in the report.

The early effects models have been substantially revised in Revision 1 and address four causes of mortality and nine categories of morbidity. These models are based upon two-parameter Weibull functions. They permit evaluation of the influence of dose rate and address the issue of variation in radiosensitivity among the population. The revisions to the late somatic and genetic effects section of the report were relatively minor.

Linear and linear-quadratic models are recommended for estimating cancer risks. Parameters are given for analyzing the risks of seven types of cancer in adults. Models of childhood cancers due to *in utero* exposure are also provided. The models of cancer risk are derived largely from information summarized in the BEIR III report—with some adjustments to reflect recently reported results from the follow-up on survivors of the Hiroshima and Nagasaki bombings—and permit analysis of both morbidity and mortality.

The new models for genetic effects allow prediction of genetic risks in each of the first five generations after an accident and include information on the relative severity of various classes of genetic effects. In addition, the impact of radiation-induced genetic damage on the incidence of pre-implantation embryo losses is discussed.

The uncertainty in modeling radiological health risks is accommodated by providing central, upper, and lower estimates of all model parameters. Data that should enable analysts to consider the timing and severity of each type of health risk are provided.

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 doses 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 both 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 1987 revealed that about 240,000 persons were monitored, of whom about 50 percent received measurable doses. The workers received a collective dose of 44,000 person-rems or an average annual dose of about 0.4 rem-per-worker among those receiving a measurable dose. 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 1988 indicate that the collective dose remained at about the 1987 value of 41,000 person-rems, even though five new plants reported.

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 480,000 persons, most of whom have 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 78,000 persons terminating employment during 1985 revealed that 7,000 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.

Changes to Radiation Protection Guidelines—Revision of 10 CFR Part 20. In 1989, the staff completed and sent to the Commission a revision of 10 CFR Part 20, Standards for Protection Against Radiation, which contains the basic requirements for protecting workers and members of the public from radiation resulting from NRC-licensed activities.

Major changes from the current Part 20 include:

(1) Elimination of quarterly dose limits for workers.

(2) Elimination of the age-prorated cumulative dose limit (5(N-18)).

(3) Requirements for limiting the sum of both internal and external doses when both components exceed 10 percent of the dose limits.

(4) An explicit dose limit for members of the public.

(5) Updated intake and concentration limits for both workers and members of the public.

Several issues regarding the application of revised air concentration limits had to be examined before the Commission could act on the rule; these were resolved by the staff in September 1989.

Residual Contamination Criteria. During fiscal year 1989, a contract with Pacific Northwest Laboratories (PNL) was continued to provide a technical basis for a revision of criteria for residual contamination of soils and structures during decommissioning. Drafts of a NUREG/CR publication were reviewed, and a final document was expected early in fiscal year 1990. The NRC staff expects to use this technical information as the foundation for revisions of Regulatory Guide 1.86 and various branch technical positions.

Proposed Rule on Large Irradiators. A proposed rule on large irradiators was prepared and sent to the Commission for its approval to publish. 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. Publication of the proposed rule is expected in early 1990. The final rule is scheduled for publication in early 1991.

Interpretation of Bioassay Measurements. A Regulatory Guide is under preparation that would facilitate the implementation of the revised 10 CFR Part 20 and endorse the methodology presented in NUREG/CR-4884 for the estimation of intakes based on *in vivo* and *in vitro* bioassay measurements. The guide will fulfill the need for a consistent approach to the interpretation and assessment of individual intakes of radioactive material by exposed persons.

Safety Requirements for Industrial Radiographic Equipment. A final rule incorporating the performance requirements of American National Standard N432, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography"—plus a number of other requirements designed to improve safety—was approved by the Commission and was expected to be published shortly after the close of the report period. The rule, intended to reduce the number of overexposures to radiographers and to provide additional protection to the public, features reduced radiation levels for radiographic exposure devices, improved source-to-cable connectors, automatic source locking upon retraction, and a requirement for radiographers to wear "alarming" dosimeters.

Certification of Industrial Radiographers. A proposed rule that would recognize a radiation safety certification program being developed by the American Society for Nondestructive Testing (ASNT) has been sent to the Commission for consideration. The NRC staff believes the ASNT program would focus more attention on radiation safety and contribute to more uniform performance in the radiography industry.

Coordination With Worker Groups. The coordination and information exchange effort with unions representing utility workers was continued. Several members of the NRC staff from three NRC offices conducted a lecture and discussion program at the 1989 Utility Workers Union of America (UWUA) nuclear conference. Presentations were made on the "hot particle" problem, human factors, the proposed new radiation protection regulations, the "below regulatory concern" issue, low-level waste disposal, the NRC/Occupational Safety and Health Administration (OSHA) working relationships, the NRC enforcement program, and current fitness for duty issues.

Particular interest was expressed with respect to NRC/union meetings; the availability of NRC reports; working hours; drug testing; the impact of Regulatory Guides, policy statements, and generic letters on workers; transient worker dose; worker dose and dose limits; limitations on OSHA inspector access to nuclear power plants; OSHA and NRC requirements related to worker safety; and the effects of enforcement on individual workers. In addition to UWUA workers, representatives of the Oil, Chemical and Atomic Workers Union and the International Brotherhood of Electrical Workers were present.

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 1989, 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 195 NRC staff members serve on working groups organized by technical and professional societies.

Proceedings and Litigation

Chapter 9

This chapter covers the three major areas of the NRC's litigatory and judicial activity, during fiscal year 1989: (1) reports, with discussion of significant decisions, from the NRC's Atomic Safety and Licensing Board Panel and from the NRC's Atomic Safety and Licensing Appeal Panel; (2) noteworthy Commission decisions in cases under litigation; and (3) a "judicial review" of litigation involving the NRC during the period, including cases pending and closed.

Office of the Secretary. The Secretary of the Commission is charged under 10 CFR Part 2 with establishing and maintaining 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, the Commission's Atomic Safety and Licensing Boards, and the Atomic Safety and Licensing Appeal Board. 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 on which the NRC seeks views before taking final action.

The Docketing and Service Branch also serves orders of the Commission, the Atomic Safety and Licensing Board, and the Atomic Safety and Licensing Appeal Board on the parties to a proceeding and certifies indexes of the dockets to the courts.

ATOMIC SAFETY AND LICENSING BOARDS

The Atomic Energy Act of 1954, as amended, requires that a hearing precede every issuance of a construction permit for a nuclear power plant or related facility. In addition, the Act requires an opportunity for a hearing in connection with any other licensing proceeding conducted under the Act. 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.

Adjudicatory hearings under the Atomic Energy Act are conducted before a board whose members are drawn from the Atomic Safety and Licensing Board Panel ("the panel"), created by the Commission under authority of Section 191 of the Act. Under other sections of the Act, or by Commission rules, an opportunity for a hearing must be provided on such matters as antitrust issues, enforcement actions, civil penalties, or other matters, as directed by the Commission. These hearings are the Commission's principal public forum in which individuals and organizations can voice their concerns in a particular licensing, enforcement, or other matter, and have those interests adjudicated by an independent tribunal.

Licensing and construction permit hearings for commercial nuclear power reactors and related facilities are conducted before Licensing Boards consisting of three administrative judges chosen from the panel. In other matters, hearings may be conducted before a single administrative judge or administrative law judge from the panel. (See "The Licensing Process" in Chapter 2.) Appointment to the panel itself by the Commission is based upon recognized experience, achievement and independence in the appointee's field of expertise. The Commission or the panel's Chief Administrative Judge assign individual judges to particular hearings where their professional expertise will assist in resolving the particular technical and legal matters at issue in a proceeding. As of September 30, 1989, the panel included 37 administrative judges (15 full-time and 22 part-time). By profession, they include 13 lawyers, 11 public health and environmental scientists; 10 engineers or physicists, one medical doctor, one chemist, and one economist. See Appendix 2 for current membership.

ASLBP Caseload

During the fiscal year ending September 30, 1989, the panel's proceedings comprised 22 cases involving 13 different nuclear power plants or related facilities and 16 proceedings involving other Commission licensees. A total of 105 days of hearings (78 trial and 27 pre-hearing conference days) were held. Nineteen proceedings were closed and 17 new proceedings were docketed.

Besides dealing with its on-going caseload, the panel must anticipate and prepare for its future caseload burden. In connection with the expected construction of a high-level nuclear waste repository, the panel took an active supporting role in the development of the procedural rules and support systems intended to govern the proceeding. With Commission adoption of those rules, the panel undertook to share its experience and expertise in "electronic dockets" with the Office of the Licensing Support System (OLSS), the office created by the Commission to oversee the state-of-the-art, full text-and-image, computerized document retrieval system, to be used by the parties and the panel in conducting the high-level waste proceeding.

Case Management and Litigation Support

Because of continued restrictions in the number of support personnel and as part of the panel's on-going program to reduce delays in the licensing process, the panel had moved rapidly in recent years towards achieving an "electronic" office, particularly for management of its voluminous and complex hearing records. In fiscal year 1989, the goal was realized. Important administrative tasks—travel arrangements, timekeeping, etc.—have been computerized. All professional and support staff now routinely work at individual computer workstations. Judges and professional support staff can, from their desks, draft, share and comment on proposed decisions; access and quickly search either the panel's electronic docket or the Commission's document retrieval system; conduct legal research through LEXISand/or WESTLAW; and communicate with one another or with other agency personnel through the Commission's electronic mail system.

To exploit the full utility of its computer facilities, the panel continued to expand and enhance INQUIRE, an electronic docket conceived, developed and maintained by the panel. INQUIRE is composed of an adjudicatory data base and a companion search and retrieval interface and currently operates on an IBM 9370 mini-computer, physically located at the Commission's White Flint One facility. Several offices, including the Commission, are wired directly to the mini-computer containing INQUIRE, thereby permitting quick and continuous access to the system. The panel is directly connected to the White Flint mini-computer through a communications controller located at the panel's Bethesda offices. Other authorized users may access INQUIRE from any location using a personal computer equipped with a modem.

By the end of the day on which any document related to any proceeding is received, the document has been abstracted and entered into the ASLBP's adjudicatory data base. In selected complex cases, the full text of significant documents—such as pre-filed testimony and hearing transcripts—will be electronically indexed and added to the adjudicatory data base. As of the close of the fiscal year, approximately 100,000 pages of hearing transcripts and related materials had been loaded onto the panel's adjudicatory data base. Finally, all Licensing Board Panel and Atomic Safety and Licensing Appeal Board decisions are added to the adjudicatory data base in full-text form, generally on the date those decisions are issued. Where appropriate, discrete portions of the data base concerning a specific proceeding can be loaded onto the hard disk of one of the panel's portable computers for use by judges conducting hearings in the field.

The INQUIRE system, cited above, uses a search and retrieval logic similar to that employed by the LEXIS and WESTLAW legal research systems. However, to permit easy access to the system by a potentially wide range of users with varying degrees of expertise, INQUIRE employs a series of "user-friendly" form fill-in screen panels. Based on information provided through these panels by the user, regarding the nature, scope and form of search desired, INQUIRE automatically generates and executes the necessary search and retrieval logic i.e., the appropriate commands. INQUIRE also produces formatted and indexed reports according to user-defined layouts which give information on the types of documents contained on the system, and particular documents can be down- loaded for printing or word processing.

Hearing Procedure

Besides these measures to computerize the licensing process, the panel continues as before to explore and implement traditional case management tools and techniques, in an effort to streamline, focus and resolve contested licensing matters. Where appropriate, boards frequently structure their hearing schedule into distinct phases, each dealing with discrete groupings of related issues. In the case of a complex proceeding involving several topics and multiple issues under each, the panel has sometimes created separate, parallel licensing boards and assigned one or more of those topics to each board. Besides saving time through the parallel adjudications, a particular board can be made up of panel members whose expertise matches the issues to be resolved.

Licensing Boards have also taken an active role in shaping the issues before them through a thorough review and, if appropriate, in consolidating admissible contentions, in monitoring the discovery portion of the proceeding, and in 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; one consequence over the last three years is the increasing percentage of the panel's cases that have been settled before final adjudication.

As the need for initial operating license proceedings for power reactors winds down, the panel is turning its attention to the increasing number of enforcement and informal proceedings on its docket. This caseload reflects the maturing of the nuclear industry from plant licensing to plant operation, as well as a demand for NRC staff oversight of over 7,000 materials licensees. Informal proceedings, which typically involve materials licenses, rely heavily on the active involvement of a single Presiding Officer in creating and shaping the record of the proceeding. In such proceedings, a hearing is conducted only as to those issues that the Presiding Officer cannot resolve based on the written submissions by the parties, and/or additional information the Presiding Officer has deemed relevant.

Finally, in the case of proceedings before a single administrative judge (for example, enforcement proceedings under 10 C.F.R. Subpart B or informal proceedings under 10 C.F.R. Subpart L), the panel has adopted a policy of assigning a legal or technical administrative judge from the panel as an assistant to the designated Presiding Officer. This step helps preserve the benefits of informal procedure while maintaining the availabiltiy of expertise associated with the traditional three-member licensing boards.

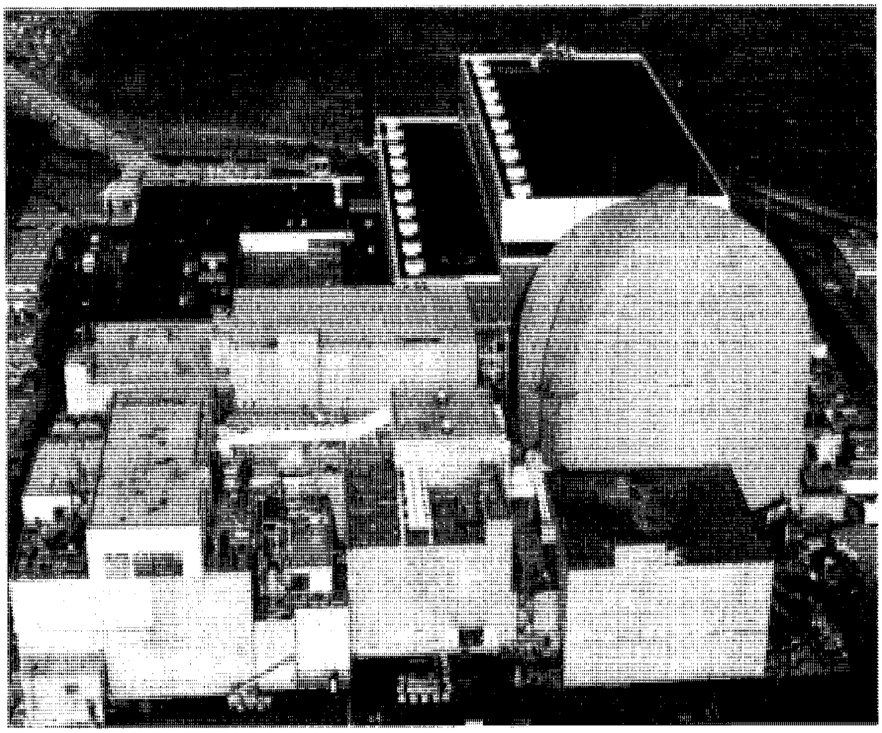
Seabrook Nuclear Power Plant

Challenges to Seabrook Emergency Plan Resolved. During the fiscal year, the Seabrook proceeding, like the Shoreham proceeding in fiscal year 1988, occupied a major portion of the panel's attention, requiring the services of two separate icensing Boards, and generating an extraordinary number of formal issuances.

In LBP-88-32, a 350-page decision focusing on the State-developed "New Hampshire Radiological Emergency Response Plan (NHRERP)"—the plan intended to cover the New Hampshire portion of the Seabrook (N.H.) nuclear power plant's emergency planning zone—the Licensing Board concluded that, subject to several board-imposed conditions, the NHRERP satisfied the Commission's emergency planning requirements. In reaching this conclusion, the board addressed and resolved in the applicants' favor the remaining 26 of 122 contentions originally filed by intervenors, challenging almost every aspect of the NHRERP. The last 26 contentions questioned the adequacy of the plan's supporting letters of agreement, emergency response personnel, transportation resources, decontamination and reception centers, public notification and communication procedures, human behavior assumptions, sheltering procedures for beach populations, and evacuation time estimates. (28 NRC 667 (1988).)

Earthquake. In LBP-89-3, the Licensing Board rejected a late-filed petition to intervene and to admit contentions based on a 1988 earthquake in the vicinity of Quebec, Canada. Because the "Quebec event" was, according to the petitioner, a 6.4 magnitude earthquake, while the Seabrook design-basis "safe shutdown earthquake" (SSE) was only a 6.0 magnitude earthquake, the petition sought intervention for the purpose of revising upward the Seabrook SSE, and to challenge the adequacy of the Seabrook emergency plan in the event of a joint accident/earthquake. To the extent that the proposed contentions sought to litigate the adequacy of an emergency plan in the event of a radiological emergency accompanied by a greater than SSE earthquake, the board noted that the Commission had already held, and had been judicially sustained, that such a scenario need not be considered in the context of emergency planning. As to the petitioners claim that the Seabrook SSE should be revised upward in light of the Quebec earthquake, the board noted that the petitioner had been a party in the Seabrook construction permit stage; had unsuccessfully litigated just such a contention there; and thus was generally precluded, absent "changed circumstances," from re-litigating that issue at the operating license stage. Because the petitioner failed to advance any evidence indicating that the Quebec earthquake was generated in the same tectonic province governing the Seabrook site, the board concluded that no "changed circumstances" related to the Seabrook SSE existed, and thus re-litigation of that issue was barred under the doctrine of *res judicata*. (29 NRC 51 (1989).)

On-site Emergency Plan. In LBP-89-4, the Licensing Board rejected intervenors' attempt, after the record in the proceeding had closed, to litigate contentions challenging the adequacy of applicants' on-site emergency plan for the Seabrook plant, based on an NRC staff evaluation suggesting an inadequate training program. The board first rejected intervenors' petition, filed almost three months after the exercise that gave the basis for the new contentions, as inexcusably late and not otherwise justified by the "compelling" showing on other factors required in 10 CFR §2.714(a)(1). The board went on to conclude that the intervenors had failed to establish the grounds necessary to justify reopening the record. In doing so, the board rejected what it termed "barren allegations" of bad faith on the part of NRC inspectors, holding



The Seabrook plant in New Hampshire was the subject of multiple and complex actions involving separate Licensing Boards, the Appeal Panel and the Commission, during fiscal year 1989. The Public Service Company of New Hampshire received a license to operate Seabrook Unit 1 at low power, in May 1989.

that, in the absence of clear evidence to the contrary, it would presume that NRC inspectors properly discharged their official duties. The board also rejected speculation about the credibility of the applicants' employees as affiants. The board noted that "[o]nly facts raising a significant safety issue, not conjecture or speculation, can support a reopening motion." Finally, the board concluded that the motion to reopen neither addressed a significant safety or environmental issue nor raised a factual triable issue, thereby precluding it from finding that the newly proffered information would likely produce a materially different result. (29 NRC 62 (1989).)

Police Powers. In LBP-89-8, the board granted applicants' motion for summary disposition on two joint intervenors' contentions challenging the legal premise underlying the applicant-developed "Seabrook Plan for the Massachusetts Communities (Seabrook Plan)." The contentions questioned whether, in the event the agencies of the Commonwealth of Massachusetts are unable or unwilling to respond adequately to a radiological emergency, the Governor could lawfully delegate to the applicants' police powers necessary to implement their emergency plan. Turning to the Massachusetts Civil Defense Act, which had been amended to include nuclear accidents among the type of disasters triggering its provisions, the board noted that the statute lodged considerable authority with the Governor to take whatever action is necessary—including cooperating with private agencies in the exercise of police powers—to ensure the public health and safety. That being the case, the board held that "it is not rational to believe that, during [a nuclear accident]...the Governor may not and would not delegate, command, direct, or cooperate with private agencies of whatever nature in the defense of Massachusetts citizens." Accordingly, the Governor or his designee had the lawful authority, in the event of inaction on the part of the State agencies, to delegate to the applicants' Off-site Response Organization those police powers necessary to implement the Seabrook Plan. (29 NRC 193 (1989).)

Bankruptcy. In LBP-89-10, the board rejected arguments that the financial condition of the Seabrook applicants (i.e., the bankruptcy of one co-owner and the default of another in its financial obligations to the Seabrook project) warranted a waiver of the Commission's regulatory exclusion of electric utilities from financial qualification reviews at the operating license stage. While recognizing that the bankruptcy of the utility, Public Service of New Hampshire (PSNH), was probably a "special circumstance" not considered by the Commission when it promulgated the regulatory exclusion, the board affirmed that such a finding was less than sufficient to warrant a waiver of a rule. Summarizing the applicable Commission law, the Licensing Board noted that a party must advance a prima facie case that (1) "special circumstances" exist which (2) undercut the rationale of the rule sought to be waived, and that (3) a waiver is needed to address a significant safety problem. The board pointed out that nothing in the record established that the costs necessary to safely operate the Seabrook power plant, guaranteed under State law regardless of any prior or existing bankruptcy, would not be authorized by the relevant State authority, once the Seabrook facility entered the rate base. That being the case, the rationale which led the Commission to adopt a regulatory exclusion from financial qualifications reviews was still applicable to the Seabrook plant. As to the claim that a financially strapped utility might not operate the plant in a safe manner, the board found the applicants' response persuasive that the NRC's continuous monitoring of its operations, together with the fact that PSNH had but one of five votes on the committee of owners that made safety-related decisions, rendered a degradation of safety a highly unlikely consequence of the PSNH bankruptcy. (29 NRC 297 (1989).)

Mobile Sirens. In LBP-89-17, the Licensing Board rejected challenges of "too loud" and "too slow" to the applicants' proposal to use trucks equipped with sirens mounted on hydraulic telescoping booms to notify Massachusetts residents in the event of an emergency. The use of mobile sirens became necessary because of actions on the part of local governments that made reliance on fixed sirens untenable. While acknowledging that the particular type of siren proposed to be used by the applicants was rated at a decibel level slightly higher than the Commission's maximum permissible level, the board pointed out that a siren's sound level was, like a beam of light, at its peak directly in front of the speaker. Because the proposed sirens were intended to be sounded from a boom extended to 47-51 feet, the sound heard at ground level would be within the Commission's guidance. Moreover, the board held that temporary and/or slight increases in sound level over the maximum were not significant for emergency planning purposes. As to the argument that the use of mobile sirens did not provide reasonable assurance that public notification would be made in a timely manner, the board noted that the Commission's rules did not require absolute guarantees that a siren system would notify all members of the public under every possible circumstance within 15 minutes or less. Having evaluated the evidence offered on each of the functions that together produced the average time needed to initiate a siren alert from all locations, the board concluded that the applicants' public notification system and procedures satisfied the Commission's 15-minute rule. (29 NRC 519 (1989).)

Shoreham Nuclear Power Plant

Final OL Decisions in Shoreham. Two decisions of significance were issued regarding the Shoreham (N.Y.) nuclear power plant prior to the Commission's affirming, in CLI-89-2, 29 NRC 211 (1989), the Licensing Board's dismissal of "government intervenors" from all Shoreham-related proceedings, as a sanction for willful disobedience of board orders. (See LBP-88-24, *Long Island Lighting Co.*, 28 NRC 311 (1988).)

Recusal Requested. In LBP-88-29 (Shoreham), two judges considered and rejected motions to recuse themselves from the proceeding for prejudice. While finding that the petitioners lacked standing to seek their recusal, the two Administrative Judges concluded, nevertheless, that the seriousness of any motion for recusal, and the potential cloud such motions cast on the integrity of a proceeding, justified a response. The judges went on to find that the grounds for recusal—i.e., pervasive bias or prejudice—must flow from extra-judicial sources and cannot be based on the fact that parties disagree with adverse rulings, even where those rulings reflect views strongly held and expressed by a judge. In the instant case, the petitioner failed to advance any grounds warranting recusal. (28 NRC 637 (1988).)

Emergency Exercise Contentions. In LBP-89-1, the Licensing Board granted in part and denied in part intervenors' motion to admit contentions related to the 1988 exercise of the applicant's off-site emergency plan for the Shoreham plant. Applying the ALAB-903 "two-prong" test—requiring that exercise-based contentions demonstrate a fundamental flaw calling for significant change—the board admitted portions of five of 20 proffered contentions. The admitted portions asserted, with sufficient basis and specificity, that the 1988 exercise of the applicant's off-site emergency plan for the Shoreham facility revealed a failure in an essential element of the plan the correction of which required significant revisions in the plan. In reaching this result, the board admitted or denied portions of contentions which, based on materially similar facts, had been previously admitted or denied in earlier litigation concerning a similar 1986 exercise. Because Commission rules required that litigation over exercises must be completed within two years (not possible for the 1986 exercise), the board certified its decision for immediate appeal, in an effort to avoid a continuous circle of litigation. (29 NRC 5 (1989).)

Severe Accident Mitigation Design Alternative

SAMDA Remand Litigation Scope Defined. In LBP-89-19, the Licensing Board defined the kinds of "severe accident mitigation design alternatives," or SAMDAs, that were proper subjects for litigation under the National Environmental Policy Act (NEPA), following judicial rejection of Commission policy generically excluding such issues from licensing proceedings. The board first pointed out that, on remand, the intervenor did not have a right to an entirely new proceeding based on new information. Rather, it had a right to litigate only those aspects of its contention which were originally supported by adequate bases and specificity, and which thus would have been considered before "but for" the Commission's severe accident policy. Those aspects were SAMDAs involving containment heat removal, core residue capture, and venting, supported by the then current NRC-sponsored studies on severe accident mitigation originally invoked by the intervenor. Moreover, while new information about originally identified and supported SAMDAs could be considered, new information identifying potentially different SAMDAs could not. Contentions on newly identified SAMDAs would have to satisfy the Commission's "late-filed contentions" test. Finally, the Licensing Board concluded that in considering severe accident mitigation issues, only matters focused on reducing the consequence of a severe accident, and not matters focusing on reducing the probability of such accidents (e.g., training), were proper. *(Philadelphia Electric Co.* (Limerick), 29 NRC 55 (1989).)

TMI Accident-Generated Water

NEPA Burdens Clarified. In LBP-89-7, the Licensing Board approved a proposal by the licensee for the Three Mile Island (TMI) facility in Pennsylvania to evaporate a volume of radioactive water left from the 1979 accident at TMI and subsequent cleanup. The Licensing Board first noted that, under the National Environmental Policy Act (NEPA), it was bound to approve the applicants' proposal unless an alternative was found which was obviously superior. However, once an intervenor has satisfied its burden to propose an alternative, the burden shifts to the applicant to establish by a preponderance of the evidence that the intervenor's alternative is not obviously superior. In this particular case, the board found that the intervenors had failed to advance any specific alternative to the licensee's proposal and thus were to be viewed as generally arguing in favor of a "no action" alternative, i.e., indefinite on-site storage. Based on a detailed examination of the scientific and health considerations, including competing risks and uncertainties attendant on each proposal, the board concluded that while some dose-saving through radioactive decay would result after 30 years under the "no-action" alternative, the minor benefit of that was outweighed by its estimated cost of over $800,000. In these circumstances, the intervenors' alternative could not be judged obviously superior to the licensee's proposal. *(General Public Utilities Nuclear Corp.*, 29 NRC 138 (1989).)

Informal Proceedings

Standing Rules Explained. In LBP-89-23, the Presiding Officer clarified the manner in which the Commission's rules on standing, developed in the context of formal proceedings, should be applied in informal proceedings. While recognizing that the Commission had modified its traditional presumptions on standing for application in informal proceedings, the Presiding Officer concluded that the Commission did not intend for those modifications to make it more difficult for a person to intervene in an informal proceeding than in a formal proceeding. Rather, the Commission intended standing to be determined in a flexible manner, consistent with the informal nature of the proceeding, taking account of the reasonably expectable amount of information available locally to assist a person in drafting a petition to intervene. In an informal proceeding, where the amount and scope of publicly and locally available information is less than that available at the start of a formal proceeding, the Presiding Officer held that standing can be shown by a concise statement of how one's interests may plausibly be affected by concerns germane to the proceeding (i.e., concerns falling generally within the range of matters that are subject to challenge in the proceeding). The merits, if any, of the identified concerns are separate questions, to be resolved later in the proceeding. *(Combustion Engineering, Inc.*, 29 NRC 140 (1989).)

Equal Access To Justice Act: Attorneys' Fees

Board Retains Jurisdiction and Authority Decided. In LBP-89-11, the Licensing Board held that it retained authority to issue a declaratory judgment notwithstanding a conditional revocation of an immediately effective license suspension order, giving rise to the proceeding. The board went on to hold, in a case of first impression, that it had authority under the Equal Access to Justice Act (EAJA) to award attorneys' fees, in appropriate cases, to licensees who were successful in whole or in part in defending against proposed staff enforcement orders.

While recognizing that the issuance of a declaratory judgment was dependent upon the existence of a live controversy, the board rejected the argument that the conditional revocation of the underlying suspension order rendered the case moot. In the board's view, the staff's power to directly affect the actions and economic viability of licensees through enforcement orders, the continuing controversy over the facts supporting the original suspension order, and the likelihood that the particular licensee involved could be subject to the same action in the future operated in unison to place this case within the ambit of the exception permitting review where an injury is ''capable of repetition, yet evading review.''

As to attorneys' fees, the board found that the EAJA applied generally to the Commission and specifically to enforcement proceedings. It went on to reason that since licensees who challenge staff enforcement orders participate before Licensing Boards as *petitioners*, Congressional restrictions on assistance to *intervenors* do not limit the board's otherwise proper authority under the EAJA to award attorneys' fees. (*Advanced Medical Systems, Inc.*, 29 NRC 306 (1989).)

Drug Use: Operator's License

License Revocation Sustained. In LBP-89-26, the Licensing Board sustained, in a case of first impression, NRC staff action initially suspending and ultimately refusing to renew a reactor operator's license because of a pattern of off-duty marijuana use. The board first found that the duties of a reactor operator are complex and require the continuous exercise of clear judgment. That being the case, the board concluded that any impairment of that judgment constitutes a threat to public health and safety. Based on the testimony of medical experts, the board acknowledged that while the routine duties of a reactor operator might not be affected at all by some levels of marijuana, an operator's performance of his or her duties in complex situations would be unpredictable. Because the Commission must have reasonable assurance that an operator can competently and safely operate a nuclear power reactor in both routine and non-routine situations, unpredictability in non-routine situations (e.g., accident scenarios) as a result of marijuana use justifies the suspension and/or revocation an operator's license. The board also turned away challenges to the specific tests used by the licensee to determine marijuana use, claims of passive inhalation of marijuana smoke, and urine sample chain-of-custody issues. (*In re Maurice P. Acosta, Jr.*, 29 NRC 195 (1989).)

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

An Atomic Safety and Licensing Appeal Board consists of three members—drawn from the Atomic Safety and Licensing Appeal Panel—who may be either legal or technical experts; both kinds are represented on each board, usually by one legal and two technical members. (See Appendix 2 for the membership of the Appeal Panel.) On behalf of the Commission, the Appeal Boards review decisions issued by Atomic Safety and Licensing Boards, presiding officers, and administrative law judges in a wide range of formal and informal adjudicatory proceedings. These proceedings concern both licensing and enforcement actions, involving commercial nuclear power reactors and other facilities that use or possess byproduct, source, or special nuclear material. An Appeal Board's decision becomes the final agency order unless the Commission, in its discretion, decides to review it. In the absence of any Commission action, the decision is the agency's final position, though it may be subject to judicial review in a Federal court of appeals. The more significant Appeal Board decisions are published in the permanent collection of NRC licensing and other decisions, titled Nuclear Regulatory Commission Issuances.

Fiscal year 1989 again saw a major portion of Appeal Board attention devoted to the proceedings on the operating license applications for the Seabrook (N.H.) and Shoreham (N.Y.) nuclear power plants. These two proceedings alone accounted for 16 published Appeal Board decisions.

Seabrook Nuclear Power Plant

Most of the matters coming before the Appeal Board in connection with the operating license application for the Seabrook facility involved aspects of the emergency response plan for persons located within the 10-mile Emergency Planning Zone (EPZ), the postulated ''plume exposure'' pathway surrounding Seabrook. The board was, however, also called upon to deal with a remaining technical issue, one which pertained to the environmental qualification of certain coaxial cable used in the facility. In one decision, the Appeal Board affirmed a Licensing Board determination that the coaxial cable issue need not be resolved prior to facility operation at levels at or below 5 percent of rated power. In reaching this judgment, the Appeal Board rejected the argument of the appellant, the

New England Coalition on Nuclear Pollution, that the Commission lacks the authority to permit low-power operation of a facility in advance of the resolution of all contested safety issues. Subsequently, on the merits of the coaxial cable issue, the Appeal Board reviewed and affirmed the Licensing Board's grant of summary disposition in the utility applicants' favor.

On the emergency planning front, an earthquake occurring in the Province of Quebec, Canada, prompted an individual's effort to intervene in the *Seabrook* proceeding, in order to litigate the significance of that earthquake in the emergency response plan for the facility. The Appeal Board affirmed the denial of the intervention petition on the ground, among others, that it did not provide any basis for believing that the Quebec earthquake might have safety significance insofar as Seabrook operation and emergency response planning were concerned.

Another Seabrook matter came to the Appeal Board as the result of a Licensing Board action expunging, because of an asserted lack of jurisdiction, a portion of a previously admitted contention of the intervenor Massachusetts Attorney General. In the portion expunged, the intervenor contended that a June 1988 exercise of emergency response plans for the Seabrook facility had revealed that the computer model used to develop protective action recommendations contained fundamental flaws. Deciding that there was reason to review the action on an interlocutory basis, the Appeal Board went on to decide that the Licensing Board's jurisdictional ruling was in error. Accordingly, the Appeal Board ordered the reinstatement of the expunged portion of the contention.

Still further in this area, the Appeal Board affirmed the denial of a joint motion filed by several Seabrook intervenors seeking to admit a new emergency preparedness exercise contention or, in the alternative, to reopen the record. The Appeal Board found that the long established standards for the grant of such relief had not been met.

Yet another controversy arose from the Licensing Board's partial disposition of an issue involving evacuation time estimates as affected by commuters who might return *to* the Seabrook EPZ during an emergency. The threshold question confronting the Appeal Board was the timeliness of the Massachusetts Attorney General's appeal from that disposition. The Appeal Board dismissed the appeal as premature, concluding that the Attorney General would have to defer his challenge to the Licensing Board's partial ruling until that board disposed of the remainder of the so-called "returning commuter" issue. Similarly, the Appeal Board dismissed as premature an appeal from a Licensing Board ruling that had denied the Attorney General's motion to supplement an issue already in contest, involving the public notification system at Seabrook.

The bankruptcy of the Public Service Company of New Hampshire (the lead Seabrook applicant) led the Massachusetts Attorney General and a second intervenor, the Seacoast Anti-Pollution League (SAPL), to seek once again to litigate the question of the Seabrook owners' financial qualifications to operate the facility in a safe manner. This time the issue arose in connection with consideration of the issuance of a *full-power* license. (Earlier, as discussed in the *1988 NRC Annual Report*, p. 181, the intervenors had sought to litigate the issue of financial qualifications with regard to *low-power* operation.) The Attorney General requested a waiver of the provisions of the Commission's 1984 rule exempting an applicant for an operating license, for a facility such as Seabrook, from the general requirement that it demonstrate the financial qualifications to conduct safe operation. SAPL maintained that the waiver sought had already been implicitly granted by the Commission, in its 1988 decision (CLI-88-10) concerned with financial qualifications in the context of a low-power testing authorization. (See discussion of *Public Service Company Of New Hampshire*, under "Commission Decisions, below.) On appeals from the Licensing Board's denial of all relief, the Appeal Board rejected the SAPL claim but concluded that the Attorney General had made a *prima facie* case of entitlement to a waiver. On the basis of that conclusion, and as required in such circumstances by the Commission's Rules of Practice, the Appeal Board certified the Attorney General's waiver petition to the Commission for an ultimate determination as to whether a waiver should be granted.

Shoreham Nuclear Power Plant

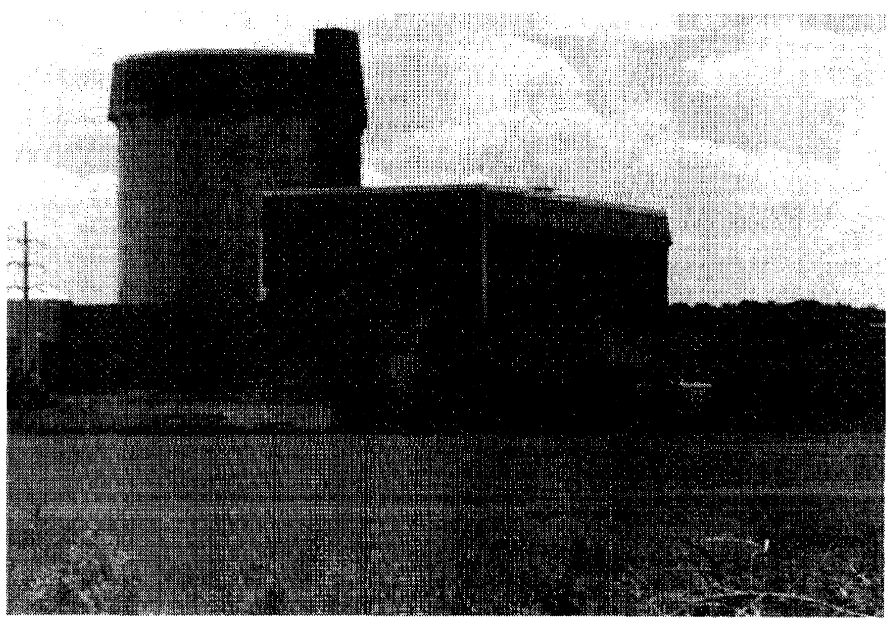
The protracted proceedings involving the Shoreham Unit 1 facility were brought to a close in 1989, but not before the resolution of a number of appeals involving significant emergency planning issues. These issues surfaced in the context of vigorous opposition from State and local governments to the operation of the Shoreham plant. Following the refusal of these governments (namely, the State of New York, Suffolk County, and the Town of Southampton) to participate in emergency planning, the applicant Long Island Lighting Company (LILCO) developed its own emergency plan, using LILCO employees and outside support organizations. Thus it was that the litigation in the concluding stages of this proceeding focused primarily on the adequacy of that ILCO emergency plan and two exercises conducted to test the plan's implementation.

In one instance—a LILCO appeal from a Licensing Board ruling—the Appeal Board was asked to interpret and apply certain guidance provided earlier by the Commission. In 1986, the Commission ruled that hearings on the results of an emergency planning exercise should be limited to issues concerned with whether the exercise revealed any "fundamental flaw" in the emergency plan. (See *1986 NRC Annual Report*, p. 200.) The Appeal Board elaborated on this standard and determined that a fundamental flaw, as disclosed by an exercise of the emergency plan, has two principal and necessary components: first, it reflects a failure of an essential element of the plan, and, second, it can be remedied only through a significant revision of the plan. Whether an essential element is involved is to be ascertained by reference to the basic emergency planning standards and other requirements specified in the Commission's regulations. The Appeal Board also ruled that, although minor or isolated problems occurring on the day of the exercise would not constitute fundamental flaws in the plan, some deficiencies may be considered collectively as tantamount to a fundamental flaw if they are pervasive or indicative of a pattern of repeated failures of an essential plan element.

In another appeal, this time by the intervening State and local governments, the principal issue before the Appeal Board was the suitability of three reception centers designated by LILCO for monitoring, decontaminating, and sheltering evacuees in the event of a radiological emergency at Shoreham. The Appeal Board concluded that the monitoring planning basis upheld by the icensing Board—i.e., 12-hour monitoring capacity for 20 percent of the EPZ population—was not adequately supported by the evidence of record. It therefore vacated this part of the Licensing Board's decision and remanded the matter to that board for further consideration and record development. In the same decision, it also directed the Licensing Board to consider the effect on the reception center issue of a recent New York State court judgment, enjoining the use of one of LILCO's proposed receptions centers.

Charging bias against them, the intervening governments asked two Licensing Board judges assigned to the *Shoreham* proceeding to recuse themselves from further participation in the matter. The judges declined and, in accordance with Commission procedure, they referred their decision to the Appeal Board. Relying on well established precedent, the Appeal Board concluded that the strong language used by the Licensing Board judges in an earlier decision imposing sanctions on the governments and dismissing them from the proceeding was neither attributable to extrajudicial conduct nor pervasive. Consequently, the Appeal Board affirmed the Licensing Board judges' decision not to step down.

Having encountered difficulties in obtaining a full-power operating license as a result of certain unresolved emergency planning issues, LILCO sought interim authorization from the Licensing Board to operate at 25 percent power. Because the same Licensing Board had earlier dismissed the governments from the proceeding as a sanction for their failure to comply with discovery orders, that board determined that LILCO's 25 percent power request was unopposed and accordingly granted it. Upon challenge by the governments, the Appeal Board concluded that the Licensing



Protracted proceedings involving the Shoreham facility on Long Island, N.Y., continued during fiscal year 1989. A number of appeals related to significant emergency planning issues engaged both the Appeals Panel and, ultimately, the Commission. This view of the plant is from the southwest.

Board's action did not comply with the Commission's regulations. Specifically, the Appeal Board noted that, while the governments had been dismissed from a part of the *Shoreham* proceeding, they were still viable litigants in another portion pending before a different Licensing Board. Thus, it was the Appeal Board's view that LILCO's 25 percent power request was not, in fact, unopposed, and that therefore it was incumbent upon the Licensing Board to take into account any contested issues still pending elsewhere before other boards. Acknowledging that there was a novel issue at stake, with complex procedural circumstances, the Appeal Board certified its own ruling to the Commission for final review.

The Commission itself directly reviewed the Licensing Board's September 1988 decision dismissing the intervening governments from the proceeding as a sanction for certain conduct (see *1988 NRC Annual Report*, pp. 177, 181), and it ultimately upheld their dismissal (see discussion under "Commission Decisions," below). The Commission had previously directed the Appeal Board, however, to resolve on the merits the governments' remaining appeals on two issues—the adequacy of the emergency broadcast system (EBS), and "role abandonment" by school-bus drivers during a radiological emergency. Once the Commission had dismissed the governments from the entirety of the proceeding, the Appeal Board necessarily dismissed the governments' appeals on these two issues and terminated several other remaining emergency-exercise appeals. Nonetheless, under a long established, Commission-endorsed prerogative—by which the Appeal Board reviews on its own a final Licensing Board disposition of significant safety or environmental issues—the Appeal Board reviewed the Licensing Board's resolution of the Shoreham EBS and school-bus driver issues. The Appeal Board concluded that the State-established, multi-station EBS network on Long Island provides adequate emergency broadcast coverage, and it therefore affirmed the Licensing Board's disposition of this issue in LILCO's favor. As for the school-bus driver issue, the Appeal Board found evidence in the record that some role abandonment was a real possibility but that it could not be quantified with any certainty. The board thus affirmed the Licensing Board's ultimate decision, subject to a modification: using LILCO's "150 percent planning assumption" for its own drivers, the board found that the number of drivers LILCO would need to supply as back-up to the drivers employed by the school districts and their contractors was calculable.

Other Noteworthy Proceedings

Litigation resulting from the accident 10 years ago at the Three Mile Island Unit 2 (Pa.) facility continued in 1989. The Appeal Board was asked to stay an operating license amendment authorized by the Licensing Board permitting the operator of that facility, General Public Utilities Nuclear Corporation, to dispose of some 2.3 million gallons of contaminated water, generated during the March 1979 accident. The water was to be processed first in order to reduce its radioactive content, followed by forced evaporation and subsequent release to the atmosphere over a period of 15-to-24 months. The solid residue left after evaporation would then be shipped to a low-level waste disposal site. The Licensing Board found that this procedure would result in extremely small radiation exposure to both plant workers and the general public, with correspondingly negligible or non-existent health consequences. The Appeal Board denied the joint intervenors' motion to stay the decision, concluding that they had failed to show any irreparable harm. The merits of the intervenors' appeal, however, remained pending before the Appeal Board at the close of the report period.

Two other operating license amendment proceedings before the Appeal Board involved applications by nuclear power plant licensees to permit expansion of the capacity of their spent fuel pools by means of "re-racking," or rearranging the spent fuel elements in the pool.

At issue in the *Vermont Yankee* proceeding was an intervenor's claim that an environmental impact statement was required to consider a purportedly increased risk of fire in the zircaloy cladding surrounding the fuel elements stored in the spent fuel pool. According to the intervenor, such a fire could be triggered by a severe reactor accident that could lead to containment failure and hydrogen detonation in the reactor building, where the Vermont Yankee spent fuel pool is located. The Licensing Board ruled that such a contention was appropriate for litigation but referred its ruling to the Appeal Board for review. The Appeal Board reversed, concluding that the technical documents on which the intervenor's contention relied did not provide a basis for the sequential occurrence of the two "worst case" accidents, of very low probability, hypothesized by the intervenor. Referring to the "rule of reason" applicable to environmental issues, the Appeal Board determined that the intervenor's contention could properly be rejected at the threshold and need not be litigated.

In another spent fuel pool expansion proceeding, the Appeal Board affirmed the Licensing Board's decision approving re-racking of the pool at the St. Lucie Unit 1 (Fla.) facility. The principal issue in the intervenor's appeal concerned the effects of heat and radiation on certain material (Boraflex) used in the racks. The

Appeal Board found that the intervenor had presented no facts or argument to undercut the expert witnesses' testimony and analyses showing the ability of the racks to withstand the conditions expected in the spent fuel pool.

New Member Appointed to Appeal Panel

The Commission appointed G. Paul Bollwerk, III, as the newest member of the Appeal Panel, effective July 9, 1989. Judge Bollwerk was formerly a senior attorney in the NRC Office of the General Counsel.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1989 are discussed below. Commission action on export licensing is treated in Chapter 7.

Seabrook Nuclear Power Plant

During fiscal year 1989, the Commission issued several decisions on the question of whether a license to conduct low-power operation for testing purposes should be granted the operator of the Seabrook (N.H.) nuclear power plant.

In *Public Service Company Of New Hampshire* (Seabrook Station Units 1 and 2), CLI-88-10, 28 NRC 573 (1988), the Commission decided all of the pending financial qualification questions in Seabrook. These questions included certain matters "of first impression," presented in unprecedented factual circumstances. The Commission chose a course that would protect the health and safety of the public and allow the complicated litigation over financial qualifications for low-power testing to be brought to a close.

The Commission required applicants to provide reasonable assurance that $72.1 million in funds were available for decommissioning before licensing for low-power testing. Financial assurance for this amount in the form of a pre-paid external account, surety, or other guarantee method was acceptable to the Commission. However, the Commission also accepted the applicant's plan to fund, before receipt of a license for low-power testing, a separate and segregated account held by its Disbursing Agent, provided that the amount was $72.1 million (rather than the $21.1 million suggested by the applicants) and that at least two of the applicants, among those whose financial health had not been called into question and who owned substantial shares of Seabrook, should jointly and severally guarantee to make up any deficiency in the fund caused by disbursements for a non-decommissioning expense.

The Commission also ruled on petitions to waive its 1984 financial qualifications rule and consequently require a financial qualifications review and finding before low-power licensing. The Commission found that with decommissioning expenses reasonably assured, there were no remaining significant financial safety problems that needed to be addressed. Since a rule waiver was not needed to resolve any significant safety problem, the waiver petition was denied.

Finally, the Commission ruled that a low-power testing license could be issued after the applicants had satisfied Staff that all the decommissioning terms of this decision had been met, subject to these qualifications: (1) Unit 1 should not exceed 5 percent of full-power levels, nor exceed 0.75 percent effective full-power hours of such operation without additional Commission approval; (2) a pending motion to litigate additional on-site emergency planning issues and any litigation on additional on-site issues before the Licensing Board shall have been resolved; and (3) 10 days have elapsed from the date of completion of the later of the two foregoing requirements.

In *Public Service Company Of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-89-07, 29 NRC 395 (1989), the Commission denied a reconsideration of CLI-88-10, on grounds that intervenors had fundamentally misperceived the purpose and nature of the decommissioning *funding* requirements and thus had failed to make a case for reconsideration. Intervenors had asked that the Commission remand the issue of low-level waste generation and disposal to the Licensing Board for litigation based on factual allegations of the unavailability of low-level waste disposal sites. Intervenors claimed that denials of access to low-level-waste regional disposal facilities at Barnwell, S. C., and Richland, Wash., along with an expected denial from Beatty, Nev., ensured that low-level waste generated by low-power operation at Seabrook could not be shipped off-site. Intervenors failed to assert what the projected increase in the costs of lengthier on-site storage would be but said they must be explored. The Commission disagreed.

The Commission saw no need to alter its decision in CLI-88-10, even in the event that all three waste disposal sites were barred to Seabrook and the state of New Hampshire did not move to meet its obligations under the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPA). The Commission held that no demonstration had been made to cause it to believe that the sum it ordered to be set aside in CLI-88-10, which included a contingency in excess of $14 million, was inadequate to provide the requisite assurance for the limited additional potential costs of continued on-site storage for the term of years until the state of New Hampshire itself becomes responsible for the waste, pursuant to Federal law.

In *Public Service Company Of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-89-08 (1989), the Commission had before it three separate motions seeking to stay authorization to conduct low-power testing at Seabrook. The Commission denied the motions after analyzing the four factors relevant to consideration of stay motions. The analysis did not favor a stay. The Commission found that intervenors' claims of harm did not meet the standards of irreparable harm, and intervenors did not demonstrate how the irreversible effects of irradiating the reactor constituted harm to them. The Commission found further that intervenors did not make a strong showing that they were likely to prevail on the merits, in that (1) intervenors erred in interpreting the Atomic Energy Act to bar *any* operation of a nuclear reactor until *all* issues material to the issuance of a full-power license are decided; (2) low-power operation was not a new circumstance, or a separate Federal action, either of which could require further Environmental Impact Statement analysis under the National Environmental Protection Act (NEPA); (3) delay of corrective measures involving three items of the Safety Parameter Display System until as late as the first refueling outage would not result in a lack of reasonable assurance of public health and safety. The Commission also found that delay would harm the applicants and would not serve the public interest.

Finally, in *Public Service Company Of New Hampshire* (Seabrook Station, Units 1 and 2), CLI-89-19, ______ NRC ______ (September 15, 1989), the Commission denied applicants' request for an exemption from the requirement to conduct an exercise of the licensees' onsite emergency plans within one year before issuance of a full-power operating license. The Commission noted that the Atomic Safety and Licensing Board set November 30, 1989, as its target date for issuance of a decision that will decide all remaining currently admitted contentions, including contentions regarding the last full participation emergency exercise, conducted in June 1988.

The Commission's regulation from which the applicants' sought relief requires that, if more than a year has passed since since the required exercise (as was the case in this instance), a new exercise must be held. Intervenors stated they wanted an opportunity to litigate the results of this second exercise. Applicants were concerned that protracted litigation could delay the issuance of the license, eventuating in a need for a third exercise in June 1990, and beginning an endless loop of litigation.

The Commission held that the applicants' assertions were speculative. The Commission also had reservations whether *indirect* costs—such as costs attendant on ensuing litigation or costs flowing from delays caused by litigation—were properly considered in evaluating an exemption request, because the very grounds that would most support the need for an adjudication would be those most likely to cause delays.

The Commission also held that generalized projections of need for power were outweighed by the public interest which underlies the safety provisions of the Commission's emergency planning rules.

Limerick Nuclear Power Plant

In fiscal year 1989, the Commission granted licensing authorization for both low-power and full-power operation of the Limerick Unit 2 (Pa.) nuclear power plant.

In *Limerick Ecology Action v. NRC*, 869 F.2d 719 (3d Cir. 1989), the United States Court of Appeals for the Third Circuit declared that, as part of its National Environmental Policy Act (NEPA) responsibilities, the Commission must give consideration to certain "Severe Accident Mitigation Design Alternatives" (SAMDAs) and remanded the case to the agency. The court also ruled that the Commission must consider an emergency planning contention submitted by inmates of the Graterford Correctional Institute. The Commission directed the Licensing Board to initiate proceedings on both issues.

While these proceedings were underway, applicant Philadelphia Electric Company (PECO) petitioned the Commission for clarification of the licensing status of Unit 2. PECO requested that the Commission authorize the NRC staff to grant low-power and full-power licenses pending completion of the adjudicatory proceeding mandated in this matter. (See "Significant Judicial Decisions," below.)

In *Philadelphia Electric Company* (Limerick Generating Station, Units 1 and 2), CLI-89-10, ______ NRC ______ (1989), the Commission held that the *Limerick Ecology Action* (LEA) decision did not preclude Commission authorization of a low-power license, because the issues on remand were not relevant to low-power operation. The Commission held that the Licensing Board's earlier full-power authorization and the existence of a final environmental impact statement adequately supported issuance of a low-power license, without Commission review, once the necessary NRC staff findings had been made, pursuant to 10 CFR 50.57. The Commission deferred ruling upon full-power operation until it conducted its immediate effectiveness review, in accordance with 10 CFR 2.764(f)(2).

Licensing of the Limerick nuclear power plant—which received both a low-power and full-power operating license during fiscal year 1989—entailed a number of Commission and judicial decisions, discussed here and under "Significant Judicial Decisions," below. In the photo, NRC Resident Inspector for the Limerick facility, Michele Evans, is shown in conversation with a utility employee in the control room.

After that immediate effectiveness review—in *Philadelphia Electric Company* (Limerick Generating Station, Units 1 and 2), CLI-89-17, _____ NRC _____ (1989)—the Commission authorized the staff, once it made the requisite findings under 10 CFR 50.57, to issue a full-power license for Unit 2, subject to amendment as a result of the SAMDA proceeding. The Commission found that the balance of factors set forth in 10 CFR 2.764(f)(2)(i), as well as consideration of environmental matters under NEPA, favored commencement at that time over a delay until after the Licensing Board SAMDA proceeding.

Intervenor Limerick Ecology Action claimed that the Licensing Board hearing must be completed before the Commission can authorize a license, and that the Commission must allow discovery and full adjudicatory hearings before it could make a NEPA determination. In rejecting these claims, the Commission explained that, in accordance with 10 CFR 2.764(f)(2), it reviews all Licensing Board decisions authorizing issuance of full-power operating licenses. That review does not give rise to hearing rights, nor does the Atomic Energy Act prevent a licensing decision from becoming effective prior to appellate review on the merits *(Oystershell Alliance v. NRC,* 800 F.2d 1201, 1206 (D.C. Cir. 1986)). The Commission also noted that NEPA itself does not always require resolution of all contested environmental issues and completion of the entire NEPA review process before the license can issue. (See 40 CFR 1506.1).

After seeking information from the parties, the Commission analyzed the potential environmental impacts of permitting full-power operation before, rather than after, conclusion of the Licensing Board's SAMDA proceeding. It examined five areas: (1) potential increase in occupational exposure from installing SAMDAs after operation had begun; (2) increased environmental effects from risk of severe accidents without SAMDAs in place; (3) environmental effect of nuclear versus non-nuclear generation; (4) whether installation of SAMDAs would be precluded once the plant was generating electricity; and (5) the cost of delay. It found that the potential increase in occupational radiation exposure from installation of SAMDAs after full-power operation had begun would be relatively small, as would be the potential negative environmental effects.

The Commission next evaluated the environmental effects of the generation of electrical energy equivalent to one fuel cycle's full-power operation of Unit 2, in terms of mortality and morbidity for both plant workers and the general public, comparing nuclear generation with non-nuclear generation. It found that nuclear generation had the lesser environmental impact. The Commission also found that the operation of Unit 2 would not make installation of the SAMDAs under consideration physically impossible, and that the only difficulties expected concerned matters of increased cost and increased occupational radiation exposure. It found that the occupational dose associated with installing any of the SAMDAs was the same as or less than that associated with typical maintenance activities.

Finally, the Commission analyzed the increased dollar cost of delay were the issuance of the full-power license deferred until the completion of the SAMDA litigation. Using intervenor LEA's minimal assertions of a 7½ month delay, and taking into account seasonal differences in power use, the Commission found that the delay would cost PECO, its shareholders and ratepayers between $100 million and $271 million.

The Commission concluded that both the public interest and relative environmental impacts favored the issuance of a full-power license.

Shoreham Nuclear Power Plant

In *Long Island Lighting Company* (Shoreham Nuclear Power Station Unit 1, CLI-89-2, 29 NRC 211 (1989)), the Commission held that intervenors' willful defiance of Licensing Board orders caused great harm and delay to applicant's efforts to demonstrate the sufficiency of its emergency plan and to the integrity of the Commission's adjudicatory process. Accordingly, in view of all the circumstances, the Commission dismissed Suffolk County, the State of New York, and the Town of Southampton as parties from all pending proceedings.

In its *Statement of Policy on Conduct of Licensing roceedings*, 13 NRC 452 (1981), the Commission established a graduated scale of sanctions to be employed when necessary in licensing proceedings, including—in the event of a participant's severe failure to meet its obligations—dismissal from the proceeding. It identified the following factors to be considered in deciding what sanction to impose: "the relative importance of the unmet obligation, its potential for harm to the other parties or the orderly conduct of the proceeding, whether its occurrence is an isolated incident or part of a pattern of behavior, the importance of the safety or environmental concerns raised by the party, and all the circumstances" (13 NRC at 454).

The Commission found that the continuing failure of Suffolk County to produce, in response to discovery requests, an emergency plan, despite a declared intention to do so dating back to 1983, and the County's announcement in June 1988 that it would no longer comply with the Licensing Board's discovery orders, meant that the hearing was one in which one party controlled the information to be disclosed. That being the case, the proceeding had become so unfair and biased as hardly to amount to a hearing at all. The Commission held that the obstructionist tactics of various governmental entities and the County's refusal to comply with discovery obligations, as ordered by the board, were patently unfair to the applicant and had effectively halted the proceeding in its tracks.

In determining whether sanctions should be imposed on the governments, the Commission noted that the record contained ample demonstration that the governments had engaged in a pattern of resistance to Licensing Board orders and authority. Taking all the circumstances into account, the Commission determined that the only sanction that would mitigate the harm caused by the obstructions, and would improve compliance both in this case and future cases, was dismissal of the obstructing parties from the entire proceeding.

JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1989 is summarized below.

Pending Cases

American Mining Congress v. NRC, No. 88-1040 (10th Cir.).

Quivira Mining Company, et al. v. NRC, No. 88-1041 (10th Cir.).

Environmental Defense Fund, et al. v. NRC, No. 88-1001 (10th Cir.).

These actions challenge the Commission's November 1987 amendments to its uranium mill tailings regulations bringing NRC's requirements into conformity with standards set by the Environmental Protection Agency. The industry petitioners argue that the NRC was obliged to re-evaluate EPA's standards, before conforming with them, in two respects: (1) whether they are supported by a cost-benefit analysis independently performed by the NRC; and (2) whether they are improperly based on EPA's regulations for low-volume, high-toxicity chemical waste, set out under Subtitle C of the Solid Waste Disposal Act (SWDA), rather than—as the industry petitioners contend they properly should be—based on regulations for high-volume, low-toxicity mining wastes yet to be proposed by EPA under Subtitle D of SWDA. The environmental petitioners assert that NRC's amended regulations do not fully conform to EPA's standards because the NRC did not adopt EPA's standards governing "point of compliance" and "detection monitoring systems," and that the NRC failed to conduct the rulemaking allegedly required by Section 84a(3) of the Atomic Energy Act. Oral argument was held in these cases on September 25, 1989. No decisions had been issued by the close of the report period.

Significant Judicial Decisions

Florida Power & Light Co. v. United States, 846 F.2d 765 (D.C. Cir. 1988), *certiorari denied*, 109 S. Ct. 1952 (1989).

The United States Court of Appeals for the District of Columbia affirmed the Commission's 1986 rule which charged nuclear power facilities approximately $1,000,000-per-year as a "user fee" under the Consolidated Omnibus Budget Reconciliation Act (COBRA).

The majority accepted every argument the NRC raised in defense of the rule. It deferred to the NRC interpretation of COBRA; it found compliance with the APA's notice-of-comment requirements; and it upheld the constitutionality of the Act as being a lawful delegation of authority from Congress to the Commission.

The Supreme Court denied the utilities' petitions for a writ of *certiorari* on May 1, 1989 (109 S. Ct. 1952).

Martin v. NRC, Nos. 85-3444 and 87-3190 (3d Cir.).
Limerick Ecology Action, Inc. v. NRC, Nos. 85-3431 and 86-3314 (3d Cir.).
Anthony v. NRC, No. 85-3606 (3d Cir.).
Limerick Ecology Action, Inc. v, NRC, No. 87-3508 (3d Cir.).
Martin v. NRC, No. 87-3565 (3d Cir.).

These seven consolidated cases challenged various orders issued by the NRC in the completed Limerick operating license proceeding. Limerick Ecology Action (LEA) argued that the Commission violated the National Environmental Policy Act (NEPA) when it refused to consider certain matters in the Limerick Environmental Impact Statement (EIS). In particular, LEA argued that the EIS should have contained a discussion of (1) how severe accident mitigation devices could have reduced the environmental consequences of a severe accident at Limerick; (2) the possible environmental consequences of sabotage; and (3) the socio-economic effects of a severe accident beyond the one year cut-off selected in the EIS. Thomas Martin, an inmate at the Graterford Prison (which is within the 10-mile emergency planning zone for Limerick), also argued that the Commission erred in rejecting a number of his contentions relating to emergency planning for the Graterford Prison.

The court's lengthy opinion, issued one year after oral argument in the case, found merit in two of the attacks raised by petitioners. First, it agreed with LEA that the Commission violated NEPA by not discussing in the Limerick EIS certain severe accident mitigation design alternatives. Secondly, it agreed with Martin that the Commission erred in refusing to let him litigate whether the bus drivers that would evacuate Graterford would *receive* adequate training; the Commission had held Martin to the letter of his contention, which was that the bus drivers would not be *offered* adequate training.

The Commission sought rehearing and rehearing *en banc*, which was summarily denied by the court. In consultation with the Department of Justice, the Commission decided not to seek Supreme Court review.

On April 14 and May 5, 1989, the Commission issued orders instructing the Chairman of the Licensing Board Panel to convene Licensing Boards to conduct further proceedings on the Graterford contention and the SAMDA issue respectively. Those proceedings were completed by agreement of the parties on August 11 and August 30, 1989. A full-power operating license for Limerick Unit 2 was issued on August 30, 1989.

NRC v. Federal Labor Relations Authority, 859 F.2d 302 (4th Cir. 1988), *enforcement denied*, 879 F.2d 1225 (4th Cir. 1989).

The NRC filed an appeal in the Fourth Circuit seeking to overturn a decision of the Federal Labor Relations Authority which held that the agency must negotiate with the employee union concerning pay increases for "bargaining unit" employees. A three-judge panel of the Fourth Circuit rejected the agency's challenge in an opinion dated October 5, 1988. The Department of Justice, with NRC support, filed a motion for reconsideration by the entire Fourth Circuit, *en banc*. The motion was granted. In its July 14, 1989 opinion, the Fourth Circuit reversed the decision of the panel. It held *en banc* that, as a general matter, Federal employee pay may not be a subject of negotiation between agencies and employee unions. The court also held that the authority of the NRC under Section 161(d) of the Atomic Energy Act, as amended, to depart from government-wide pay scales, if deemed necessary to accomplish the NRC mission, is within the exclusive discretion of the agency and may not be negotiated. Finally, the court found that requiring an agency to negotiate employee pay would be an improper interference with the exclusive statutory right of an agency to determine its budget.

Quivira Mining Company, et al. v. NRC, 866 F.2d 1246 (10th Cir. 1989).
Environmental Defense Fund, et al. v. NRC, 866 F.2d 1263 (10th Cir. 1989).

On January 27, 1989, the Tenth Circuit issued decisions in these two cases upholding the NRC's uranium mill tailings regulations, 10 CFR Part 40, Appendix A, as amended in October 1985 (50 FR 41852 (1985)), finding them in conformity with general standards promulgated by the Environmental Protection Agency (EPA). The court rejected all challenges raised in these cases to NRC's regulations. The *Quivira* petitioners sought rehearing or rehearing *en banc*, both of which were denied on March 31, 1989.

In *Quivira*, the industry petitioners primarily argued that the NRC violated the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), as amended, by failing to support its regulations with a cost-benefit analysis allegedly required by the statute. The court, while agreeing with the petitioners that UMTRCA requires both NRC and EPA to perform cost-benefit analyses, drew a distinction, which proved decisive, between "cost-benefit optimization" and "cost-benefit rationalization." The court observed that the former analysis, the "optimization," requires quantification of costs and benefits and a mathematical balancing of the two to determine the optimum result, while the latter, the "rationalization," requires only a consideration and comparison of the costs and benefits of various approaches, and the choice of an approach in which costs and benefits are reasonably related. The court then held that UMTRCA required only the latter type of cost-benefit analysis, and that NRC had performed the requisite cost-benefit rationalization in 1980, when the regulations were originally issued. Moreover, the court held that the NRC had met its statutory cost-benefit duty when it amended its regulation in 1985 and relied on EPA's cost-benefit analysis for its general standards, because the 1985 amendments simply duplicated EPA's standards.

In *EDF*, the the second action cited above, environmental petitioners had raised two related issues: whether the NRC was correct in claiming authority under Section 84c of the Atomic Energy Act to deviate, on a site-specific basis, from EPA's general standards, and whether the NRC could accept, at a particular site, alternate concentration limits of a hazardous constituent or "delist" hazardous constituents without EPA's concurrence. The court gave an affirmative answer to both questions.

Management and Administrative Services

Chapter 10

This chapter covers internal NRC matters, such as changes in the Commission membership, consolidation of the agency's offices, major aspects of personnel management, NRC's information resources, license fees levied and collected, activities of the Office of the Inspector General, and activities of the Office of Small and Disadvantaged Business Utilization and Civil Rights.

Changes Within the Commission

Two changes occurred on the Commission during the year. In July, Commissioner Kenneth M. Carr began his tenure as Chairman of the NRC, succeeding Chairman Lando W. Zech, Jr., whose term had expired. Chairman Carr was first appointed to the Commission in August 1986. The vacancy created by the retirement of Chairman Zech was filled after the close of the fiscal year, when Dr. Forrest J. Remick, former Chairman of the Advisory Committee on Reactor Safeguards, was sworn in on December 1, 1989, bringing the Commission back to its full complement of five. Other changes and appointments at the senior staff level are reported in Chapter 1.

Consolidation of NRC Headquarters

The first phase of the NRC Headquarter's consolidation effort was completed in April 1988 when the Commissioners moved into One White Flint North, located at 11555 Rockville Pike, Rockville, Md. At that time, the NRC was able to move out of five of the 12 buildings it had been leasing in the Washington metropolitan area and collocate approximately 1,400 of its employees.

The second phase of consolidation, consisting of the construction and occupancy of a second building adjacent to One White Flint North, has been delayed. Completion and occupancy were previously scheduled for 1991. Montgomery County's zoning, site plan and building permit review processes have taken years longer to complete than anticipated. That delay caused the developer to seek changes to the terms of the contract under which the building would be constructed and leased to the Government, for use by the Nuclear Regulatory Commission. Negotiations on the matter were continuing at the close of the report period.

The second building will house the 1,000 NRC Headquarter's staff currently working out of various buildings in the Bethesda and Rockville area; and it will be the new site of such vital support facilities as the NRC Emergency Operations Center, as well as a central computer facility, the Public Document room, a central library, an auditorium and a day-care center for infants and toddlers.

PERSONNEL MANAGEMENT

NRC Staff Ceilings

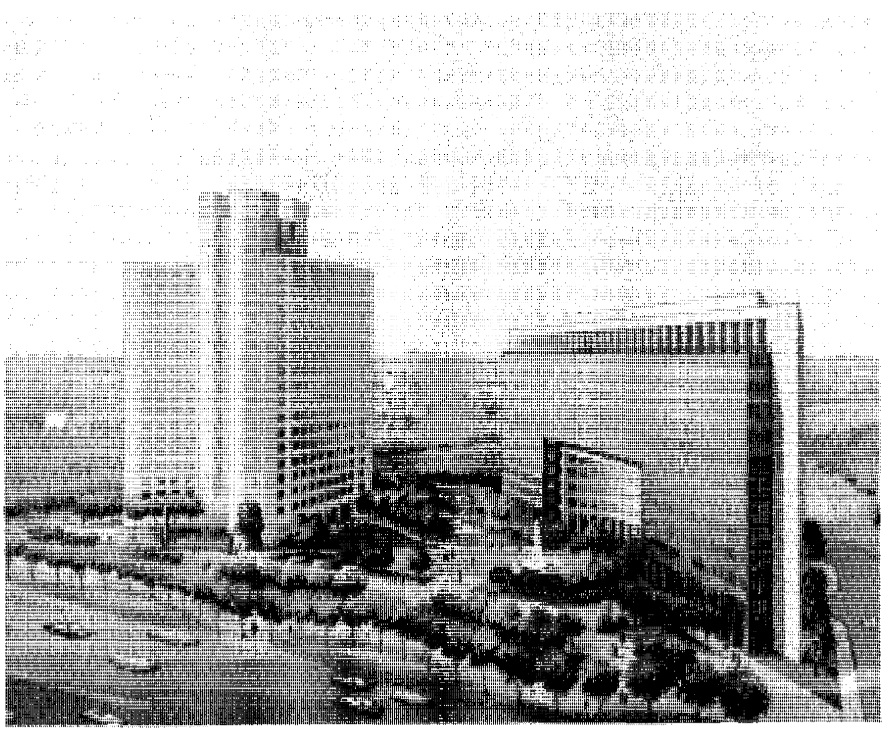
During fiscal year 1989, the NRC expended a total of 3,209 staff-years in carrying out its mission, 0.9 percent above the budget ceiling of 3,180 staff years. Included in the total staff-years expenditure are permanent full-time staff, part-time and temporary workers, and consultants.

Recruitment

In fiscal year 1989, the NRC hired 163 employees and lost 178 permanent full-time employees, representing an attrition rate of 5.7 percent. The NRC recruited at 27 college campuses and participated in approximately 17 other "job fairs." The NRC also sponsored its own secretarial job fair to facilitate the hiring of secretarial and clerical employees.

Performance and Incentive Awards

A substantial number of NRC personnel were recognized for their high quality work in fiscal year 1989, receiving 7 Distinguished Service Awards, 36 Meritorious Service Awards, 655 Special Achievement Awards, 419 High-Quality Performance Salary Increases, 348 Certificates of Appreciation, 2 Presidential Distinguished Executive Rank Awards, 9 Presidential Meritorious Executive Rank Awards, 86 Senior Executive Service (SES) Bonuses, 29 SES Pay Level Increases, and 10 Suggestion Awards.



Consolidation of NRC Headquarters staff encountered unexpected delays during fiscal year 1989. The start of construction of a second building, shown to the right in the artist's conception, awaits successful negotiation of building permit review processes in Montgomery County, Md. The building at left is One White Flint North, at 11555 Rockville Pike in Rockville, Md., currently housing the Commission offices and most of the Headquarters staff.

Labor Relations

The present contract between the NRC and the National Treasury Employees Union (NTEU) was implemented in 1987, with the exception of articles pertaining to "Performance Appraisal," "Reduction in Force (RIF)," and "Salary." Performance Appraisal and RIF have been referred to third party mediation, while Salary has been declared non-negotiable by the 4th Circuit Court of Appeals. In addition, negotiations concerning "Merit Selection," "Reorganization and Moves," and "Drug Testing" are at an impasse and are slated for mediation.

Training and Development

The NRC provides more than 60 different technical courses in reactor and reactor-related technology, end-user computer applications, and probabilistic risk assessment for its technical and administrative personnel. Twenty-nine on-site courses are also provided to improve executive, management, supervisory and administrative skills. NRC employees also participate in a wide range of private sector, college and university, and government-wide educational and development programs designed to improve performance and to assure up-to-date technical proficiency.

In fiscal year 1989, the NRC continued its emphasis on upward mobility programs and the use of Individual Development Plans to help employees clarify their career goals and improve their job skills and performance. A career counseling program continued to be available to all NRC employees, and a graduate level multi-disciplinary program provided participants with technical backgrounds an opportunity to gain or sharpen systems, technology and management skills. A Certified Professional Secretary Program, an Administrative Skills Enhancement Program, and a Computer Science Development Program were available as vehicles by which secretarial/clerical/administrative personnel might expand their sphere of training and advancement opportunities. NRC employees also participated in two formal development programs sponsored by the Office of Personnel Management: the Women's Executive Leadership Program and the Interagency Executive Potential Program for Mid-Level Employees. NRC employees also participated in the Congressional Fellowship Program, which is sponsored by the American Political Science Association. The program allows senior-level Federal executives to work as congressional aides for nine months and to gain a working knowledge of the legislative branch.

The NRC offers extensive supervisory and management development programs for current staff members. Supervisory development training is mandatory for new supervisors. A course in supervising human resources covers all aspects of supervision, and an NRC Management Workshop enables managers to evaluate and analyze their current managerial effectiveness. The NRC has also developed a leadership Training for Technical and Professional Engineers Program. Courses in this program have been designed to provide inspectors with a better understanding of managerial principles.

Rotational Assignments

In fiscal year 1989, the NRC further expanded its use of rotational assignments for the career development of employees and for satisfying organizational and staffing needs. Managers and supervisors were actively involved in identifying employees for rotational assignments. A total of 118 employees participated in rotational assignments during fiscal year 1989.

Executive Leadership Development

Members of the Senior Executive Service continued their participation in a very active rotational assignment program. These assignments, designed to develop a broader understanding of all aspects of the agency's regulatory activities, involved exchanges of assignments between Headquarters and the Regions, and also intra-office exchanges at Headquarters. Many of the executives attended the Government's premier executive development program at the Federal Executive Institute located in Charlottesville, Va., as well as other programs designed to provide insight into budgetary and policy decisions of the Administration and Congress. A senior management conference for all executives was held during the report period, providing a forum for further analysis and understanding of the NRC Strategic Plan and Five-Year Plan, and for addressing salient technical and management issues.

Voluntary Leave Transfer Program

This program is intended to provide 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 1989, 12 NRC employees received voluntary leave donations.

Employee Assistance Program

In fiscal year 1989, the NRC Employee Assistance Program (EAP) conducted 30 training sessions for approximately 600 headquarters, regional, and field supervisors and managers, to the end of maintaining a drug-free workplace. The EAP continued to provide individual counseling and referral services for NRC employees for such problems as chemical dependency, job or family stresses, and chronic illness. Employee awareness lectures were presented on substance abuse, eating disorders, and alcoholism in the family. EAP staff initiated and monitored a contract to provide the services of the Drug Rehabilitation Assessment Coordinator required by the NRC Drug Testing Plan (see next heading).

Drug Testing Policy

The Commission's drug testing policy—communicated to all employees in a policy statement dated July 9, 1987—declares that the use of illegal drugs by NRC employees is unacceptable and that the agency maintains ''zero tolerance'' of such use. (See the *1988 NRC Annual Report* p. 190.)

Random drug testing was instituted in November 1988, involving 575 ''non-bargaining unit'' employees in testing-designated positions. Provision for such testing was made at NRC Headquarters, at all Regional Offices, and at resident inspector stations. The agency has set up a100-percent-random testing procedure for designated positions, based on a series of selections randomly occurring 12 times a year among employees at Headquarters, Regional Offices and other sites. Significant improvements in the program introduced during the report period include on-site specimen collection and the use of two separate testing laboratories in the analysis of each collection. Besides random testing of employees, the NRC plan allows for testing based on reasonable suspicion of abuse, the testing of applicants for testing-designated positions, voluntary testing, and follow-up testing.

Drug testing involving ''bargaining unit'' employees is subject to negotiations with union representatives, which were under way at the end of the fiscal year.

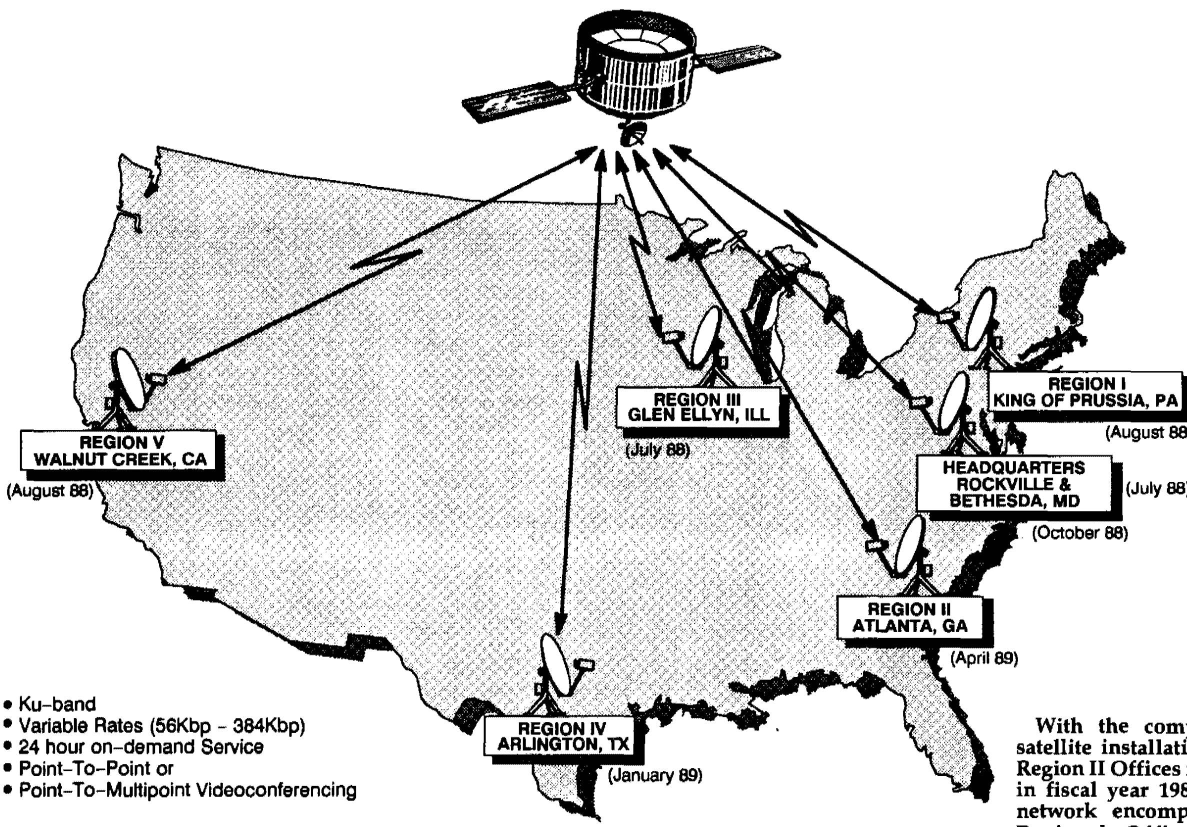
FACILITIES MANAGEMENT

The Facilities Management Branch—part of the Division of Contracts and Property Management in the NRC Office of Administration—meets the facilities and transportation needs of Headquarters staff and provides technical guidance to the Regional Offices in facility management and transportation. During fiscal year 1989, the facilities management staff carried out over 400 internal facility reconfigurations and relocations, as required by new or expanded functions and venues of various NRC offices.

NRC INFORMATION RESOURCES

The Office of Information Resources Management (IRM) is responsible for developing, providing and administering information resources throughout the

NRC SATELLITE NETWORK



With the completion of the satellite installation at the NRC Region II Offices in Atlanta, Ga., in fiscal year 1989, the satellite network encompassed all five Regional Offices and Headquarters. Two-way audio/video telecommunications are now possible between and among Headquarters and the Regions.

agency in the areas of computer operations, telecommunications, and similar centralized information services. Such services include data base management, office automation, computer hardware and software, systems development, nationwide telecommunications equipment and services, an Information Technology Services Support Center, user training, document control and management, mail and distribution, and other information support to all NRC offices and elements.

Shared Information Network (SINET)

Beginning in 1984, IRM introduced a fundamental change in the agency's approach to the development of new computer systems, looking to integrate independent and often redundant data structures into a single shared data base. The concept grew out of the conviction that information, like human and monetary resources, is an invaluable agency asset which must be managed for maximum benefit and effect, in support of organizational objectives. A basic tenet of the shared data approach is that data are not the property of any individual organizational element, but rather are common agency property which should be easily accessible by all NRC components. Under the new approach, data are to be managed in an integrated hardware/software structure, linked by the agency's telecommunications network. The structure, formerly known as the Safety Information Network, is now called the Shared Information Network, SINET. Shared data are defined as data from various sources that the agency must have, use and preserve in performing its regulatory functions, and that are needed by more than one Headquarters or Regional Office.

SINET consists o three main elements—a shared data base, data access tools, and applications systems. All SINET data are stored in a single data base. Each item is created exactly once, so the possibility of discrepancies is eliminated. Because SINET data are scrutinized by a broad audience of users, as well as by a data quality assurance group, the creators of data receive the feedback that ensures a prompt detection and correction of errors. The data access tools include an executive information system for NRC managers, a menu-driven system for displaying and printing data on specific dockets, and a flexible reporting tool that can search, sort and format data to satisfy many kinds of *ad hoc* analysis and reporting needs. SINET is accessible from almost any NRC terminal, for immediate display or print-out. Training in the uses of SINET is furnished by the Information Technology Services Training Laboratory. Applications systems are the means for getting data into the SINET data base and for providing customized reports and display screens meeting the needs of the creators of data.

More than 20 separate subject areas can be accessed through SINET, such as "Enforcement," "Facility," "Inspection," "Radioactive Materials," and so forth. Each of these is divided into several data entities, i.e., a person, thing, organization, or event about which data are stored. Each entity, in turn, is broken down into a sometimes large number of data elements. Data selected for earliest inclusion in SINET are mainly safety-related, but the long range goal is that all systems which create shared data use SINET as the data base. Among the computer systems whose data are stored in and retrieved from SINET, as of the close of report period, were:

- Inspection Procedure Authority System, maintained by the Office of Nuclear Reactor Regulation to create and store information useful in the Master Inspection Planning System (see next item).
- Master Inspection Planning System, used by the Regional Offices to create and maintain master inspection plans for all commercial nuclear power plants and to manage the inspection program.
- Probabilistic Risk Assessment Study Information System, used by the Office of Research to create and store information and produce reports on completed probabilistic risk assessment studies for commercial nuclear power plants.
- Systematic Assessment of Licensee Performance (SALP) Scheduling and Reporting System, used by the Regional Offices to maintain information about planned and completed SALPs (see Chapter 2).
- Unit Daily Status System, used by the Office for Analysis and Evaluation of Operational Data to record and disseminate the operational status of each commercial nuclear power plant as received at the NRC Operations Center (see Chapter 3) each morning.
- Event/Unit Data Maintenance System, used to create and/or maintain the event- and unit-related data in the SINET data base.

Nuclear Documents System (NUDOCS)

The NRC's Nuclear Documents System (NUDOCS) is the agency's centralized document search-and-retrieval system, encompassing a vast amount of information related to the licensing and inspection of nuclear reactors and materials, as well as extensive data regarding regulatory, adjudicatory, high-level and low-level nuclear waste issues.

Besides the normal growth over time of the NUDOCS data base, a number of improvements have been realized during the report period. The quantity of documents identified for full text processing—such as the Licensee Event Reports (see Chapter 3) and Title 10 of the *Code of Federal Regulations* (10 CFR)—increased by the day. The categories of documents designated for full text search-and-retrieval have also expanded, with the addition of Generic letters, Information Notices, and Information Bulletins. Work continues on the Nuclear Documents System—Advance Design (NUDOCS/AD), which promises still greater capacity and usability.

Integration into the NUDOCS system of new full text systems—the Waste Management Transitional Licensing Support System (TLSS) and the Congressional Correspondence Retrieval System (CCS)—has been completed, as has that portion of the Atomic Safety Licensing Board Panel Proceedings System related to the Seabrook (N.H.) nuclear power plant.

Not only did the size and content of the NUDOCS data base increase during the report period, external user access was also greatly enhanced. During the fiscal year, access to the publicly available portions of the NUDOCS data base has been permitted and available to such potential users as the utilities, National Laboratories, universities, and the technical libraries of various State governments.

During the fiscal year, at the Nuclear Information and Records Management Association Conference, the NRC announced the inauguration of the new NRC Electronic Document Exchange (NEDEX) Program. NEDEX is a demonstration program designed to evaluate the feasibility and potential benefits of the NRC's receiving submitted documents in machine-processable format. The two vehicles by which the program will be effected are the "electronic file transfer" and "magnetic media submittals." In conjunction with the NEDEX program, NRC will provide WordPerfect 4.2 templates for computerized preparation of certain documents, such as License Event Reports. Thus utilities may prepare their reports using the templates and submit the computer diskette to the NRC. Having these documents submitted in this machine-processable format should bring about a more efficient and timely incorporation of documents into NUDOCS and into other NRC technical data bases, such as SINET (see above).

NRC Satellite Telecommunications

Remarkable strides were made during the fiscal year in the area of NRC telecommunications. As a result of this progress, communications between Headquarters and the Regions will never be the same: at the end of the report period, all five Regional Offices were equipped for satellite-transmitted two-way audio/video telecommunications with Headquarters and with one another. Events such as Commission discussion of issues related to a particular licensed operation are transmitted to the Regional Office dealing with that licensee. Headquarters program offices are also in touch with the Regions via satellite.

The NRC has in recent years carefully researched the potential benefits of installing a videoconferencing network and creating a satellite link between Headquarters and the Regions. After some preliminary testing involving Headquarters and Region III near Chicago, it was decided to go ahead with a full-fledged Pilot Program and begin transmitting live audio/video coverage of Commission meetings and other events to the Regions. After some months of the pilot effort, a survey was taken by IRM of NRC senior management. The responses were very positive: the videoconferencing capability was perceived as both an immediately useful resource and a tool of extraordinary potential, especially in the areas of training, inspection and enforcement, as well as for regular interchange between and among NRC offices and with licensees.

Accordingly, IRM offered both near term and prospective applications of the technology for the consideration of agency decision-makers. In the near term, it was proposed that the satellite communications network be made a part of the Operation Center's Emergency Telecommunications System (see Chapter 3). Another near term objective presented was satellite linkage with the National aboratories, and possibly other contractors.

OFFICE OF THE INSPECTOR GENERAL

During the report period, the former Office of Inspector and Auditor (OIA) was discontinued and, pursuant to the Inspector General Act Amendments of 1988, a statutory Inspector General was established at the NRC. The new Office of the Inspector General (OIG) began operations on April 15, 1989. (See Chapter 1.) From its inception to the close of the fiscal year, OIG issued five audit reports and 24 investigation reports to NRC management, and referred five matters to the Department of Justice. At the close of the report period, 14 audits and 41 active investigations were under way in the OIG.

Following are brief descriptions of the audits and select investigations completed in fiscal year 1989.

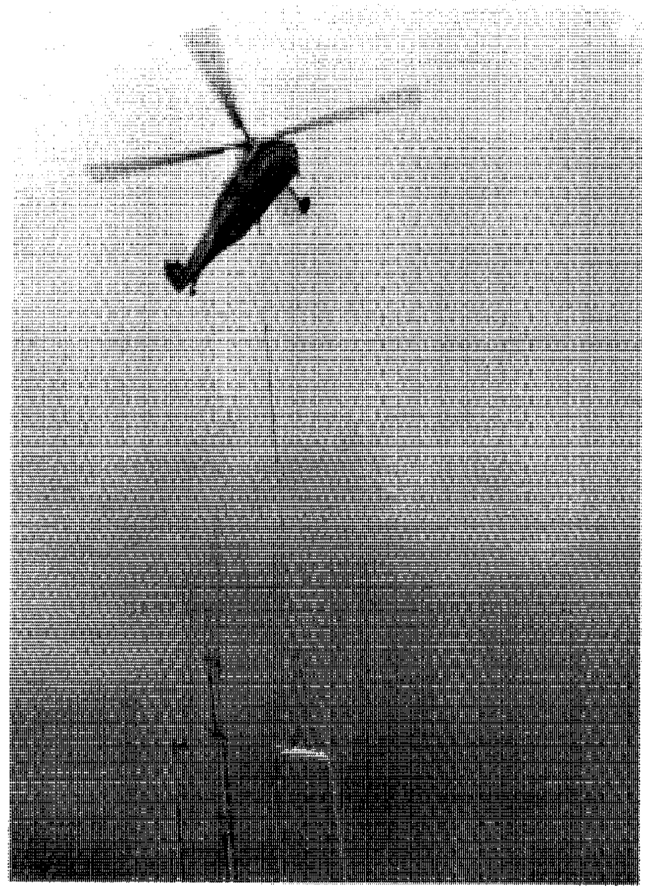
OIG FY 1989 Audits

Enforcement Program. The OIG audit of the NRC's enforcement program found that—while the Office of Enforcement (OE; see Chapter 1) was doing a creditable job of managing the program—certain weaknesses in administrative procedures and a need for better headquarters oversight were in evidence. Among the recommendations coming out of the audit were that a sampling procedure be instituted by the Director of OE to determine if Regional Offices were consistently adhering to agency enforcement policy.

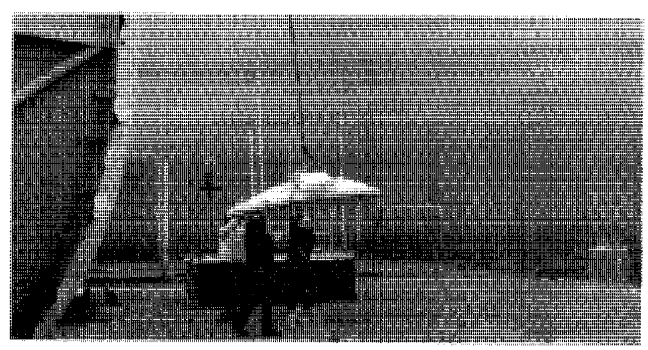
Reporting of SALP Findings. Under the NRC's Systematic Assessment of Licensees Performance program, called SALP (see Chapter 2), nuclear power plants are individually and comprehensively appraised and rated. An OIG review of the program disclosed a lack of consistency among the Regional Offices in reporting SALP findings—attributable to the use of differing formats. Since SALP ratings are supposed to be consulted in the process of allocating agency resources, the NRC needed to develop a system to ensure that this process was taking place reliably and equitably. OIG recommendations were aimed mainly at introducing a greater degree of standardized reporting in the program.

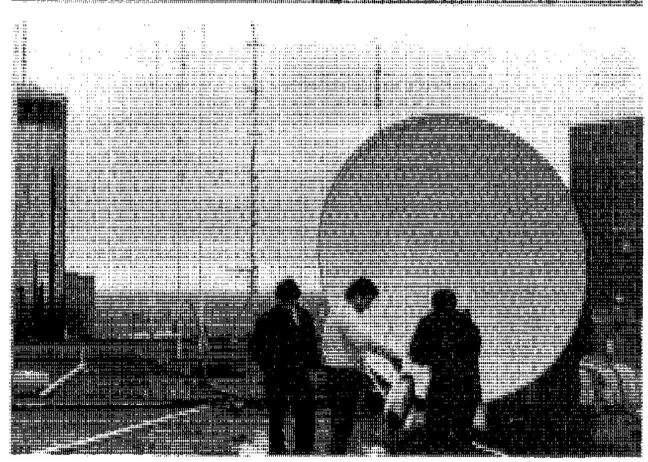
Incident Investigation Program. The Incident Investigation Program (IIP) is an inspection effort undertaken in response to the more significant unplanned operational events in nuclear power plants (see Chapter 3). An OIG study of the program revealed that procedures were not in place to ensure that corrective actions had been taken, by either the NRC or the licensee, in following up inspection findings. It was also determined that changes were needed to provide for an independent review of NRC staff action taken in response to IIP findings, especially those critical of staff performance. It was recommended that the Executive Director for Operations develop more specific guidance to staff regarding follow-up to IIP determinations and that steps be taken to ensure independent confirmation of staff action on team conclusions critical of staff performance.

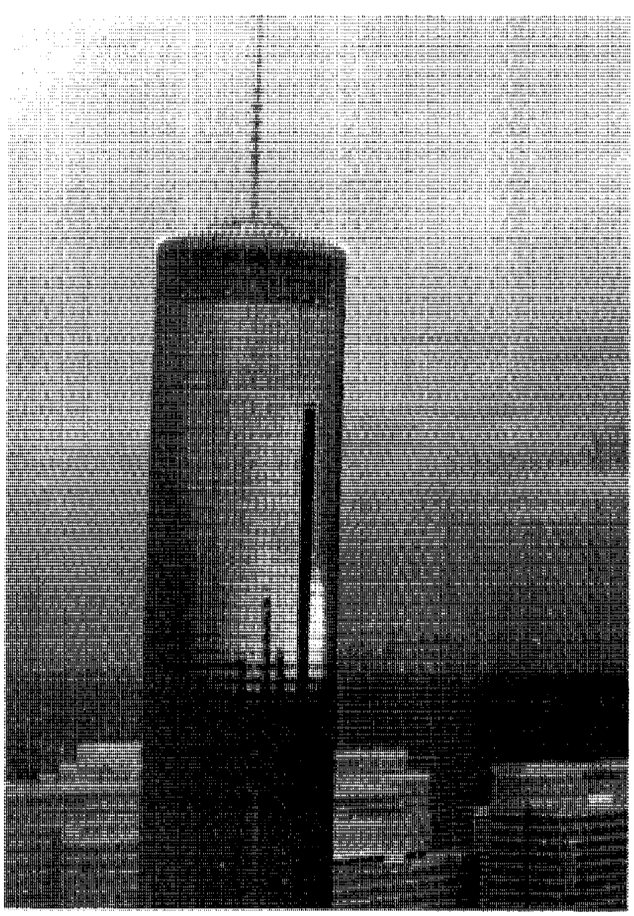
Handling of Cash Receipts. Acting on a referral from the NRC's Division of Accounting and Finance, in the Office of the Controller, the OIG discovered that certain imprest funds were unaccounted for, besides those that prompted the referral, and determined that numerous internal control weaknesses were present in the way the funds were handled. An investigation ensued from the audit (see below). A dozen separate recommendations regarding handling of the funds were made by OIG and adopted by management.



Installing the satellite dish on the roof at the Region II Office in Atlanta, Ga., involved the use of a helicopter. Clockwise from top left, the dish is removed from the truck; the helicopter hoist is under way; the dish is lowered to the crew on the roof; the dish is secured to the roof; and the setting sun signals the end of a banner day in NRC communications.







Use of Personal Computers. An examination of NRC's control and management of personal computers led to several conclusions, among them that personal computers were not being fully utilized in the agency, and that the NRC staff lacked awareness regarding agency personal computer and software policy. OIG recommendations were accepted and were being implemented at the close of the report period.

Select OIG FY 1989 Investigations

Theft of Imprest Funds. As noted above, an audit of the handling of imprest funds led to an investigation, during which an individual cashier admitted diverting funds for personal use. The cashier resigned in lieu of dismissal.

Conflict of Interest. An OIG investigation was launched after allegations were made in the media that a former NRC resident inspector at the Shoreham (N.Y.) nuclear power plant had accepted employment with the firm that built the reactor at that facility, this in violation of NRC regulations. Agreeing that regulations governing the negotiation of employment with outside firms had been violated by the individual and the agency, OIG found no evidence to indicate that the person had compromised his inspection activities at the Shoreham plant. Further action in the matter was pending at the close of the report period.

Outsider Use of Long Distance Telephone. An OIG investigation revealed that an NRC employee had rigged his government telephone to give his wife direct access to a Federal Telecommunications System line, through which she made over 2,000 personal calls, including a large number in connection with her private business. The value of the calls was placed at about $3,800. Action by the agency was pending at the close of the report period.

Unauthorized Calls to 976 Exchange. At the request of agency management, OIG opened an inquiry into the origin of unauthorized calls placed to "976" exchanges over agency telephones. The exchange is used exclusively for reaching various kinds of pre-recorded messages. The investigation disclosed that over 600 such calls had been made over a two-month period, but, since there were made after hours from vacant offices, no identification of the callers could be made. The agency has restricted access to the exchange through agency telephone lines.

Misuse of Inside Information. An allegation conveyed through a Regional Office that a utility's security force was receiving advance information of unannounced NRC inspections, and that the source was an NRC secretary whose husband was a utility employee, led to this investigation. OIG could not confirm that there had, in fact, been unauthorized disclosure of the

On November 22, 1989, David C. Williams, at right, was sworn in as the NRC's first Inspector General. The Office of Inspector General supplants the former Office of Inspector and Auditor. Administering the oath, at left, is NRC Chairman Kenneth M. Carr; at center is Samuel J. Chilk, Secretary of the Commission.

Table 1. License Fee Collections 1989

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR 171	$125.3 million		$125.3 million
10 CFR 170	54.5 million	$3.6 million	58.1 million
DOE Fund	15.0 million		15.0 million
TOTAL FEES	$194.8 million	$3.6 million	$198.4 million

kind alleged, but did establish that the opportunity for such existed. Notification procedures related to unannounced inspections at the plant involved were subsequently altered by the Regional Office.

CONTRACTING

In fiscal year 1989, NRC contracting with commercial firms for technical assistance, research work, and general purchases totaled approximately $64,800,000. Contracts under the Small Business Innovative Research Program came to $1,231,579, and grants under cooperating agreements with education and non-profit institutions totaled $3,003,193.

NRC LICENSE FEES

In fiscal year 1989, $125.3 million was collected in reactor operating license fees, under 10 CFR 171; $51.8 million was collected from facility and materials licensing and inspection fees, under 10 CFR 170, and $15 million from the Department of Energy's Waste Fund. The total of fees collected, $192.1 million, constitutes 45.7 percent of the NRC budget for the period of $420 million. In addition, payment from the prior year's billings totalling $6.3 million was received, producing an overall total collection in fiscal year 1989 of $198.4 million. Section 5601 of the Omnibus Budget Reconciliation Act of 1987 (OBRA; Public Law 100-203) requires the Commission toassess and collect not less than 45 percent of its fiscal year 1989 budget, which means not less than $189 million. As determined by the fiscal year 1989 appropriation, the NRC deposited $198.4 million into the General Treasury, thus reducing its final appropriation for the fiscal year to $221.6 million.

As indicated above, the Commission uses three different approaches in collecting these fees. First, Public Law 100-203 authorizes the agency to assess annual fees to utilities licensed to operate nuclear power plants. These fees are established under 10 CFR 171, in the Commission's regulations. Second, under Title V of the Independent Offices Appropriation Act of 1952, the NRC is authorized to collect fees for processing applications, permits, licenses and approvals, and for both routine and non-routine safety inspections. These fees are established under 10 CFR 170. Third, under a Memorandum of Understanding with the Department of Energy, dated July 20, 1988, the NRC is reimbursed from the Nuclear Waste Fund for pre-application activities related to the disposal of high-level radioactive waste and spent fuel in a geologic repository. Table 1 shows the totals for the three categories cited; it includes collections from the prior year's billings.

New Fee Schedule

The Commission adopted a revised schedule of fees, effective January 30, 1989, designed to more completely recover NRC's costs for providing services to identifiable recipients. Under 10 CFR 170, the fee ceilings for reactor and major fuel cycle permits, licenses, amendments and inspections were eliminated and the hourly rate assessed for regulatory services was revised. With respect to 10 CFR 171, the method of assessing annual fees was changed. Previously, one annual fee applicable to all operating power reactors was assessed. Under the revised rule, annual fees were established and assessed on the principle that licensees who require the greatest expenditure of NRC resources shall pay the greatest fee. Thus, the annual fee established takes into account the kind of reactor, its location, and other considerations related to the generic research and other costs associated with power reactor regulation.

Litigation Concerning Fees

The Commission published a Final Notice of Rulemaking in the *Federal Register* on September 18, 1986, establishing annual fees for power reactors with operating licenses (10 CFR 170), which became effective on October 20, 1986. The rule was challenged and upheld in its entirety, in *Florida Power and Light Co., et. al. v. United States,* 846 F. 2nd 765 (D.C. Circuit, 1988). A petition for writ of certioari challenging that decision was filed with the Supreme Court, in *Florida Power and Light Co. v. United States,* No. 88-234. The Supreme Court denied the petition on May 1, 1989, 109 S. Ct. 1952 (1989).

On December 29, 1988, the Commission published a Final Notice of Rulemaking in the *Federal Register* revising the fees in 10 CFR 170 and 10 CFR 171. Several lawsuits were filed challenging the fee schedules. As a result of the Supreme Court decision in *Skinner v. Mid-America Pipeline Co.,* No. 87-2098, decided on April 25, 1989, and the denial of certiorari in *Flordia Power and Light,* all of the lawsuits have been withdrawn.

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 response to provisions of Public aw 95-507, amending the Small Business Investment Act of 1957. During fiscal year 1989:

- It was estimated that $65,156,450 in total prime contracts would be awarded in fiscal year 1989. The actual total prime contracts dollar award was $60,832,671.

- It was estimated that small business prime awards would be $29,971,967, or 46 percent of the total estimate. The actual achievement for small business prime awards was $19,665,039, or 32.32 percent of the dollars reflected in the item above.

- The NRC estimated that awards to "8(a) firms" would be $8,441,670, or 12.96 percent in fiscal year 1989. Awards to 8(a) firms were actually $8,796,999, or 14.46 percent of the total dollar amount of all prime contracts, regardless of dollar value.

- The goal for prime contract awards to small disadvantaged business firms other than 8(a) was $183,520, or 0.28 percent. The actual achievement was $537,206, or 0.88 percent of the dollars reported in the first item above.

- The estimate for prime contract awards to small business concerns owned and controlled by women was $2,000,000, or 3.07 percent. Awards to such firms were $2,189,001, or 3.60 percent of the total dollar amount of all prime contracts, regardless of dollar value.

- The goal for subcontract awards to small businesses was $1,590,000, or 63.6 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $2,227,285, or 74.74 percent of total subcontracts awarded. The NRC's total subcontract dollar awards goal in fiscal year 1989 was $2,500,000. The NRC's total subcontract dollar awards in the fiscal year was $2,980,042.

- The goal for subcontract awards to small disadvantaged businesses was $200,000, or 8 percent. Subcontracting awards to small businesses were $370,025, or 12.42 percent of total subcontract dollars awarded.

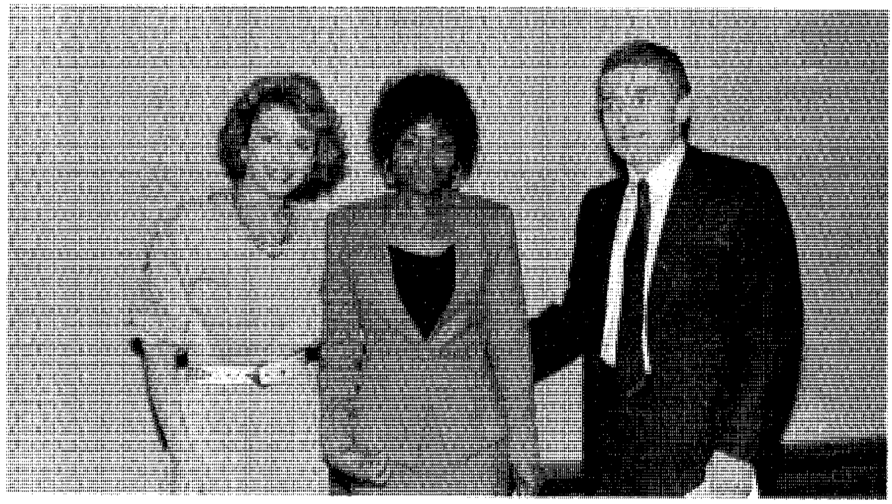
During the year, 120 interviews were conducted with firms wanting to do business with the NRC, and 45 follow-up meetings were arranged with NRC technical personnel. The Office of Small and Disadvantaged Business Utilization and Civil Rights staff also participated in five major small business conferences. Most noteworthy among them were the Small Business Week observance in May 1989, and Minority Enterprise Development Week, in October 1989.

Federal Women's Program

The Federal Women's Program (FWP) efforts continued to be successful through fiscal year 1989 in promoting the utilization and advancement of women. Despite limitations on outside hires, which slowed recruitment activities, and a planned reduction in the overall work force, women continue to comprise one-third of the NRC work force. Women moved into managerial positions, as the representation of women in grades GG-11 and above went from 379 to 392, an increase from 17 percent to 18.4 percent of that category, and another woman was added to the ranks of the Senior Executive Service. Contributing to these gains were the agency's affirmative action activities and its special emphasis personnel programs. Other notable events taking place during fiscal year 1989 included:

- Seven women were given upward mobility positions.

- Forty-one women went on rotational assignments and details for development and training.



Among the many events presented under the aegis of the Federal Women's Program is the appearance as guest speakers at the NRC of women prominent in government, business, culture and the media in the Washington, D.C. metro area. One such is Roberta Baskin, at left in the photo, Investigative Reporter for TV Station WJLA, Channel 7 in Washington, who served as Keynote Speaker for Women's History Month in March 1990. At center is Era Marshall, Director of the Federal Women's Program at the NRC, and at right is James M. Taylor, NRC's Executive Director for Operations.

- Three out of the five employees selected for the Mid-Level Executive Potential Program were women.
- Three women were accepted for training in the Women's Executive Leadership Program.
- Four women were accepted at the Federal Executive Institute.
- One woman was selected to participate in the Congressional Fellowship Program.
- Three hundred women received non-governmental training (over 50 percent of the 500 total participants).
- Women made up half of the Office of Nuclear Reactor Regulation's Intern Program, a pilot recruitment/development effort designed to secure a cadre of professionals for future reactor licensing and regulating responsibilities, including leadership roles.

Underlying these gains are the several special emphasis programs developed and carried out with the assistance of the FWP Advisory Committee and the NRC regional FWP coordinators, who reported outstanding management support for the wide array of program presentations.

National Women's History Month was observed during the fiscal year in the Regions and at Headquarters. The highlight of the celebration at Headquarters was the presentation by the world-renowned physicist and pioneer in computer science, Dr. Grace Hopper, as the keynote speaker.

National Secretaries Week was also celebrated throughout the NRC, with special "recognition and awareness" luncheons attended by hundreds of secretaries and their managers. Agency efforts in support of NRC's secretaries included the presentation of several "lunch and learn" self-help programs, career counseling, career planning, and other programs.

The annual FWP conference, which took place toward the close of the report period, was attended by representatives of NRC management, by regional and headquarter FWP coordinators, and by personnel representatives and management officials from the personnel and civil rights elements of the Department of Transportation, the National Aeronautics and Space Administration, the Department of Labor, the Office of Personnel Management, and the Bureau of Labor Statistics.

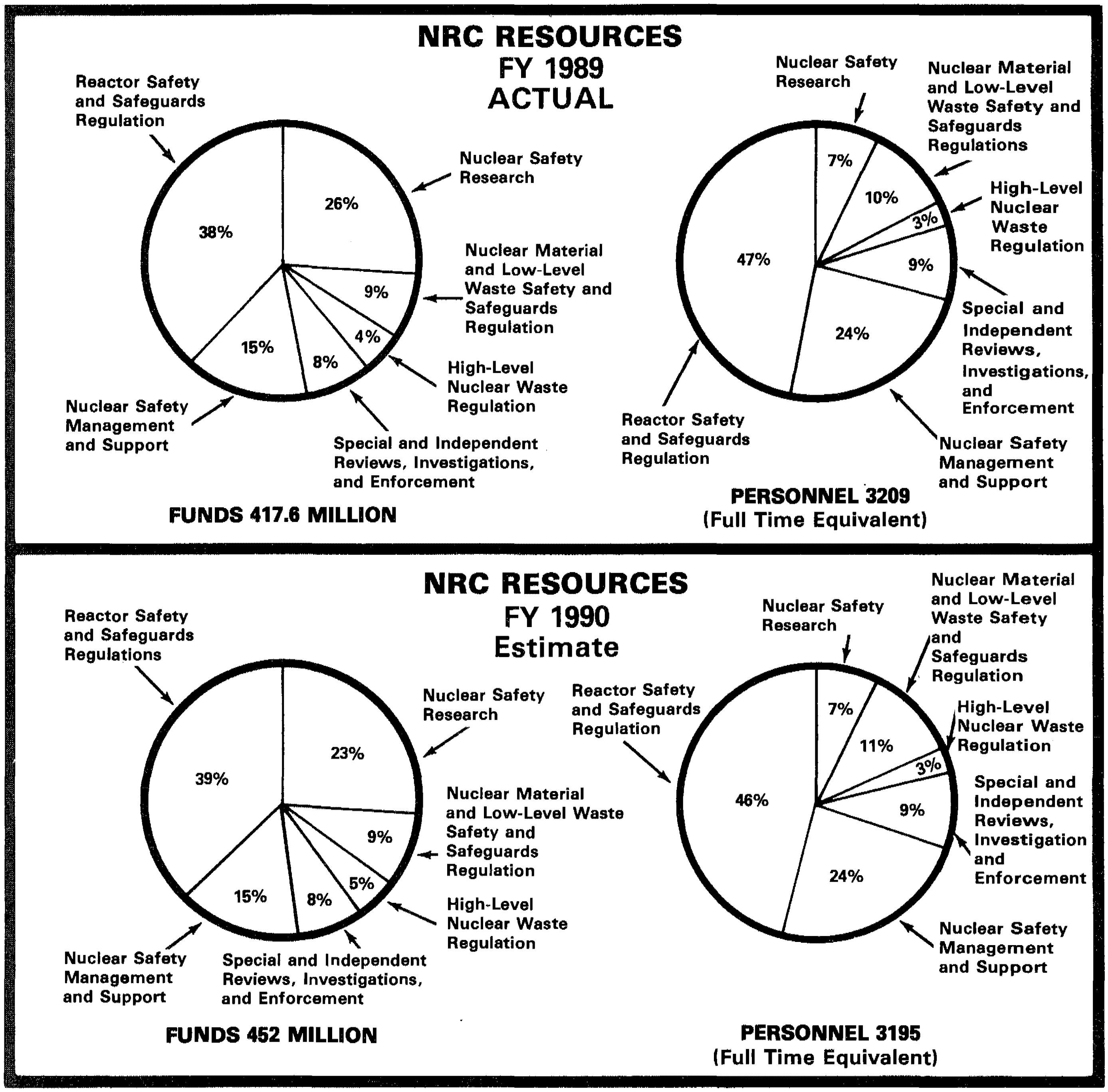
Civil Rights Program

During fiscal year 1989, the NRC Multi-year Affirmative Employment Plan was approved by the Chairman and forwarded to the Equal Employment Opportunity Commission. The Commission was briefed on February 2 and August 15, 1989, concerning the NRC's EEO and Affirmative Employment programs, goals and accomplishments.

An analysis of the EEO accomplishment report, submitted annually by Office Directors and Regional Administrators to the Director, OSDBU/CR, was provided to the NRC Executive Director for Operations to apprise him of the performance of managers in achieving assigned goals. The Director, OSDBU/CR, continues to function as a non-voting, *ex-officio* member of the SES Performance Review Board.

The Civil Rights Program staff sponsored a training seminar for EEO Counselors from Headquarters and the Regional Offices. The well attended event was held at Westminster, Md.

Training sessions for new supervisors and managers were conducted at Headquarters and the Regional Offices, during the report period. Civil Rights Program staff were in attendance at all locations where training was held.



Source: **USNRC Budget Estimates,Fiscal Years 1991,** "Summary of Headquarters and Regional Resources by Mission Area," pg. 180, January 1990.

FY 1988/1989 NRC Financial Statements

Balance Sheet (in thousands)

	September 30, 1989	*September 30, 1988 (Note 5)*
Assets		
Cash:		
Appropriated Funds in U.S. Treasury	$ 151,821	$ 119,472
Other—Notes 1 & 3	95,846	91,565
Imprest Fund Balance	354	346
	248,021	211,383
Accounts Receivable:		
Federal Agencies	—0—	—0—
Miscellaneous Receipts—Note 2	718	7,512
Other	83	1,091
Less: Allowance For Uncollectibles	(307)	(307)
	494	8,296
Plant:		
Completed Plant and Equipment	34,314	33,562
Less: Accumulated Depreciation	(23,462)	(21,104)
	10,852	12,412
Advances and Prepayments:		
Federal Agencies	—0—	—0—
Other	5,533	5,636
	5,553	5,636
	$ 264,900	$ 237,727

	September 30, 1989	*September 30, 1988*
Liabilities and NRC Equity		
Liabilities:		
Funds Held for Others—Notes 1 & 3	$ 95,846	$ 91,565
Accounts Payable and Accrued Expenses:		
Federal Agencies	16,645	23,687
Other	8,430	3,796
Accrued Annual Leave of NRC Employees	16,205	15,345
Deferred Revenue—Note 3	—0—	—0—
Total Liabilities	$ 137,126	$ 134,393
NRC Equity: Balance at October 1	103,334	119,874
Additions:		
Funds Appropriated—Net—Note 6	420,000	392,800
Non-Reimbursable Transfers from Other Gov't Agencies	—0—	—0—
	523,334	512,674
Deductions:		
Net Cost of Operations	389,804	404,009
Funds Returned to U.S. Treasury—Note 2	5,756	5,331
	395,560	409,340
Total NRC Equity	127,774	103,334
Total Liabilities and NRC Equity	$ 264,900	$ 237,727

Note 1. As of September 30, 1988, includes $3,493,656.55 of funds received under cooperative research agreements involving NRC, DOE, Euratom, France, Federal Republic of Germany, Japan, Austria, the Netherlands, Belgium, and the United Kingdom.
Also included is $88,567,906.00 funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processing the applications has been finalized and, accordingly, are not available for NRC use.

Note 2. These funds are not available for NRC use.

Note 3. On March 24, 1978, 10 CFR 1 was revised. Contained therein by category of license are maximum fee amounts to be paid by applicants at the time a facility or material license is issued. Also, after the review of the license application is complete, the expenditures for professional manpower and appropriate support services are to be determined and the resultant fee assessed. In no event will the fee exceed the maximum fee for that license category, which generally has been paid. This could involve the refunding of a significant portion of the initial amount paid. Therefore, the revenue is recorded in a Deferred revenue account at the time of billing and is removed from this account and recorded in Funds Held for Others when the bill is paid. The balance in the Deferred revenue account consists of deferred revenue on billings issued but not collected. See Note 1.

Note 4. Represents current year cost of plant and equipment acquisition for use at DOE facilities.

Note 5. Prior year figures, in this column, have been restated to reflect adjusted balances at 9/30/88.

Note 6. PL 100-371 authorizes NRC to retain revenues derived from license fees, inspection services, and other services. These funds are to be appropriated back to NRC to be used for salaries and expenses.

Statement of Operations (in thousands)

	Fiscal Year, 1989 (October 1, 1988, thru September 30, 1989)	Fiscal Year, 1988 (October 1, 1987, thru September 30, 1988 —Note 5)
Personnel Compensation	$ 171,587	$ 158,028
Personnel Benefits	27,164	23,009
Program Support	126,410	123,974
Administrative Support	49,184	72,110
Travel of Persons	11,423	18,743
Equipment (Technical)—Note 4	—0—	1
Construction—Note 4	—0—	—0—
Taxes and Indemnities	127	167
Refunds to Licensees	—0—	—0—
Representational Funds	20	16
Reimbursable Work	1,649	491
Increase in Annual Leave Accrual	860	1,011
Depreciation Expense	2,358	3,736
Equipment Write-Offs and Adjustments	—0—	—0—
Allowance for Uncollectibles	—0—	—0—
Total Cost of Operations	$ 390,782	$ 401,291
Less Revenues:		
Reimbursable Work for Other Federal Agencies	(1,649)	(491)
Fees (Deposited in U.S. Treasury as Miscellaneous Receipts—Note 6)		
Material Licenses	(3,651)	—0—
Facility Licenses	(47,958)	(35,007)
Other	(5,871)	(3,714)
Total Revenue	(59,129)	(39,212)
Net Cost of Operations Before Prior Year Adjustments	331,653	362,079
Transfer to Appropriated Funds—Note 6	58,151	41,930
Net Cost of Operations	$ 389,804	$ 404,009

Government Investment in the Nuclear Regulatory Commission (in thousands)

Appropriation Expenditures:	
Fiscal Year 1975 (January 19, 1975 through June 30, 1975)	$ 52,792
Fiscal Year 1976 (July 1, 1975 through September 30, 1976)	226,248
Fiscal Year 1977 (October 1, 1976 through September 30, 1977)	230,559
Fiscal Year 1978 (October 1, 1977 through September 30, 1978)	270,877
Fiscal Year 1979 (October 1, 1978 through September 30, 1979)	309,493
Fiscal Year 1980 (October 1, 1979 through September 30, 1980)	377,889
Fiscal Year 1981 (October 1, 1980 through September 30, 1981)	416,867
Fiscal Year 1982 (October 1, 1981 through September 30, 1982)	441,902
Fiscal Year 1983 (October 1, 1982 through September 30, 1983)	514,613
Fiscal Year 1984 (October 1, 1983 through September 30, 1984)	462,084
Fiscal Year 1985 (October 1, 1984 through September 30, 1985)	467,902
Fiscal Year 1986 (October 1, 1985 through September 30, 1986)	420,946
Fiscal Year 1987 (October 1, 1986 through September 30, 1987)	392,624
Fiscal Year 1988 (October 1, 1987 through September 30, 1988)	410,663
Fiscal Year 1989 (October 1, 1988 through September 30, 1989)	387,644
	$5,383,103
Unexpended Balance of Appropriated Funds in U.S. Treasury September 30, 1989	151,821
Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975	429
Funds Appropriated—Net	5,535,353
Less:	
Funds Returned to U.S. Treasury—Note 2	336,119
Assets and Liabilities Transferred from Other Federal Agencies Without Reimbursement	1,673
Net Cost of Operations from January 19, 1975 through September 30, 1989	5,069,783
Total Deductions	5,407,579
NRC Equity at September 30, 1989 as Shown on Balance Sheet	$ 127,774

Appendix 1

NRC Organization

(As of December 31, 1989)

COMMISSIONERS

Kenneth M. Carr, Chairman
Thomas M. Roberts
Kenneth C. Rogers
James R. Curtiss
Forrest J. Remick

The Commission Staff

General Counsel, William C. Parler
Office of Governmental and Public Affairs, Harold R. Denton, Director
Office of the Inspector General, David C. Williams, Inspector General
Office of the Licensing Support System Administrator, Lloyd J. Donnelly, Administrator
Secretary of the Commission, Samuel J. Chilk

Other Offices

Advisory Committee on Nuclear Waste, Dade W. Moeller, Chairman
Advisory Committee on Reactor Safeguards, Carlyle Michelson, Chairman
Atomic Safety & Licensing Appeal Panel, Christine N. Kohl, Chairman
Atomic Safety & Licensing Board Panel, B. Paul Cotter, Jr., Chairman

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 (as of 4-1-90)
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 the Controller, Ronald M. Scroggins, Controller
Office of Consolidation, Michael L. Springer, Director
Office of Enforcement, James Lieberman, Director
Office of Information Resources Management, Joyce A. Amenta, 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

Regional Offices

Region I—Philadelphia, Pa., William T. Russell, 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 of1954,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 ofwhom 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. NRR 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 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 Governmental and Public Affairs maintains communications between the NRC and governmental entities at all levels within the United States, and with the nations and organizations that make up the international nuclear community; in the latter area, the office coordinates and licenses export-import activity. The office also administers the agency's program of public information.

The Office of the Inspector General, by mandate of the Inspector General Act Amendments of 1988 (P.L. 100-504), was established effective April 15, 1989, supplanting the former Office of Inspector and Auditor. Investigations and audits conducted by the office 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 the Licensing Support System Administrator is responsible for ensuring that the Agency 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 the Secretary provides general management-services to support the Commission and to implement Commission decisions, advises and assists the Commission and staff on the planning, scheduling and conduct of Commission business; prepares for and records Commission meetings; manages the Commission staff paper system and monitors the status of all items requiring action; integrates automated data processing and office automation initiatives into the Commission's administrative system, maintains a forecast of matters for future Commission consideration; processes and controls Commission correspondence; maintains the Commission's official records; maintains the official Commission adjudicatory and rulemaking dockets and serves Commission issuances in all adjudicatory matters and public proceedings; administers the NRC Historical Program; and directs and administers the NRC Public Document Room.

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 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 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 Enforcement develops policies and programs for the enforcement of NRC requirements, manages major enforcement actions, and 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.

OTHER ORGANIZATIONAL ELEMENTS

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.

The Advisory Committee on Reactor Safeguards is a statutory committee of 15 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 the ACRS Fellowship Program.

The Atomic Safety and Licensing Appeal Panel is a panel from which three-member Appeal Boards are selected to exercise the authority and perform the review functions which would otherwise be carried out by the Commission in certain licensing proceedings. Licensing Board decisions are reviewable by an Appeal Board, either in response to an appeal or on its own initiative. The Appeal Board's decision is also subject to review by the Commission in response to an appeal for discretionary review or on its own initiative.

The Atomic Safety and Licensing Board Panel is a panel of lawyers and others with expertise in various technical fields from which three-member censing 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.

Appendix 2

NRC Committees and Boards

Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS) 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 1989, the members were:

CHAIRMAN: DR. FORREST J. REMICK, Associate Vice-President for Research and Professor of Nuclear Engineering, The Pennsylvania State University, University Park, Pa. (Note: Dr. Remick was appointed to the Nuclear Regulatory Commission in December 1989, filling the vacancy created when the term of Chairman Lando W. Zech, Jr., ended in July 1989.)

VICE-CHAIRMAN: MR. CARLYLE MICHELSON, retired Principal Nuclear Engineer,Tennessee Valley Authority, Knoxville, Tennessee, and retired Director, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.

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 of Nuclear Engineering, University of Michigan, Ann Arbor, Mich.

DR. HAROLD W. LEWIS, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.

DR. PAUL G. SHEWMON, Professor, Metallurgical Engineering Department, Ohio State University, Columbus, Ohio.

DR. CHESTER P. SIESS, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.

MR. DAVID A. WARD, Research Manager, retired, E.I. du Pont de Nemours & Company, Savannah River Laboratory, and Consulting Engineer, North Augusta,S.C.

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., ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Executive) ROBERT M. LAZO, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Technical) FREDERICK J. SHON, ASLBP Engineer, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE GEORGE C. ANDERSON, Marine Biologist, University of Washington, Seattle, Wash.

JUDGE CHARLES BECHHOEFER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE PETER B. BLOCH, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE GLENN O. BRIGHT, ASLBP Engineer (retired), U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE A. DIXON CALLIHAN, Physicist (retired), Union Carbide Corporation, Oak Ridge, Tenn.

JUDGE JAMES H. CARPENTER, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE RICHARD F. COLE, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE GEORGE A. FERGUSON, Engineer (retired), Washington, D.C.

JUDGE HARRY FOREMAN, Medical Doctor (retired), University of Minnesota, Minneapolis, Minn.

JUDGE RICHARD F. FOSTER, Environmental Scientist, Sunriver, Ore.

JUDGE JOHN H. FRYE, III, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE JAMES P. GLEASON, Attorney, Silver Spring, Md.

JUDGE CADET H. HAND, JR., Marine Biologist, University of California, Bodega Bay, Cal.

JUDGE JERRY HARBOUR, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE DAVID L. HETRICK, Nuclear Engineer, University of Arizona, Tucson, Ariz.

JUDGE ERNEST E. HILL, Nuclear Engineer, Hill Associates, Livermore, Cal.

JUDGE FRANK F. HOOPER, Marine Biologist (retired), University of Michigan, Ann Arbor, Mich.

JUDGE HELEN F. HOYT, Attorney (retired), U.S. Nuclear Regulatory Commission, Bethesda, Md.

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 MICHAEL A. KIRK-DUGGAN, Economist, University of Texas, Austin, Tex.
JUDGE JERRY R. KLINE, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JAMES C. LAMB, III, Environmental Engineer, George Washington University, Washington, D.C.
JUDGE GUSTAVE A. LINENBERGER, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE EMMETH A. LUEBKE, ASLBP Physicist (retired), U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MORTON B. MARGULIES, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE KENNETH A. McCOLLOM, Electrical Engineer (retired), Oklahoma State University, Stillwater, Okla.
JUDGE GARY L. MILHOLLIN, Attorney, Catholic University of America, Washington, D.C.
JUDGE MARSHALL E. MILLER, Attorney (retired), U.S. Nuclear Regulatory Commission, Daytona Beach, Fla.
JUDGE OSCAR H. PARIS, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission Bethesda, Md.
JUDGE DAVID R. SCHINK, Oceanographer, Texas A&M University, College Station, Tex.
JUDGE IVAN W. SMITH, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MARTIN J. STEINDLER, Chemist, Argonne National Laboratory, Argonne, Ill.
JUDGE SEYMOUR WENNER, Administrative Law Judge (retired), Postal Rate Commission, Chevy Chase, Md.
JUDGE SHELDON J. WOLFE, Attorney (retired), U.S. Nuclear Regulatory Commission, Bethesda, Md.

PROFESSIONAL STAFF:

C. SEBASTIAN ALOOT, 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.
CHARLES N. KELBER, ASLBP Senior Technical Advisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JACK G. WHETSTINE, Hearing Support Supervisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Atomic Safety and Licensing Appeal Panel

On September 18, 1969, an Atomic Safety and Licensing Appeal Board was established and delegated authority to review decisions of Atomic Safety and Licensing Boards in certain licensing proceedings that would otherwise have been reviewed by the Atomic Energy Commission (AEC). Due to an increase in the number of proceedings subject to administrative appellate review, on October 25, 1972, the AEC established the Atomic Safety and Licensing Appeal Panel, from whose membership three-person Appeal Boards could be designated. Pursuant to subsection 201(g)(1) of the Energy Reorganization Act of 1974, the functions performed by Appeal Boards were transferred to the Nuclear Regulatory Commission. The Commission appoints members to the Appeal Panel, and the Chairman of the panel establishes one or more three-member Appeal Boards for each proceeding. Appeal Board review authority encompasses formal adjudicatory proceedings involving the licensing of commercial power reactors (including license amendments) and enforcement actions such as show-cause and civil penalty proceedings. In 1989, the Commission also delegated review authority to Appeal Boards in informal adjudications concerned with so-called materials licenses. The Commission retains discretionary review authority over decisions and actions of Appeal Boards.

As of September 30, 1989, the Appeal Panel was composed of the following members and professional support staff.

FULL-TIME MEMBERS:

CHRISTINE N. KOHL, Appeal Panel Chairman, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS S. MOORE, Appeal Panel Member (Legal), U.S. Nuclear Regulatory Commission, Bethesda, Md.
HOWARD A. WILBER, Appeal Panel Member (Technical), U.S. Nuclear Regulatory Commission, Bethesda, Md.
G. PAUL BOLLWERK, III, Appeal Panel Member (Legal), U.S. Nuclear Regulatory Commission, Bethesda, Md.

PART-TIME MEMBERS:

ALAN S. ROSENTHAL, Attorney, Kensington, Md.
DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

PROFESSIONAL STAFF:

STEPHEN M. GOLDBERG, Technical Advisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Advisory Committee on Nuclear Waste

The Advisory Committee on Nuclear Waste was established by the Commission to provide advice on all aspects of nuclear waste management within the purview of NRC responsibility.

As of September 30, 1989, 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. CLIFFORD V. SMITH, JR., Chancellor, University of Wisconsin, Milwaukee, Wis.
DR. MARTIN J. STEINDLER, Director, Chemical Technology Division, Argonne National Laboratory, Argonne, Ill.

Advisory Committee on 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 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 30, 1989, the members were:

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, 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. GERALD M. POHOST, Director, Division of Cardiovascular Disease, University of Alabama at Birmingham, Birmingham, Ala.
DR. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.

In addition, four candidates have been nominated by the Commission:

DR. PETER K. LEICHNER, Associate Professor of Oncology, Johns Hopkins University, Baltimore, Md.
DR. CAROL S. MARCUS, Director, Nuclear Medicine Outpatient Clinic, Los Angeles County Harbor—UCLA Medical Center, Torrance, Cal.
MS. JOAN A. McKEOWN, R.T., Radiation Safety Technologist, Presbyterian-University of Pennsylvania Medical Center, Philadelphia, Pa.
DR. BARRY A. SIEGEL, Professor of Medicine, Washington University School of Medicine, St. Louis, Mo.

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 afford State officials the opportunity to participate in the Commission's decision-making process regarding cleanup of the damaged nuclear facility. The panel consists of the following members representing agencies of the Commonwealth of Pennsylvania, local government officials, the scientific community, and persons having their principal place of residence in the vicinity of the Three Mile Island nuclear power plant.

ARTHUR E. MORRIS, Panel Chairman, 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. Hershey Medical Center, Hershey, Pa.
FREDERICK S. RICE, Resident of Harrisburg, Pa.
GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.
JOEL ROTH, resident of Harrisburg, Pa.
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.

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 each 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. Some of these are temporary ''mini-LPDRs,'' designated ''(temp.),'' which maintain selected data collections for a limited time, usually in support of an NRC hearing procedure. (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

- Mrs. Maude S. Miller,
  Head Librarian
  Athens Public Library
  South Street
  Athens, Ala. 35611
  Browns Ferry nuclear plant
  Browns Ferry low-level waste storage

- Ms. Yvonne Cooper,
  Reference Librarian
  Houston-Love Memorial Library
  212 W. Burdeshaw Street
  P.O. Box 1369
  Dothan, Ala. 36302
  Joseph M. Farley nuclear plant

- Ms. Nancy Stover
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte nuclear plant

ARIZONA

- Ms. Stefanie Moritz, Librarian II
  Business and Science Division
  Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde nuclear plant

ARKANSAS

- Mrs. Delores Pollard,
  Serials Librarian
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One nuclear plant

CALIFORNIA

- Ms. Cecelia Riddle
  Chatsworth Branch Library
  21052 Devonshire Street
  Chatsworth, Cal. 91311
  Rockwell International

- Ms. Margaret J. Nystrom
  Documents Librarian
  Eureka-Humboldt County Library
  421 I 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 (temp.)
  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

- Miss Shirley Soenksen
  Greeley Public Library
  City Complex Building
  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

- Ms. Carolyn Greene
  Waterford Public Library
  49 Rope Ferry Road (temp.)
  Waterford, Conn. 06385
  Millstone nuclear plant

FLORIDA

- Ms. Julie DeBusk
  Coastal Region Library
  8619 W. Crystal Street
  Crystal River, Fla. 32629
  Crystal River nuclear plant

- Ms. Jimmie Anne Nourse, Librarian
  Charles S. Miley Learning Resources Ctr.
  Indian River Community College
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie nuclear plant

- Ms. Karlinne Wulf, Librarian
  Miami-Dade Public Library
  Homestead Branch
  700 North Homestead Blvd. (temp.)
  Homestead, Fla. 33030
  Turkey Point 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

- Mrs. Wynell Bush, Librarian
  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 (temp.)
  Byron, Ill. 61010
  Byron nuclear plant

- Ms. Susan Bekaires
  University of Illinois Documents Library
  1408 W. Gregory (temp.)
  Urbana, Ill. 61801
  Clinton nuclear plant

- Mrs. Malinda Evans
  Vespasian Warner Public Library
  120 West Johnson Street
  Clinton, Ill. 61727
  Clinton nuclear plant

- Mr. Earl R. Shumaker, Head
  Government Publications Department
  Founder's Memorial Library (temp.)
  Northern Illinois University
  DeKalb, Ill. 60115
  Byron 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 Trotter
  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. Jean Beeman
  Business, Science and Technology Dept.
  Rockford Public Library
  215 North Wyman Street
  Rockford, Ill. 61101
  Byron nuclear plant

- Ms. Nancy Barbour, Librarian
  Government Documents Collection
  Wilmington Public Library
  201 South Kankakee Street
  Wilmington, Ill. 60481
  Braidwood nuclear plant

- Mrs. Gail Dever
  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, Reference Librarian
  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 Division
  William Allen White Library
  Emporia State University
  1200 Commercial Street
  Emporia, Kans. 66801
  Wolf Creek Generating Station

- Mr. David Ensign, Assistant Director
  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

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
  Fourth Street
  P.O. Box 405
  Prince Frederick, Md. 20678
  Calvert Cliffs nuclear plant

MASSACHUSETTS

- Mrs. Margaret E. Howland, Director
  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
  11 North Street
  Plymouth, Mass. 02360
  Pilgrim nuclear plant

MICHIGAN

- Dr. Carol Juth, Reference Librarian
  Van Wylen Library
  Hope College
  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
  Monroe, Mich. 48161
  Enrico Fermi nuclear plant

- Ms. Bea Rodgers, Library Assistant
  Maude 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

MISSISSIPPI

- Mrs. Gayle Keefe
  Library Technical Assistant
  George M. McLendon Library
  Hinds Community College
  Main Street
  Raymond, Miss. 39154
  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
  Skinker and Lindell Boulevards
  St. Louis, Mo. 63130
  Callaway nuclear plant

NEBRASKA

- Mrs. Trudy Peaslee
  Auburn Public Library
  1118 15th Street
  P.O. Box 324
  Auburn, Neb. 68305
  Cooper nuclear plant

- Mr. Patrick R. Esser, Librarian
  Business, Science and Technology Dept.
  W. Dale Clark Library
  215 S. 15th Street
  Omaha, Neb. 68102
  Fort Calhoun nuclear plant

NEVADA

- Ms. Susan Jarvis, Head
  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

- 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, Director
  Exeter Public Library
  Founders Park
  Exeter, N.H. 03833
  Seabrook nuclear plant

NEW JERSEY

- Mrs. Amy Allen, Librarian
  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. Erick 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 (temp.)
  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. Valerie Rogers,
  Library Assistant
  Sallisaw City 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

- Mr. John E. Geschwindt, Head
  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. Sharon Reilly
  Apollo Memorial Library
  219 N. Pennsylvania Avenue
  Apollo, Pa. 15613
  Babcock & Wilcox Parks Township
  and B&W Apollo

- Mr. Janette Neal
  Assistant Department Head
  Government Publications
  Department
  Free Library of Philadelphia
  19th and Vine Streets (temp.)
  Philadelphia, Pa. 18103
  Limerick nuclear plant

- Mrs. Julia Albright
  Interlibrary Loan Librarian
  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. Diane H. Smith, Head
  Government Documents
  Pattee Library (temp.)
  Room C 207
  Pennsylvania State University
  University Park, Pa. 16802
  Beaver Valley nuclear plant
  Susquehanna steam electric
  station

- 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. Mary Toll, Reference Librarian
  Technical Services Department
  South Carolina State Library
  1500 Senate Street (temp.)
  Columbia, S.C. 29201
  Catawba nuclear plant
- Ms. Virginia Warr, 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 McCall, Librarian
  Oconee County Library
  501 W. South Broad Street
  Walhalla, S.C. 29691
  Oconee nuclear plant
- Ms. Sarah D. McMaster, Director
  Fairfield County Library
  Garden and Washington Streets
  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. Pamela A. Morris, Head
  Library—Documents
  University of Texas at Arlington
  701 South Cooper (temp.)
  P.O. Box 19497
  Arlington, Tex. 76019
  Comanche Peak steam electric station
- Mr. Tim Wilder, Library Aide
  Austin History Center
  Austin Public Library
  810 Guadalupe Street (temp.)
  P.O. Box 2287
  Austin, Tex. 78701
  South Texas Project
- Ms. Peggy Oldham
  Librarian
  Glen Rose-Somervell Library
  Barnard and Highway 144
  P.O. Box 417
  Glen Rose, Tex. 76043
  Comanche Peak steam electric station
- Mr. John R. Deosdade
  Documents Librarian
  Business and Science Dept.
  San Antonio Public Library
  203 S. St. Mary's Street (temp.)
  San Antonio, Tex. 78205
  South Texas Project
- 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 Truhler
  Reference Librarian
  Richland Public Library
  Swift and Northgate Streets
  Richland, Wash. 99352
  WPPSS Nuclear Projects 1, 2, & 4
  Richland low-level waste burial site

WISCONSIN

- Mrs. Kathy Pletcher, Head
  Government Documents Section
  Library Learning Center
  University of Wisconsin
  2420 Nicolet Drive
  Green Bay, Wis. 54301
  Kewaunee nuclear plant
- Ms. Noreen Fish
  Reference Librarian
  LaCrosse Public Library
  800 Main Street
  LaCrosse, Wis. 54601
  LaCrosse nuclear plant
- Ms. Linda Tebo
  Adult Services Assistant
  Joseph Mann Library
  1516 16th Street
  Two Rivers, Wis. 54241
  Point Beach nuclear plant

Appendix 4

Regulations and Amendments—Fiscal Year 1989

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Licensee Announcements of Inspectors—Part 50

On October 25, 1988 (53 FR 42939), the NRC published an amendment to its regulations to ensure that the presence of NRC inspectors on nuclear power reactor sites is not widely communicated or broadcast to licensee and contractor personnel without the expressed request to do so by the inspector. The amendment, effective immediately, will allow the NRC inspectors, badged at the facility, to observe ongoing activities as they are being performed without advanced notification of the inspection to licensee and contractor personnel.

Relocation of NRC's Public Document Room; Other Minor Nomenclature Changes— Parts 1, 2, 7, 9, 15, 19, 20, 21, 30, 35, 40, 50, 51, 53, 55, 60, 61, 70, 71, 72, 73, 74, 75, 81, 100, 110, 140, 150, 170, and 171

On October 27, 1988 (53 FR 43419), the NRC published an amendment to its regulations to indicate that its Public Document Room has moved to a new location in the District of Columbia. These amendments, effective immediately, also make minor nomenclature changes in NRC organization to reflect new internal organizational titles.

Safeguards Requirements for Fuel Facilities Possessing Formula Quantities of Strategic Special Nuclear Material—Parts 2, 70, and 73

On November 10, 1988 (53 FR 45447), the NRC published an amendment to its physical protection and security personnel performance regulations and its design basis threat for fuel facilities possessing formula quantities of strategic special nuclear material to require protection equivalent to that in place at comparable Department of Energy fuel facilities. The amendment, effective December 12, 1988, provides greater assurance that physical protection measures at these fuel facilities provide adequate protection against theft.

Alternative Method for leakage Rate Testing—Part 50

On November 15, 1988 (53 FR 45890), the NRC published an amendment to its regulations, effective immediately, to modify the requirements applicable to the leakage testing of containments of light-water-cooled nuclear power plants.

Revision of Fee Schedules—Parts 170 and 171

On December 29, 1988 (53 FR 52632), the NRC published an amendment to its regulations that revises its fee schedules contained in 10 CFR Parts 170 and 171. As a result of the amendment, effective January 30, 1989, the power reactor, fuel cycle facility, and materials applicants and licensees who require the greatest expenditure of NRC resources will pay the greatest fees.

Reorganization of Functions Within the Office of Administration and Resources Management and Minor Corrective Amendments—Parts 1, 2, 9, and 73

On December 30, 1988 (53 FR 52993), the NRC published an amendment to its regulations to codify nomenclature changes required by a reorganization of NRC staff activities within the Office of Administration and Resources Management. The amendments, effective immediately, are necessary to reflect the reorganization of functions reporting to the Deputy Director for Administration.

Criteria and Procedures for Emergency Access to Non-Federal and Regional Low-Level Waste Disposal Facilities—Part 62

On February 3, 1989 (54 FR 5409), the NRC published an amendment to its regulations, effective March 6, 1989, establishing criteria and procedures for fulfilling its responsibilities associated with acting on requests by low-level radioactive waste generators, or State officials on behalf of those generators, for emergency access to operating, non-Federal or regional, low-level radioactive waste disposal facilities under section 6 of the Low-Level Radioactive Waste Policy Amendments Act of 1985.

Centralization of Material Control and Accounting Licensing and Inspection Activities for Non-Reactor Facilities—Parts 70 and 74

On February 15, 1989 (54 FR 6876), the NRC published an amendment to its regulations, effective immediately, to reflect a management action to centralize material control and accounting licensing and inspection activities in NRC Headquarters, Rockville, Md., for non-reactor facilities.

Licensee Action During National Security Emergency—Part 50

On February 17, 1989 (54 FR 7178), the NRC published an amendment to its regulations to allow a licensee to take action that departs from approved technical specifications in a national security emergency. The amendment, effective March 20, 1989, is necessary to specify in the regulations that for a national security emergency a licensee is permitted to take a needed action although it may deviate from technical specifications.

Issuance or Amendment of Power Reactor License or Permit Following Initial Decision—Part 2

On February 23, 1989 (54 FR 7756), the NRC published an amendment to its regulations that specifies when a license, permit, or amendment can be issued following an initial adjudicatory decision resolving all issues before the presiding officer in favor of authorizing the licensing action. The amendment, effective March 27, 1989, deletes outdated language in the existing regulation emanating from Three Mile Island-related regulatory policies upon which action has now been completed. This action is necessary to clarify existing procedures.

Informal Hearing Procedures for Materials Licensing Adjudications—Part 2

On February 28, 1989 (54 FR 8269), the NRC published an amendment to its regulations, effective March 30, 1989, to provide rules of procedure for the conduct of informal adjudicatory hearings in materials licensing proceedings. The Atomic Energy Act of 1954 requires that the NRC afford an interested person, upon request, a "hearing" in any proceeding for the granting, suspending, revoking, or amending of an NRC license, including a license involving source, byproduct, and special nuclear materials.

Freedom of Information Act; Appeal Authority for Deputy Executive Director—Part 9

On March 10, 1989 (54 FR 10138), the NRC published an amendment to its regulations to reflect the reorganization within the Office of the Executive Director for Operations. This amendment, effective immediately, will permit a Deputy Executive Director to respond to Freedom of Information Act appeals in lieu of the Executive Director for Operations.

Extension of Time for the Implementation of the Decontamination Priority and Trusteeship Provisions of Property Insurance Requirements—Part 50

On March 17, 1989 (54 FR 11161), the NRC published an amendment to its regulations, effective immediately, amending the implementation schedule to change the effective date for the stabilization and decontamination priority and trusteeship provisions of its property insurance regulations.

Flow Control Conditions for the Standby Liquid Control System in Boiling Water Reactors—Part 50

On April 3, 1989 (54 FR 13361), the NRC published an amendment to its regulations to set forth conditions and considerations for determining reactivity control capacity for boiling water reactor standby liquid control systems. The changes, effective May 3, 1989, are necessary to clarify existing regulations.

Emergency Preparedness for Fuel Cycle and Other Radioactive Material icensees —Parts 30, 40, and 70

On April 7, 1989 (54 FR 14051), the NRC published an amendment to its regulations to require approximately 30 major NRC fuel cycle and other radioactive material licensees to maintain emergency plans for coping with serious accidents involving licensed radioactive materials for which responses by off-site response organizations (such as police, fire, and medical organizations) might be needed. This action, effective April 7, 1990, is intended to ensure that these licensees are prepared to take action to protect public health and safety if an accident occurs.

Submission and Management of Records and Documents Related to the Licensing of a Geologic Repository for the Disposal of High-Level Radioactive Waste—Part 2

On April 14, 1989 (54 FR 14925), the NRC published an amendment to its rules of practice for the adjudicatory proceeding on the application for a license to receive and possess high-level radioactive waste at a geologic repository operations area pursuant to 10 CFR Part 60. This action, effective May 15, 1989, establishes the basic procedures for the licensing proceeding, including procedures for the use of the Licensing Support System, an electronic information management system, in the proceeding.

Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors—Parts 2, 50, 51, 52, and 170

On April 18, 1989 (54 FR 15372), the NRC published an amendment to its regulations adding a new part which provides for issuance of early site permits, standard design certifications, and combined construction permits and operating licenses with conditions for nuclear power reactors. This action, effective May 18, 1989, is intended to achieve the early resolution of licensing issues and enhance the safety and reliability of nuclear power plants.

Access to Safeguards Information—Part 73

On April 25, 1989 (54 FR 17703), the NRC published an amendment to its regulations for access to Safeguards Information to be consistent with "The Omnibus Diplomatic Security and Anti-Terrorism Act of 1986." The Act requires nuclear power reactor applicants and licensees to conduct Federal Bureau of Investigation criminal history checks of certain individuals with access to information protected as Safeguards Information. This action, effective May 25, 1989, is necessary to ensure that applicable NRC regulations reflect this requirement.

Disposal of Radioactive Wastes—Part 61

On May 25, 1989 (54 FR 22578), the NRC published an amendment to its regulations to require disposal of greater-than-Class C (GTCC) low-level radioactive wastes in a deep geologic repository unless disposal elsewhere has been approved by the Commission. This action, effective June 26, 1989, is necessary to ensure that GTCC wastes are disposed of in a manner that would protect the public health and safety and eliminate the need for altering existing classifications of radioactive wastes as high-level or low-level.

Financial Protection Requirements and Indemnity Agreements; Miscellaneous Amendments Necessitated By Changes in the Price-Anderson Act—Part 140

On June 6, 1989 (54 FR 24157), the NRC published an amendment to its regulations to conform to changes made to the Price-Anderson Act by "The Price-Anderson Amendments Act of 1988," which was enacted on August 20, 1988. The NRC is also amending its regulations to increase the level of the primary layer of financial protection required of certain indemnified licensees. These amendments, effective July 1, 1989, would provide additional insurance to pay public liability claims arising out of a nuclear incident.

Fitness-For-Duty Programs—Parts 2 and 26

On June 7, 1989 (54 FR 24468), the NRC published an amendment to its regulations to require licensees authorized to construct or operate nuclear power reactors to implement a fitness-for-duty program. This amendment, effective July 7, 1989, is intended to create an environment which is free of drugs and the effects of such substances.

Manner of Service of Pleadings Upon the Secretary of the Commission—Part 2

On June 26, 1989 (54 FR 26730), the NRC published an amendment to its regulations that requires all parties in NRC proceedings to file copies of all pleadings filed with any agency adjudicatory tribunal with the Office of the Secretary in the same or equivalent manner in which they were filed with the tribunal. This amendment, effective July 26, 1989, will result in the Office of the Secretary receiving the pleading on approximately the same day as the tribunal.

Advisory Committees; Policies and Procedures—Part 7

On June 27, 1989 (54 FR 26947), the NRC published an amendment to its regulations that defines the policies and procedures to be followed by the NRC in the establishment, utilization, and termination of advisory committees. This amendment, effective immediately, is intended to reflect administrative and management changes that have taken place since NRC's regulations were published in 1975 and to maintain consistency between NRC regulations and those issued by the General Services Administration in 1987.

NEPA Review Procedures for Geologic Repositories for High-Level Waste— Parts 2, 51, and 60

On July 3, 1989 (54 FR 27864), the NRC published an amendment to its regulations to adopt procedures for the implementation of the National Environmental Policy Act with respect to geologic repositories for high-level radioactive waste. In accordance with the Nuclear Waste Policy Act of 1982, as amended, the Commission will adopt, to the extent practicable, the final environmental impact statement prepared by the Department of Energy that accompanies a recommendation to the President for repository development. The amendment, effective August 2, 1989, sets out the standards and procedures that would be used in determining whether adoption of the Department's final environmental impact statement is practicable.

Rules of Practice for Domestic Licensing Proceedings—Procedural Changes in the Hearing Process—Part 2

On August 11, 1989 (54 FR 33168), the NRC published an amendment to its regulations to improve the hearing process with due regard for the rights of the parties. The amendment, effective September 11, 1989, requires a person seeking to participate as a party in an NRC proceeding to file a list of contentions with the presiding officer together with a brief explanation of the bases for each contention, a concise statement of the alleged facts or expert opinion that support the contention and which, at the time of the filing, the person intends to rely upon in supporting the contention at the hearing, and references to the specific sources and documents of which the person is aware and upon which he or she intends to rely to establish such facts or expert opinions.

Duplication Fees—Part 9

On September 5, 1989 (54 FR 36757), the NRC published an amendment to its regulations to revise the charges for copying records publicly available at the NRC Public Document Room in Washington, DC. 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.

REGULATIONS AND AMENDMENTS PROPOSED

Debt Collection Procedures—Part 15

On October 7, 1988 (53 FR 39480), the NRC published a notice of proposed rulemaking that would amend its regulations concerning the procedures that the NRC uses to collect the debts which are owed to it. The proposed action is intended to allow the NRC to further improve its collection of debts due the United States.

Flow Control Conditions for the Standby Liquid Control System in Boiling Water Reactors—Part 50

On October 24, 1988 (53 FR 41607), the NRC published a notice of proposed rulemaking that would amend its regulations concerning the flow control conditions for the standby liquid control system in a boiling water reactor. The proposed action would set forth conditions and considerations for determining the reactivity control capacity of a BWR standby liquid control system.

Rule on the Submission and Management of Records and Documents Related to the icensing of a Geologic Repository for the Disposal of High-Level Radioactive Waste—Part 2

On November 3, 1988 (53 FR 44411), the NRC published a notice of proposed rulemaking that would amend its rules of practice for the adjudicatory proceeding on the application for a license to receive and possess high-level radioactive waste at a geologic repository operations area pursuant to 10 CFR Part 60. The proposed revisions would establish the basic procedures for the licensing proceeding, including procedures for the use of the Licensing Support System, an electronic information management system, in the proceeding.

Criteria and Procedures for the Reporting of Defects—Parts 21 and 50

On November 4, 1988 (53 FR 41594), the NRC published a notice of proposed rulemaking that would amend its regulations on the reporting of safety defects. The proposed amendments would eliminate duplicative reporting of defects, clarify the criteria for reporting defects, and would establish uniform time periods for reporting and uniform requirements for the content of reports of defects.

Sequestration of Witnesses Interviewed Under Subpoena—Part 19

On November 14, 1988 (53 FR 45768), the NRC published a notice of proposed rulemaking that would amend its regulations to provide that all persons compelled to appear before NRC representatives under subpoena in connection with an agency investigation (and their counsel, if any) will, unless otherwise authorized by the NRC official conducting the investigation, be sequestered from other interviewees in the same investigation. The proposed rule is intended to clarify and delineate the rights and responsibilities of the agency, interviewees, and licensees during the conduct of agency investigations and inspections.

Ensuring the Effectiveness of Maintenance Programs for Nuclear Power Plants—Part 50

On November 28, 1988 (53 FR 47822), the NRC published a notice of proposed rulemaking that would amend its regulations to require commercial nuclear power plant licensees to strengthen their maintenance activities in order to reduce the likelihood of failures and events caused by the lack of effective maintenance. The proposed rule would require plant maintenance programs to include specific activities, including the monitoring of the effectiveness of plant maintenance programs.

Financial Protection Requirements and Indemnity Agreements; Miscellaneous Amendments Necessitated by Changes in the Price-Anderson Act—Part 140

On December 20, 1988 (53 FR 51120), the NRC published a notice of proposed rulemaking that would amend its regulations to conform to changes made to the Price-Anderson Act by "The Price-Anderson Amendments Act of 1988," which was enacted on August 20, 1988. The NRC is also proposing to amend its regulations to increase the level of the primary layer of financial protection required of certain indemnified licensees. This proposed change would provide additional insurance to pay public liability claims arising out of a nuclear incident.

Education and Experience Requirements for Senior Reactor Operators and Supervisors at Nuclear Power Plants—Parts 50 and 55

On December 29, 1988 (53 FR 52716), the NRC published a notice of proposed rulemaking that would amend its regulations concerning operating personnel at nuclear power plants. The proposed amendment would require additional education and experience requirements for senior operators and supervisors. In consideration of the comments received on this proposed rule and the status of industry initiatives to enhance the education level of its operating personnel, the Commission concluded that the proposed rule should be withdrawn. The notice of withdrawal was published in the Federal Register on August 15, 1989 (54 FR 33568).

Enforcement of Nondiscrimination on the Basis of Handicap in Federally Assisted Programs—Part 4

On March 8, 1989 (54 FR 9966), the NRC published a notice of proposed rulemaking that would amend its regulations concerning enforcement of section 504 of the Rehabilitation Act of 1973, as amended, in Federally assisted programs or activities to include a cross-reference to the Uniform Federal Accessibility Standards.

Palladium-103 for Interstitial Treatment of Cancer—Part 35

On April 6, 1989 (54 FR 13892), the NRC published a notice of proposed rulemaking that would amend its regulations governing the medical use of byproduct material. This proposed amendment would add palladium-103 as a sealed source in seeds to the list of brachytherapy sources permitted for use in the treatment of cancer.

Informal Hearing Procedures for Nuclear Reactor Operator Licensing Adjudications—Part 2

On April 26, 1989 (54 FR 17961), the NRC published a notice of proposed rulemaking that would amend its regulations to provide rules of procedure for the conduct of informal adjudicatory hearings in nuclear reactor operator licensing proceedings. This proposed amendment would include reactor operator licensing proceedings under the informal hearing procedures already established for materials licensing proceedings.

Storage of Spent Nuclear Fuel in NRC-Approved Storage Casks at Nuclear Power Reactor Sites—Parts 50, 72, and 170

On May 5, 1989 (54 FR 19379), the NRC published a notice of proposed rulemaking that would amend its regulations to provide, as directed by the Nuclear Waste Policy Act of 1982, for the storage of spent fuel at the sites of power reactors without, to the maximum extent practicable, the need for additional site-specific approvals. This proposed amendment contains criteria for obtaining an NRC Certificate of Compliance for spent fuel storage casks.

Preserving the Free Flow of Information to the Commission—Parts 30, 40, 50, 60, 70, 72, and 150

On July 18, 1989 (54 FR 30049), the NRC published a notice of proposed rulemaking that would amend its regulations governing the conduct of all Commission licensees and license applicants. This proposed rule would prohibit the use of provisions which would inhibit the free flow of safety information to the Commission in agreements related to employment.

Minor Amendments to the Physical Protection Requirements—Parts 70, 72, 73, and 75

On August 15, 1989 (54 FR 33570), the NRC published a notice of proposed rulemaking that would amend its regulations that cover the physical protection of special nuclear material. The proposed amendments are necessary to reflect the results of a systematic review of NRC's safeguards regulations.

Credit Checks; Expanded Personnel Security Investigative Coverage—Parts 11, 25, and 95

On September 21, 1989 (54 FR 38863), the NRC published a notice of proposed rulemaking that would amend its regulations to (1) expand the investigative scope for license "R" special nuclear material access authorization and "L" security clearance applicants by adding a credit check; and (2) revise the corresponding fee schedules to recover the additional cost of each credit check. This proposed amendment would achieve a higher degree of assurance that licensee "R" and "L" applicants are reliable, trustworthy, and do not have any significant financial problems which may cause them to be susceptible to pressure, blackmail, or coercion to act contrary to the national interest.

Procedures Applicable to Proceedings for the Issuance of Licenses for the Receipt of High-level Radioactive Waste at a Geologic Repository—Part 2

On September 26, 1989 (54 FR 39387), the NRC published a notice of proposed rulemaking that would amend its regulations for the licensing proceeding on the disposal of high-level radioactive waste at a geologic repository (HLW proceeding). This proposed amendment would facilitate 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, while providing for a thorough technical review of the license application and the equitable treatment of the parties to the hearing.

Consideration of Environmental Impacts of Temporary Storage of Spent Fuel After Cessation of Reactor Operation—Part 51

On September 28, 1989 (54 FR 39765), the NRC published a notice of proposed rulemaking that would amend its regulations concerning its generic determinations on the timing of availability of a geologic repository for commercial high-level radioactive waste and spent fuel and the environmental impacts of storage of spent fuel at reactor sites after the expiration of reactor operating licenses. This proposed amendment reflects proposed findings of the Commission reached in a five-year update and supplement to its 1984 "Waste Confidence" rulemaking proceeding, which was published on September 28, 1989 (54 FR 39767).

ADVANCE NOTICES OF PROPOSED RULEMAKING

Indemnification of Licensees that Manufacture, Produce, Possess, or Use Radiopharmaceuticals or Radioisotopes for Medical Purposes—Part 140

On October 14, 1988 (53 FR 40233), the NRC published a notice of intent establishing the schedule and format for a negotiated rulemaking proceeding. Section 19 of the Price-Anderson Amendments Act of 1988 requires the NRC to conduct a "negotiated rulemaking" to determine whether to enter into indemnity agreements with persons licensed by the Commission or by an Agreement State for the manufacture, production, possession, or use of radioisotopes or radiopharmaceuticals for medical purposes. The convenor of the negotiated rulemaking proceeding recommended that the NRC not extend the Price-Anderson indemnification to radiopharmaceutical licensees, and therefore terminate the rulemaking proceeding. The Federal Register notice terminating this rulemaking proceeding was published in the Federal Register on May 24, 1989 (54 FR 22444).

Acceptance of Products Purchased for Use in Nuclear Power Plant Structures, Systems and Components—Part 50

On March 6, 1989 (54 FR 9229), the NRC published an advance notice of proposed rulemaking (ANPRM) announcing its intent to develop regulations requiring enhanced acceptance procedures including, but not limited to, receipt inspection and testing of products purchased for use in nuclear power plant structures, systems and components. This ANPRM is intended to solicit comments on the need for additional regulatory requirements and to obtain an improved understanding of alternatives to regulatory requirements that could provide assurance that structures, systems and components procured for use in nuclear power plants will perform as expected to protect public health and safety.

Appendix 5

Regulatory Guides—Fiscal Year 1989

NRC regulatory guides describe methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations and, 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. 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, 1988, to September 30, 1989.

Division 1—Power Reactor Guides

1.74 Withdrawn. Quality Assurance Terms and Definitions

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 26)

1.85 Materials Code Case Acceptability—ASME Section III, Division 1 (Revision 26)

1.114 Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Revision 2)

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 7)

1.157 Best-Estimate Calculations of Emergency Core Cooling System Performance

1.158 Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants

Division 2—Research and Test Reactor Guides

None

Division 3—Fuels and Materials Facilities Guides

3.44 Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type) (Revision 2)

3.45 Nuclear Criticality Safety for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Materials (Revision 1)

3.48 Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage) (Revision 1)

3.50 Standard Format and Content for a License Application To Store Spent Fuel and High-Level Radioactive Waste (Revision 1)

3.61 Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask

3.62 Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks

3.64 Calculation of Radon Flux Attenuation by Earthen Uranium Mill Tailings Covers

3.65 Standard Format and Content of Decommissioning Plans for Licensees Under 10 CFR Parts 30, 40, and 70

Division 4—Environmental and Siting Guides

None

Division 5—Materials and Plant Protection Guides

None

Division 6—Product Guides

None

Division 7—Transportation Guides

7.8 Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material (Revision 1)

Division 8—Occupational Health Guides

8.12 Criticality Accident Alarm Systems (Revision 2)

Division 9—Antitrust and Financial Review Guides

None

Division 10—General Guides

10.9 Guide for the Preparation of Applications for Licenses for the Use of Self-Contained Dry Source-Storage Irradiators (Revision 1)

DRAFT GUIDES

Division 1

DG-1001 Maintenance Programs for Nuclear Power Plants

DG-1003 Assuring the Availability of Funds for Decommissioning Nuclear Reactors

DG-1005 Standard Format and Content for Decommissioning Plans for Nuclear Reactors

DG-1006 Records Important for Decommissioning of Nuclear Reactors

RS 802-5 Proposed Revision 3 to Regulatory Guide 1.9, Selection, Design, Qualification, Testing, and Reliability of Diesel Generator Units Used as On-site Electric Power Systems at Nuclear Power Plants

SC 708-4 Withdrawn. Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety

Division 3

DG-3001 Records Important for Decommissioning for Licensees Under 10 CFR Parts 30, 40, 70, and 72

Division 7

DG-7001 Fracture Toughness Criteria for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of Four Inches (0.1 m)

DG-7002 Fracture Toughness Criteria for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than Four Inches (0.1 m)

Appendix 6

Civil Penalties and Orders—Fiscal Year 1989

CIVIL PENALTY ACTIONS IN FISCAL YEAR 1989 (Organized according to EA number)

Licensee	*Action*	*Summary*
Advanced Medical Systems (Geneva, Ohio) EA 85-060	$6,250 proposed in FY 85; imposed in FY 89; pending	Violations involving significant weaknesses in management control of the radiation protection program.
Sacramento Municipal Utility District (Rancho Seco) EA 86-110	$100,000 proposed and paid in FY 89	Violations involving significant breakdown in the management oversight of the program to properly implement and control the plant's radioactive effluent releases.
U.S. Testing Company (Modesto, California) EA 87-052	$280,000 proposed in FY 89	Violations involving significant breakdown in management protection program.
Baltimore Gas and Electric (Calvert Cliffs) EA 87-077	$300,000 proposed in FY 88; imposed and paid in FY 89	Violations of the equipment qualification requirements.
Consumers Power Company (Big Rock Point) EA 87-080	$187,500 proposed in FY 88; imposed and paid in FY 89	Violations of the equipment qualification requirements.
Commonwealth Edison Company (Dresden) EA 87-081	$150,000 proposed in FY 88; $75,000 imposed and paid in FY 89	Violations of the equipment qualification requirements.
Iowa Electric Power & Light (Duane Arnold) EA 87-083	$50,000 proposed and paid in FY 89	Violations of the equipment qualification requirements.
H&G Inspection Company (Houston, Texas) EA 87-145	$7,500 proposed and imposed in FY 88; $3,000 paid in FY 89	Radiological overexposure to a radiographer.
Boston Edison Company (Pilgrim) EA 87-164	$50,000 proposed and paid in FY 89	Violations involving access control, corrective actions, and a deliberate material false statement.
Carolina Power & Light (Brunswick) EA 87-165	$50,000 proposed in FY 88; imposed and paid in FY 89	Violations of the equipment qualification requirements.
Precision Logging (Cleveland, Oklahoma) EA 87-184	$1,000 proposed in FY 88; $500 imposed in FY 88; paid in FY 89	Radiation safety violations involving surveys, unsecured material, posting, records, labels, papers, and storage.
G.P.U. (Oyster Creek) EA 87-185	$50,000 proposed and paid in FY 89	Violations involving the failure to maintain at least two recirculation loop discharge valves in open position.
Payne and Payne (Shawnee, Oklahoma) EA 87-205	$1,600 proposed in FY 88; paid in FY 89	Violations involving calibration, surveys, evaluation of personnel dosimetry, and leak testing and inventory of sealed sources.

Licensee	*Action*	*Summary*
Florida Power Corporation (Crystal River) EA 87-216	$100,000 proposed in FY 88; $50,000 imposed and paid in FY 89	Violations involving access control into high radiation areas, training, procedures, provision of radiation monitoring device to employee, and follow radiation control procedures.
Biomedical Diagnostic Service (Troy, Michigan) EA 87-231	$750 proposed and paid in FY 89	Failure to follow procedures concerning the use of dosimetry, testing of incoming packages, quarterly linearity tests, supervision of personnel.
Georgia Institute of Technology (Atlanta, Georgia) EA 88-032	$5,000 proposed and paid in FY 89	Violations involving failure of management to assure that procedures are followed, surveys, and evaluation of the extent of radiological hazards.
Florida Power Corporation (Crystal River) EA 88-034	$50,000 proposed in FY 88; $25,000 imposed and paid in FY 89	Failure to take appropriate corrective action.
V.A. Hospital Loma Linda (Loma Linda, California) EA 88-039	$6,500 proposed and paid in FY 89	Violations involving radiation exposures, training, surveys, dose calibrator results, and review and renewal of research projects.
United States Air Force (Wright-Patterson Air Force Base) EA 88-087	$102,500 proposed and paid in FY 89	Violations involving a significant spill of americium-241, including a willful failure to report the event and the internal exposure to an individual in excess of NRC quarterly limits.
Carolina Power & Light (H.B. Robinson) EA 88-088	$50,000 proposed in FY 88; imposed and paid in FY 89	Violations indicating little or no effort to develop and maintain a program for ensuring compliance.
Illinois Power Company (Clinton) EA 88-090	$75,000 proposed in FY 88; imposed and paid in FY 89	Failure to ensure that electrical equipment important to safety was environmentally qualified.
Commonwealth Edison Company (Braidwood) EA 88-091	$50,000 proposed in FY 88; imposed and paid in FY 89	Deficiencies involving design control which resulted in the Control Room Ventilation System being in a degraded condition.
Detroit Edison Company (Fermi) EA 88-104	$200,000 proposed in FY 88; $175,000 imposed and paid in FY 89	Violations involving containment isolation provisions for primary containment radiation monitoring system and operation of non-interruptible air system in a degraded mode, leading to the violation of two Technical Specifications.
Alabama Power Company (Farley) EA 88-113	$100,000 proposed in FY 88; $75,000 imposed in FY 89	Failure to determine system operability of the high level safety injection system upon evidence of design deficiency.
Carolina Power & Light (Brunswick) EA 88-131	$75,000 proposed in FY 88; $25,000 imposed and paid in FY 89	Violations involving leaving a control rod fully withdrawn with the reactor protection system shortinglinks installed, failure to have required system alignments, and failure to maintain proper shutdown coolant alignment.
Duke Power Company (Catawba) EA 88-132	$50,000 proposed and paid in FY 89	Violations of equipment qualification requirements.
Consumers Power Company (Palisades) EA 88-138	$150,000 proposed and paid in FY 89	Violations of requirements for environmental qualification for certain items of electrical equipment.

Licensee	Action	Summary
Consumers Power Company (Palisades) EA 88-140	$75,000 proposed and paid in FY 89	Failure to satisfy fire protection requirements and failure of management to take aggressive and timely corrective action.
Consolidated Edison Company (Indian Point 2) EA 88-142	$75,000 proposed and paid in FY 89	Violations of requirements for environmental qualification for certain items of electrical equipment.
Pennsylvania Power & Light (Susquehanna) EA 88-143	$50,000 proposed in FY 88; paid in FY 89	Violations of equipment qualification requirements.
Louisiana Power & Light (Waterford) EA 88-144	$50,000 proposed in FY 88; imposed and paid in FY 89	Event in which inacurate reactor vessel water level indication twice resulted in cavitation of and subsequent loss of the operating shutdown cooling pump.
Omaha Public Power District (Ft. Calhoun) EA 88-145	$50,000 proposed and paid in FY 89	Violations involving installation of improper check valves and failure to install a test cap on a containment pressure sensing line.
Power Authority of New York (Indian Point 3) EA 88-148	$75,000 proposed in FY 88; paid in FY 89	Violations of the equipment qualification requirements.
Carolina Power & Light (Brunswick) EA 88-149	$75,000 proposed and paid in FY 89	Failure to take prompt and adequate corrective action with respect to equipment deficiencies.
Bill Miller, Incorporated (Henryetta, Oklahoma) EA 88-155	$8,000 proposed in FY 88; $4,000 imposed in FY 89; being paid in monthly installments	Failure to maintain surveillance, post, and rope off area where industrial radiography was performed, resulting in two members of the public receiving radiation exposures.
Nebraska Public Power District (Cooper) EA 88-159	$150,000 proposed and paid in FY 89	Violations of the equipment qualification requirements.
Commonwealth Edison Company (Quad Cities) EA 88-161	$125,000 proposed in FY 88; $50,000 imposed and paid in FY 89	Violations involving a shared diesel being unable to start using an ungrounded battery system in a grounded condition.
Computalog, Incorporated (Drumright, Oklahoma) EA 88-169	$1,000 proposed in FY 88; and paid in FY 89	Violations involving a breakdown in management oversight and control over licensed activities, including four repeat violations.
Commonwealth Edison Company (Braidwood) EA 88-174	$50,000 proposed in FY 88; and paid in FY 89	Violations involving two examples of inattentiveness on the part of security guards assigned as compensatory measures.
Arizona Publice Service (Palo Verde) EA 88-182	$250,000 proposed and paid in FY 89	Violations involving an event in which both trains of the essential chilled water system were rendered inoperable, an event resulting in an individual receiving a cumulative whole body dose in excess of regulatory limits, and a breakdown in the licensee's control of high radiation areas.
Shadyside Hospital (Pittsburgh, Pennsylvania) EA 88-188	$2,500 proposed and paid in FY 89	Violations involving the failure to recover and return nuclear pacemaker to the manufacturer.
Penn Inspection (Chickasha, Oklahoma) EA 88-189	$2,500 proposed and paid in FY 89	Violations involving a source hang-up incident during radiography operations.

Licensee	Action	Summary
Arkasas Power & Light (ANO) EA 88-192	$75,000 proposed in FY 89; pending	Violations of the equipment qualification requirements.
Combustion Engineering (Windsor, Connecticut) EA 89-193	$12,500 proposed and paid in FY 89	Violations involving the performance of adequate surveys, failure of some individuals to wear gloves while handling certain contaminated equipment, measurement of U-235 content of waste materials, and shipment of radioactive waste without meeting all shipping requirements.
Commonwealth Edison Company (Braidwood) EA 88-198	$75,000 proposed and paid in FY 89	Violations involving failure to adequately implement equipment qualification requirements.
Baltimore Gas & Electric (Calvert Cliffs) EA 88-202	$150,000 proposed in FY 88; paid in FY 89	Events in which two Technical Specification Limiting Conditions for Operation were violated.
G.P.U. (Oyster Creek) EA 88-203	$50,000 proposed and paid in FY 89	Violations of the equipment qualification requirements.
Wise Appalachian Hospital (Wise, Virginia) EA 88-204	$1,250 proposed in FY 88; and paid in FY 89	Violations involving labelling syringes, syringe shields, and vial radiation shields, posting, annual review of radiation safety program, training, quarterly dose calibrator linearity tests, and written permission to visiting physician as an authorized user.
Virginia Electric & Power Company (Surry) EA 88-215	$50,000 proposed and paid in FY 89	Violations involving failure to adequately maintain cleanliness and foreign material exclusion controls.
B&W Navy (Lynchburg, Virginia) EA 88-225	$6,250 proposed and paid in FY 89	Violations involving failure to follow nuclear criticality safety double-contingency policy, control of materials, posting, conduct of activities in accordance with procedures, review and approval by Nuclear Licensing Board, and interpretations of nuclear safety limits.
Maryview Hospital (Portsmouth, Virginia) EA 88-227	$1,250 proposed and paid in FY 89	Violations involving formal annual review, test and calibration of equipment, conduct of bioassay test, conduct of radiation surveys, performance of radiation exposure rate measurements, inventory, and records.
Duke Power Company (Oconee) EA 88-228	$25,000 proposed and paid in FY 89	Failure to provide adequate procedural guidance to ensure that high pressure safety injection system would remain operable for all required accident conditions.
Davis Great Guns Logging (Wichita, Kansas) EA 88-231	$500 proposed and paid in FY 89	Violations involving inadequate control of licensed material, failure to identify and post radiation areas, to conduct semiannual equipment inspections and visual checks of logging equipment, and failure to maintain records.
Toledo Edison Company (Davis-Besse) EA 88-234	$80,000 proposed and paid in FY 89	Violations involving discrimination against individual for raising safety concerns.

Licensee	Action	Summary
Philadelphia Electric Co. (Peach Bottom) EA 88-237	$50,000 proposed and paid in FY 89	Violations involving numerous examples of failure to take compensatory measures, issuance of a vital area key to an unauthorized person, maintenance of assessment capabilities.
Public Service Electric and Gas (Salem) EA 88-238	$50,000 proposed and paid in FY 89	Violations of the equipment qualification requirements.
Duke Power Company (McGuire) EA 88-247	$37,500 proposed and paid in FY 89	Violations involving failure to ensure that containment hydrogen skimmer systems could perform their intended functions.
E. L. Conwell (Bridgeport, Pennsylvania) EA 88-248	$1,950 proposed and paid in FY 89	Violations involving failure to secure and maintain surveillance of a nuclear gauge, failure to secure source lock when not in use, transport of of gauge without adequate shipping papers, and transmittal of inaccurate information to the NRC.
Northeast Utilities (Millstone) EA 88-24	$50,000 proposed and paid in FY 89	Violations involving equipment qualification requirements.
Ford Motor Company (Sterling Height, Michigan) EA 88-255	$500 proposed and paid in FY 89	Violations involving the improper disposal of a generally licensed nuclear gauge containing one curie of americium-241.
C&R Laboratories (Pearl City, Hawaii) EA 88-256	$2,000 proposed FY 89; being paid in monthly installments	Violations involving exposure to radiation in excess of regulatory limits, use of an inoperable survey instrument, and failure to secure radiation source in shielded position.
Honolulu Medical Group (Honolulu, Hawaii) EA 88-257	$2,500 proposed and paid in FY 89	Violations involving dose calibrator accuracy tests, performance of annual audits, training, maintenance of required information on waste records, and establishment of radiation dose rate trigger levels for surveys.
Carolina Power & Light (Shearon Harris) EA 88-261	$25,000 proposed and paid in FY 89	Violations involving several failures to provide vital area barriers.
Basin Testing Labortories (Williston, North Dakota) EA 88-265	$5,000 proposed in FY 89; pending	Violations involving conduct of radiographic activities by unqualified individual, notification of NRC, transportation of radioactive materials, providing inaccurate information to NRC.
Commonwealth Edison Company (Byron) EA 88-266	$50,000 proposed and paid in FY 89	Violations involving a loss of one train of the residual heat removal system.
Florida Power & Light Company (Turkey Point) EA 88-267	$100,000 proposed and paid in FY 89	Violations involving the improper repositioning of a security guard posted as a compensatory measure.
Urban Engineers, Inc. (Erie, Pennsylvania) EA 88-274	$500 proposed and paid in FY 89	Violations involving surveillance of nuclear-density gauges, use of gauge by an individual without proper training certification, shipping papers not kept with gauge being shipped, performance of leak tests at required frequency, and maintenance of records.

Licensee	Action	Summary
Portland General Electric (Trojan) EA 88-277	$75,000 proposed and paid in FY 89	Violations involving inadequate vital area barriers and compensatory measures, badging and escort of visitor, and record of visits.
Detroit Edison Company (Fermi) EA 88-281	$50,000 proposed and paid in FY 89	Failure to establish adequate design control measures to ensure that motor operated valve torque switches were properly installed and set.
Kansas Gas & Electric Company (Wolf Creek) EA 88-282	$50,000 proposed and paid in FY 89	Event in which erosion reduced pipe wall thickness.
Arkansas Power & Light (ANO) EA 88-283	$25,000 proposed and paid in FY 89	Violations involving failure to control an individual's occupational exposure and failure to perform adequate surveys to identify radiation sources that resulted in an individual exceeding quarterly whole body radiation exposure limits.
Arkansas Power & Light (ANO) EA 88-284	$175,000 proposed and paid in FY 89	Violations involving the failure to take adequate corrective actions for various identified conditions adverse to quality and failure to properly control safety-related equipment.
Rappahannock Hospital (Kilmarnock Virginia) EA 88-287	$2,500 proposed, imposed, and paid in FY 89	Violations involving fabrication of minutes for required meeting that was not held.
MQS Inspection (Elk Grove Village, Illinois) EA 88-288	$5,000 proposed and paid in FY 89	Violations resulting from an individual receiving a dose in excess of regulatory limits and various failures associated with the radiation safety program.
AFRRI (Bethesda, Maryland) EA 88-289	$2,500 proposed, imposed, and paid in FY 89	Violations involving failure to perform written safety evaluations, training, discrimination, and adherence to procedures.
Pesara P. Reddy (Butler, Pennsylvania) EA 88-291	$1,500 proposed and paid in FY 89	Violations involving training, package receipt surveys, dose calibrator constancy and linearity tests, analysis of survey wipe samples, and notification of change of mailing address.
Maine Yankee Atomic Power (Maine Yankee) EA 88-295	$75,000 proposed and paid in FY 89	Violations involving positive control of vital area keys, lighting in the isolation zone, vehicle escort, maintenance of of the isolation zone, and personnel and package search.
Virginia Electric & Power Company (Surry) EA 88-296	$500,000 proposed and paid in FY 89	Violations involving the self-identification of deficiencies, evaluations, and corrective actions.
West Virginia University Hospital (Morgantown, West Virginia) EA 88-297	$2,500 proposed and paid in FY 89	Violations involving the control of licensed materials, inventory of radioactive sources, notification to NRC of loss of sources, and wearing of proper dosimetry.
St. Agnes Medical Center (Philadelphia, Pennsylvania) EA 88-298	$2,500 proposed, imposed, and paid in FY 89	Violations involving dose calibrator constancy checks, surveys, and use of dose calibrator after tests indicated erroneous responses.

Licensee	*Action*	*Summary*
Hemphill Corporation (Tulsa, Oklahoma) EA 88-301	$500 proposed and paid in FY 89	Violations involving allowing individuals to use gauging devices without proper training, leak tests on sealed sources, having a person serve as radiation safety officer other than the one who was named in license requirements, and storage of material.
General Electric (Wilmington, South Carolina) EA 88-302	$20,000 proposed and paid in FY 89	Violations involving discrimination against an employee for engaging in protected activities.
Power Authority of New York (Fitzpatrick) EA 88-304	$75,000 proposed and paid in FY 89	Violations involving the licensee's failure to properly analyze two potential safety issues.
South Carolina Electric & Gas (Summer) EA 88-305	$62,500 proposed and paid in FY 89	Violations involving multiple significant failures to control access to the protected area of the plant.
T.V.A. (Sequoyah) EA 88-307	$50,000 proposed, imposed, and paid in FY 89	Violations involving inadequate identification and correction of conditions adverse to quality.
St. Mary's Hospital (Richmond, Virginia)	$1,250 proposed and paid in FY 89	Violations involving written procedures to ensure no one is present in teletherapy room during testing, access control to high radiation area, and training on procedures.
Virginia Electric & Power Company (North Anna) EA 88-311	$25,000 proposed and paid in FY 89	Violations involving the failure to take corrective actions for identified deficiencies in the control room ventilation and instrument air systems.
Professional Service (Lombard, Illinois) EA 88-313	$20,000 proposed, imposed, and paid FY 89	Violations involving failure to secure or maintain continuous surveillance over an unsecured moisture-density gauge in an unrestricted area.
Advex (Hampton, Virginia) EA 88-315	$2,000 proposed and paid in FY 89	Violations involving failure to control an individual's occupational exposure within regulatory limits.
Carolina Power & Light (Brunswick) EA 88-316	$150,000 proposed and paid in FY 89	Violations involving two events that individually resulted in a loss of secondary containment integrity during fuel sipping operations.
Entela, Inc. (Grand Rapids, Michigan) EA 88-318	$1,250 proposed and paid in FY 89	Violations involving willful failure to have eight individuals trained before using moisture-density gauges without supervision, failure to maintain adequate surveillance and control over a moisture-density gauge, failure to notify NRC of event, and failure to perform required leak testing.
Brigham & Womens (Boston, Massachusetts) EA 88-319	$5,000 proposed and paid in FY 89	Violations involving failure to maintain adequate surveillance and control of a source.
Houston Light & Power Company (South Texas) EA 89-001	$50,000 proposed in FY 89; pending	Violations involving failure to install vortex suppressors prior to plant licensing.
Computerized Medical Imaging (Eau Claire, Wisconsin) EA 89-014	$2,500 proposed and paid in FY 89	Violations involving failure to secure or maintain constant surveillance and immediate control of radioactive materials in an unrestricted area.

Licensee	Action	Summary
Cornish Wireline (Chanute, Kansas) EA 89-015	$500 proposed and and paid in FY 89	Violations involving willful failure to provide personnel dosimetry to well-logging operators.
Portland General Electric (Trojan) EA 89-016	$75,000 proposed and paid in FY 89	Violations involving failure to adequately assure the quality of equipment and components purchased commercial grade for use in safety-related systems.
Isomedix, Inc. (Parsippany, New Jersey) EA 89-019	$28,500 proposed, imposed, and paid in FY 89	Violations involving deliberate bypass of the radiation monitor interlock system and bypass of a safety system designed to protect individuals from radiation-produced toxic gases.
St. Joseph's Hospital (Hunrinfburg, Indiana) EA 89-020	$2,500 proposed and paid in FY 89	Violations including the failure to check and test the dose calibrator on a timely basis.
Jeffrey Weisman, MD (Wilmington, Delaware) EA 89-023	$1,250 proposed and paid in FY 89	Violations involving failure of the Radiation Safety Officer to implement the radiation safety program, provide training, and perform instrument calibration checks.
C. Chinwuba, MD (Washington, DC) EA 89-027	$250 proposed, imposed, and paid in FY 89	Violations involving instrument calibration checks, training, establishment of procedures, failure of Radiation Safety Officer to ensure that radiation safety activities were being performed in accordance with procedures and requirements.
Duke Power Company (Oconee) EA 89-032	$25,000 proposed and paid in FY 89	Violations involving failure to have two independent reactor building cooling trains operable.
Mayaguez Hospital (Puerto Rico) EA 89-033	$5,000 proposed in FY 89; $500 imposed in FY 89; pending	Violations involving failure to perform audits, adhere to possession limits, limit molybdenum breakthrough, perform dose calibrator tests, survey packages, perform bioassays, establish survey trigger levels and post regulations.
Lee County Hospital (Pennington Gap, Virginia) EA 89-044	$2,500 proposed, imposed, and paid in FY 89	Violations involving an incident in which the minutes of a quarterly Radiation Safety Committee meeting were falsified by copying minutes of another meeting.
Duke Power Company (Catawba) EA 89-046	$75,000 proposed, imposed, and paid in FY 89	Violations involving inoperable Unit 2 Containment Air Return and Hydrogen Skimmer System Train caused by a design modification wiring error and licensee's failure to report.
Anna Jaques Hospital (Newburyport, Massachusetts) EA 89-048	$1,250 proposed and paid in FY 89	Violations involving failure to perform source leak tests and inventories, establish a program for semi-annual visual inspection and maintenance of equipment, perform surveys, comply with transportation requirements, calibrate instruments, maintain required records and maintain a storage facility as described in license.
Toledo Edison Company (Davis-Besse) EA 89-049	$50,000 proposed and paid in FY 89	Violations identified as a result of an inspection following an improper reactor startup.

Licensee	*Action*	*Summary*
Orange Hospital (Orange, New Jersey) EA 89-051	$3,750 proposed and paid in FY 89	Violations involving surveys, control and surveillance of certain unsecured materials, constancy, linearity, and geometrical variation tests on a dose calibrator, wearing of appropriate protective clothing while handling radioactive material.
Illinois Power Company (Clintonb) EA 89-059	$75,000 proposed and paid in FY 89	Violations involving failure to take adequate corrective actions for environmental qualification deficiencies.
Grand Haven Light & Power (Grand Haven, Michigan) EA 89-060	$500 proposed, imposed, and paid in FY 89	Violations involving use of unauthorized individuals to remove gauges, perform source leak tests and device on-off mechanism tests, maintain gauge labels legible, maintain records, and furnish transfer report.
Brand X Perforators (Woodward, Oklahoma) EA 89-061	$1,125 proposed in FY 89; $750 imposed FY 89; pendin	Violations involving failure to perform source leak tests and inventories, establish program for inspection and maintenance of equipment, perform surveys, comply with transportation requirements, calibrate instruments, maintain records, and maintain storage facility as described in the license.
James River Corporation (Easton, Pennsylvania) EA 89-062	$1,250 proposed and paid in FY 89	Violations involving an incident in which a generally-licensed device was inadvertently disposed of in a sanitary landfill.
Niagara/Wisc. Paper Corp. (Niagra, Wisconsin) EA 89-065	$750 proposed and paid in FY 89	Violations involving failure to have licensed individuals remove gauge from installed location and transfer of radioactive gauge to unlicensed metal salvage yard.
Louisiana Power & Light (Waterford) EA 89-069	$50,000 proposed in FY 89; pending	Violations involving failure to properly evaluate ASME Section XI test results for the B High Pressure Safety Injection pump.
Power Authority of New York (Indian Point 3) EA 89-075	$25,000 proposed and paid in FY 89	Violations involving the actual entry of an individual to the Protected Area who had recently been terminated for cause.
Pacific Gas and Electric Co. (Diablo Canyon) EA 89-085	$75,000 proposed and paid in FY 89	Violations which resulted in having less than the required number of auxiliary feedwater pumps and failure to take corrective actions for identified conditions adverse to quality.
Arizona Public Service (Palo Verde) EA 89-088	$250,000 proposed in FY 89; pending	Violations involving inadequate corrective actions related to atmospheric dump valves, training, and inadequate emergency lighting.
Cleveland Electric (Perry) EA 89-091	$37,500 proposed and paid in FY 89	Violations related to equipment qualification requirements.
Ellis Fischel Cancer Center (Columbia, Missouri) EA 89-092	$5,000 proposed in FY 89; $4,583 imposed and paid in FY 89	Violations involving the replacement of Radiation Safety Officer and Chairman of Radiation Safety Committee without NRC approval, failures of the radiation safety program, failure of radiation monitor to indicate that teletherapy source was partially exposed, and use of licensed material by unauthorized individuals.

Licensee	Action	Summary
Texas Nuclear Corporation (Austin, Texas) EA 89-093	$2,500 proposed and paid in FY 89	Violations involving notification of health physics personnel to perform a prejob assessment in accordance with procedures, inadequate surveys, posting and access restriction, failure to utilize personnel radiation exposure devices, failure to perform dose evaluation, and failure to notify NRC of potential radiation overexposure.
Boston Edison Company (Pilgrim) EA 89-095	$25,000 proposed and paid in FY 89	Violations involving the overpressurization of the reactor core isolation cooling system due to failure to carry out station equipment tagout requirement.
Photon Field Inspection (Saginaw, Michigan) EA 89-098	$7,500 proposed in FY 89; pending	Violation involving failure to obtain NRC authorization prior to facility relocation, provide annual retraining, perform audits and inventories, calibrate survey instruments, leak test sealed sources, and maintain records.
Baltimore Gas & Electric (Calvert Cliffs) EA 89-107	$75,000 proposed and paid in FY 89	Violations involving events in which core alterations took place without having proper level of containment integrity, and failure to perform proper safety evaluations prior to making temporary plant modifications.
Bucks Diagnostic Center (Levittown, Pennsylvania) EA 89-113	$500 proposed and paid in FY 89	Violations involving records of instrument calibration checks, surveys, checking survey meters, inventories, and training.
Florida Power Corporation (Crystal River) EA 89-118	$100,000 proposed in FY 89; pending	Violations of the equipment qualification requirements.
Yale New Haven Hospital (New Haven, Connecticut) EA 89-119	$2,500 proposed in FY 89; pending	Violations involving improper disposal of source, failure to survey waste prior to disposal, failure to perform adequate inventory.
Northeast Utilities (Millstone) EA 89-124	$25,000 proposed and paid in FY 89	Violations involving failure to establish adequate procedures to prevent contamination to hydrolaze equipment and other violations of transportation requirements that resulted.
Philadelphia Electric Co. (Limerick) EA 89-124	$75,000 proposed and paid in FY 89	Violations involving inability of operations staff to properly escalate emergency classifications and make appropriate protective action recommendations, and failure to correct deficiencies identified in audits.
General Electric (Cleveland, Ohio) EA 89-127	$24,000 proposed and paid in FY 89	Violations involving surveys, decontamination of areas, breathing-zone air sampling, process or engineering controls to limit airborne radioactivity, and posting.
University of Oklahoma (Oklahoma City, Oklahoma) EA 89-128	$7,500 proposed and paid in FY 89	Violations involving preparation and distribution of unauthorized byproduct material for human use and failure to maintain fume hoods used for storing and processing volatile liquid iodine-131.
Yale University (New Haven, Connecticut) EA 89-131	$12,000 proposed in FY 89; pending	Violations involving reported overexposure to the finger of of a researcher and additional violations indicative of a significant breakdown in management control over licensed activities.

Licensee	Action	Summary
Bradley Memorial Hospital (Southington, Connecticut) EA 89-134	$625 proposed in FY 89; pending	Violations involving training, constancy tests, determination of molybdenum-99 breakthrough, calibration of instruments, establishment of trigger levels, measurement of ventilation rates, and written procedures.
St. Joseph's Hospital (St. Paul, Minnesota) EA 89-140	$4,375 proposed and paid in FY 89	Violations involving surveys, exceeding regulatory limits radiation levels in an unrestricted area, wipe tests of packages, training, linearity and accuracy tests, leak tests inventory of sealed sources, weekly surveys, disposal of radioactive material, and record of diagnostic misadministration.
Bluefield Hospital (Bluefield, West Virginia) EA 89-142	$5,000 proposed in FY 89; pending	Violations involving storage of licensed material in unrestricted area, assignment of dosimetry, training, dose calibrator tests, adherence to possession limits, surveys, calibration and checks on instruments, inventories of radioactive materials, posting procedures, maintenance of certification, records of receipt and transfer of material, teletherapy practice drills, checks on teletherapy unit, maintenance of required information, and post radiation warning signs.
South Carolina Electric & Gas (Summer) EA 89-143	$25,000 proposed and paid in FY 89	Violations involving allowing a licensed senior reactor operator who failed a portion of his requalification exam to assume duties prior to retraining.
U. S. Testing Company (Modesto, California) EA 89-148	$5,000 proposed in FY 89; pending	Violations of NRC requirements designed to protect members of the public against exposure in excess of Part 21 limits.
McDowell & Associates (Ferndale, Michigan) EA 89-149	$750 proposed in FY 89; pending	Violations involving surveys in unrestricted areas, securing of licensed material in unrestricted area, use of material in unauthorized location, leak tests and inventory of sources, packaging during transit, shipping papers, storage of shipping papers, locking of moisture-density gauge.
T.V.A. (Sequoyah) EA 89-152	$87,500 proposed in FY 89; pending	Violations involving failure to implement or adhere to safety review program requirements.
Commonwealth Edison Company ((Zion) EA 89-153	$75,000 proposed in FY 89; pending	Violations involving failure to ensure that vital area barriers were capable of deterring intrusion.
New York Power Authority (Indian Point 3) EA 89-155	$50,000 proposed in FY 89; pending	Violations related to access control, personnel and package search, compensatory measures, and illumination of the Protected Area.
Precision Components (York, Pennsylvania) EA 89-175	$5,000 proposed in FY 89; pending	Violations involving the exposure in excess of regulatory limits, and failures to lock source assembly, perform surveys, stop work when dosimeter went off-scale, and notify both the individual and the NRC of radiation exposure received by the individual.

Licensee	Action	Summary
ORDERS ISSUED IN FISCAL YEAR 1989 (Organized According to EA Number)		
Penn Inspection (Chickasha, Oklahoma) EA 88-189	Order Modifying License, Effective Immediately	To prevent the individual involved from acting as a radiographer for the licensee without specific NRC approval.
Syncor International Corp. (Chadsworth, California) EAs 88-194 and 88-242	Order Modifying Licenses (Effective Immediately)	To require the provisions of the Confirmatory Action letters be maintained and that the licensee perform an assessment of licensed activities at its facilities.
Hole Truth, Inc (Oklahoma City, Oklahoma) EA 88-212	Order Modifying License (Effective Immediately)	To require the licensee to obtain independent consulting services to perform audits to evaluate adherence to NRC requirements, to observe and evaluate performance, and to assess record quality and accuracy.
Toledo Edison Company (Davis-Besse) EA 88-234	Order Modifying License	To direct the licensee to notify the NRC if a certain QC supervisor becomes reinvolved in safety-related activities authorized under the utility license.
E. L. Conwell (Bridgeport, Pennsylvania) EA 88-248	Order to Show Cause Why License Should Not Be Modified	Violations involving failure to secure and maintain surveillance of a nuclear gauge, failure to secure source lock when not in use, transport of of gauge without adequate shipping papers, and transmittal of inaccurate information to the NRC.
American Testing and Inspection, Inc. (Joliet, Illinois) EA 88-290	Order to Show Cause and Order Suspending License	Based on findings that ATI willfully conducted licensed activities in violation of regulatory requirements and that the President of ATI had made material false statements and had knowingly and willfully violated NRC license conditions or permitted them to be violated.
Saturn Services, Inc. (Tulsa, Oklahoma) EA 89-007	Order Confirming Transfer of Regulated Material (Effective Immediately)	To require SSI to confirm that all licensable radioactive materials have been transferred to an authorized recipient and that no such material remains in SSI's possession.
Safety Light Corporation (Bloomsburg, Pennsylvania) EA 89-029	Order Modifying Licenses (Effectve Immediately)	To establish funding for implementation of a site characterization plan and for taking necessary immediate remedial actions.
Michael F. Dimun, M.D. (Carnegie, Pennsylvania) EA 89-052	Order to Cease and Desist Use of Regulated Material (Effective Immediately)	To require confirmation that all licensable radioactive materials have been transferred to an authorized recipient and that no such material remains in Dr. Dimun's possession.
James River Corporation (Easton, Pennsylvania) EA 89-062	Order Modifying License	To require the licensee to conduct an audit for compliance with the terms of the general license at each James River facility where generally-licensed materials are used or stored.
P&L Trucks (Wetumka, Oklahoma) EA 89-067	Order Suspending and Revoking License (Effective Immediately)	To require the transfer of all licensed material to an authorized recipient.
Nuclear and Radiologic Imaging Physicians (Troy, Michigan) EA 89-068	Order Suspending License (Effective Immediately) and Revoking License	To require the licensee to transfer all licensed material material to an authorized recipient.
Yale University (New Haven, Connecticut) EA 89-131	Order Modifying Licenses	To require the licensee to conduct an assessment of deficiencies in radiation safety program and develop detailed plan for correction.

Appendix 7

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1989)

The following is a listing of the 122 nuclear power reactor electrical generating units which were in operation or under construction in the United States as of December 31, 1989, representing a total capacity of approximately 113,000 MWe, of which about 12,000 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 122 reactor units listed, 82 are PWRs and 40 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, 1989. At that time, there were 112 commercial nuclear reactors in the United States with operating licenses, and 10 units for which construction permits were in effect (although construction of some of these has been postponed indefinitely). See the last page of this appendix for an alphabetic listing of all nuclear plants in the United States, with information on power ratings and dates of licensing.

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ALABAMA						
Decatur	Browns Ferry Unit 1 nuclear power plant	1,065	BWR	OL 1973	Tennessee Valley Authority	1974
Decatur	Browns Ferry Unit 2 nuclear power plant	1,065	BWR	OL 1974	Tennessee Valley Authority	1975
Decatur	Browns Ferry Unit 3 nuclear power plant	1,065	BWR	OL 1976	Tennessee Valley Authority	1977
Dothan	Joseph M. Farley Unit 1 nuclear power plant	804	PWR	OL 1977	Alabama Power Co.	1977
Dothan	Joseph M. Farley Unit 2 nuclear power plant	814	PWR	OL 1981	Alabama Power Co.	1981
Scottsboro	Bellefonte Unit 1 nuclear power plant	1,235	PWR	CP 1974	Tennessee Valley Authority	1993
Scottsboro	Bellefonte Unit 2 nuclear power plant	1,235	PWR	CP 1974	Tennessee Valley Authority	1995
ARIZONA						
Wintersburg	Palo Verde Unit 1 nuclear power plant	1,304	PWR	OL 1984	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Unit 2 nuclear power plant	1,304	PWR	OL 1985	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Unit 3 nuclear power plant	1,304	PWR	OL 1987	Arizona Public Service Co.	1988
ARKANSAS						
Russelville	Arkansas Nuclear One 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

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
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
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

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ILLINOIS						
Morris	Dresden Unit 2 nuclear power plant	772	BWR	OL 1969	Commonwealth Edison Co.	1970
Morris	Dresden Unit 3 nuclear power plant	773	BWR	OL 1971	Commonwealth Edison Co.	1971
Zion	Zion Unit 1 nuclear power plant	1,040	PWR	OL 1973	Commonwealth Edison Co.	1973
Zion	Zion Unit 2 nuclear power plant	1,040	PWR	OL 1973	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Unit 1 nuclear power plant	769	BWR	OL 1972	Comm. Ed. Co. -Iowa-Ill. Gas & Elec. Co.	1973
Cordova	Quad-Cities Unit 2 nuclear power plant	769	BWR	OL 1972	Comm. Ed. Co. -Iowa-Ill. Gas & Elec. Co.	1973
Seneca	LaSalle Unit 1 nuclear power plant	1,078	BWR	OL 1982	Commonwealth Edison Co.	1984
Seneca	LaSalle Unit 2 nuclear power plant	1,078	BWR	OL 1983	Commonwealth Edison Co.	1984
Bryon	Byron Unit 1 nuclear power plant	1,120	PWR	OL 1984	Commonwealth Edison Co.	1985
Byron	Byron Unit 2 nuclear power plant	1,120	PWR	OL 1986	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 1 nuclear power plant	1,120	PWR	OL 1986	Commonwealth Edison Co.	1988
Braidwood	Braidwood Unit 2 nuclear power plant	1,120	PWR	OL 1987	Commonwealth Edison Co.	1988
Clinton	Clinton Unit 1 nuclear power plant	950	BWR	OL 1986	Illinois Power Co.	1987
IOWA						
Pala	Arnold Unit 1 nuclear power plant	515	BWR	OL 1974	Iowa Elec. Power & Light Co.	1975
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

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
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	Yankee nuclear power plant	175	PWR	OL 1960	Yankee Atomic Electric Co.	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	64	BWR	OL 1962	Consumers Power Co.	1963
South Haven	Palisades nuclear power plant	635	PWR	OL 1971	Consumers Power Co.	1971
Laguna Beach	Fermi Unit 2 nuclear power plant	1,093	BWR	OL 1985	Detroit Edison Co.	1988
Bridgman	Cook Unit 1 nuclear power plant	1,044	PWR	OL 1974	Indiana & Michigan Electric Co.	1975
Bridgman	Cook Unit 2 nuclear power plant	1,082	PWR	OL 1977	Indiana & Michigan Electric Co.	1978
MINNESOTA						
Monticello	Monticello nuclear power plant	525	BWR	OL 1970	Northern States Power Co.	1971
Red Wing	Prairie Island Unit 1 nuclear power plant	503	PWR	OL 1973	Northern States Power Co.	1973
Red Wing	Prairie Island Unit 2 nuclear power plant	500	PWR	OL 1974	Northern States Power Co.	1974
MISSISSIPPI						
Port Gibson	Grand Gulf Unit 1 nuclear power plant	1,250	BWR	OL 1982	Mississippi Power & Light Co.	1985
Port Gibson	Grand Gulf Unit 2 nuclear power plant	1,250	BWR	CP 1974	Mississippi Power & Light Co.	Indef.
MISSOURI						
Fulton	Callaway Unit 1 nuclear power plant	1,188	PWR	OL 1984	Union Electric Co.	1985

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
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 1986	Public Service of New Hampshire	Indef.
NEW JERSEY						
Toms River	Oyster Creek Unit 1 nuclear power plant	620	BWR	OL 1969	GPU Nuclear Corp.	1969
Salem	Salem Unit 1 nuclear power plant	1,079	PWR	OL 1976	Public Service Electric & Gas Co.	1977
Salem	Salem Unit 2 nuclear power plant	1,106	PWR	OL 1980	Public Service Electric & Gas Co.	1981
Salem	Hope Creek Unit 1 nuclear power plant	1,067	BWR	OL 1986	Public Service Electric & Gas Co.	1986
NEW YORK						
Indian Point	Indian Point Unit 2 nuclear power plant	864	PWR	OL 1973	Consolidated Edison Co.	1974
Indian Point	Indian Point Unit 3 nuclear power plant	891	PWR	OL 1975	Power Authority of the State of New York	1976
Scriba	Nine Mile Point Unit 1 nuclear power plant	610	BWR	OL 1969	Niagara Mohawk Power Co.	1969
Scriba	Nine Mile Point Unit 2 nuclear power plant	1,080	BWR	OL 1986	Niagara Mohawk Power Co.	1988
Ontario	Ginna Unit 1 nuclear power plant	470	PWR	OL 1969	Rochester Gas & Electric Co.	1970
Brookhaven	Shoreham nuclear power plant	820	BWR	OL 1984	Long Island Lighting Co.	Indef.
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

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
OHIO						
Oak Harbor	Davis-Besse Unit 1 nuclear power plant	874	PWR	OL 1977	Toledo Edison-Cleveland Electric Illuminating Co.	1977
Perry	Perry Unit 1 nuclear power plant	1,205	BWR	OL 1986	Toledo Edison-Cleveland Electric Illuminating Co.	1987
Perry	Perry Unit 2 nuclear power plant	1,205	BWR	CP 1977	Toledo Edison-Cleveland Electric Illuminating Co.	Indef.
OREGON						
Prescott	Trojan Unit 1 nuclear power plant	1,080	PWR	OL 1975	Portland General Electric Co.	1976
PENNSYLVANIA						
Peach Bottom	Peach Bottom Unit 2 nuclear power plant	1,051	BWR	OL 1973	Philadelphia Electric Co.	1974
Peach Bottom	Peach Bottom Unit 3 nuclear power plant	1,035	BWR	OL 1974	Philadelphia Electric Co.	1974
Pottstown	Limerick Unit 1 nuclear power plant	1,065	BWR	OL 1984	Philadelphia Electric Co.	1986
Pottstown	Limerick Unit 2 nuclear power plant	1,065	BWR	OL 1989	Philadelphia Electric Co.	1990
Shippingport	Beaver Valley Unit 1 nuclear power plant	810	PWR	OL 1976	Duquesne Light Co. Ohio Edison Co.	1976
Shippingport	Beaver Valley Unit 2 nuclear power plant	852	PWR	OL 1987	Duquesne Light Co. Ohio Edison Co.	1987
Goldsboro	Three Mile Island Unit 1 nuclear power plant	776	PWR	OL 1974	GPU Nuclear Corp.	1974
Berwick	Susquehanna Unit 1 nuclear power plant	1,052	BWR	OL 1982	Pennsylvania Power & Light Co.	1983
Berwick	Susquehanna Unit 2 nuclear power plant	1,052	BWR	OL 1984	Pennsylvania Power & Light Co.	1985
SOUTH CAROLINA						
Hartsville	Robinson Unit 2 nuclear power plant	665	PWR	OL 1970	Carolina Power & Light Co.	1971
Seneca	Oconee Unit 1 nuclear power plant	860	PWR	OL 1973	Duke Power Co.	1973
Seneca	Oconee Unit 2 nuclear power plant	860	PWR	OL 1973	Duke Power Co.	1974
Seneca	Oconee Unit 3 nuclear power plant	860	PWR	OL 1974	Duke Power Co.	1974
Broad River	Summer Unit 1 nuclear power plant	900	PWR	OL 1982	So. Carolina Electric & Gas Co.	1984
Lake Wylie	Catawba Unit 1 nuclear power plant	1,145	PWR	OL 1984	Duke Power Co.	1985
Lake Wylie	Catawba Unit 2 nuclear power plant	1,145	PWR	OL 1986	Duke Power Co.	1986

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
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	CP 1974	Texas Utilities	1988
Glen Rose	Comanche Peak Unit 2 nuclear power plant	1,150	PWR	CP 1974	Texas Utilities	1989
Bay City	South Texas Unit 1 nuclear power plant	1,250	PWR	OL 1987	Houston Lighting & Power Co.	1988
Bay City	South Texas Unit 2 nuclear power plant	1,250	PWR	OL 1989	Houston Lighting & Power Co.	1989
VERMONT						
Vernon	Vermont Yankee nuclear power plant	504	BWR	OL 1972	Vermont Yankee Nuclear Power Corp.	1972
VIRGINIA						
Gravel Neck	Surry Unit 1 nuclear power plant	775	PWR	OL 1972	Virginia Electric & Power Co.	1972
Gravel Neck	Surry Unit 2 nuclear power plant	775	PWR	OL 1973	Virginia Electric & Power Co.	1973
Mineral	North Anna Unit 1 nuclear power plant	865	PWR	OL 1976	Virginia Electric & Power Co.	1978
Mineral	North Anna Unit 2 nuclear power plant	890	PWR	OL 1980	Virginia Electric & Power Co.	1980
WASHINGTON						
Richland	WPPSS No. 1 (Hanford) nuclear power plant	1,266	PWR	CP 1975	Wash. Public Power 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	515	PWR	OL 1973	Wisconsin Public	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.
Cook 1 (Mich.) = pwr, 1020, 3/69, 10/74.
Cook 2 = pwr, 1060, 3/69, 12/77.
Cooper (Neb.) = bwr, 764, 6/68, 1/74.
Crystal River 3 (Fla.) = pwr, 821, 9/68, 1/77.
Davis-Besse ((Ohio) = pwr, 860, 3/71, 4/77.
Diablo Canyon 1 (Cal.) = pwr, 1073, 4/68, 11/84.
Diablo Canyon 2 = pwr, 1087, 12/70, 8/85.
Dresden 2 (Ill.) = bwr, 772, 1/66, 12/69
Dresden 3 = bwr, 773, 10/66, 3/71.
Duane Arnold (Iowa) = bwr, 515, 6/70, 2/74.
Farley 1 (Ala.) = pwr, 813, 8/72, 6/77.
Farley 2 = pwr, 823, 8/72, 3/81.
Fermi 2 (Mich.) = bwr, 1093, 9/72, 7/85.
Fitzpatrick (N.Y.) = bwr 778, 5/70, 10/74.
Fort Calhoun 1 (Neb.) = pwr, 478, 6/68, 8/73.
Ginna (N.Y.) = pwr, 470, 4/66, 12/84.
Grand Gulf 1 (Miss.) = bwr, 1142, 9/74, 11/84.
Haddam Neck (Conn.) = pwr, 569, 5/64, 12/74.
Harris 1 (N.C.) = pwr, 860, 1/78, 1/87.
Hatch 1 (Ga.) = bwr, 860, 9/69, 10/74.
Hatch 2 = bwr, 768, 12/72, 6/78.
Hope Creek 1 (N.J.) = bwr, 1067, 11/74, 7/86.
Indian Point 2 (N.Y.) = pwr, 849, 10/66, 9/73.
Indian Point 3 = pwr, 965, 8/69, 4/76.
Kewaunee (Wis.) = pwr, 503, 8/68, 12/73.
LaSalle 1 (Ill.) = bwr, 1036, 9/73, 8/82.
LaSalle 2 = bwr, 1036, 9/73, 3/84.
Limerick 1 (Pa.) = bwr, 1055, 6/74, 8/85.
Limerick 2 = bwr, 1065, 6/74, 7/89.
Maine Yankee = pwr, 810, 10/68, 6/73.
McGuire 1 (N.C.) = pwr, 1129, 2/73, 7/81.
McGuire 2 = pwr, 1129, 2/73, 5/83.
Millstone 1 (Conn.) = bwr, 654, 5/66, 10/86.
Millstone 2 = pwr, 863, 12/70, 9/75.
Millstone 3 = pwr, 1142, 8/74, 1/86.
Monticello (Minn.) = bwr, 536, 6/67, 1/81.
Nine Mile Point 1 (N.Y.) = bwr, 610, 4/65, 12/74.
Nine Mile Point 2 = bwr, 1080, 6/74, 7/87.
North Anna 1 (Va.) = pwr, 915, 2/71, 4/78.
North Anna 2 = pwr, 915, 2/71, 8/80.
Oconee 1 (S.C.) = pwr, 846, 11/67, 2/73.
Oconee 2 = pwr, 846, 11/67, 10/73.
Oconee 3 = pwr, 846, 11/67, 6/74.
Oyster Creek (N.J.) = bwr, 620, 12/64, 8/69.
Palisades (Mich.) = pwr, 730, 3/67, 10/72.
Palo Verde 1 (Ariz.) = pwr, 1221, 5/76, 6/85.
Palo Verde 2 = pwr, 1221, 5/76, 4/86.
Palo Verde 3 = pwr, 1221, 5/76, 11/87.
Peach Bottom 2 (Pa.) = bwr, 1051, 1/68, 12/73.
Peach Bottom 3 = bwr, 1035, 1/68, 7/74.
Perry 1 (Ohio) = bwr, 1205, 5/77, 11/86.
Pilgrim 1 (Mass.) = bwr, 670, 8/68, 9/72.
Point Beach 1 (Wis.) = pwr, 485, 7/67, 10/70.
Point Beach 2 = pwr, 485, 7/68, 3/73.
Prairie Island 1 (Minn.) = pwr, 503, 6/68, 4/74.
Prairie Island 2 = pwr, 503, 6/68, 10/74.
Quad Cities 1 (Ill.) = bwr, 769, 2/67, 12/72.
Quad Cities 2 = bwr, 769, 2/67, 12/72.
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.
Shoreham (N.Y.) = bwr, 820, 4/73, 4/89.
South Texas 1 = pwr, 1250, 12/75, 3/88.
South Texas 2 = pwr 1250, 12/75, 12/88.
St. Lucie 1 (Fla.) = pwr, 839, 7/70, 3/76.
St. Lucie 2 = pwr, 839, 5/77, 6/83.
Summer (S.C.) = pwr, 885, 3/73, 11/82.
Surry 1 (Va.) = pwr, 781, 6/68, 5/72.
Surry 2 = pwr, 781, 6/68, 1/73.
Susquehanna 1 (Pa.) = bwr, 1032, 11/73, 11/82.
Susquehanna 2 = bwr, 1032, 11/73, 6/84.
Three Mile Island 1 (Pa.) = pwr, 776, 5/68, 4/74.
Trojan (Ore.) = pwr, 1095, 2/71, 11/75.
Turkey Point 3 (Fla.) = pwr, 666, 4/67, 7/72.
Turkey Point 4 = pwr, 666, 4/67, 4/73.
Vermont Yankee = bwr, 504, 12/67, 2/73.
Vogtle 1 (Ga.) = pwr, 1079, 6/74, 3/87.
Vogtle 2 = pwr, 1165, 6/74, 2/89.
Washington Nuclear 2 = bwr, 1095, 3/73, 4/84.
Waterford 3 (La.) = pwr, 1075, 11/74, 3/85.
Wolf Creek 1 (Kans.) = pwr, 1128, 5/77, 6/85.
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 12/31/89 = 112.

Reactor projects for which construction permits were in effect† as of 12/31/89 (cp date shown):

Bellefonte 1 (Ala.) = pwr, 1235, 12/74.
Bellefonte 2 = pwr, 1235, 12/74.
Comanche Peak 1 (Tex.) = pwr, 1150, 12/74.
Comanche Peak 2 = pwr, 1150, 12/74.
Grand Gulf 2 (Miss.) = bwr, 1250, 9/74.
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 12/31/89 = 10

*Name of plant; type of plant: pressurized water reactor = pwr, boiling water reactor = bwr; power output in megawatts (MWe); date of construction permit (cp) issuance; date of operating license (ol) issuance.

†Construction has been halted on a number of these projects.

INDEX