U.S. Nuclear Regulatory Commission

1987 Annual Report

UNITED STATES NUCLEAR REGULATORY COMMISSION

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UNITED STATES NUCLEAR REGULATORY COMMISSION

July 22, 1988

The President
The White House
Washington, D.C. 20500

Dear Mr. President:

This Annual Report for 1987 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.

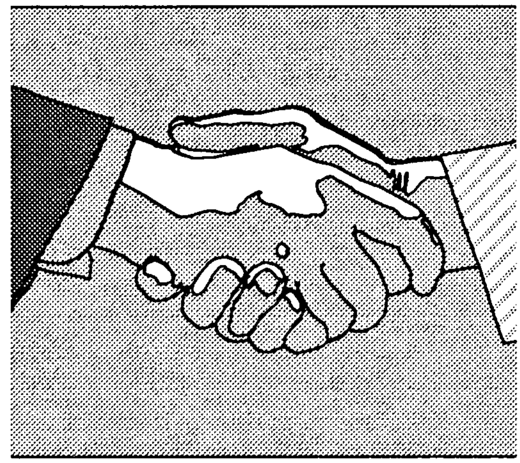
The report is devoted mainly to coverage of events and activities occurring in fiscal year 1987, with additional treatment of events after that period where circumstances warranted.

Respectfully

Lando W. Zech, Jr.

Lando W. Zech, Jr.
Chairman

NRC



WELCOME TO THE

U.S. NUCLEAR REGULATORY COMMISSION

1987 Annual Report

PREVIOUS REPORT 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

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NRC Annual Report
Statutory Reporting Requirements

ENERGY REORGANIZATION ACT OF 1974, AS AMENDED

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

"...the short-range and long-range goals, priorities, and plans of the Commission as they related to the benefits, costs, and risks of nuclear power." (See Chapter 1 for discussion of NRC reorganization and consolidation; digest of fiscal year 1987 policy statements; regulation of special nuclear projects and enforcement activities. Specific goals concerning nuclear power reactors are also discussed in Chapters 2 and 3; operating experience and the evaluation thereof in Chapter 4; fuel cycle concerns in Chapter 5; safeguards in Chapter 6; waste management in Chapter 7; nonproliferation concerns Chapter 8; and nuclear regulatory research in Chapter 9.)

"...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, 3 and 9); for materials facilities, devices and transportation packaging, see Chapters 5 and 9; for waste disposal facilities, see Chapters 7 and 9.)

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

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle...." (See Chapters 6, 8, and 9.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing contingency plans for dealing with such incidents...." (See Chapters 1, 6, and 9.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities...." (See Chapters 7 and 9.)

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities...." (See Chapters 2, 5, and 7.)

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 the plan. (See Chapter 9.)

Section 209 requires the Commission to include in each Annual Report a chapter describing the status of the NRC's domestic safeguards program. (See Chapter 6.)

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

NUCLEAR NONPROLIFERATION ACT OF 1978

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

ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 170(i) directs the Commission to report annually on indemnity actions implementing the Price-Anderson Act which provides a system to pay public liability claims in the event of a nuclear incident. (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 or permit and for the inspection of any facility." (See Chapter 11.)

PUBLIC LAW 97-415

Section 10(c) requires that the "Commission include as a separate chapter a description of the collaborative efforts...by the Commission and the Department of Energy with respect to the decontamination, repair or rehabilitation of facilities at Three Mile Island Unit 2...." (See Chapter 3.)

1987 Highlights / Special Reports

Chapter 1

This is the 13th annual report of the U.S. Nuclear Regulatory Commission (NRC), covering events and activities occurring in fiscal year 1987 (with some treatment of events, where warranted, in the last quarter of calendar 1987).

The NRC was created by enactment of 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 non-military 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 the national security. The NRC accomplishes its purposes through the licensing of nuclear reactor operations and other possession and use of nuclear materials, including transport and disposal of nuclear materials and wastes; the safeguarding of nuclear materials and facilities from theft and sabotage; and inspection and enforcement actions.

This report covers the major activities, events, decisions, and planning that took place during fiscal year 1987 (October 1986 through September 1987) within the NRC or involving the NRC. The report is prepared 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 deals with highlight events and actions of the report period, including in particular the reorganization of the agency and consolidation of offices in a new venue, as well as other noteworthy topics. The chapter also carries reports on the activities of two new NRC offices—the Office of Special Projects and the Office of Enforcement—and of the Office of Investigations. (See description of new offices under "Major NRC Reorganization," below.)

Changes Within Commission and Senior Staff

The following changes occurred on the Commission and at senior staff level during the report period:

- In August 1987, Commissioner Kenneth C. Rogers was appointed to the Commission, filling a vacancy created when former Commissioner James K. Asselstine completed his five-year term.
- In February 1987, Hugh L. Thompson was appointed Director of the Office of Nuclear Material Safety and Safeguards, succeeding John G. Davis.
- In February 1987, James G. Keppler was appointed Director of the new Office of Special Projects.

In April 1987, the following appointments were made in connection with the reorganization described under the next heading in this chapter:

- Harold R. Denton was appointed Director of the new Office of Governmental and Public Affairs.
- James M. Taylor was appointed Deputy Executive Director for Regional Operations.
- William G. McDonald was appointed Director of the new Office of Administration and Resources Management.
- James Lieberman was appointed Director of the new Office of Enforcement.
- Paul E. Bird was appointed Director of the new Office of Personnel.
- Thomas E. Murley was appointed Director of the Office of Nuclear Reactor Regulation, succeeding Harold R. Denton.
- William T. Russell was appointed Regional Administrator of Region I (Philadelphia), succeeding Thomas E. Murley.
- A. Bert Davis was appointed Regional Administrator of Region III (Chicago), succeeding James G. Keppler.

Major NRC Reorganization

As noted briefly in last year's annual report (pp. 8 and 10), the NRC undertook a sweeping reorganization of agency offices and reassignment of functions in late 1986, at the start of fiscal year 1987. The new deployment of the NRC's human and material resources was a natural and necessary response to the reality that, while the number of operating nuclear power plants continued to grow, the number of units under construction has decreased sharply in recent years, with no new projects on the horizon. In 1975, when the NRC was established, there were fewer than

Dr. Kenneth C. Rogers, a physicist who had served for 15 years as President of the Stevens Institute of Technology in Hoboken, N.J., prior to his appointment to the Nuclear Regulatory Commission, was sworn in for a five-year term on August 7, 1987. Dr. Rogers, author of numerous technical papers in such areas as plasma physics, particle accelerators, optical spectroscopy, and particle physics, served in research appointments at Cornell University before joining the Stevens Institute in 1957. His academic degrees, all in physics, include a Bachelor of Science from St. Lawrence University and Master of Science and Doctor of Philosophy degrees from Columbia University, as well as honorary doctorates from St. Lawrence and Stevens.

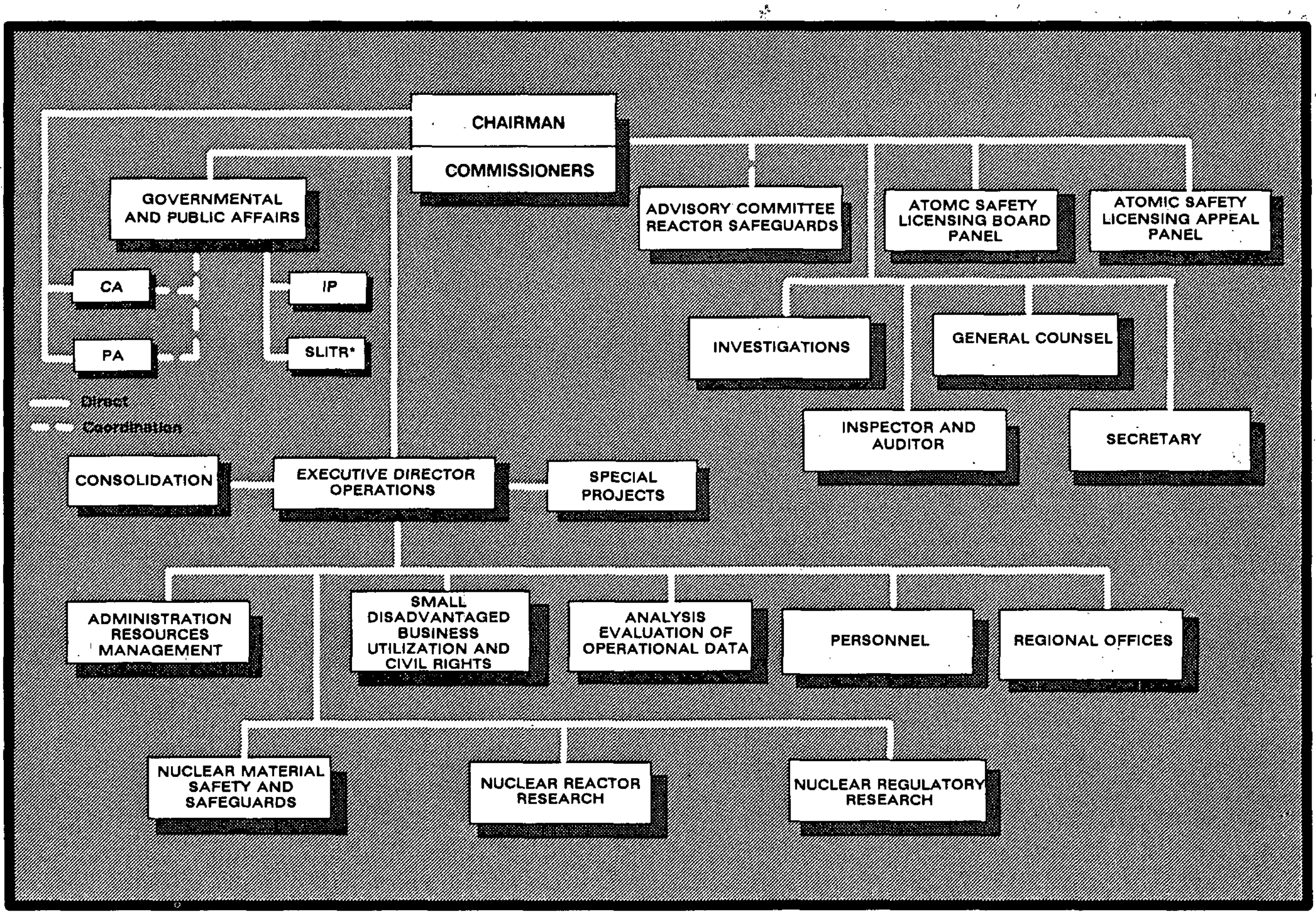
50 nuclear plants in actual operation and over 70 under construction. At the close of fiscal year 1987, there were 109 plants in operation and 15 under construction. That massive shift in the regulated industry clearly called for a corresponding shift within the regulatory body. In the words of NRC Chairman Lando W. Zech, Jr., when announcing the reorganization in a message to all NRC personnel: "As the plants presently in the final stages of construction are completed, we will have progressively less regulatory actions with large complex construction facilities and much more involvement with plant operations, maintenance, life extension and other operational issues. The new organization will focus NRC's major program offices on the day-to-day safety of operational facilities and make them more accountable for our safety programs."

The NRC reorganization, which was effected as planned by April 1987, entailed the abolition or merger of eight existing offices and the creation of five new offices, as the following moves were carried out.

- The inspection activities formerly administered by the Office of Inspection and Enforcement (IE) were reassigned to the Office of Nuclear Reactor Regulation (NRR), if reactor-related, and to the Office of Nuclear Materials Safety and Safeguards (NMSS), for non-reactor related inspections.
- Enforcement actions have been entrusted to the new Office of Enforcement (OE), reporting to the Deputy Executive Director for Regional Operations.
- The former Office of Administration (ADM), the Office of Resource Management (RM), and the Office of Information Resources Management (IRM) were combined into the new Office of Administration and Resources Management (ARM).
- The former Office of State Programs (SP), the Office of International Programs (IP), the Office of Congressional Affairs (CA), and the Office of Public Affairs (PA) were incorporated into the new Office of Governmental and Public Affairs (GPA). Some SP and IP functions were assigned to NMSS.
- The personnel management functions of the agency, formerly a responsibility of the Office of Administration, were assigned to the new Office of Personnel (OP).
- A new Office of Special Projects (OSP) was created to deal specifically with certain licensees presenting unusually complex regulatory concerns calling for sharply focused, comprehensive, and continuous attention by NRC (see discussion of OSP activities later in this chapter). The office currently exercises licensing and regulatory authority pertaining to facilities of the Tennessee Valley Authority and to the Comanche Peak (Tex.) nuclear power plant.

In summation, the new NRC elements are these offices: Administration and Resources Management (replacing ADM, RM, and IRM), Enforcement (replacing IE, in part), Governmental and Public Affairs (replacing SP, IP, CA, PA), Personnel, and Special Projects. All of the new offices report to the Executive Director for Operations (EDO) except for GPA, which reports directly to the Commission.

Besides these structural changes, a number of special operations or activities have been reassigned under the reorganization. Thus, management and support for the work of the Committee to Review Generic Requirements (CRGR),



* STATE, LOCAL & INDIAN TRIBE PROGRAMS

formerly a function of the EDO's office, now belongs to the Office for Analysis and Evaluation of Operational Data (AEOD). Continuing work on Unresolved Safety Issues and the resolution of other generic issues related to reactors and plant systems design—formerly carried out under the aegis of NRR—is assigned to the Office of Nuclear Regulatory Research (RES). AEOD has also been given responsibility for the NRC incident response program and the agency's technical training center, as well as the tracking of licensee performance indicators (all former functions, either in toto or as "lead office," of IE). RES has additional tasks in the area of rulemaking procedures and probabilistic risk assessments. Also, the evaluations of reactor licensees' Quality Assurance and Emergency Preparedness programs (former IE tasks) are to be conducted by NRR. Indemnification agreements related to the Price-Anderson Act and other matters related to licensee financial qualification (formerly administered by SP) have also been assigned to NRR. (See Appendix 1 for descriptions of all NRC offices and their scope of operations.) Activities of new and existing offices for fiscal year 1987 are covered in the appropriate chapters of this report, generally as follows: NRR activities, in Chapters 2 and 3; AEOD in Chapter 4; NMSS in Chapters 5, 6, and 7; GPA in Chapter 8; RES in Chapter 9; ARM in Chapter 11 (Chapter 10 deals with Litigation and Commission decisions).

The basic purpose underlying these wide-ranging changes within the agency is to promote the most effective and efficient disposition of resources within a structure which comports with the realities of the regulated industry.

Consolidation of NRC Headquarters

Major progress was made during the year toward the longtime objective of consolidating all of the NRC's Washington headquarters operations at a single location. Occupancy of the new 18-storey building, One White Flint North, in Rockville, Md., began in mid-December of 1987. By early spring of 1988, some 1,400 agency employees—or 60 percent of the Headquarters total—were to be housed in a modern facility incorporating the latest in facilities and office design.

Progress toward consolidation dates from November 1986, when the General Services Administration (GSA) completed formal purchase of the One White Flint North building.

Agreement also was reached to lease and subsequently purchase a second building to be constructed adjacent to the existing structure. At year's end, the private developer of the White Flint North complex was awaiting approval from the Montgomery County (Md.) government for the necessary zoning changes and permits. The County's review and approval process was expected to extend over several months. Occupancy of the second building by the remaining 1,000 headquarters NRC employees could take place within two years of receiving the necessary county approval.

As early as the end of fiscal year 1986, there was unmistakable evidence of the determination of the NRC and GSA to make One White Flint North a showcase project for the housing of Federal government agencies. Design of the interior of the building incorporates open planning and contemporary modular furniture, in order to make maximum use of space and move toward GSA's new, government-wide standards for allocating office space. To meet current needs of the NRC's highly technical activities, One White Flint North includes built-in communications wiring and other facilities for state-of-the-art computer and video applications. Closed circuit television and local area networks will greatly expand the range and ease of communication among individual staff members and units.

While the primary objective of consolidation at One White Flint North is to bring most of the major operating components of the NRC together, the new building will offer several amenities not available in existing NRC facilities. A cafeteria will be operated by an outside contractor on the first floor, and there will be shower and changing facilities for employees. The building includes some 375 indoor parking spaces. Another 350 to 450 parking spaces were promised by the developer at a nearby shopping mall for use of the NRC staff.

Reaching the point of imminent occupancy of the new building involved numerous policy decisions with respect to both the building itself and the agency's operations. During the year, the major reorganization of the NRC's structure (see above) mandated some revisions in building plans to accommodate the realignment of offices and their responsibilities. The selection of equipment, furniture, and decorations required extensive exploration and evaluation of the available options, in keeping with the overall goal of setting the standard for other government agencies.

The first phase of agency consolidation, the occupancy of One White Flint North, will permit the NRC to vacate four of the 11 office buildings it occupied in 1987, and parts of several others.

Facilities are provided at One White Flint North for the Office of Nuclear Material Safety and Safeguards and the Office of Nuclear Reactor Regulation. The third program unit, the Office of Nuclear Regulatory Research, will be consolidated in a nearby leased building until the second new building is completed. One White Flint North will permit the Commissioners and their supporting staff to move from downtown Washington and, for the first time in the history of the agency, share a common location with staff offices. The new facility also will provide for support and administrative personnel.

Early in 1987, the NRC acknowledged the concern of local government about traffic congestion in the area. It has joined with Montgomery County government, the building developer, and others in developing plans that will mitigate the agency's impact on traffic volume, in keeping with its desire to be a good neighbor in its new home. Changes in the NRC's work hours, encouragement of ride-sharing, and convenient on-site sale to employees of reduced-price transit tickets are all part of this effort.

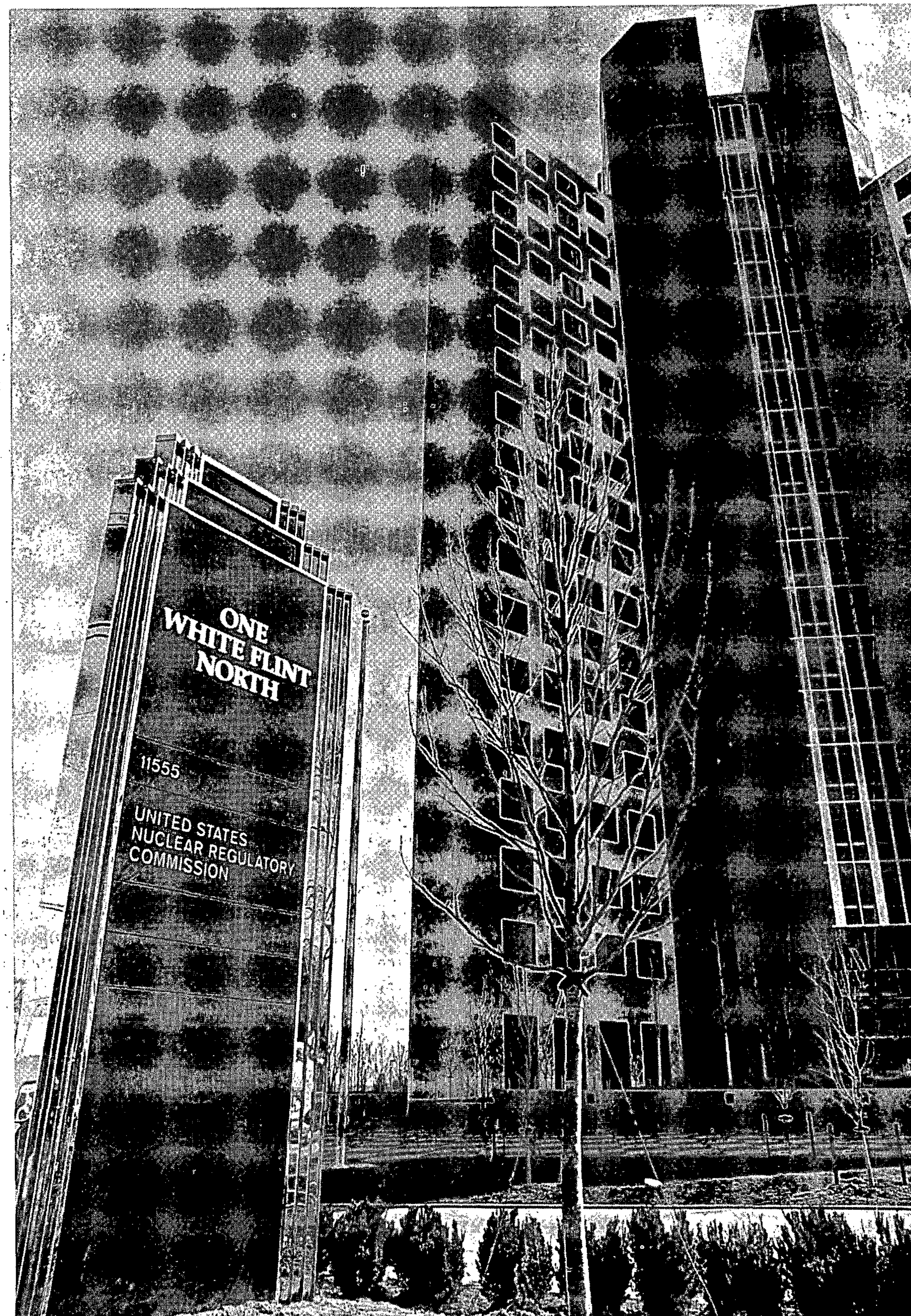
Consolidation of the headquarters staff has been sought by the NRC since its creation in 1975 as a means to improving operational efficiency and to eliminating costly duplication of services at multiple locations. With the massive relocation of staff and resources into One White Flint North, this objective is approaching full realization. The NRC anticipates completion of the move, with all associated benefits, over the next three years.

Licensing Activity In Fiscal Year 1987

During the fiscal year, the NRC issued nine low-power or fuel-loading licenses to eight utilities; full-power licenses were subsequently issued, also during the report period, for six of these reactors and for two others which had received low-power licenses in fiscal year 1986. Three other plants received low-power licenses only, and one a fuel-loading license. One reactor was shut down during the year (the LaCrosse facility in Wisconsin, operating since 1969). The 10 units authorized to operate at low or full power in the United States brings the total to 108, as of September 30, 1987 (110, as of the end of calendar year 1987; see Appendix 7). There were no new applications for operating licenses, construction permits or manufacturing licenses during the fiscal year. There are 15 nuclear power plants still technically under construction in the United States, though some of them are delayed indefinitely.

Commission Policy Statements In Fiscal Year 1987

The Commission proposed several new policy statements for comment during the report period and issued a number of revisions to existing policy statements.



The NRC's new office building at 11555 Rockville Pike in Rockville, Md., stands 18 storeys tall. The new facility, located near a Washington metro-subway stop, permits the NRC to vacate four of the 11 buildings formerly occupied, and parts of several others.

License Renewals for Nuclear Plants. On November 3, 1986, the Commission issued a formal request for comment on policy then being developed regarding the extension of existing nuclear power plant licenses beyond 40 years. The governing regulation, Chapter 10, Section 103(c) of the Atomic Energy Act of 1954, states that licenses shall be issued for a period to be determined by the Commission appropriate to the activity to be licensed, "but not exceeding forty years," though the license "may be renewed upon the expiration of such period." The matter remained under consideration at the close of the report period.

Revised Guidelines on Reviewing Agreement State Programs. On November 13, 1986, proposed minor revisions to the policy of December 1981 entitled "Guidelines for NRC Review of Agreement State Radiation Control Programs" were issued for comment. The policy statement is intended to inform the public of the the indicators and guidelines which the Commission uses in reviewing the radiation control programs of the 29 Agreement States, which regulate certain by-product, source, or special nuclear materials by agreements with the NRC (see Chapter 8). On June 3, 1987, the final revisions were promulgated, updating the guidelines and incorporating editorial and format changes.

Policy on Deferred Nuclear Projects. In an issuance of March 11, 1987, the Commission proposed a policy statement on deferred nuclear power plants, i.e., plants for which construction has been authorized but on which, for economic or other reasons, work has been suspended. The number of such projects was about five at the close of the report period, with a possibility that more would be added to that number. A deferred plant is one for which a construction permit is in effect but on which construction activity has ceased or been reduced to a maintenance level (in contrast to a terminated project, which is one for which the licensee has announced a permanent cessation of construction activity, even though a construction permit remains in effect). On October 7, 1987, a final policy statement was issued, giving specific response to comments received in the interim and setting forth procedures to be followed at plants in a deferred status or undergoing reactivation of construction activity; procedures for terminated plants were contained in the final statement.

Drug Testing Policy. Chairman Zech announced on July 14, 1987, that the Commission had approved a drug testing policy for NRC employees that provided for the random testing of persons in sensitive positions. The program was adopted in accordance with the President's Executive Order of September 15, 1986, on achieving a drug-free Federal workplace. (See Chapter 11.)

Standardization of Nuclear Plants. In a revised policy statement issued for comment on September 9, 1987, the Commission again encouraged the use of standard plant designs for future nuclear projects and set forth information on the certification of plant designs that are essentially complete in scope and level of detail. The Commission reaffirmed its support of the "reference system" concept, by which an entire nuclear plant design or a major portion of the design is deemed acceptable for incorporation by reference in individual license applications. The Commission expressed the belief that standardization through certification of the reference system design by rulemaking would not only enhance safety but contribute to the stability and predictability of the regulatory process.

Enforcement Policy Revised. At the end of the fiscal year, on September 28, 1987, the Commission announced a revision of its policy and procedure for enforcement actions. The revision was intended to accomplish four aims: (1) to explain enforcement actions against individuals more fully; (2) to lay out the criteria under which a closed enforcement action may be reopened; (3) to provide for the exercise of agency discretion in certain circumstances to refrain from issuing notices of violations or proposed civil penalties; and (4) to make some minor changes in the language of the policy statement. The revised policy became effective September 28, 1987, though comment on the statement was invited up to November 27.

OFFICE OF SPECIAL PROJECTS

The Office of Special Projects (OSP) was created to ensure that facilities with particularly complex regulatory problems are dealt with in the most comprehensive and timely manner with a strengthened and integrated staff organization, clear and direct lines of management responsibility and authority, and appropriate high-level direction. The mission of this office is short-term and is expected to be completed by the end of fiscal year 1988.

The Office is responsible for the Tennessee Valley Authority (TVA) and the Texas Utilities Electric Company (TU Electric) projects. This office was established to:

(1) Assure the identification and resolution of those problems which occasioned the shutdown of the TVA Sequoyah (Tenn.) and Browns Ferry (Ala.) facilities and identification and implementation of steps necessary to prevent their recurrence.

(2) Evaluate the TVA Watts Bar (Tenn.) and the TU Electric Comanche Peak (Tex.) facilities to confirm compliance with the NRC's regulatory requirements.

(3) Assess whether identified problems are on a path toward an acceptable solution, and, where not, to identify solutions that are acceptable and sufficient for the staff to complete its licensing reviews of these facilities, consistent with the NRC statutory mandate to protect the health and safety of the public.

The Office has the authority to issue orders and to develop and implement policies and procedures related to licensing and inspection for the facilities under its jurisdiction. The Office was organized along project lines. Of approximately 95 people in the office during fiscal year 1987, two-thirds were assigned to the TVA Projects Division and one-third were assigned to the Comanche Peak Project Division.

TVA Projects

TVA holds operating licenses for five nuclear power units: two at Sequoyah near Chattanooga, Tenn., and three at Browns Ferry near Decatur, Ala. TVA also has four units under construction: two at Watts Bar near Spring City, Tenn., and two at Bellefonte near Scottsboro, Ala. One of the Watts Bar units is 99 percent complete; the other is approximately 75 percent complete. Completion of construction at Bellefonte has been delayed to 1993 for one unit and to 1995 for the other.

TVA Operations and Management

Extensive technical problems at the TVA nuclear facilities, both the operating units and those under construction, and also problems with management culminated in the shutdown of all operating units and lengthy delays in licensing Watts Bar. All three Browns Ferry units were shut down by TVA in March 1985 because of poor operational performance, coupled with management and equipment problems. The two Sequoyah units were shut down in August 1985, because available documentation could not substantiate the environmental qualification of electrical equipment. The licensing of Watts Bar was delayed to allow completion of fire protection modifications, as well as to resolve certain allegations and a large number of employee concerns. The operating plants remained shut down throughout 1987.

The NRC staff requested that TVA provide information, pursuant to 10 CFR 50.54(f), that would describe the actions TVA is taking to address the identified problems. TVA has developed a Nuclear Performance Plan (NPP) describing TVA proposed activities to address the technical and management problems at its plants. TVA has submitted Volume 1 of the NPP for Corporate activities, Volume 2 for the Sequoyah facility, and Volume 3 for the Browns Ferry facility; TVA is developing Volume 4 for the Watts Bar facility.

Corporate Activities

The staff review of the Corporate Nuclear Performance Plan activities is complete. The staff issued its conclusions on July 28, 1987, by letter to the TVA Board of Directors and in NUREG-1232, Volume 1, "Safety Evaluation Report (SER) on Tennessee Valley Authority Revised Corporate Nuclear Performance Plan," in July 1987. The staff concluded that TVA has acceptably addressed the corporate-level concerns raised by the staff in its 10 CFR 50.54(f) letter to TVA.

The staff monitors the TVA corporate-level nuclear activities to ensure that TVA continues to seek out root causes of problems and conducts critical independent reviews or self-reviews of its nuclear organization. The staff will continue to monitor the TVA performance to determine if the past technical and management problems that led to the plant shutdowns recur.

Sequoyah

TVA submitted Revision 2 of the Sequoyah Nuclear Performance Plan on July 2, 1987. Since January 1987, the NRC staff and the TVA have made substantial progress on the resolution of key issues affecting Sequoyah Unit 2. The staff has completed its evaluation of several major TVA programs, covering the following activities:

(1) Operator training program.
(2) Quality assurance program.
(3) Maintenance program.
(4) Environmental qualification of safety-related equipment.
(5) Design basis verification program.
(6) Cable tray supports.
(7) Quality assurance program for replacement part procurement.
(8) Analyses related to moderate energy line break flooding.
(9) Analyses related to Emergency Core Cooling System water loss.
(10) Operational readiness of the Sequoyah facility staff.

The significant issues that have the greatest potential to affect the restart of Sequoyah Unit 2 include those related to the adequacy of silicon rubber insulated cables and to the regeneration of non-retrievable civil engineering design calculations, as well as issues that evolved from the Integrated Design Inspection of the Essential Raw Cooling Water System conducted by the staff, especially civil calculations and related design issues. Still another significant issue to be dealt with before restart of Sequoyah Unit 2 is the restart test program.

Schedule for Sequoyah Unit 2 Restart. The staff approved, by letter dated June 9, 1987, the criteria used by TVA to identify issues that must be resolved prior to the restart of any Sequoyah or Browns Ferry units. Having approved the restart criteria, OSP identified the staff activities to be completed before the restart of Sequoyah Unit 2.

The Integrated Design Inspection (IDI). Although the TVA's efforts to identify and resolve the issues to be addressed prior to Sequoyah restart were considerable, the identification of issues has remained an evolving and somewhat fragmented process. TVA efforts did not include a vertical review through one or more safety systems to provide assurance that all major design and construction problems have been identified and resolved prior to the restart of Sequoyah Unit 2. Accordingly, the staff determined that an independent design and construction verification—including significant aspects of the interactions and interfaces throughout design, engineering and construction of at least one safety-related system—should be undertaken. The staff provided TVA the option of conducting this review with an independent contractor or having the NRC undertake the task. TVA estimated that an independent contractor would require at least nine months to complete the task, causing unreasonable delay in a decision on the Sequoyah Unit 2 restart, and would not commit to such a program at that time. The NRC staff subsequently performed the review.

The Essential Raw Cooling Water System was selected as the safety-related system for the "vertical slice" review. The inspection was conducted from July 8, 1987, through September 11, 1987. The report documenting the inspection review, which uncovered some inconsistencies between the design and as-built plant configuration, was completed on November 6, 1987.

Adequacy of Cable Installation. Concerns were raised by TVA employees regarding cable installation. Potential problems with the integrity of the cable insulation were cited for three installation situations: cable pullbys, cable jamming, and long vertical cable runs supported at the top. To resolve these concerns, a testing program was developed by TVA and submitted on April 8, 1987. Based on initial test results and questions regarding the test conditions, TVA undertook an effort to re-evaluate the proposed test program and submitted a revised program on July 31, 1987. The NRC staff was reviewing the revised program at the close of the report period. The original cable problems have been resolved; however, cable failures later identified during testing may require replacement of silicon rubber insulated cable.

Design Calculations Review. TVA instituted a review of essential design calculations to verify that they had been done and were adequate to support the design of Sequoyah. TVA completed an initial review of a sample of engineering calculations in the electrical, nuclear, mechanical, and civil engineering areas. Although no significant problems were disclosed by the nuclear and mechanical engineering calculations, there were significant problems identified in the electrical and civil areas. TVA regenerated all electrical calculations. When, in the civil area, about 5,000 pipe support calculations were determined to be missing, the staff required TVA to regenerate the missing civil calculations, to the extent practicable, prior to restart. The staff will confirm the regeneration of the calculations and of assumptions used in the electrical calculations.

Restart Test Program. In response to all of the programs under way since the shutdown of the Sequoyah plant, the staff is requiring a comprehensive restart test program to ensure that plant safety systems are functional, before restart of Sequoyah. A summary description of the program and a listing and schedule of required testing identified by this program were submitted by TVA to the NRC staff on May 26 and July 6, 1987, respectively. Inspections have been conducted on this program and the staff is in general agreement with the TVA program. The staff will closely follow the restart test program.

Sequoyah Unit 1 Restart. TVA currently plans to restart Sequoyah Unit 1 approximately six months after the restart of Sequoyah Unit 2. The NRC staff is currently developing a schedule of issues, other than the Sequoyah Unit 2 issues, that will have to be addressed by TVA prior to restart of Sequoyah Unit 1.

Browns Ferry

TVA submitted the Revision 1 to the Browns Ferry Nuclear Performance Plan on July 1, 1987. TVA has scheduled the restart for Browns Ferry Unit 2 for no earlier than mid-1988. Browns Ferry Units 1 and 3 will require additional time before restart. Since January 1987, the NRC staff and TVA have made progress toward the resolution of key issues affecting Browns Ferry Unit 2. These issues include configuration management/design control, fire protection, seismic issues, environmental qualification of safety-related electrical equipment, electrical issues, and the Browns Ferry restart test program.

Watts Bar

On June 30, 1987, the NRC issued an order granting the extension of the construction permits for the Watts Bar facility, at the request of TVA. The extension for Watts Bar Unit 1 is to September 1, 1988 and for Unit 2 is to January 1, 1990. The pacing items are expected to be the completion of the Design Baseline and Licensing Verification Program, the re-analysis of piping and supports, and the resolution of employee concerns. Detailed schedules for resolving these issues have not been provided by TVA; the submittal date for the Watts Bar portion of the Nuclear Performance Plan (Volume 4) remains uncertain.

Bellefonte

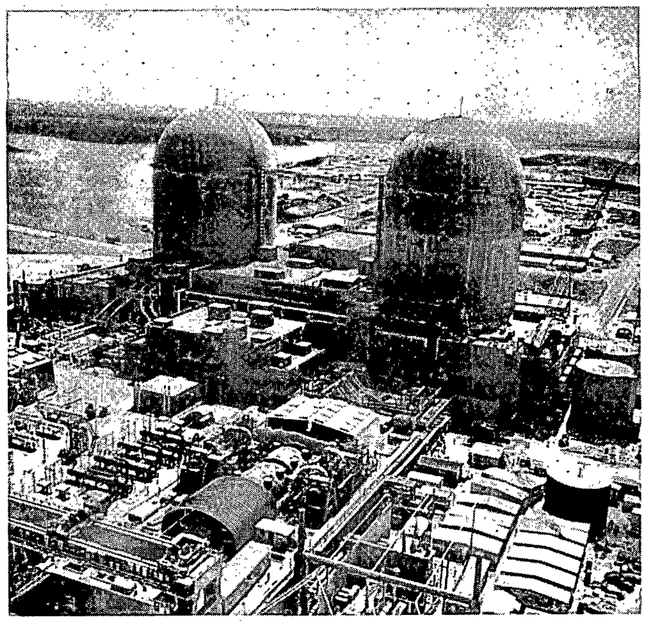
On June 30, 1987, the NRC issued an order granting the extension of the construction permits for the Bellefonte facility at the request of TVA. The extension for Bellefonte Unit 1 is to July 1, 1994, and for Unit 2 is to July 1, 1996.

Comanche Peak Project

During fiscal year 1987, TU Electric began implementing a comprehensive corrective action program to address concerns related to the design and construction adequacy of the Comanche Peak facility. This program encompasses prior efforts and is intended to resolve the adequacy of Comanche Peak design and construction. The corrective action program consists of a complete design re-verification; hardware validation, including hardware re-inspection and modifications; and design and "as-built" reconciliation in a number of areas including:

(1) Large bore piping supports.
(2) Small bore piping supports.
(3) Conduit supports for trains A, B, and C which are greater than two inches and for train C which are equal to or less than two inches.
(4) Cable tray hangers.
(5) Heating, ventilation, and air conditioning equipment.
(6) Mechanical equipment.
(7) Electrical equipment.
(8) Instrumentation and control program.
(9) Civil and structural program.
(10) Equipment qualification.

The OSP staff has prepared a comprehensive review and inspection plan addressing Comanche Peak licensing activities and is continuing to assess the adequacy of the corrective action program, both to monitor program implementation and to resolve routine licensing issues. TU Electric has indicated that all corrective actions would be completed by mid-1988 and is planning for Unit 1 fuel loading in August 1988. The fuel loading schedule is dependent on a motion pending before the Atomic Safety and Licensing Board to resume the licensing hearing.



Texas Utilities (TU), parent company of the Comanche Peak nuclear power plant near Glen Rose, Tex., initiated a comprehensive corrective action program to address NRC concerns about design and construction of that facility, shown above. Early during the plant's construction, as a condition of the NRC construction permit, TU removed large blocks from the plant excavation to preserve five well defined acrocanthosaurus tracks, of interest to paleontologists. One of the tracks is pictured below.

OFFICE OF INVESTIGATIONS

The Office of Investigations (OI) continues to perform investigations of allegations of wrongdoing by individuals or organizations other than NRC employees or NRC contractors (including licensees, applicants and vendors, or their contractors), as described in the 1985 NRC Annual Report, pp. 193-195.

In fiscal year 1987, OI opened 79 new cases and closed 78 cases. Nineteen of the closed cases were referred to the Department of Justice for consideration and possible prosecution.

Indictments

On September 18, 1987, a Federal Grand Jury in Cheyenne, Wyo., acting on information developed by an OI Field Office in Region IV (Dallas), indicted five individuals previously employed by Western Stress, Inc., Evanston, Wyo. The individuals were charged with having made false statements to the NRC, conspiring to make false statements to the NRC, and "impeding, impairing, obstructing, and defeating" the lawful functions of the NRC.

In February 1987, the President of American Testing Laboratories (ATL), Salt Lake City, Utah, and the ATL Radiation Safety Officer were indicted on Federal charges of making false statements, conspiring to make false statements, and improper use of byproduct material. The charges stemmed from an investigation by the OI Field Office in Region IV.

Convictions

In September 1987, the President of ATL (see above) was placed under one year of supervision in lieu of prosecution. The Radiation Safety Officer pled guilty to violation of Federal statutes associated with the improper use of byproduct material and was sentenced to one year probation and a $500 fine.

On October 29, 1986, International Nutronics, Inc. (INI), and the Vice President of INI were convicted of one count of conspiracy, one count of concealing and covering up a material fact, two counts of wire fraud, three counts of mail fraud, and two counts of failing to make timely NRC notification. INI was fined $35,000. The Vice President received two years probation on all counts. Upon appeal, the charges of mail and wire fraud were dismissed against the Vice President.

On March 20, 1987, YOH Security, Inc., was convicted and fined $100,000 on violations of conspiracy and false statements.

In April 1987, the YOH Security Project Manager at Limerick (Pa.) was convicted and sentenced to six months in jail and assessed a $30,000 fine for the violations of conspiracy and making false statements. And, on January 16, 1987, the YOH Site Captain for Operations was convicted on violations of conspiracy and false statements and placed on five years probation.

The actions against YOH Security were the result of an OI investigation into possible falsification of Training and Qualification records of armed guards employed by YOH Security, the primary site security contractor employed by the Philadelphia Electric Co. at its Limerick Unit 1 nuclear power plant.

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 under the revised enforcement policy (10 CFR Part 2, Appendix C, 52 FR 36215 (1987)) 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 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 deal with licensees who do not respond to civil penalties or to deal with violations that constitute a significant threat to public health and safety or to the common defense and security. In the latter case, the order may be made effective immediately.

During the NRC reorganization of 1987, the enforcement functions were transferred from the Enforcement Staff of the Office of Inspection and Enforcement to the new Office of Enforcement. The Regional Administrators retained their authority to issue Notices of Violation not involving civil penalties and Notices of Violation proposing civil penalties, with the concurrence of the Director of the Office of Enforcement and the Deputy Executive Director for Regional Operations (DEDRO). However, with the abolition of the Office of Inspection and Enforcement (IE), the enforcement authorities previously delegated to the Director of IE have been transferred to the DEDRO, who 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 DEDRO in his absence or as otherwise directed.

Appendix 6 provides a listing and brief summary of the 115 civil penalty actions taken during fiscal year 1987, and also a brief description of the 17 enforcement Orders issued during fiscal year 1987.

Nuclear Reactor Regulation

Chapter 2

The Office of Nuclear Reactor Regulation (NRR) has, from the inception of the NRC in 1975, been responsible for regulating operating nuclear reactors, for reviewing applications for construction permits and operating licenses for new reactors, and for issuing such permits and licenses where appropriate.

During fiscal year 1987, the NRC underwent a major reorganization and realignment of agency offices and components, the better to accommodate the changed reality in the regulatory workload (see Chapter 1). 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 has brought about a substantial shift in overall NRC activity. From the traditional preoccupation with reviewing applications for construction permits and operating licenses, NRR staff energies will currently, and for the foreseeable future, be directed mainly to the regulation of the more than 100 nuclear power plants licensed for operation in the United States. This change has necessarily occasioned realignments in the internal structure of NRR. And, as part of the agency-wide 1987 reorganization, NRR has been given responsibility for reactor inspection programs and reactor safeguards programs. Emergency preparedness and incident response have also been moved under the NRR aegis. The resolution of generic safety issues, formerly a function of NRR, has been transferred to the Office of Nuclear Regulatory Research (see Chapter 9).

NRR activities during fiscal year 1987 are treated in this chapter under the following headings:

- Status of Licensing
- Improving the Licensing Process
- Inspection Programs
- Appraisal Programs
- Quality Assurance
- Emergency Preparedness
- Human Factors
- Safety Reviews
- Antitrust Activities
- Indemnity, Financial Protection and Property Insurance
- The Advisory Committee on Reactor Safeguards

STATUS OF LICENSING

Applications for Permits or Licenses

The licensing process for nuclear power plants encompasses a number of phased review procedures that are performed by the Office of Nuclear Reactor Regulation. (See "The Licensing Process," on the following page.) The NRC received no new applications for operating licenses, construction permits or manufacturing licenses during fiscal year 1987. Two utilities were issued two fuel load and pre-critical test licenses. Also, eight Low-Power Licenses (permitting fuel load at 0 percent power or low-power operation at 5 percent power) were issued during fiscal year 1987. In addition, eight Full-Power Operating Licenses were issued to seven utilities.

One licensed power reactor was permanently closed down during the report period—the LaCrosse (Wis.) plant of the Dairyland Power Corp., which had been operating since 1967.

Table 1 is a numerical summary of NRR activity in power reactor licensing during fiscal year 1987. Table 2 identifies the licensees and facilities licensed, with additional information.

At the close of the report period, the staff was reviewing applications for operating licenses for the 15 nuclear units still under construction; the schedules for these reviews are consistent with the projected plant completion dates. Construction of some of these units has been delayed indefinitely.

Licensing Actions for Operating Power Reactors

At the end of fiscal year 1987, 108 power reactors were licensed to operate in the United States. 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

THE LICENSING PROCESS

Obtaining an NRC construction permit—or a limited work authorization (see discussion below) prior to a decision on issuance of a construction permit—is the first objective of a utility or other company seeking to operate a nuclear power reactor or other nuclear facility under NRC licensing authority. The process is set in motion with the filing and acceptance of the application, generally comprising 10 or more large volumes of material covering both safety and environmental factors, in accordance with NRC requirements and guidance. The second phase consists of safety, environmental, safeguards and antitrust reviews undertaken by the NRC staff. Third, a safety review is conducted by the independent Advisory Committee on Reactor Safeguards (ACRS); this review is required by law. Fourth, a mandatory public hearing is conducted by a three-member Atomic Safety and Licensing Board (ASLB), which then makes an initial decision as to whether the permit should be granted. This decision is subject to appeal to an Atomic Safety and Licensing Appeal Board (ASLAB) and could ultimately go to the Commissioners for final NRC decision. The law provides for appeal beyond the Commission in the Federal courts.

As soon an initial application is accepted, or "docketed," by the NRC, a notice of that fact is published in the Federal Register, and copies of the application are furnished to appropriate State and local authorities and to a local public document room (LPDR) established in the vicinity of the proposed site, as well as 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 local newspapers which provides 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.

The NRC staff's safety, safeguards, environmental and antitrust reviews proceed in parallel. With the guidance of the Standard Format (Regulatory Guide1.70), the applicant for a construction permit lays out the proposed nuclear plant design in a Preliminary Safety Analysis Report (PSAR). If and when this report has been made sufficiently complete to warrant review, the application is docketed and NRC staff evaluations begin. Even prior to submission of the report, NRC staff conducts a substantive review and inspection of the applicant's quality assurance program covering design and procurement. The safety review is performed by NRC staff in accordance with the Standard Review Plan for Light-Water-Cooled Reactors, initially published in 1975 and updated periodically. This plan sets forth the acceptance criteria used in evaluating the various systems, components and structures related to safety and in assessing 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 his analysis and evaluation in sufficient depth and breadth to support staff approval with respect to safety. When the staff is satisfied that the acceptance criteria of the Standard Review Plan have been met by the applicant's preliminary report, a Safety Evaluation Report is prepared by the staff which summarizes the results of its review regarding the anticipated effects of the proposed facility on public health and safety.

Following publication of the staff Safety Evaluation Report, the ACRS completes its review and meets with staff and applicant. The ACRS then prepares a letter report to the Chairman of the NRC presenting the results of its independent evaluation and recommending whether or not a construction permit should be issued. The staff issues a supplement to the Safety Evaluation Report incorporating 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 safety aspects of the licensing decision.

In appropriate cases, the NRC may grant a Limited Work Authorization to an applicant in advance of the final decision on the construction permit in order to allow certain work to begin at the site, saving as much as seven months time. The authorization will not be given, however, until NRC staff has completed environmental impact and site suitability reviews and the appointed ASLB has conducted a hearing on environmental impact and site suitability with a favorable finding. To realize the desired saving of time, the applicant must submit the environmental portion of the application early.

The environmental review begins with an assessment of the acceptability of the applicant's Environmental Report (ER). If the ER is judged sufficiently complete to warrant review, it is docketed, and an analysis of the consequences to the environment of the construction and operation of the proposed facility at the proposed site is begun. Upon completion of this analysis, a Draft Environmental Statement is published and distributed with specific requests for review and comment by Federal, State and local agencies, other interested parties and members of the public. All of their comments are then 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 respective publication. During this same period, the NRC is conducting an analysis and preparing a report on site suitability aspects of the proposed licensing action. Upon completion of these activities, a public hearing—with the appointed ASLB presiding—may be held on environmental and site suitability issues related to the proposed licensing action. (Or a single hearing on both safety and environmental matters may be held, if that is indicated.)

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 those on safety and environmental aspects.

About two or three years before construction of a plant is scheduled to be completed, the applicant files an application for an operating license. A process similar to that for the construction permit is followed. The application is filed, the NRC staff and the ACRS review it, a Safety Evaluation Report and an updated Environmental Statement are issued. A public hearing is not mandatory at this stage, but one may be held if requested by affected members of the public or at the initiative of the Commission. Each license for operation of a nuclear reactor contains technical specifications which set forth the particular safety and environmental protection measures to be imposed upon the facility and the conditions that must be met for the facility to operate.

Once licensed, a nuclear facility remains under NRC surveillance and undergoes periodic inspections throughout its operating life. In cases where the NRC finds that substantial, additional protection is necessary for the public health and safety or the common defense and security, the NRC may required "backfitting" of a licensed plant, i.e., the addition, elimination or modification of structures, systems or components of the facility.

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, it has also included plant-specific actions needed to close out allegations or other post-licensing items. These activities, and the growth in the number of operating reactors,have resulted in a relatively large number of new actions and pending actions in the inventory. During fiscal year 1987, NRR and the Office of Special Projects completed about 2,900 licensing actions. About 85 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 remained relatively constant in the past, but recently the inventory has decreased from about 4,000 to 3,300 licensing actions under review.

Licensing Actions for Non-power Reactors

As of September 30, 1987, 53 non-power reactors licensed for operation by the NRC were in use for research, training, and testing. Table 3 summarizes the licensing status and fiscal year 1987 licensing actions for non-power reactors.

HEU/LEU Conversion. The first order to convert to low-enriched uranium (LEU) fuel from high-enriched uranium (HEU) fuel was issued to the RensselaerPolytechnic Institute in July 1987. The conversion is in response to the HEU/LEU rule, published in February 1986, whose purpose is to promote the common defense and security by reducing the risk of theft or diversion of HEU fuel used in non-power reactors. The Department of Energy (DOE) has provided funding for conversion to 10 other licensees, and nine of these are planning to submit safety analysis reports for HEU/LEU conversion in fiscal year 1988. (Two NUREG reports were published that evaluate and qualify low-enriched fuels for use in licensed non-power reactors. These are NUREG-1281, "Evaluation of the Qualification of SPERT Fuel for Use in Non-Power Reactors," and NUREG-1282, "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors." (SPERT fuel is fuel used in the Special Power Excursion Reactor Test)).

Petition for Rulemaking Concerning Use of Graphite. The NRC approved for publication in the *Federal Register* a response to a petition which requested that regulations be developed to protect against graphite fires in reactors. The petition hypothesized that the graphite could store sufficient energy from neutron irradiation to constitute a fire hazard. The petition was denied because graphite burning is considered to be a very low-probability (i.e., "non-credible") event in NRC licensed reactors, and its potential is essentially independent of stored energy in graphite. Another relevant consideration is that NRC licensees must and do have approved plans for dealing with emergencies, under existing reactor regulations.

Special Cases

Peach Bottom 2 and 3. On March 31, 1987, the Peach Bottom nuclear power plant, Units 2 and 3, in York County, Pa., were ordered by the NRC to be shut down because of reactor operators sleeping on duty, and other instances of inattentiveness while on duty, which led the NRC to question utility management effectiveness in operating the facility.

The licensee has undertaken a number of actions in response to the Commission's order. These actions are described in the licensee's report, "Peach Bottom Commitment to Excellence Action Plan," as submitted to the NRC on August 7, 1987. The licensee's report describes analyses of the Peach Bottom operations by various licensee and industry groups. The report discusses root causes for the problems cited and identifies areas where changes are being made which address these concerns. The areas include changes in the plant management, attitudinal reassessment and training, development of additional operators and resources, plant procedures, quality assurance, and management involvement and communication.

As of October 1987, the NRC staff is continuing its review of the licensee's submittals and has not reached a conclusion regarding the acceptability of the plan or of the licensee's scheduled plans for resuming power operations at Unit 2. Peach Bottom Unit 3 begins a nominal one year outage in October 1987 to replace residual heat removal and recirculation system piping and for refueling.

Re-evaluation of the GE "Reed Report"

In 1975, the General Electric Company (GE) published an internal product-improvement study titled, "Nuclear Reactor Study," also known as the "Reed Report." GE considered the report a proprietary document and thus, under NRC regulations, exempt from mandatory public disclosure. Members of the NRC staff reviewed the document in 1976 and determined that it did not raise any significant new safety issue.

Recently, in the discovery phase of a lawsuit involving GE and the owners of the Zimmer (Ohio) facility, excerpts from the Reed Report and other internal GE documents were included in documents exchanged between the parties to the lawsuit. The material came into the possession of a newspaper reporter who divulged some of the contents in a news article. Some newspaper accounts alleged or implied that the NRC had conspired with GE to keep this report "secret" from the public. The articles reasoned that the report must contain information that would be damaging to GE if it were disclosed and that there are alleged serious weaknesses in the safety of GE boiling water reactors (BWRs). These articles—together with interest aroused in Congress, officials from the State of Ohio, and concerned citizens—prompted the NRC staff to initiate a thorough re-evaluation of the Reed Report.

The purpose of the NRC staff re-evaluation of the Reed Report was to reconsider the issues and concerns identified in the report. On June 2,1987, NRC established a special task group to perform the re-evaluation.

Taking into account the current knowledge about nuclear power, present regulatory practice, and operating plant experience in the 12 years since the Reed Report was written, the NRC concluded:

(1) The Reed Report does not identify any matters that would support a decision to curtail the operation of any GE BWRs now licensed.

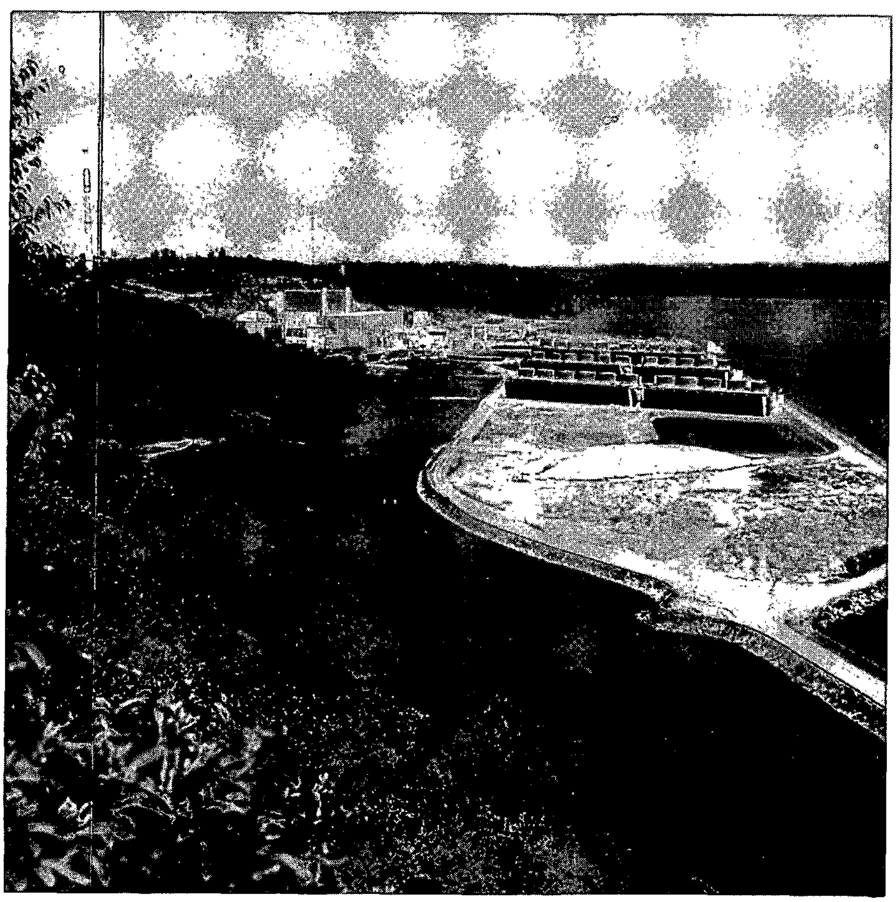
(2) The Reed Report does not identify any new safety issues of which the staff is unaware.

(3) While certain issues addressed by the Reed Report are still being studied by the NRC and industry, there is a basis for permitting continued plant operations while those issues are being resolved.

The results of this review are documented in NUREG-1285, "NRC Staff Evaluation of the General Electric Company Nuclear Reactor Study ("Reed Report")."

Spent Fuel Storage at Nuclear Plants. At the time a utility applies for an operating license, the NRC reviews the capability of the proposed nuclear power reactor facility to safely store spent fuel. All plants have the capacity to store a certain amount of spent fuel on-site. Initially, the nuclear industry anticipated that the spent fuel could be shipped off-site to be stored or reprocessed on a regular basis, and thus only a minimal on-site storage capability would be necessary. However, away-from-reactor storage of spent fuel has proved to be a limited option, and fuel is not being reprocessed in the United States. As a result, the volume of spent fuel at nuclear power plants has been steadily increasing, and the available on-site storage space has correspondingly been shrinking. In order to use available storage space more efficiently, the racks holding the spent fuel have been redesigned. The redesigned racks (known as high density racks) employ a closer spacing of spent fuel assemblies (bundles) and use a boron poison material within the rack to maintain a safe storage configuration. Expansion of storage capacity in this manner is reviewed and approved by NRC staff prior to installation.

The proposed modification to increase the number of fuel assemblies that may be stored in the spent fuel pool is reviewed in a number of areas, including (1) the ability of the cooling water systems to remove the heat produced by the spent fuel so as to maintain a proper pool-water temperature, (2) the ability of the rack to maintain a subcritical condition, and (3) the ability of the rack to main-



Philadelphia Electric Co., owner of the Peach Bottom nuclear power plant in York County, Pa., was ordered by the NRC to shut down Units 2 and 3 of the station in March 1987, because of operator lapses and management problems. In August, the licensee submitted a corrective action plan and a schedule for the restart of Unit 2. At year's end, the NRC staff continued review and evaluation of the plan.

Table 1. Power Reactor Licensing—FY 1987

Fuel-load and Pre-critical Test Operating Licenses issued	2
Low-Power Operating Licenses issued	8
Full-Power Operating Licenses issued	8
Safety Evaluation Reports issued	0
Draft Environmental Impact Statements issued	0
Final Environmental Impact Statements issued	0
Operating Licenses under review	15
Applications cancelled	0
Construction Permits issued	0
Construction Permits under review	0
Manufacturing Licenses issued	0
Manufacturing Licenses under review	0

tain safe spent-fuel storage conditions in the event of an earthquake or the accidental dropping of a fuel assembly. Such modifications are necessary to avoid an unacceptable release of radioactivity to the environment or produce an unsatisfactory working environment at the pool.

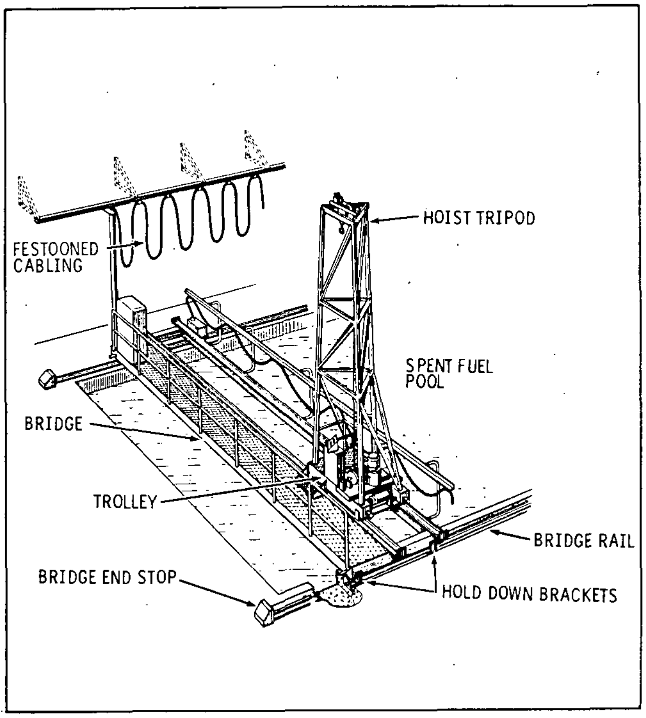
Currently, most reactor facilities have already increased their spent fuel storage capacity at least once. The physical process of increasing storage capacity is to remove the existing fuel storage racks and to replace them with the new high density storage racks. The process is called reracking. The use of high density racks permits more spent fuel to be stored in the spent fuel pool without actually changing the size of the pool. A proposal to increase the actual reactor basin storage capacity is also under review.

At present, licensees are, for the most part, arranging for on-site spent fuel storage capability to accommodate power operation beyond the year 2000. With yet additional storage capacity at the reactor site obtained through modular dry storage (see Chapter 7)—and with provision under the Nuclear Waste Policy Amendments Act of 1987 for a DOE Monitored Retrievable Storage facility—further increases in the storage capability for plants which can store fuel and maintain operation through the year 2005 may prove unnecessary.

Rancho Seco Restart. Following an overcooling event at the Rancho Seco (Cal.) nuclear power plant on December 26, 1985, the unit was brought to cold shutdown. Because of a history of poor operating performance before the overcooling event, the NRC decided not to allow restart of the plant until the factors contributing to poor performance were identified and corrected. Numerous meetings and discussions were held between the licensee and NRC. As a result, the licensee proposed a comprehensive program to improve operating performance and correct long-standing deficiencies at the plant.

During the past year, corrective action on the part of the licensee progressed well. Management at Rancho Seco has been extensively reorganized and restructured to improve management control and accountability; new systems and equipment necessary for more stable plant operation have been integrated; an extensive retraining program for the plant staff has been instituted; a strong maintenance program has been installed; and changes have been made which enhance reliability and ensure better control of the plant during normal operation and upset conditions.

The NRC continues to monitor the progress of the corrective action program by the Sacramento Municipal Util-



Increased requirements for on-site storage of spent fuel at nuclear plants have resulted in the redesigning of storage facilities to accommodate a larger number of spent fuel assemblies in existing pools. This undertaking involves closer spacing of the assemblies and the use of boron "poison" in the storage liquid to ensure safety. These diagrams show, at left, a spent fuel handling apparatus and, below, a two-region high-density storage module, installed at the Florida Power and Light Company's St. Lucie nuclear power plant near Ft. Pierce, Fla.

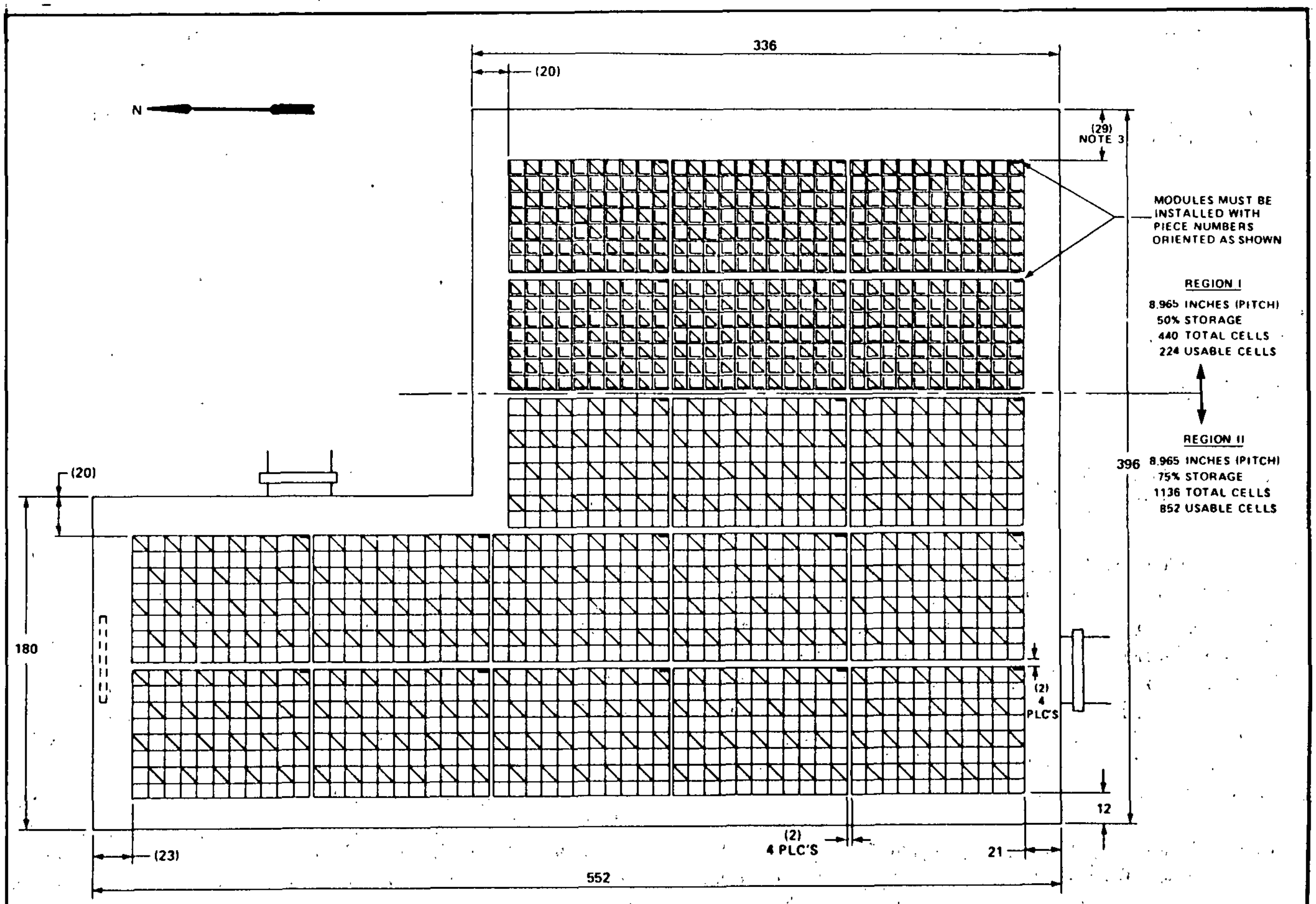


Table 2. Licenses Issued in FY 1987 for Operation of Nuclear Power Plants

Applicant	*Facility*	*Low-Power License*	*Full-Power License*	*Location*
Public Service Co. of New Hampshire	Seabrook 1	10/17/86*		13 Miles South of Portsmouth, NH
Commonwealth Edison	Braidwood 1	05/21/87**	07/02/87	24 Miles SSW of Joliet, IL
Carolina Power & Light	Harris 1	10/24/86	01/12/87	20 Miles SW of Raleigh, NC
Niagara Mohawk Power Corporation	Nine Mile Point 2	10/31/86	07/02/87	8 Miles NE of Oswego, NY
Commonwealth Edison	Byron 2	11/06/86	01/30/87	17 Miles SW of Rockford, IL
Georgia Power Co.	Vogtle 1	01/16/87	03/16/87	25 Miles SSE of Augusta, GA
Duquesne Power & Light Company	Beaver Valley 2	05/28/87	08/14/87	5 Miles East of East Liverpool, OH
Houston Power & Lighting Company	South Texas 1	08/21/87		12 Miles SSW of Bay City, TX
Arizona Public Service	Palo Verde 3	03/25/87	36 Miles	West of Phoenix, AZ
Cleveland Electric Illuminating Co.	Perry1	03/18/86	11/13/86	35 Miles NE of Cleveland, OH
Illinois Power Company	Clinton	09/29/86	04/17/87	22 Miles NE of Decatur, IL

* License authorizes fuel load and precriticality testing only.
** License authorizing fuel load and precriticality testing only, issued on October 17, 1986.

ity District (SMUD) at Rancho Seco. Most aspects of the program have been reviewed for acceptability and have been or are being implemented. An inspection to confirm proper implementation will be made by the NRC, and a detailed safety evaluation report (SER) addressing all areas of activity associated with the improvement program at the Rancho Seco plant will be issued by the staff. The corrective action program is nearly complete; SMUD projects startup of the plant early in 1988.

Erosion/Corrosion at Surry. On December 9, 1986, Unit 2 at the Surry (Va.) nuclear power plant experienced a catastrophic failure of a main feedwater pipe, which resulted in fatal injuries to four contractor workers. Investigation of the accident and examination of data by the licensee, NRC, and others led to the conclusion that failure of the piping was caused by erosion/corrosion of the carbon steel pipe wall. Although erosion/corrosion pipe failures have occurred in other carbon steel systems—particularly in small diameter piping in two-phase systems and in water systems containing suspended solids—there have been few previously reported failures in large diameter systems containing high-purity water. As was the general industry practice at the time of the event, the licensee did not have a regular inspection program for examining the thickness of the walls of feed-water and condensate piping.

Table 3. Licensing Status and Actions for Non-power Reactors—FY1987

(OL = operating license)

Non-power reactor operating licenses	53
Non-power reactor possession only licenses	12
Non-power reactor construction permits	1
Non-power reactor licenses under dismantling orders	5
OL renewals issued for operation	1
OL renewals issued for possession only	2
Orders issued to decommission/dismantle	1
High-enriched uranium to low-enriched uranium conversion orders issued	1
Licenses terminated	1
Facilities planning decommissioning/dismantlement	4
OL renewals under review	2
Other license amendments issued	9

An Information Notice (86-106) was issued on December 16, 1986, to report the event. Two supplemental information notices were issued as the investigation progressed. On July 9, 1986, the staff issued NRC Bulletin 87-01 requiring all plants to submit information regarding their programs for monitoring the thickness of pipe walls in carbon steel piping systems. This information will be compiled early in fiscal year 1988, and as assessment will be made of the status of industry practices for maintaining integrity of high energy carbon steel piping.

Following the Surry event, a number of licensees undertook examinations of their high energy carbon steel piping. Some of these inspections revealed pipe degradation. The most serious condition was that reported from the Trojan (Ore.) nuclear power plant and communicated in Information Notice 87-36 on August 4, 1986.

Industry organizations, such as the Nuclear Management and Resources Council and the Electric Power Research Institute, and individual licensees as well are developing and implementing programs to identify, examine, and repair pipe degradation caused by erosion/corrosion. The results of these programs from a large number of plants will be available in 1988. At that time, the NRC will make a determination as to whether industry practices are adequate to minimize the potential for rupture of high energy carbon steel piping. If industry practices are found inadequate, additional regulatory actions will be taken.

IMPROVING THE LICENSING PROCESS

Standardization

The Commission strongly endorses regulatory policies which encourage industry to pursue standardization of power reactor designs. It is expected that standard designs will benefit public health and safety in a number of ways—concentrating industry resources on common approaches to design problems that will have wide application, stimulating adoption of sound construction practices and quality assurance, fostering constantly improving maintenance and operation procedures, and permitting a more efficient and effective licensing and inspection process. In this regard, on September 15, 1987, the Commission issued a Statement of Policy on Nuclear Power Plant Standardization. The policy reflects the experience the agency has acquired in its review of standard designs, the applicable provisions of the Commission's Severe Accident Policy Statement and of the proposed standardization legislation, and the current views of the Commission and industry on standardization. The focus of the policy is the reference system, design certification, a regulatory instrument that would fulfill the ultimate goal of licensing the construction of plants of certified designs on a pre-approved site. To implement the policy a rule will be proposed. The proposed rule, which would provide a regulatory framework for certification of standard designs, addresses the following subjects: relationship of the new regulatory framework to the existing provisions of Appendices M, N, and O to Part 50; filing requirements; contents of applications; design certification and renewal fees; design certification rulemaking procedures; referral of applications to the Advisory Committee on Reactor Safeguards (ACRS); duration and renewal of design certifications; changes to certified standard designs; and provisions for plant-specific variances.

EPRI Advanced Light Water Reactor Program. The NRC continues to work with the Electric Power Research Institute (EPRI) on an advanced LWR standard plant program. To date EPRI has submitted for NRC review the first four chapters of a 13-chapter "requirements document," treating performance specifications for a total plant in the range of 600 MWe to 1,350 MWe power output. The NRC expects to complete its review of all 13 chapters, and revisions thereof by EPRI, in early 1991.

GE Advanced BWR. General Electric (GE), in cooperation with its international technical associates, is developing a new boiling water reactor design—the Advanced Boiling Water Reactor (ABWR). The ABWR is an advanced design incorporating innovative features from BWR plants around the world. Conceptual work was done in 1978 and 1979, and design development and confirmatory testing proceeded in the years 1980-1985, in a joint effort by GE, Hitachi, and Toshiba. GE receives support for the certification of the ABWR under the Department of Energy's Design Verification Program. It will be the first BWR to be reviewed against the criteria of the Electric Power Research Institute's Advanced Light Water Reactor (ALWR) Requirements Document.

In August 1987, the NRC completed the GE Advanced Boiling Water Reactor (ABWR) Licensing Review Bases document. This document is intended to establish the licensing bases for the design certification review by the staff of the ABWR in accordance with the Commissions Standardization Policy Statement. In September 1987, GE submitted an application for design certification and four volumes of the ABWR Final Safety Analysis Report, covering standard review plan chapters 4, 5, 6, and 15. Design certification for the ABWR is expected tobe completed in late 1991.

Westinghouse RESAR SP/90. The staff continues to review the Westinghouse Electric Corporation application for the Preliminary Design Approval (PDA) for its RESAR SP/90 Nuclear Power Block design, docketed on May 19, 1984. Westinghouse intends to pursue Final Design Approval (FDA) and Design Certification for its RESAR SP/90 design following the issuance of the PDA.

CESSAR-DC, SYSTEM 80+. In March 1987, Combustion Engineering (CE) initiated discussions with the NRC in preparation for the review of the System 80^{S} Nuclear Steam Supply System design. A Final Design Approval (FDA) is expected in late 1990 and Design Certification (DC) by rulemaking in 1991. Thus, CE could reference these approvals in new construction permits and operating license applications in the 1990's. First submittals of the System 80^{S} CESSAR-DC were made late in fiscal year 1987. It will be the first PWR to be reviewed against the criteria of the Electric Power Research Institute's Advanced Light Water Reactor (ALWR) Requirements Document.

CESSAR-F SYSTEM 80. During fiscal year 1987, the NRC continued to review Combustion Engineering's (CE) application to amend the Final Design Approval (FDA) for their CESSAR-F System 80 Nuclear Steam Supply System design. CE plans to reference the FDA in new construction permits and operating license applications. The CESSAR-F FDA, issued on December 21, 1983, applied only to those plants whose construction permit application referenced the CESSAR Preliminary Design Approval (PDA) at the construction permit stage of the licensing process. The staff continues to review Combustion Engineering's amendment request. A decision is expected by the middle of fiscal year 1988.

Integrated Implementation Schedules

Formal scheduling processes have now been incorporated into the licenses for three operating facilities, providing for the implementation of both existing and new requirements according to their relative importance to safety. The staff is considering a similar provision for priority scheduling at a number of operating facilities.

Generic Letter 85-07, issued on May 2, 1985, described the staff's intentions with respect to integrated schedules and solicited industry comments on the development and application thereof. Responses were varied. Some respondents for the industry saw considerable benefit in the orderly scheduling of the implementation of regulatory requirements, according to priorities established through a systematic NRC-approved methodology. Others did not view an integrated implementation schedule as an improvement and expressed no interest in developing such schedules. The staff is considering these responses in developing the policies and practices appropriate to the use of integrated implementation schedules at all operating reactor facilities.

The staff has solicited comment and feedback from industry groups and is working with them on Integrated Implementation Schedules. And the Commission is considering a Policy Statement on this subject, as proposed by the staff in late fiscal year 1987. The Policy Statement will offer options in addition to those discussed in Generic Letter 85-07, as well as definitive guidance in establishing an integrated schedule program.

Backfitting

On August 4, 1987, the U.S. Circuit Court of Appeals for the D.C. Circuit rendered a decision vacating the revised rule in 10 CFR 50.109 on the backfitting of nuclear power plants. The court concluded that the rule—when considered along with certain declarations in the preamble to the rule, as published in the *Federal Register*—failed to speak unambiguously in the language constraining the Commission from considering economic costs when establishing standards assuring adequate protection of the public health and safety. At the same time, the court agreed with the Commission that, once an adequate level of safety protection had been achieved, the Commission was fully authorized to consider and take economic costs into account in ordering further safety improvements beyond this minimum threshold.

The Commission published a proposed amendment to the rule in the *Federal Register* on September 10, 1987, and will amend NRC Manual Chapter 0514 so that it conforms unambiguously to the court's opinion. The proposed revision to 10 CFR 50.109 clarifies the backfit rule to reflect actual NRC policy and practice, i.e., that in determining whether to adopt a backfit requirement, economic costs will be considered only when the backfit in question addresses safety requirements beyond those needed to ensure the adequate protection of public health and safety. Such costs are not to be considered when establishing what constitutes adequate protection of public health and safety.

During fiscal year 1987, the staff considered eight licensee and five staff-identified backfits. Of these issues, 10 have been resolved and three are under review. Of the licensee-designated backfits, seven were found not to be backfits within the meaning of the Backfit Rule, and one is under NRC staff review.

Technical Specification Improvements

On February 10, 1987, the Commission issued an interim policy statement on Technical Specification improvements for nuclear power plants. The policy 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 licensee. The application of the criteria will permit the relocation of some Technical Specification requirements to licensee-controlled documents and programs. This will permit subsequent changes to those commitments without prior NRC approval, when appropriate technical evaluations are performed under approved administrative controls. Those requirements that have a more significant impact on safety will remain in the Technical Specifications, and both the requirements and their bases will be upgraded to provide greater emphasis on human factors and clarity.

The staff has been evaluating programs developed by representatives of the NSSS vendor owners groups for the development of new Standard Technical Specifications (STS). The new STS will be used by licensees to improve the Technical Specifications for individual plants. The results of the first phase of the NSSS vendor owner groups improvement programs—i.e., model specifications, trial criteria applications, and generic positions—have been reviewed and approved by the staff. This review will serve as the basis for proceeding with the next program phase involving the development of a complete new set of STS.

The NRC and industry are continuing with a program of specific line-item improvements to both the scope and substance of existing Technical Specifications, in parallel with the complete rewrite of the STS mentioned above. Thus, on June 4, 1987, the staff issued a generic letter that describes alternatives to several general Technical Specification requirements. The alternatives will remove unnecessary restrictions on plant operations. These kinds of improvements will be incorporated into the new Standard Technical Specifications.

INSPECTION PROGRAMS

In the inspection sphere for fiscal year 1987, the focus continued to be on plants with problems calling for special attention. NRC headquarters and regional office inspection personnel were integrally involved in the agency's effort to investigate and resolve various significant plant design, installation, equipment, and performance problems at plants in both construction and operational phases. Alternative approaches within the reactor inspection program were exercised to redirect inspection resources from plants with a high level of performance to plants with marginal performance.

Regional Administrators and NRC headquarters management were cooperatively involved in the agency's programs for identifying those facilities where a very good or generally poor performance on the part of the licensee justified a reduction or increase in inspection effort. Substantial progress continued to be made in developing a program of performance indicators to track the changes in each plant's performance.

Special team inspection programs, such as the Safety System Functional Inspection and the Safety System Outage Inspection—as well as implementation by the Regional Offices of the routine and reactive inspection programs in fiscal year 1987—continued to be employed as proven and effective tools in assessing the operational readiness of key plant safety systems.

The NRC reorganization in April 1987 resulted in the reassignment of responsibility for administering the reactor inspection program into activities of the Office of Nuclear

Reactor Regulation. The responsibility for developing, maintaining, and assessing the effectiveness of the reactor inspection program is now shared among NRR staff, consistent with their assigned technical responsibilities. Improvements have been made in a number of inspection programs, and measures have been taken since the reorganization to restructure the reactor inspection program so as to focus headquarters and regional inspection effort on those plant operations which contribute most to ensuring reactor safety. Current NRC plans for restructuring the reactor inspection program, and providing for the integration of the inspection and licensing programs, will be implemented in fiscal year 1988.

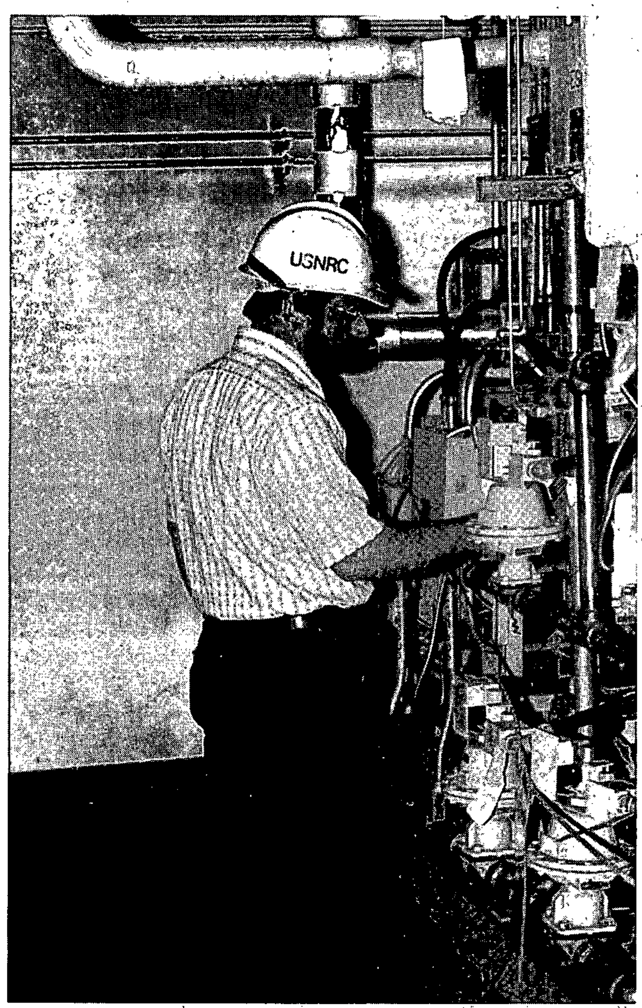
A basic element in NRC reactor regulation is the inspection of licensed reactor facilities to determine the state of reactor safety, to confirm that the operations are in compliance with the provisions of the license, and to ascertain whether other conditions exist which have safety implications serious enough to warrant corrective action. The inspection programs of the NRC are mainly carried out by the five NRC Regional Offices. As described later in this report, a limited number of inspection programs are conducted directly by NRC Headquarters. During fiscal year 1987, the Office of Inspection and Enforcement and, subsequent to the April reorganization, the Office of Nuclear Reactor Regulation, were responsible for developing inspection policies and procedures and for monitoring and assessing the effectiveness and uniformity of the programs carried out by the NRC Regional Offices. (Regional Offices are now under the supervision of the NRC Deputy Executive Director for Regional Operations.)

In addition to 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 which occur at individual plant sites or other facilities involving licensed operations ("reactive" inspections). In conducting these 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 both region-based and resident inspectors. In general, region-based inspectors are specialists, while resident inspectors are generalists. Resident inspectors provide the major on-site NRC presence for direct observation and verification of licensee 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 nuclear systems important to safe operation; and frequent plant tours to generally assess housekeeping, radiation control, security, equipment condition, and the like. The resident also acts as the primary on-site evaluator for the NRC inspection efforts related to licensee event reports (LERs), events, and incidents. Residents 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, non-destructive testing, refueling, quality assurance, training, core-physics testing, radiation protection, emergency planning, environmental protection, security/safeguards, and licensee management systems.

Development and utilization of an innovative inspection approach to appraise the functionality of safety systems at operating plants continued in fiscal year 1987. The new



Innovation and improvements in NRC inspection methods continued to prove useful in 1987. A new technique, called a Safety Systems Functional Inspection, facilitates identification of significant safety issues requiring prompt licensee corrective action. Shown here, an NRC inspector checks radiation levels in a reactor radiation waste evaporator.

methodology, termed a Safety Systems Functional Inspection (SSFI), was included in the reactor inspection program for implementation by the Regions in fiscal year 1986. It continues to prove its usefulness in regional inspections by identifying significant safety issues that require licensee corrective actions. Another approach, the Safety System Outage Modification Inspection, 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 its demonstrated success, this method will also be included in the reactor inspection program for implementation by the Regions in fiscal year 1988.

Finally, a Probabilistic Risk Assessment (PRA) pilot program, initiated in fiscal year 1986 on a trial basis, is being continued to further develop methods for using risk information to focus inspection activities on those areas of reactor plant design and operations that contribute the most to potential accident risk to the public. The results of four PRA initiatives will be evaluated in fiscal year 1988 and will be employed in future decisions on methods to be used in the routine reactor inspection program. All four of the PRA initiatives employ two interactive computer programs to calculate core-melt frequency and risk values, based on existing plant conditions. One inspection approach will be used for plants for which a PRA is available and a different one, using generic PRA information derived from surrogate plants, will be adopted for other plants.

In fiscal year 1987, the staff continued activities inaugurated in 1986 to provide for the more efficient use of limited inspection resources.

The operating reactor inspection program is divided into three sub-programs—Minimum, Basic, and Supplemental—to provide a priority for implementation. Over the past year, the inspection procedures in each subprogram were categorized into functional areas (operations, maintenance, surveillance, training and qualification, etc.). The functional areas are identical to those used to evaluate licensees in the Systematic Assessment of Licensee Performance (SALP) program (see discussion below). This scheme has allowed the allocation of inspection resources to a given plant to take account of the results of the latest SALP evaluations, in order to concentrate those resources in precisely the functional areas in need of safety improvement.

Special Verification Of Multiple Plant Actions

In fiscal year 1987, the NRC program for verification of licensee resolution of significant safety issues was greatly advanced by further development and implementation of the Safety Issues Management System (SIMS; see Chapter 11 for details). Special instructions were issued setting forth inspection requirements and giving guidance in verifying licensee responses to NRC directives related to high priority safety issues affecting multiple plants. The following multiple plant issues were addressed in these special instructions:

- Verification of compliance with the order for modification of licenses regarding primary coolant system pressure isolation valves. The order requires periodic testing of certain check valves to verify that the valves are seated properly and function as a pressure isolation device. Such testing is intended to reduce the overall risk of a loss-of-coolant accident.
- Verification that boiling water reactor (BWR) licensees with Mark I containment designs have modified their plants in accordance with formal commitments. The key safety issue here is the effect of the suppression pool hydrodynamic loads on the containment that were not considered in the original Mark I containment design basis.
- Verification that licensees have an instrumentation system which meets the criteria set forth in Regulatory Guide 1.97 for assessing plant conditions during and following an accident. Safety issues relate to instrumentation used in emergency response facilities, including the control room, the technical support center, and the emergency operations facility.
- Verification that safety-grade equipment will not be damaged by flooding resulting from the failure of non-safety-grade equipment. Examples of equipment and components whose functions may be affected by flooding are motor control centers, electrical switchgear, batteries, diesel generators, and pump and valve controls. Areas susceptible to flooding or water impingement may be adjacent to water supplies for fire suppression, general service, and cooling.
- Verification that boiling water reactor (BWR) licensees have taken long-term corrective actions to ensure scram discharge volume capability. The safety issue relates to improved reliability of the control rod drive and scram systems.
- Verification of modifications to improve the reliability of reactor trip system circuit breakers. These corrective actions were required as a consequence of the Salem Unit 1 (N.J.) event in which reactor trip system breakers failed to open upon receipt of a valid trip signal.
- Verification that pressurized water reactor (PWR) licensees have implemented programs for the control of natural circulation cooldown. The requirement involves training programs and facility procedures that deal with the prevention or mitigation of reactor vessel boiling during natural circulation cooldown.
- Verification that BWR licensees have performed inspections of stainless steel piping welds susceptible to intergranular stress corrosion cracking. This action derives from the discovery of intergranular stress corrosion cracking in large diameter recirculation and residual heat removal piping systems.

This NRC inspector is following up on his earlier finding that some bolts were missing during construction of the outside support structure at a nuclear power plant. During 1987, new emphasis was placed on the early identification and prevention of such problems at construction site.

- Verification that PWR licensees have an effective mitigation system for low-temperature overpressure transient conditions. This relates to design reviews, procedure changes, equipment modifications, operator training, and surveillance to mitigate severe pressure transients while at a relatively low temperature.
- Determination of compliance with the anticipated transients without scram (ATWS) rule (10 CFR 50.62). The determination pertains to the prescribed reduction of risk from ATWS events for light-water-cooled nuclear power plants.
- Reduction of routine inspection efforts at the top-performing operating plants. This program, called the Special Minimum Program, carries certainprecautionary features which (1) limit the number of plants in each Region that can simultaneously be placed on the program, (2) provide criteria for plant selection, (3) require periodic regional review of plant performance to justify continuation of the program at plants at which it is implemented, and (4) provide guidance regarding the scope of resident inspection to be in effect while the plant is subject to this program. During fiscal year 1987, these facilities were placed on the Special Minimum Program: Farley (Ala.), St. Lucie (Fla.), Monticello (Minn.), Prairie Island (Minn.), and Kewaunee (Wis.).
- The program begun in 1985 to place a second resident inspector at single-unit operating reactor sites continued during the report period. As of the end of fiscal year 1987, all second residents had been placed, consistent with the fiscal year 1987 staffing plan. Placing additional residents at single-unit operating reactor sites has allowed increased coverage for both routine and reactive on-site inspections.

New procedures were added to the program addressing the following areas: (1) fire protection, (2) second resident inspector at a single-unit site, (3) inservice testing, (4) masonry walls, and (5) safety system functional inspections.

At sites where reactor plants are under construction, program requirements were revised to focus greater attention on the early identification and prevention of problems at construction sites. The revision calls for increased depth of construction inspections; special emphasis on any plant area that was assigned a SALP-3 rating; earlier resolution of allegations; and the training, qualification, and performance of construction workers and inspectors.

Safety Systems Inspections. The comprehensive inspection of safety system operational readiness by the Safety Systems Functional Inspection (SSFI) teams has its roots in the Performance Appraisal Team (PAT) program. The team examines system design, maintenance of key items, modifications, system configuration, and operational history for selected systems critical to safety. The comprehensive inspection is focused on the operational readiness of the system and is oriented towards the hardware rather than the licensee's programs.

The effort continued during the report period with SSFIs performed at Cooper (Neb.) by a headquarters team and at H. B. Robinson (S.C.), Monticello (Minn.), and WNP-2 (Wash.) by the Regional Offices. In addition, a headquarters team used SSFI techniques to conduct a series of engineering-based team inspections at Rancho Seco (Cal.), as part of the staff's review of that facility's readiness for startup. Deficiencies within the safety systems inspected were identified at all the facilities visited by the SSFI teams.

The effects that modifications have on the design, configuration, and function of safety systems are determined by the Safety Systems Outage Modifications Inspection (SSOMI) teams. A multi-team approach is used to follow the licensee's progress with system and equipment changes during major outages. The teams assess the design, procurement, installation, inspection, and testing of changes made to safety systems and decide whether the systems are ready for plant startup. SSOMIs were performed at Sequoyah Unit 2 (Tenn.) and Indian Point Unit 3 (N.Y.) during the past year. The program is evolving to provide a more effective approach for assessing plants during a major outage, as more experience is gained in the inspection of modifications to safety systems. The SSOMI program is being incorporated into the routine inspection program for use by Regions as needed.

Plant Operations. The Operational Safety Team Inspection (OSTI) was developed to permit a customized team inspection based on selected plant performance indicators. Areas of licensed activities are selected for in-depth inspection by reviewing available relevant information, including Systematic Assessment of Licensee Performance (SALP) ratings, enforcement history, and licensee event reports. Although the teams' focus is on operational safety through around-the-clock control room surveillance, other areas of inspection have included design controls, plant security, corrective action systems, licensee management oversight, maintenance, and system function surveillance. The OSTI draws on the expertise and techniques developed through the PAT inspections for the operational safety aspects of the inspection. OSTIs have been conducted at Crystal River Unit 3 (Fla.) and Fermi Unit 2 (Mich.) during the past year.

Other Inspections. NRR's special inspection staff provided assistance to other offices of the NRC during 1987. Inspections were conducted at Sequoyah Unit 2 (Tenn.) to provide the Office of Special Projects (OSP) with information regarding the ability of that TVA plant to start up safely. These inspections included an SSOMI, an independent design inspection, and several other inspections tailored to investigate specific design issues. In addition, assistance was provided to OSP in evaluating TVA's resolution of employee concerns at several of TVA's plants. The staff assisted Regions II (Atlanta) and III (Chicago) with balance-of-plant systems inspections. These inspections evaluated maintenance, operation, and configuration of non-safety systems that can malfunction and challenge or disable the systems required for safe operation of the plant.

Non-destructive Examination Program

Since 1981, the NRC has been operating a mobile non-destructive examination (NDE) laboratory to conduct inspections at nuclear power plants throughout the country. The NDE van has also been used to provide independent findings in connection with the investigation of various allegations registered with the NRC. The NDE facility is operated out of the Region I Office (Philadelphia).

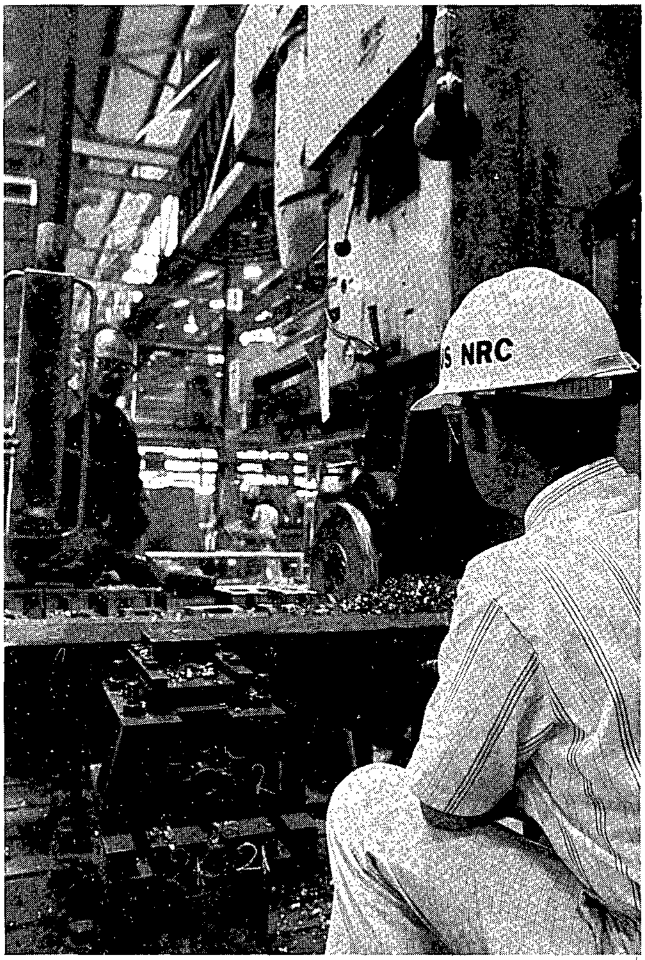
The mobile laboratory is capable of performing radiographic, ultrasonic, liquid penetrant, and magnetic particle examinations. It is also employed in carrying out visual examinations of piping, pipe support, and structural welding; along with testing of concrete and electrical cabling; the van is also equipped with a dark room for developing radiographic film. The laboratory is staffed by three NRC Region I (Philadelphia) personnel, supplemented by two contractors. The lead NRC engineer is qualified as a Level III examiner by the American Society for Nondestructive Testing (ASNT). The other two NRC personnel and the two contractors are qualified to at least ASNT Level II, in the disciplines applicable to the program.

Vendor Inspection Program

NRR vendor inspections in fiscal year 1987 continued to focus on vendor activities associated with nuclear plant operation, maintenance, and modifications. Inspection emphasis is on the quality of the vendor products, the licensee/vendor interfaces, environmental qualification of equipment, equipment problems found during operation, and corrective action in response to identified problems. Inspections of vendors and contractors are based on information from a variety of sources, including licensee construction deficiency and operating reactor event reports, vendor reports of product defects, 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 through its inspection programs.

The NRC performed approximately 100 vendor inspections during fiscal year 1987. The most frequently conducted inspections involved component manufacturers, equipment qualification test facilities, equipment problems at licensees' facilities, and licensees' equipment qualification programs. Inspections of component manufacturers primarily involved circuit breakers, electrical cable, splicing, valves and their operators, transmitters, diesel engines, pressure switches, fire protection components, electrical connectors, and fasteners. Other inspections were directed toward design organizations, fuel fabricators, material suppliers, and licensees. Eight inspections were conducted specifically to resolve allegations. In approximately one half of inspections, expert assistance was used from outside contractors, including the National Laboratories of the Department of Energy.

A significant effort was directed toward coordinating and providing technical support for the inspection of electrical equipment and qualification activities at operating reactors. The environmental qualification rule (10 CFR 50.49) requires that all safety-related electrical equipment that could be exposed to a potentially harsh environment be qualified by testing to demonstrate its operability during normal and "design basis accident" environmental conditions. Twenty-eight operating nuclear plant sites were inspected during



About 100 NRC inspections were conducted in 1987 of vendors providing safety-related hardware and components to nuclear power plants. Here, an inspector observes a milling operation at a factory where valves are made.

fiscal year 1987. Several significant qualification problems were identified and corrected. One of these involved failure by several licensees to provide adequate environmental qualification for cable splices and terminations. Information Notice 86-53, "Improper Installation of Heat Shrinkable Tubing" was issued to alert all licensees to this problem. The equipment qualification inspections are continuing, and it is anticipated that all licensed plants will be inspected by mid-1988.

Operational Safety Issues. The Vendor Inspection program has been the agency's principal instrument for dealing with several significant operational safety concerns, such as assuring the operability of check valves, verifying the quality of fasteners installed in safety systems, addressing technical deficiencies in safety-related circuit breakers, and reviewing the effect of safety-related procurement practices on the quality of installed hardware at nuclear power plants. With regard to check valves, the NRC has performed in-depth reviews of the maintenance and testing practices for check valves at five nuclear power plants, in order to provide the information necessary for decisions on the future level of NRC involvement in this area.

In the area of fastener quality, the staff has performed inspections of fastener manufacturers, suppliers, and users, and has performed mechanical and chemical destructive testing on a sample of safety-related fasteners obtained during these inspections. Serious concerns in the area of safety-related circuit breaker quality have resulted in a number of inspections at manufacturers, distributors, and nuclear power plants to investigate both electrical and mechanical problems with these components. Corrective action to resolve the deficiencies in these components was under development at the close of the report period.

The NRC has determined that the controls established and implemented by utilities do not always properly categorize plant items with regard to their safety function; do not always result in the specification of the necessary requirements to assure high quality; and do not always impose the necessary analysis, testing, or inspection efforts to properly dedicate a commercial grade item for a safety-related application.

APPRAISAL PROGRAMS

Systematic Assessment of Licensee Performance

Under the NRC program for the Systematic Assessment of Licensee Performance (SALP), 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 relevant to each facility.

The SALP process entails an integrated assessment based on manifold appraisals as to how licensee management directs, guides, and provides resources for the assurance of safety. The purpose of the SALP review is to direct both NRC and licensee attention and resources toward exactly those areas which can affect nuclear safety and 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 ultimately consists of performance evaluations in certain specific functional areas, including plant operations, maintenance, surveillance, emergency preparedness, and so forth.

The SALP process is currently being refined to improve implementation, and consideration is being given to redefining the functional areas to be rated. Changes are also being considered to ensure that the SALP report provides a useful synthesis of licensee performance information and identifies trends or characteristics of that performance.

The systematic assessment program supplements normal regulatory processes and is intended to be sufficiently diagnostic to provide meaningful guidance to utility management regarding NRC concerns about quality and safety in plant construction or plant operation. The results of the program comprise part of a data base for periodic reporting in the historical data summary which was published in "Historical Data Summary of the Systematic Assessment of Licensee Performance" (NUREG-1214, October 1986).

Special Reactor Inspections

The reassignment of inspection functions under the agency reorganization resulted in the creation of NRR's Special Inspection Branch. The Safety System Functional Inspections (SSFIs) and Safety Systems Outage Modifications Inspections (SSOMIs) developed previously have been continued under the new organization. Although Performance Appraisal Team (PAT) inspections are not now being performed, the techniques and expertise developed by the PATs are being used in a newly developed team approach to appraising the quality of plant operations. The Operational Safety Team Inspections (OSTIs) evolved from the need for a method of assessing the operation of nuclear power plants that took selected performance indicators into account.

New and innovative inspection techniques, such as the OSTI, are being developed and tried by the Special Inspection Branch. The techniques and programs that prove effective are then made a part of the regular NRC inspection program. The SSFI technique was made a part of the Operations Phase of the Reactor Inspection Program so that it may be employed by the Regions as needed.

Performance Evaluation

The NRC has initiated a program to improve its ability to evaluate the effectiveness of nuclear power plant licensee performance. This effort will integrate the input from programs and activities such as SALP, enforcement, performance indicators, trend analysis by the Office for Analysis and Evaluation of Operational Data, event evaluation, operator examinations, and licensing and inspection. The effort will identify the need for, and recommend special inspections and management attention for input to, performance evaluation.

QUALITY ASSURANCE

Quality Assurance Program Plan

NRC activity in the area of Quality Assurance (QA) during the report period continued along lines recommended in last year's QA Report to Congress entitled "Improving Quality and the Assurance of Quality in the Design and Construction of Nuclear Power Plants" (NUREG-1055), as modified following public comment and guidance from the ACRS and the Commission. The four major areas receiving staff emphasis are inspection programs, software QA, procurement of commercial grade items, and QA training. Activities and accomplishments in these areas are described below. In addition, NRR is providing QA guidance or consultation to the other program offices of the NRC on such issues as decommissioning, independent spent fuel storage installations, and QA standards development.

Inspection Programs for Quality Assurance

Readiness Reviews. Readiness Reviews were identified as a topic for further analysis in the QA Report to Congress. A Readiness Review is a formal assessment of the licensee's readiness to construct or operate a nuclear power plant. It is a comprehensive evaluation of the licensee's program for design, construction, and pre-operational activities, so that issues and problems are identified at a stage when they can best be resolved.

A Readiness Review was completed for Georgia Power Company's Vogtle Unit 1 in January 1987. The program provided significant added assurance that Vogtle Unit 1 licensing commitments and NRC regulations have been adequately implemented. The results are reported in NUREG-1278, "Vogtle Unit 1 Readiness Review Assessment of Georgia Power Company Readiness Review Pilot Program." Because of the success of this program, Georgia Power Company initiated a Readiness Review of Vogtle Unit 2 in mid-fiscal year 1987.

Quality Verification Functional Inspections. Quality Verification Functional Inspections (QVFIs) are conducted to assess the effectiveness of licensees' quality verification organizations in identifying and obtaining correction of safety-significant technical problems and deficiencies in plant systems and operations. QVFIs also serve to measure the licensees' line management effectiveness in promptly resolving identified problems and deficiencies.

QVFIs are intended to improve reactor safety by emphasizing to licensees that their qualify verification organizations should have the ability to detect and understand significant operational safety problems in a technically creditable manner, thus helping ensure that NRC requirements are satisfied. These inspections are led by NRR staff with multi-regional participation and support. QVFIs focus on technical, safety-significant issues, rather than being limited to QA programmatic reviews.

Three quality verifications have been conducted thus far, at Indian Point 2 (N.Y.), Catawba (S.C.), and Arkansas Nuclear 1. These inspections have identified safety-significant technical problems and deficiencies and have successfully emphasized to the licensees the importance of having their quality verification organizations involved in the daily activities during the operations phase to help ensure safe operations. The NRC plans to conduct six or seven QVFIs annually.

QA Inspection Procedures. Consistent with the recommendations of the QA report to Congress, the staff is reorienting the NRC QA inspection program for operating reactors to provide proper emphasis to QA program performance and effectiveness. QA inspection procedures that emphasize program implementation and QA program effectiveness are being developed and will be incorporated into the NRC inspection program.

Computer Software Quality

There has been an ever increasing use of computers in the nuclear industry, with increasingly sophisticated computers and computer software. The proliferation of mini-, micro-, and personal computers brought about the more widespread use of computers for engineering calculations and other technical applications at reactor sites. There has also been a marked increase in the number of companies supplying computers and computer software.

The NRC published a "Handbook of Software Quality Assurance Techniques Applicable to the Nuclear Industry" (NUREG/CR-4640), in August 1987. The publication prescribes good engineering practices in the application of 10 CFR 50, Appendix B requirements to assure quality in the development and use of computer software for the design and operation of nuclear power plants. Inspection procedures are being developed to address QA for software use in nuclear applications, and NRC inspections are being planned to assess the effectiveness of industry's QA programs in the development and use of computer software in nuclear applications.

Procurement Quality Assurance

NRC staff have developed draft procurement inspection guidance for determining the suitability of application of commercial grade items for use in safety-related applications. The emphasis of this effort is on guidance for evaluating the engineering effort necessary in the identification of an item's critical characteristics. Pilot inspections have been performed at Donald C. Cook (Mich.) and Peach Bottom (Pa.) to confirm that industry's procurement practices are being appropriately addressed by the guidance. After additional pilot inspections, the guidance will be incorporated in a temporary inspection procedure and in the NRC inspection program.

NRC Inspection Training

A new training course, "Inspecting for Performance," has been developed to help the agency move its Quality Assurance inspection emphasis from QA programmatic inspections toward performance-oriented, technical-based inspections. The recent assumption of QA inspection responsibility by all NRC inspectors has made the new course a timely and important addition to the NRC's training program.

A second course, "Effective Communications with Licensees," has been developed and will be presented to the inspectors during fiscal year 1988. This course is designed to enable NRC inspectors to have more effective communications with licensee personnel during inspections and inspection-related entrance and exit meetings.

EMERGENCY PREPAREDNESS

Licensing and Inspection Activities

During the report period, NRC staff continued to evaluate the adequacy of applicant on-site plans to be included in the Safety Evaluation Report, and supplements thereto, for each nuclear power plant in a near-term operating licensing status (designated NTOLs). The staff also took part in licensing hearings before Atomic Safety and Licensing Board panels and served on inspection teams appraising applicants' implementation of emergency preparedness programs and their full-participation exercises. NTOLs appraised during fiscal year 1987 included facilities at Beaver Valley 2 (Pa.), Byron 2 (Ill.), South Texas 1 (Tex.), and Palo Verde 3 (Ariz.). Pre-licensing activities also included an evaluation by the NRC and the Federal Emergency Management Agency (FEMA) of a full participation emergency preparedness exercise at the South Texas site.

The staff continued to implement the recommendations of an Emergency Preparedness Task Group formed during the previous fiscal year to evaluate the emergency preparedness inspection program for nuclear power reactors. Several revised inspection procedures were issued and the staff has coordinated with FEMA in addressing generic emergency preparedness issues. Several controversial issues were also addressed regarding the continuing Shoreham (N.Y.) and Seabrook (N.H.) licensing actions. A change in program emphasis away from the review of emergency plans prior to plant licensing toward the inspection of emergency preparedness programs at operating reactors was reinforced by the April 1987 reorganization that brought the emergency preparedness functions into the Office of Nuclear Reactor Regulation.

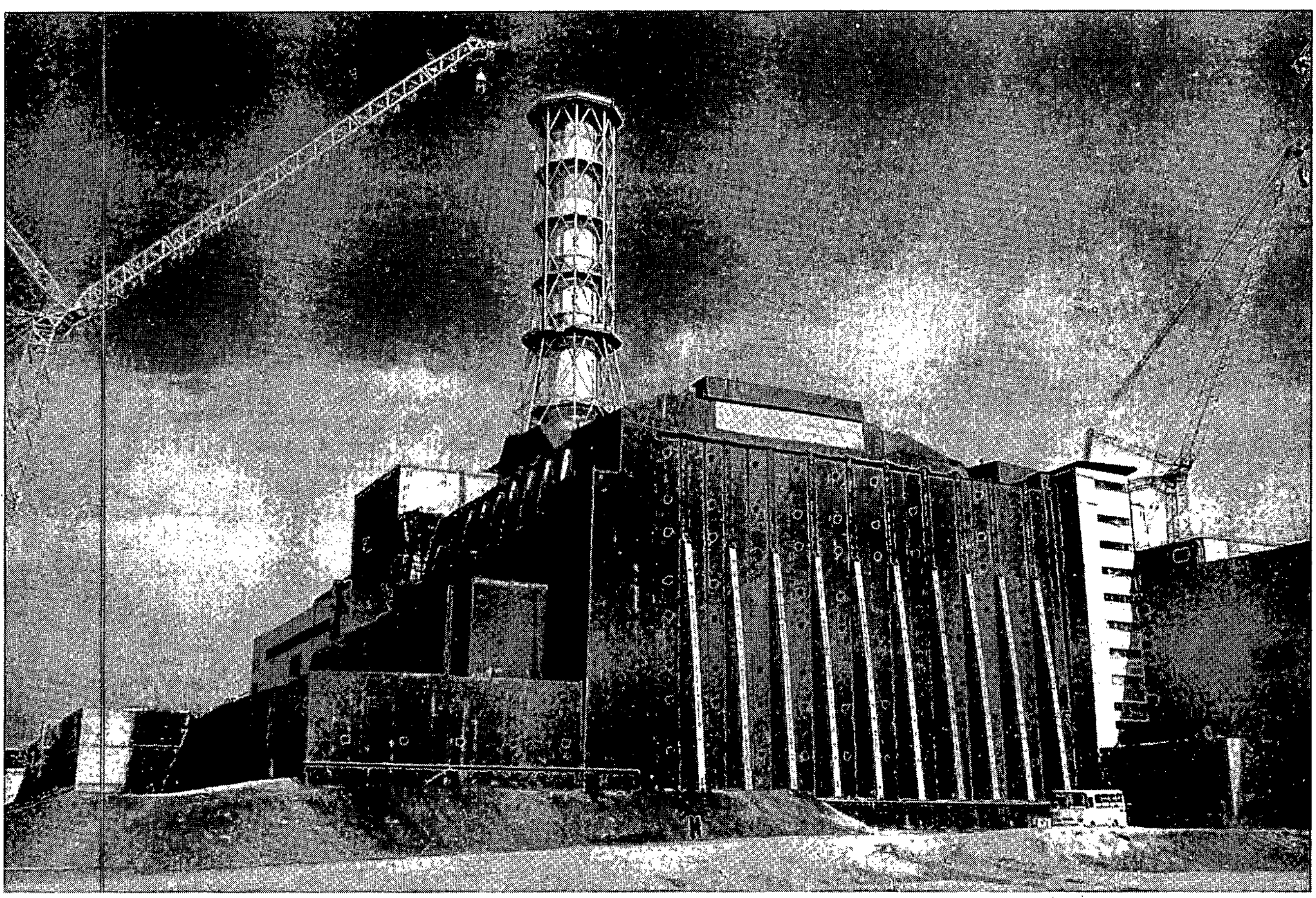
Chernobyl Accident Implications

A prime preoccupation of NRC's emergency preparedness staff during the report period was that with the agency's follow-up to the April 1986 Chernobyl accident. The staff participated in the fact-finding activities that resulted in the "Report on the Accident at the Chernobyl Nuclear Power Station" (NUREG-1250), in January 1987. In addition, an assessment of the implication of the accident as it might affect policies and practices for commercial nuclear reactors was undertaken. The findings with regard to emergency planning are presented in draft report, "Implications of the Accident of Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States" (NUREG-1251, August 1987).

Evaluation Of On-site Meteorological Data

The review of meteorological data collected continuously at operating nuclear power plant sites is an integral part of NRC monitoring to assure that plants are operated in accordance with NRC regulations. Reliable and representative meteorological data must be available for use in emergency response and in assessments of the radiological impacts of routine and accidental releases. For emergency response, the meteorological data are used in identifying effluent plume arrival and transit times, leading to recommendations on proper designation of areas for which protective actions and the deployment of environmental sampling teams and emergency personnel would be warranted.

During fiscal year 1987, in-depth evaluations of meteorological data were completed for four plants. That brings to nine the number of plants evaluated over the past two years in the pilot study. The overall quality of the meteorological data was examined, with particular attention given to wind direction, wind speed, and atmospheric stability. These are the most significant meteorological variables consulted during emergency response and in routine radiological dose assessment. The evaluations gave the staff a perspective on the effectiveness of the upgrading



Entombment of the ill-fated Chernobyl Nuclear Power Station has been completed, according to Soviet officials. The U.S.S.R. provided this photo of the sealed plant to NRC officials during an international conference held in 1987.

of meteorological facilities after the TMI-2 accident. Good meteorological monitoring programs are important to the protection of the public health and safety from potential radioactive releases from nuclear plants.

HUMAN FACTORS

Policy Issues

In 1986, the staff completed Revision 2 to Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Plant," and a corresponding change to Standard Review Plan Section 1.3.1.2, "Operating Organization." In 1987, the revised Regulatory Guide and Standard Review Plan Section were published in the *Federal Register* to obtain public comment (52 FR 1979, January 15, 1987.)

The comment period ended on March 15, 1987. The Commission directed the staff to develop a policy statement on the professional conduct of control room operators. It is expected that a proposed policy statement will be issued in early 1988.

Management and Organization

The NRC is now focusing greater attention on certain licensed operations whose management performance appears to be weak. In addition to evaluating leadership and management practices and their impact on nuclear operational performance, the NRC is also evaluating the overall "organizational environment/operator culture" to determine what effects it is having on plant performance. While leadership and management practices deal with effective management principles and skills, organizational environment/operator culture focuses on attitudes, norms, practices, and history, and their role in creating an atmosphere that affects nuclear operational performance. During the report period, NRR performed such evaluations at Peach Bottom (Pa.) and the Davis-Besse (Ohio) nuclear power plants, in response to incidents of inattentiveness to duty, and provided support in this area in a diagnostic inspection at the Dresden (Ill.) plant.

Procedures

The NRC is continuing to implement a long-term program to upgrade licensees' emergency operating procedures (EOPs). The program was initiated shortly after the Three Mile Island (Pa.) accident in 1979. The objectives of the program are to improve the technical content of EOPs and also to improve them through the application of human factors principles. Owners Groups, representing the four nuclear power plant vendors, have satisfactorily re-analyzed 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 NUREG-0899.

The staff continues to evaluate industry's efforts to upgrade EOPs by reviewing Procedures Generation Packages (PGPs) from operating reactors and license applicants.

To evaluate the effectiveness of the NRC's long-term program for upgrading emergency operating plans, the staff continued auditing the implementation of PGPs at selected plants. Based on input from PGP implementation audits, staff PGP reviews, and license examiners, the staff has identified certain problems that licensees are experiencing in implementing their PGPs. During the report period, the staff issued Information Notice 86-64, Supplement 1, to alert all facilities that significant problems continue to be identified in all major aspects of licensee EOP upgrade programs.

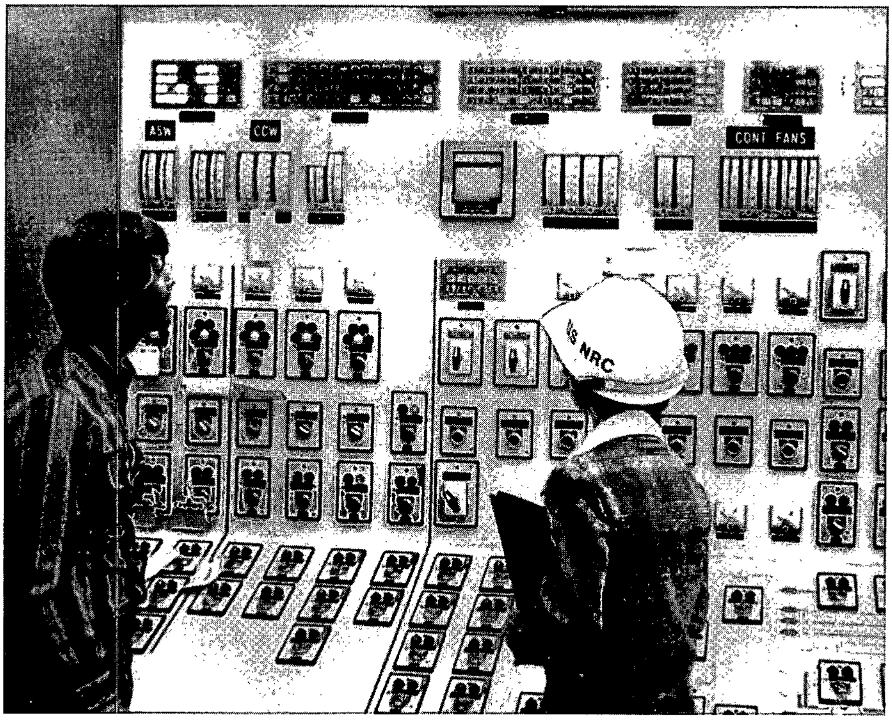
The staff has recently developed a program plan to integrate Headquarter's review of PGPs with the Regions' inspection of licensee's implementation of the PGP: "Temporary Instruction, Inspection of Emergency Operating Procedures (TI 2515/79)." The program being proposed is designed to expedite the PGP review and EOP inspection process through a team approach.

The NRC's original review of the Owners Groups' generic technical guidelines turned up certain unresolved technical issues, and the staff continues working with each Owners Group to resolve them.

Man-Machine Interface

Staff reviews on man-machine interface continued during the report period in three areas: Detailed Control Room Design Reviews (DCRDR), Safety Parameter Display Systems (SPDS), and Salem Anticipated Transients Without Scram (ATWS), Item 1.2—"Data and Information Handling Capabilities." By the end of fiscal year 1987, DCRDR Safety Evaluation Reports (SERs) were issued for approximately 70 percent of all plants, and SPDS Safety Evaluation Reports were issued for approximately 75 percent of all plants. Supplemental DCRDR and SPDS SERs closing out remaining issues will be issued for many of these plants during the next several years. It is anticipated that all plants will satisfy NRC DCRDR and SPDS requirements by the end of fiscal year 1991.

In 1986, Information Notice (86-10) was issued to the industry describing some serious weaknesses found in SPDSs during conduct of a pilot audit program at six plants. By the end of fiscal year 1987, an additional 15 SPDS Post-Implementation Audits were conducted with similar results. The staff will be conducting these audits at all plants until the end of fiscal year 1991. In fiscal year 1988, a pilot program of DCRDR Post-Implementation audits will be started to determine how well the industry has implemented commitments for control room modifications.



By the end of FY 1987, the NRC had issued Safety Evaluation Reports for approximately 70 percent of all nuclear power plants in the nation, covering such matters as control room design and safety parameter display systems. Post-implementation audits continued during the year and are expected to go on through FY 1991. Checking industry progress in meeting commitments for control room modification are an important part of these audits.

Training

During fiscal year 1987, the staff continued to evaluate the results of the Institute of Nuclear Power Operations (INPO) accreditation program, in order to determine whether the industry's voluntary efforts will ensure that training is sufficiently performance based. As part of this evaluation effort, the staff participated as observers when utilities' training programs were under examination by an INPO accreditation team. The staff also conducted five post-accreditation reviews during this report period.

The staff completed its two-year evaluation of the accreditation program and presented the findings in SECY-87-121, "Two-Year Evaluation of the Implementation of the Commission Policy Statement on Training and Qualifications" (May 11,1987). The staff concluded that significant progress is being made by industry in improving training and implementing the Commission's Policy Statement. While significant training improvements have been observed, training deficiencies and weaknesses have been identified in both accredited and non-accredited training programs. The staff recommended, therefore, that the Commission (1) continue to endorse the industry accreditation program and defer rulemaking, (2) allow the staff to continue to evaluate industry implementation of training and qualification of nuclear power plant personnel as described in the current policy statement, and (3) direct the staff to propose a revised policy statement on training and qualification of nuclear power plant personnel to incorporate the results of the two-year trial period and the results of discussions with INPO.

Maintenance and Surveillance

In response to Commission policy and planning guidance, the staff developed a Maintenance and Surveillance Program Plan (MSPP). The purpose of the MSPP is to coordinate NRC and industry programs for the evaluation of maintenance effectiveness in the nuclear power industry. The staff continued cooperative efforts in this area with the industry's Nuclear Utility Management and Resources Committee (NUMARC) and the Institute of Nuclear Power Operations (INPO).

Phase I of the MSPP, which was approved for implementation by the NRC Executive Director for Operations in January 1985, was completed in June 1986. Major findings from the MSPP Phase I efforts were published in the "Status of Maintenance in the U.S. Nuclear Power Industry: 1985" (NUREG-1212, Volumes 1 and 2). Phase II of the MSPP was approved for implementation in November 1986.

Operator Licensing

Reactor operator licensing functions continue to be administered by the NRC's five Regional Offices. During fiscal year 1987, the Regional Offices administered examinations to initial (new plant) operators and to replacement power and non-power reactor operators. Following examination, the NRC issued 355 new operator licenses and 443 new senior reactor operator licenses. Also, 623 reactor operator and 1,117 senior reactor operator renewal licenses were issued. Besides the new license examinations, 435 requalification examinations were given to currently licensed operators. In accordance with the revised 10 CFR Part 55, the NRC no longer issues instructor certifications.

Annual regional office audits and program reviews continue to be performed, in order to maintain consistency and standardization of the examination process within and between the Regions. Quarterly regional oversight audits were performed by the headquarters program office at the facility sites during the administration of the licensing examinations. These quarterly site-visits allow for the direct participation of NRC examiners from Headquarters during the operating portion of the examination, and for audits of regional and contractor examiner performance. "Operator Licensing Examiner Standards" (NUREG-1021, Rev. 4) was issued to upgrade the content and documentation guidelines for conducting operating examinations and the operator licensing examination report. And the standard was modified to reflect the changes to the operator examination and licensing process resulting from the implementation of the revised 10 CFR Part 55 rulemaking, effective May 26, 1987.

The supplement to the "Knowledge and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors" (NUREG-1122)" was completed to make the PWR catalog compatible with the BWR catalog. A workshop including both NRC and industry representatives was conducted to validate the passing point criteria of NRC examinations and to review and recommend corrections to identified inconsistencies in the proposed "Examiner's Handbook for Developing Operator Licensing Examinations" (NUREG-1121). The handbook will be issued in early fiscal year 1988 after incorporation of final NRC comments.

Arrangements have been made to obtain a mini-supercomputer to replace the current leased computer supporting the NRC Examination Question Bank. The new computer will have greater capability with reduced NRC long-term costs, because of the elimination of user fees. The system is scheduled to be operational in June 1988.

A pilot test of an alternative approach to NRC requalification examinations was completed and the results reported to the Commission in SECY-87-142 (June 1987). It was concluded that the alternative approach would provide an acceptable evaluation of a facility requalification program and would be used to complement the present NRC program.

The revised 10 CFR Part 55 and three associated regulatory guides were issued in May 1987: Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants"; Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use In Operator License Examinations"; and Regulatory Guide 1.134, "Medical Evaluation of Licensed Personnel for Nuclear Power Plants." The NRC conducted public meetings in Atlanta, Ga., Denver, Colo., Chicago, Ill., and King of Prussia, Pa., to explain provisions of the rule and respond to questions.

SAFETY REVIEWS

TMI Action Plan

The accident at Three Mile Island Unit 2 (Pa.) in 1979 led to a thorough review of NRC regulatory and licensing requirements for nuclear power. The TMI Action Plan (NUREG-0660) was issued and requirements were approved for implementation at plants in operation or under construction; the requirements were later clarified in NUREG-0737. TMI Action Plan requirements for plants under construction are implemented as part of the licensing process, while those for operating reactors are transmitted and confirmed by NRC orders. Items not covered by NUREG-0737 are addressed in NUREG-0933, the document setting priorities for generic items.

Supplement 1 of NUREG-0737 delineates the requirements for emergency response capabilities; it was sent to all licensees on December 17, 1982. Implementation schedules were discussed with the utilities at regional meetings, and, by June 12, 1984, the schedules were confirmed by issuance of Confirmatory Orders for all licensed light water reactors.

By the end of calendar year 1987, about 90 percent of those TMI Action Plan items requiring licensee action had been implemented at the 65 facilities operating at the time of the TMI accident. While the remaining items are being considered in the context of potential safety enhancement beyond adequate plant safety, a concentrated effort is under way to have NRC licensees complete those remaining items (by priorities comporting with their contribution to safety improvement) within the next five years.

Integrated Safety Assessment Program

In a policy statement published in the Federal Register on November 15, 1984, the Commission proposed a trial program to evaluate all pending licensing issues on a given operating reactor, in conjunction with operating experience,probabilistic analyses, and licensee plant improvements, so as to establish effective and efficient implementation schedules for any necessary plant modifications. This program, called the Integrated Safety Assessment Program(ISAP), was implemented in early 1985 for two plants in Connecticut, Millstone Unit 1 and Haddam Neck. The licensees for these plants volunteered to implement ISAP.

In 1985 and 1986, Northeast Utilities (agent and service organization for both plants) completed the plant-specific probabilistic safety studies (PSS) for both plants, together with safety assessments for the licensing issues pertaining to the facilities. The NRC staff completed detailed reviews of the PSS for both plants during 1986. In July 1986 and December 1986, Northeast Utilities submitted the ISAP reports for Millstone Unit 1 and Haddam Neck. These reports specified actions that could be taken to resolve safety issues and rated their relative safety significance.

The staff issued the draft Integrated Safety Assessment Report (ISAR) for Millstone Unit 1 in April 1987 and the draft ISAR for Haddam Neck in July 1987. The reports were issued for comments by the public, the licensee, a peer review group, and the ACRS. Comments on the draft reports will be incorporated into final ISARs, together with recommended integrated implementation schedules for all issues. The final ISARs for Millstone Unit 1 and Haddam Neck are to be issued in fiscal year 1988.

On August 31, 1987, the staff issued SECY-87-219 which details the progress made in ISAP. SECY-87-219 also contains a recommendation to the Commission regarding the future course of action for ISAP.

Probabilistic Risk Assessment

(PRA) was used in evaluating the risk implications of emergency preparedness at several nuclear facilities. The effects of reduced power and its concomitant reduction of risk of plant operation were shown to ease the challenge to emergency response. Risk methods were also used to evaluate the impact on public safety resulting from a reduction of the size of the emergency planning zone (EPZ).

Halfway through fiscal year 1987, the NRC reorganization brought about a heavier concentration of NRR efforts on licensing and inspection of operating reactors. The role of PRA within NRR changed, with the emphasis shifting to applications of PRA to operating reactor issues. The traditional PRA activities, such as "full scope PRAs," were transferred to the Office of Research (see Chapter 9). Specifically, PRA methods are being used to evaluate the risk significance of various types of operator errors, and to support licensing decisions that involve design or procedural modifications. PRA-based guidance is being generated to evaluate Technical Specification modifications. Currently, a major effort is under way to promote PRA-based team inspections in all five Regions and within Headquarters. A program initiated in Region I to use PRA for prioritizing the resident inspectors' activities is also being expanded to give wider coverage of the operating plants, including those without PRA studies.

Furthermore, PRA is being used by the staff to review and evaluate standardized LWR designs. It is being applied on a daily basis to the assessment of operating events and to evaluating the performance of licensees. It is expected that, as these programs mature, they will make a significant contribution to reducing risk at U.S. nuclear power plants.

Fire Protection

The NRC fire protection rule (Section 50.48 and Appendix R to 10 CFR Part 50) for nuclear power plants became effective on February 17, 1981. It required licensees of all plants holding operating licenses issued prior to January 1, 1979, to submit plans and schedules for meeting the applicable requirements, to describe proposed modifications needed to provide alternative safe-shutdown capability, or to submit requests for exemptions from specific technical requirements of the rule, if appropriate. These requirements are in addition to the fire protection guidelines contained in Branch Technical Position APCSB 9.5-1, and Appendix A to the branch technical position, both of which were developed in response to the 1975 Browns Ferry (Ala.) fire. For plants licensed after January 1, 1979, the NRC staff uses the criteria of the Standard Review Plan (NUREG-0800). These include the technical requirements of the fire protection rule, along with supplemental guidance provided in Generic Letter 81-12 and its clarifying letter, to review and evaluate nuclear plant fire protection programs. Requests for exemptions, proposed modifications, and revised safe shutdown methodologies continue to be received and reviewed by the staff.

The staff issued Generic Letter 86-10 to provide to licensees and applicants additional clarification of NRC fire protection guidelines and requirements in several areas of concern—fire area boundaries, fire barriers qualifications, automatic suppression systems, and intervening combustibles. The Generic Letter also provided for the removal of certain features of the fire protection program from individual plant technical specifications, in conjunction with the incorporation of the fire protection program in the Final Safety Analyses Report and the adoption of the standard fire protection license condition.

Two significant fires took place in the 1987. One fire involved the ignition of leaking hydraulic fluid onto hot steam piping at the facility at the Fort St. Vrain (Colo.) nuclear power plant. A second fire involved leaking fuel oil ignited by a hot cylinder head in the "A" diesel generator room at the Palo Verde (Ariz.) site. The plants were operating at the time. The fires were quickly extinguished by the plants' fire brigade and the plants were shut down. NRC inspectors went to the sites and assessed the safety impact and possible generic implications of the events. No radioactive material was released and no radiological exposure to the public or plant personnel occurred as a result of these fires.

The staff is continuing the inspection of fire protection at operating reactors to assess compliance with the applicable requirements of Appendix R and is developing a tri-annual audit program to assure long-term conformance.

Operational Safety Assessment

NRC headquarters staff actively participate with the Regions in the follow-up review of operational events that could lead to an ordered plant derating or shutdown, license amendment, or new generic concerns. These reviews involve the evaluation of events against existing safety analyses, appraisal of plant and operator performance during events, review of licensee analyses, and assessment of any need for corrective action prior to plant restart.

In fiscal year 1987, the staff, as part of the formalized program for the assessment of major reactor incidents, assigned Augmented Inspection Teams to ascertain the facts related to the following operating reactor events:

- Main Feedwater Pump Suction Line Rupture at Surry 2 (Va.) in December 1986.
- Boric Acid Buildup on Reactor Vessel Due to Instrument Seal Leakat Turkey Point 4 (Fla.) in March 1987.
- Loss of Decay Heat Removal Capability at Diablo Canyon 2 (Cal.) in April 1987.
- Loss of Off-site Power at Palisades (Mich.) in July 1987.
- Steam Generator Tube Rupture at North Anna 1 (Va.) in July 1987.
- Reactor Scram With Multiple Equipment Failures at Davis-Besse (Ohio)in September 1987.
- Violation of Safety Limit (Affecting Reactor Vessel MonitoringCapability) at Oyster Creek (N.J.) in September 1987.

Radioactive Effluents Summary / Analysis

The program for implementing Radiological Effluent Technical Specifications (RETS) at operating reactors—a program designed to keep releases of radioactive materials to unrestricted areas during normal operations as low as reasonably achievable—was completed in 1987. All nuclear power plants have now incorporated the RETS into their operating license.

As part of the RETS license requirements, licensees submit periodic reports on radioactive effluents and radiological environmental monitoring. Semiannual reports contain detailed summaries characterizing the radioactive gaseous and liquid effluents released from the plant to the environment, and also quantify solid radioactive wastes shipped off the site. These reports include calculations of the radiation doses from these effluent releases to members of the public. The NRC annually collates these individual plant summaries in two publications: "Radioactive Materials Released from Nuclear Power Plants" and "Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites." These reports document the fact that the effluents released by licensed nuclear power plants in 1987 complied with the regulations and rarely exceeded the amount considered in the Final Environmental Statements to provide very small risks to members of the public.

In addition to the semiannual effluent reports, licensees submit an annual radiological environmental operating report. This report contains the results of the extensive weekly and monthly monitoring programs required by the plant's RETS and records when, if ever, radioactive contamination above natural background is detected outside the plant boundaries. The semiannual effluent reports and the annual radiological environmental operating reports for all operating plants are available for public inspection in local Public Document Rooms (see Appendix 3).

Reassessment of B&W Reactors

Following the accident at Three Mile Island Unit 2 (Pa.) in 1979, licensees of nuclear power plants made a significant number of improvementsto upgrade performance in their facilities. Despite these improvements, the NRC has found and is concerned that the number and complexity of unplanned events in power plants with reactors designed by Babcock & Wilcox (B&W) did not decrease. The events at Davis-Besse (Ohio) in June 1985 and at Rancho Seco (Cal.) in December 1985 reinforced the staff's concern.

The NRC's Executive Director for Operations informed the Chairman ofthe Babcock & Wilcox Owners Group (BWOG), by letter dated January 24, 1986,that events at B&W-designed reactors have led the NRC staff to conclude thatthere is a need to re-examine the basic design requirements for B&W reactors.In its response, on February 13, 1986, the BWOG agreed to take the lead in aconcerted effort to define the factors in B&W plants causing the frequency of reactor trips and the complexity of post-trip response. The BWOG developed a reassessment plan. The NRC staff reviewed this plan and proposed changes that were incorporated by the BWOG. A BWOG final report, B&W 1919 "B&W Owners Group Safety and Performance Improvement Program," Revision 5, was issued in July 1987.

The NRC reassessment of B&W plants involves a review of the BWOG effort. The BWOG effort includes an assessment of the thermal-hydraulic design, instrumentation, controls and power supplies, instrument air systems, emergency and main feedwater systems, and means of pressure relief in the secondary plant. Also, the reassessment involves a review of operating experience at B&W plants and operator training and responses. The staff has performed limited independent evaluations of the B&W plant design, including review of operating experience, evaluation of inspection results, human factors affects, risk assessment, and thermal-hydraulic analysis. A staff Safety Evaluation Report (SER) is expected to be issued at the end of 1987. The SER will appraise most of the BWOG activities except for the integrated control system/non-nuclear instrumentation and the acceptability of the emergency feedwater initiation and control system. These and any remaining BWOG efforts are scheduled to be completed in 1988.

Application of Leak-Before-Break Technology

In May of 1986, a limited-scope revision of General Design Criterion 4, Environmental and Missile Design Bases, became effective. The revision was made to permit the use of advanced fracture mechanics as an alternative method to postulating pipe breaks in primary-coolant-loop piping in pressurized water reactors (PWR). This technology is referred to as "leak-before-break." The application of the technology produces a significant safety benefit both in existing plants and plants under construction. This method has led to the removal of unnecessary pipe whip restraints and jet shield and barriers, thus facilitating maintenance inside the containment structure.

Certain utilities have submitted analyses to justify removing or not installing pipe-rupture protection devices in reactor coolant systems. Theanalyses purport to demonstrate that the piping system under consideration will not be subjected to degradation by corrosion mechanisms, fluid system transients, or fatigue. They also seek to demonstrate that, if a through-way flaw did exist in the pipe, the flaw would be stable under normal-plus-design-basis-earthquake loading, and that any flow from the flaw would be readily detected. Under these considerations, pipe breaks would no longer need to be postulated in PWR primary-coolant-loop piping.

Large snubbers of the size used on the reactor-coolant-loop piping are costly to maintain, and their maintenance results in considerable radiation exposure to plant operations personnel. Inoperable snubbers can inhibit the motion of the piping system under heatup and cooldown and can, therefore, actually be detrimental to safety. The NRC staff considers removal or down-sizing of snubbers located on PWR reactor coolant loops to be an improvement in safety, by virtue of the elimination of pipe-break loading effects. This step will result in a reduction in radiation exposure to plant personnel and will provide more reliable performance under normal plant operating conditions.

The broad-scope revision of General Design Criterion 4, Environmental and Missile Design Bases, became effective on November 27, 1987. This revision of General Design Criterion 4 supersedes the limited scope revision, which became effective in May of 1986. It enlarges the scope of leak-before-break to encompass all applicable high energy piping systems not only in PWRs but boiling water reactors (BWRs) as well. The safety benefits realized under the limited scope revision are expected to be multiplied with wider application of this alternative to postulating pipe breaks.

Pipe Cracks at Boiling Water Reactors

Although intergranular stress corrosion cracking has been a recurring problem in BWR piping for many years, it was not until 1982 that cracking was found in large diameter reactor coolant piping. The NRC then required inspections at all BWR plants to determine the extent and severity of the problem. The initial inspection program (initiated by Inspection and Enforcement Bulletins 82-03 and 83-02) resulted in the discovery of significant cracking at many BWRs. The reinspection program required by Generic Letter 84-11, covering the reactor coolant and connecting systems, also resulted in the detection of cracking in essentially all BWRs. (See the *1985 NRC Annual Report*, pp. 49 and 50, for further background.)

Some utilities have chosen to replace their degraded piping with more resistant material. The replacement has been completed at the following eight BWRs: Nine Mile Point Unit 1 (N.Y.), Monticello (Minn.), Pilgrim Unit 1 (Mass.), Hatch Unit 2 (Ga.), Cooper (Neb.), Peach Bottom Unit 2 (Pa.), Vermont Yankee and Dresden Unit 3 (Ill.). It is expected that several others will make a replacement decision in the near future.

A draft Generic letter outlining the staff position regarding acceptable mitigative action was issued on July 21, 1986. Issued concurrently for public comment was a draft revision to NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." After the 60-day comment period, the comments reviewed were evaluated. The staff resolution of the public comments, including a peer review by experts from several national laboratories, has been completed, and appropriate revisions to the two documents have been made and reviewed by the Advisory Committee on Reactor Safeguards and the Committee to Review Generic Requirements. The final resolution of the Generic Letter and the technical basis report (NUREG-0313, Rev. 2) has been submitted to the Commission for approval.

Radiation Protection at Nuclear Reactors

Daily monitoring of licensee and regional reports to the NRC Operations Center alerts staff to potential problems in radiation safety, ranging from major repair problems involving highly radioactive components to contamination and cleanup from small leaks of liquid and gaseous contaminants. These initial alerts are followed up by communication with regional representatives and eventual follow-through on the health physics problems in regional inspections. Further involvement of headquarters staff in regional and licensee problems is provided by participation in routine environmental and radiological inspections, as well as special team inspections investigating major licensee problems.

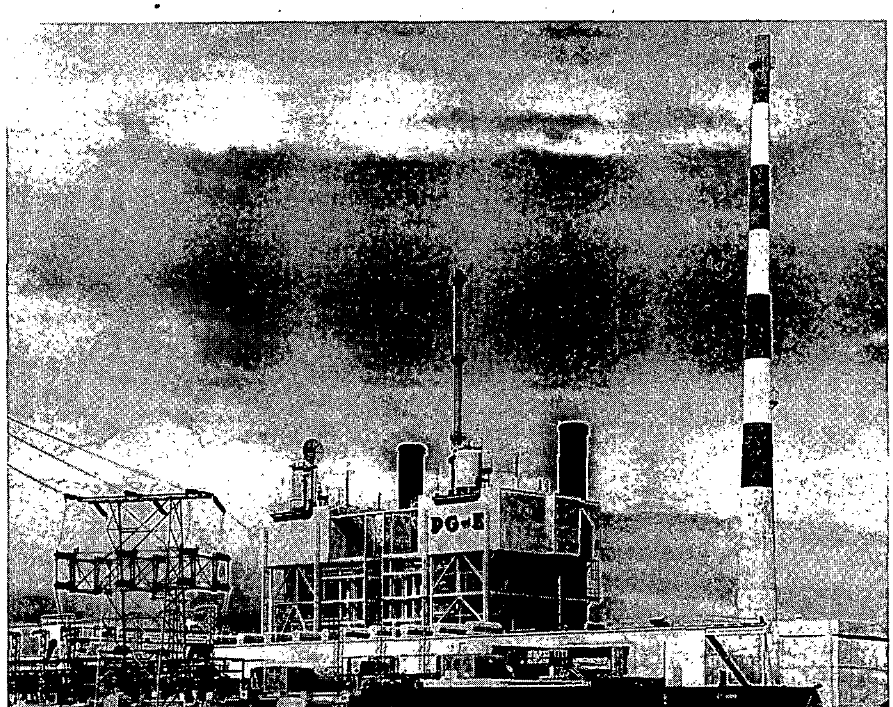
During the year, the staff has provided radiation protection support for the licensing activities of most of the operating nuclear power reactors. This support included extensive consultation on the pre-planning for steam generator repairs at the D.C. Cook (Mich.) plant, health physics support for evaluating requests for expansion of spent fuel pool capacity at St. Lucie (Fla.) and Prairie Island (Minn.), and evaluation of radiation doses and risks to members of the public from small amounts of contamination found in fish and sediments at several nuclear facilities.

An important staff function has been to provide radiation protection evaluation and perspective in the decommissioning of the Humboldt Bay Unit 3 (Cal.) reactor through the Final Environmental Statement, the Decommissioning Safety Evaluation, the SAFSTOR Technical Specifications, and support for relevant hearings. In addition, the staff has evaluated proposals for disposal of wastes contaminated with very low levels of radioactivity, such as sewage sludge, soil, water treatment residues, and waste oil. Among the plants included in these activities were Point Beach (Wis.), Surry Va.), and LaSalle (Ill.).

Hot Particle Contamination At Nuclear Power Plants

Extent of the Problem. During the first nine months of 1987, events involving skin exposures resulting from skin contamination by small, highly radioactive particles with high specific activity (hot particles) were reported at 10 different nuclear power stations. Four events since mid-1985—at McGuire (N.C.), V.C. Summer (S.C.), San Onofre (Cal.), and North Anna (Va.)—involved exposures apparently exceeding NRC regulatory limits. The NRC staff has met this problem with a vigorous campaign of information notices to licensees, cooperation with the Institute of Nuclear Power Operations, and intensifiedinspection activity.

Hot particles come primarily from two major sources: degraded fuel and neutron-activated corrosion and wear products (e.g., Stellite). The latter are referred to as activated particles, and irradiated fuel particles are referred to as fuel particles. A hot particle on the skin gives a high beta dose to a very small area. (Gamma dose rates are generally very small and are not a problem.) Any radiation dose to the skin is assumed to result in some increased risk of skin cancer, although this type of cancer is rarely fatal. In addition to any increased risk of cancer, large doses to the skin from hot particles may produce observable effects, such as reddening, hardening, peeling, or ulceration of the skin immediately around the particle. These effects appear only after a threshold dose is exceeded. The doses from hot particles required to produce these effects in the skin are not known precisely; however, it appears likely, except for a point reddening, that these effects will only be seen after doses of hundreds of rems or more for that point of skin. No such effects have been seen to date on any workers who have been exposed to hot particles, even though one exposure has been calculated as high as 512 rems.



The NRC staff continued in 1987 to provide assessment and oversight in the decommissioning of the Humboldt Bay nuclear power plant near Eureka, Cal. This boiling water reactor plant first generated power, under a provisional license from the Atomic Energy Commission, on April 18, 1963, and reached its licensed 165 thermal megawatt power level two weeks later. The plant was ordered shut down in July 1976, because of hazards posed by geologic faulting nearby, and has not operated since.

Extended power plant operation with degraded fuel (leaking fuel pins) can result in widespread dispersal of fuel particles. Some plants continue to experience fuel particle contamination problems long after leaking fuel pins have been removed because of the residual contamination of plant systems. Some plants with these problems have started programs to account for missing fuel pellets and fragments and to identify measures to recover this material.

Control Initiatives. Some plants that have operated for extended periods of time with degraded fuel and plants with activated particle problems now have instituted specialized, comprehensive training programs for plant system maintenance workers and general employees. These programs are designed to better inform and prepare the plant staff to cope with the continuing fuel particle problems. In addition, as part of comprehensive contamination control programs, special new procedures to improve surveys for detection of hot particles have been prepared and health physics technicians have been trained in their use. Decontamination and dose evaluation methods and procedures that focus on hot particles have been implemented.

Approximately 75 percent of the U.S. power reactor facilities are currently using new high-sensitivity whole-body contamination monitors. These state-of-the-art contamination monitors increase the probability of detecting hot particles on plant personnel while reducing the likelihood of inadvertently releasing particles from the plant site. Most of the particles found on personnel have been detected by these new monitors, and no significant public exposures have been reported to date.

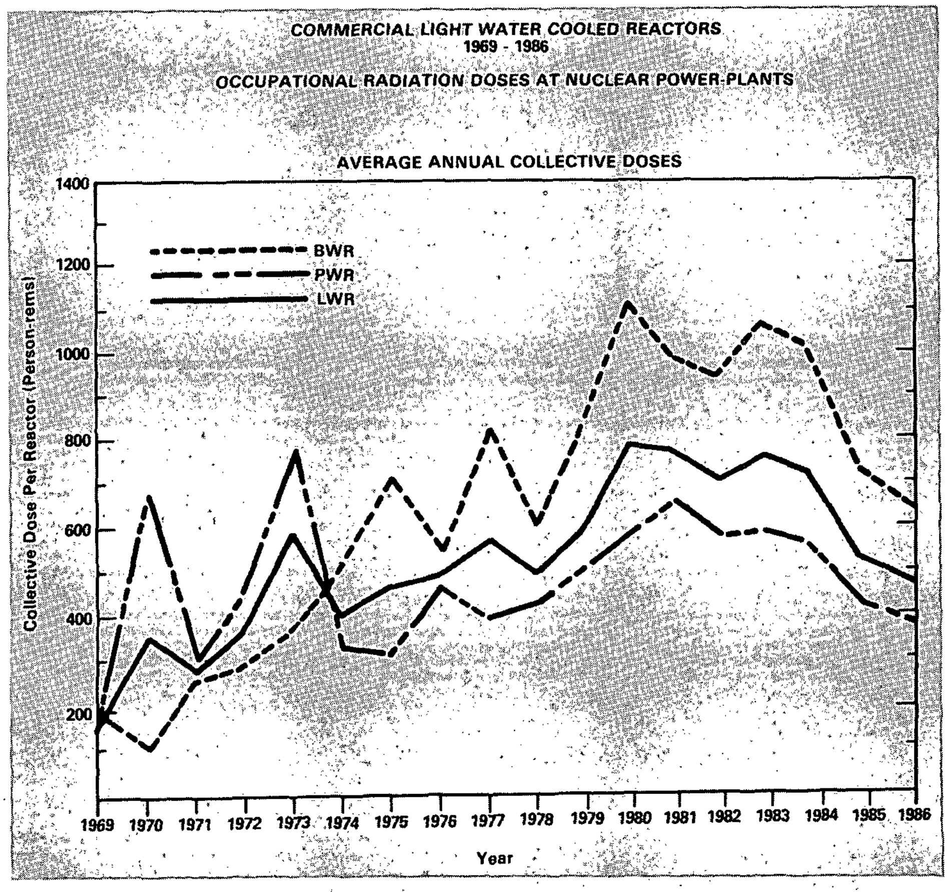
The NRC staff recognizes the need for more information on the effects of radiation on skin, particularly the effects of hot particle irradiation, and has requested that the National Council on Radiation Protection and Measurements (NCRP) study the health significance of hot particle exposures and provide recommendations based on their findings. The recommendations may result in changes in NRC requirements with respect to hot particle exposures. Additional information on hot particle contamination at nuclear power plants is available in NRC Information Notice No. 87-39, dated August 21, 1987.

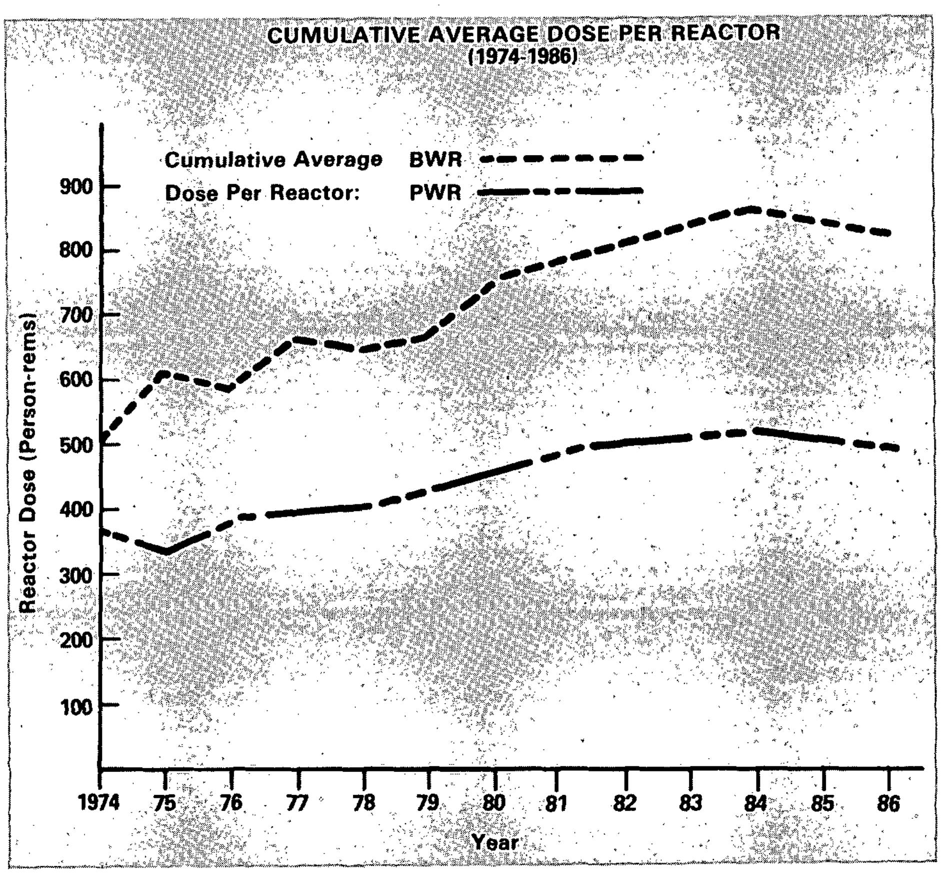
Occupational Exposure Data And Dose Reduction Studies

The staff has been collating the annual occupational doses at light water reactors (LWRs) since 1969 (see upper graph). Although the annual dose averages for both pressurized water reactors (PWRs) and boiling water reactors (BWRs) have fluctuated over the years, the overall trend between the early 1970s and 1980 was one of increasing annual dose averages. Annual dose averages peaked in the early 1980s, mainly because of the implementation 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 1986, the average dose per unit for LWRs was 480 person-rems. This is 9 percent lower than the LWR average for 1985, and is the lowest LWR average dose since 1975. In 1986, the average doses per unit for PWRs and BWRs were 392 and 653 person-rems, respectively, both down from the 1985 averages of 416 and 735 person-rems. Maintenance jobs which were large contributors to BWR doses in 1986 included replacement of, or work on, recirculation system piping, induction heating stress improvement and inservice inspection of welds, inspection for intergranular stress corrosion cracking, and Appendix R modifications. Steam generator maintenance and repair (including tube sleeving, plugging, and eddy current testing) was a major source of occupational exposure at PWRs.

The lower graph is a plot of the cumulative average exposure data for both PWRs and BWRs for the years 1974 to 1986. The year 1974 was chosen as a starting date for this analysis, since the average rated capacity for years prior to 1974 (less than 500 MWe/reactor) was much lower than the current average rated capacity of operating reactors. The graph shows that the cumulative average exposures for both PWRs and BWRs appear to have peaked in 1984. The cumulative average exposures for the 13-year period spanning 1974 to 1986 for PWRs and BWRs are 492 and 829 person-rems, respectively. The 1986 dose tabulation includes data from 59 PWRs and 30 BWRs. This total reflects the addition of five new PWRs (Byron 1 (Ill.), Catawba 1 (S.C.), Diablo Canyon 1(Cal.), Waterford 3 (La.), and Wolf Creek 1 (Kans.)), and two new BWRs (Grand Gulf 1 (Miss.) and Susquehanna 2 (Pa.)).

The NRC has several ongoing contracts with Brookhaven National Laboratory in the area of occupational dose reduction at LWRs. The objective of one of the NRR-sponsored studies is to estimate the dose and cost savings resulting from the control of contamination at nuclear power plants.


COMMERCIAL LIGHT WATER COOLED REACTORS
1969 - 1986
OCCUPATIONAL RADIATION DOSES AT NUCLEAR POWER PLANTS
AVERAGE ANNUAL COLLECTIVE DOSES
BWR
PWR
LWR
Collective Dose Per Reactor (Person-rems)
1400
1200
1000
800
600
400
200
1969 1970 1971 1972 1973 1974 1975 1976 1977 1978 1979 1980 1981 1982 1983 1984 1985 1986
Year


CUMULATIVE AVERAGE DOSE PER REACTOR
(1974-1986)
Cumulative Average BWR
Dose Per Reactor: PWR
Reactor Dose (Person-rems)
900
800
700
600
500
400
300
200
100
1974 75 76 77 78 79 80 81 82 83 84 85 86
Year

Antitrust Activities

As required by law since December of 1970, the staff has conducted pre-licensing antitrust reviews of all construction permit applications for nuclear power plants and certain other commercial nuclear facilities. In addition, applications for amendments to construction permits or operating licenses that transfer ownership interest or operating responsibility in a nuclear facility are subject to antitrust review. In fiscal year 1987, staff received three requests for license amendments pursuant to sale-leaseback proposals requiring antitrust review. The reviews associated with two of these requests have been completed, each concluding that there were no apparent antitrust problems. The third request was still under review at the close of the fiscal year. Additionally, requests were received from two licensees to amend the antitrust license conditions attached to the respective construction permits and operating licenses of each utility. Each of these requests was undergoing staff review at the close of fiscal year 1987.

An application for an operating license is not subject to formal antitrust review unless the staff first determines that "significant changes" in the licensee's activities have occurred since the review of the application for a construction permit (see NUREG-0970, "Procedures For Meeting NRC Antitrust Responsibilities"). During fiscal year 1987, three significant change analyses were completed. In each instance, the finding was that the changes that had occurred were not significant in an antitrust context. The staff also received a request to re-evaluate a "no significant change" finding reached in fiscal year 1986. After re-evaluation by the staff, the finding was affirmed.

Remedies to antitrust problems usually take the form of conditions attached to licenses, and the NRC has the responsibility to enforce compliance with these antitrust conditions. During the latter part of fiscal year 1986, the staff issued a Notice of Violation (pursuant to the provisions of 10 CFR 2.201 of the NRC's Rules of Practice) against the principal owner of the Farley (Ala.) nuclear power plant. The Notice of Violation pertained to the antitrust license condition which directed the principal applicant to offer ownership access to the Farley plant. At the close of fiscal 1987, after several negotiating sessions involving the staff and each of the parties, the licensee had submitted its response to the Notice and the staff was in the process of evaluating the response.

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 nuclear incident causing personal injury or property damage.

The first layer requires all licensees of commercial nuclear power plants rated at 100 electrical megawatts 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 $160 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 $160 million. In the event of such an incident, each licensee of a commercial reactor rated at 100 electrical megawatts or more would be assessed a prorated share of damages up to the statutory maximum of $5 million-per-reactor-per-incident. At present, the secondary financial protection layer is $555 million (a figure derived from the 111 power reactors rated over 100 MW(e) which had been licensed to operate prior to the close of the report period times $5 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 $5 million for each new commercial reactor licensed.

Price-Anderson Renewal

New bills were introduced in the 100th Congress to modify and extend the Price-Anderson Act, which expired on August 1, 1987. The Commission testified on these bills on March 27, 1987, at joint hearings before the House Subcommittees on Energy and the Environment and Energy and Power. On April 30, 1987, the Commission testified before the Senate Subcommittee on Nuclear Regulation. H.R. 1414 was reported out by the House Interior and Insular Affairs, Energy and Commerce, and Science and Technology Committees, and was passed as H.R. 2994 by the House on July

30, 1987. Although each of the two Senate Committees with oversight over Price-Anderson reported out a Price-Anderson bill, neither bill had been brought to the Senate floor by the close of fiscal year 1987.

Indemnity Operations

As of September 30, 1987, 237 indemnity agreements with NRC were in effect. Indemnity fees collected by the NRC from October 1, 1986, through September 30, 1987, total $127,439. Fees collected since the inception of the program total $23,343,834. 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 30 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 Underwriter—paid policyholders the 21st 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 1987 totalled $6,752,362, which is approximately 38.5 percent of all premiums paid on the nuclear liability insurance policies issued in 1977 and covers the period 1977-1987. The refunds represent 74.2 percent of the premiums placed in reserve in 1977.

Utility Financial Qualifications 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 applicants, 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 involving the utilities and outside investors, and (2) the formation of holding companies and utility subsidiaries.

Incentive Regulation of Electric Utilities

Economic performance incentives established by State public utility commissions (PUCs) are applicable to the construction or operation of about 45 nuclear power reactors owned by 30 utilities in 17 States. (For background, see the *1986 NRC Annual Report,* p. 150.) The NRC 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.

Property Insurance

Since 1982, the NRC has required power reactor licensees to carry on-site property damage insurance. It is intended that the proceeds of such insurance would be used to help pay for cleanup and decontamination of a reactor plant following an accident. The NRC believes that such insurance should be required so that the potential impact of financing on the pace and thoroughness of cleanup following an accident does not become a threat to public health and safety.

On August 5, 1987, the Commission published a final rule in the *Federal Register* (52 FR 28963) that increases the amount of on-site property damage insurance required to be carried by power reactor licensees to slightly over $1 billion. In addition, the rule requires that any proceeds from this insurance must be expended first to stabilize, decontaminate, and clean up the reactor following an accident, when 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 disperse funds for decontamination and cleanup. The rule also made a determination that the Federal Government can pre-empt State law that prohibits certain public utilities from buying insurance offered by mutual companies or insurance requiring payment of a retrospective premium.

The fifth annual property insurance reports submitted by power reactor licensees indicated that, of the 74 sites insured as of April 1, 1987, 63 are covered for at least the $1.06 billion required in the rule. Four additional sites are exempt from NRC's full property insurance requirement. The remaining sites were required to be fully insured as of October 4, 1987, the effective date of the revised rule.

The NRC has been informed by the nuclear property insurers that, as of September 15, 1987, capacity increased to $1.395 billion. As of January 1, 1988, capacity is expected to grow to $1.525 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 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, for each application for a construction permit or an operating license for power reactors, and for applications for licenses to construct or operate test reactors, spent fuel reprocessing plants, and waste disposal facilities.

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, which is drawn from scientific and engineering disciplines, includes individuals experienced in metallurgical engineering, electrical engineering, mechanical engineering, structural engineering, reactor operations, reactor physics, and environmental health.

During fiscal year 1987, the Committee completed its annual report to Congress on the NRC Safety Research Program for fiscal year 1988 and reported to the Commission on proposed future action on the Safety Research Program and Budget.

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

- Protective Action Guides for Nuclear Emergency Planning.
- DOE's Final Environmental Assessment for High-Level Waste Repository Sites.
- Implications of the Accident at Chernobyl Nuclear Power Station, Unit 4.
- Improved Safety for Future Light Water Reactors.
- Testing of Charcoal Adsorption Capacity.
- The Proposed Nuclear Waste Advisory Committee.
- Proposed Research to Reduce Source Term Uncertainty.
- Disposal of Mixed Waste.
- Quality Assurance Programs for a High-Level Waste Repository.
- International Cooperation on Research Related to Radiation Protection.
- Control Room Habitability.
- Embrittlement of Structural Steel.
- The Integrated Safety Assessment Program.
- Research on Continuous Containment Leakage Monitoring.
- The Fire Risk Research Scoping Study.
- Developments in Emergency Planning.
- Uncertainties Associated with the Use of Realistic ECCS Evaluation Models.

The Committee's activities during the period included reports on the Resolution of ACRS Comments on the Clinton Nuclear Power Station and the Westinghouse SP-90 design.

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

- NRC Policy on Nuclear Power Plant Standardization.
- Containment Spray as a Fission Product Cleanup System and Fission Product Control Systems.
- Application of NRC's Safety Goals in Licensing Issues.
- Determination of Rupture Locations and Dynamic Effects Associated with Pipe Rupture.
- Analysis of French PWR Designs Compared to Current U.S. PWR Designs.
- Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for PWRs.
- Proposed Policy Statement on Deferred Plants.
- Electrical Penetration Assemblies in Containment Structures.
- Environmental and Missile Design Bases.
- Proposed Mark I Containment Requirements for Severe Accidents.
- Preparation of a License Application for a Low-Level Waste Disposal Facility.
- Proposed Rulemaking on the Definition of High-Level Radioactive Waste.
- Environmental Qualification of Connection Assemblies for Nuclear Power Plants.
- Implementation Plan for the Safety Goal Policy.
- NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants.
- Proposed Generic Letter on Individual Plant Examinations for Severe Accident Vulnerabilities.
- Proposed Resolution of USI A-44, "Station Blackout."
- Seismic Qualification of Electrical and Mechanical Equipment.

- Qualification of Lead Storage Batteries.
- Leak-Before-Break Evaluation Procedures.
- Degree Requirements for Senior Operators.
- Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.
- Draft Reactor Risk Reference Document.

The Committee commented in reports on the NRC Staff's proposed priority rankings for generic issues.

The Committee also prepared two classified reports on aspects of the Naval Reactors Program.

In performing the reviews and preparing the reports cited above, the ACRS held 12 full Committee meetings and 53 subcommittee meetings. In addition to these subcommittee meetings, a small delegation of ACRS members met in Rome, Italy, with the Technical Committee to the ENEA (Italian Commission for Nuclear and Alternative Energy Sources) to discuss decay heat removal systems, severe accidents, advanced LWRs, the training of reactor operators, and emergency planning; the delegation toured the Alto Lazio Nuclear Power Station, which is now under construction.

In the first such meeting of its kind, nuclear power plant safety representatives from the Federal Republic of Germany, France, Japan, and the United States assembled in Racine, Wis., during the week of October 20-23, 1986, to discuss subjects of mutual interest. The meeting, organized under the leadership and direction of David A. Ward, Chairman, Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, was held at the Wingspread Conference Center of the Johnson Foundation. Attending were approximately 40 representatives of the several countries. The discussions were candid and provided the participants an opportunity to share thoughts and information on nuclear safety concerns and solutions.

Cleanup At Three Mile Island

Chapter 3

During fiscal year 1987, significant progress was made with the cleanup of the damaged reactor at Unit 2 of the Three Mile Island nuclear power plant (TMI-2) near Harrisburg, Pa. Decontamination and dose-reduction activities continued in parallel with defueling operations, as did the processing and shipment of radioactive wastes.

Workers using long-handled tools performed defueling operations at Unit 2 from a shielded platform located nine feet above the reactor vessel flange. This deployment allowed for the removal of damaged fuel and structural debris in the reactor vessel at a greater rate than before. As of the end of September 1987, a total of 162,451 pounds of damaged fuel and debris had been removed from the reactor vessel. That volume represents more than 55 percent of the post-accident core inventory and includes the remnants of 132 of the total of 177 original fuel assemblies. Defueling of the original core region was expected to be completed by the end of 1987, with the removal of the remaining fuel assemblies. The next areas to be defueled are the lower internals and the lower head (below the normal core region). These areas contain a mixture of loose material and solidified, once molten, material. The areas between the baffle plates (outside the normal core region) and the core barrel will also have to be defueled. Other cleanup activities in fiscal year 1987 included defueling of the "A" steam generator, which yielded about 10 pounds of debris. The decay heat drop line was also found to contain a significant quantity of fuel and will be cleaned along with the remainder of the reactor coolant system piping. The completion of defueling is expected by the end of calendar year 1988.

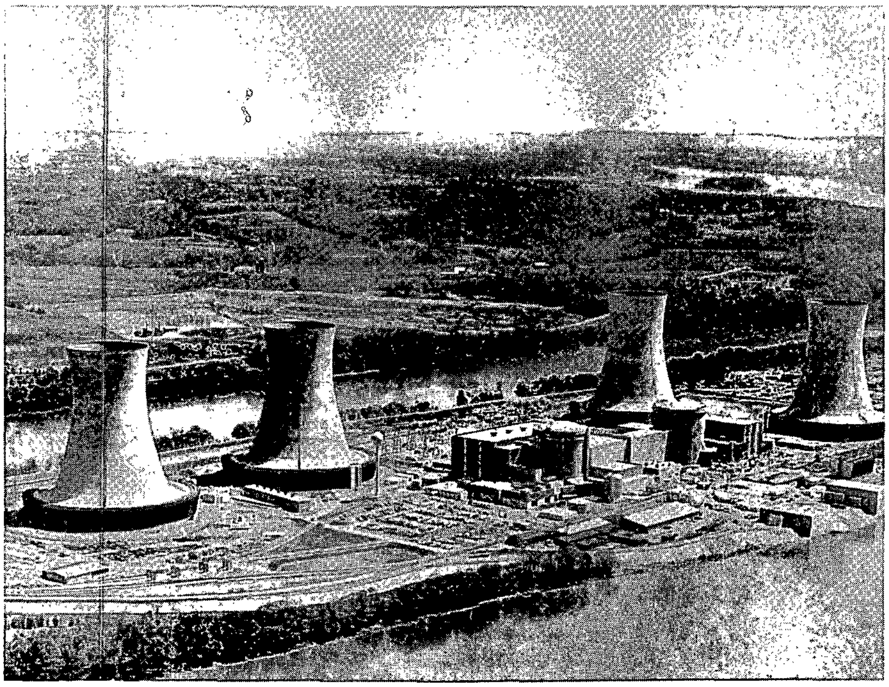
Dose rates to personnel during defueling were low and remained low throughout the report period. The rates averaged less than 10 mrem/hr on the shielded platform and less than 40 mrem/hr near the shielded core debris canisters during canister transfer. Projected cumulative worker dose for calendar year 1987 is 1,027 person-rem. This is below the licensee's goal of 1,175 person-rem and just 120 person-rem (13 percent) above the 907 person-rem total for calendar year 1986.

Shipment of damaged core material from the TMI site to the Idaho National Engineering Laboratory (INEL) continued throughout the period. A total of 16 shipments of debris have been made to INEL, 15 of them occurring in fiscal year 1987. These shipments comprise 112,348 pounds of debris, which is more than 37 percent of the total amount to be removed from the reactor vessel. General Public Utilities Nuclear Corporation (GPUNC) made arrangements to use a third shipping cask to help expedite shipments to INEL.

During the report period, the Submerged Demineralizer System (SDS) and the EPICOR-II system were used to process radioactive water. The two systems processed about 352,518 and 609,515 gallons of water, respectively. Currently, the EPICOR-II system handles all processing of contaminated water, with the SDS in a standby mode. Twenty-nine EPICOR-II dewatered liners were shipped to Richland, Wash., for burial during this same period.

In July 1986, GPUNC submitted a proposal for disposing of approximately 2.1 million gallons of slightly radioactive water. This water was contaminated either during the accident of April 1979 or during subsequent cleanup operations. The proposed method involves the forced evaporation of the water at the TMI site over a two and one-half year period. The residue from this operation—containing small amounts of the radioactive isotopes cesium-137 and strontium-90, and larger amounts of non-radioactive boric acid and sodium hydroxide—would require solidification and disposal as low-level waste. The staff evaluated the licensee's proposal together with eight alternative approaches, giving consideration to the risk of radiation exposure to workers and to the general public; the probability and consequences of potential accidents; the necessary commitment of resources, including costs; and regulatory constraints. The results of the staff evaluation were presented in the June 1987 Final Supplement No. 2 to the "Programmatic Environmental Impact Statement" (NUREG-0683), dealing with disposal of accident-generated water. The staff concluded that the licensee's proposal to dispose of the water by forced evaporation to the atmosphere, followed by on-site solidification of the remaining solids and disposal thereof at a low-level waste facility, was an acceptable plan. The staff also concluded that no alternative method of disposing of the contaminated water was clearly preferable to the GPUNC proposal. An opportunity for a prior hearing to consider removing the prohibition on the disposal of the contaminated water was offered, and the matter was pending before the Atomic Safety and Licensing Board at the end of fiscal year 1987.

Throughout 1987, GPUNC performed decontamination and dose-reduction activities aimed at maintaining worker radiation exposures at a level as low as reasonably achievable. Scabbling (the mechanical removal of a layer of concrete), water flushing, vacuuming, painting, and hands-on techniques such as wiping and scrubbing were the primary means for decontaminating areas in the reactor building and the auxiliary and fuel-handling buildings (AFHB). Sludge removal from the auxiliary building sump and the reactor building was completed, and a flushing of the reactor building begun in September 1987. Seventy-five percent of the



The Three Mile Island (TMI) nuclear power plant, located on an island in the Susquehanna River in Dauphin County, Pa., was the scene of the nation's most serious nuclear accident when, on March 28, 1979, a partial meltdown of the reactor core to TMI Unit 2 occurred. (Unit 2 is the cylindrical containment building to the right in the photo.) Although no one was killed or injured in the accident, it remains a major traumatic episode in the history of the technology. In the years since the event, many far-reaching changes in regulatory requirements and procedures have been introduced.

previously contaminated areas (462,708 square feet) of the AFHB has been decontaminated. Of 143 contaminated cubicles in the AFHB, 107 have been decontaminated. Twenty-three of the remaining 36 cubicles were expected to be cleaned up during the last quarter of calendar year 1987.

The NRC continued to monitor the day-to-day cleanup operations of the licensee. The staff at TMI performed numerous reviews and issued approvals of the licensee's detailed defueling procedures and conducted periodic inspections of systems and of equipment used in the cleanup. In conjunction with headquarters staff, the NRC staff at the TMI site performed safety and technical reviews of licensee proposals for major cleanup efforts, in order to assure that they would genuinely contribute to the safe and expeditious cleanup of the plant.

The 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 1987, the panel held seven public meetings, in Harrisburg and Lancaster, Pa., and met with the NRC Commissioners in Washington, D.C. Among the topics addressed by the panel during these meetings were: TMI-2 health effects studies presented by the Pennsylvania Department of Health and local citizens, the status of the ongoing defueling operations, the Department of Energy's plans for off-site shipment and storage of fuel, the licensee's proposal for the disposal of accident-generated water, and the NRC's continuing oversight and enforcement activity.

Financial Aspects of TMI-2 Cleanup

Funding by GPU. (For background, see the *1986 NRC Annual Report,* p. 150.) Revenues collected by General Public Utilities Corporation's three operating subsidiaries in Pennsylvania and New Jersey continued to be expended on cleanup during 1987. Customer funding of the cleanup amounted to about $48 million in 1987 and is estimated to total approximately $250 million over the course of the cleanup effort. GPU continues to provide cash advances from internal sources to alleviate any cash flow problem related to cleanup activities. The total 1987 advance is estimated at $37 million. The GPU projections provided to NRC indicate a continuing GPU commitment to provide such cash advances as needed. Continued improvement in GPU's financial condition and cash flow position gives greater assurance that such cash advances will be made.

Cost Sharing Plan. During 1987, GPU 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 (see *1986 NRC Annual Report,* p. 150). The Edison Electric Institute's (EEI) industry cost-sharing program paid its committed $26 million annual contribution in 1987, the third year of industry contributions through the EEI program. The NRC will continue to monitor the cleanup funding situation closely.

Operational Experience

Chapter 4

ANALYSIS AND EVALUATION OF OPERATIONAL DATA

The NRC Office for Analysis and Evaluation of Operational Data (AEOD) was strengthened and expanded in 1987, in accordance with the Commission's emphasis on operational safety matters. The office serves as the focal point for the continuing independent assessment of operational data and manages the review, analysis, and evaluation of reactor plant safety performance. Under the NRC reorganization (see Chapter 1), it is also responsible for the Incident Response Program, the Diagnostic Evaluation Program, the NRC's Technical Training Center, and the management of the Committee to Review Generic Requirements (CRGR); AEOD continues to manage the Incident Investigation Program (IIP). The office reports directly to the Executive Director for Operations (EDO).

AEOD's specific functions include the following:

- Analyzes and evaluates operational safety data associated with all NRC-licensed activities and identifies safety issues which require NRC staff actions.
- Develops, in consultation with the Office of Nuclear Reactor Regulation (NRR) and Office of Nuclear Material Safety and Safeguards (NMSS), the NRC policy for response to incidents and emergencies and ensures that the NRC response is consistent with the NRC role and licensee responsibilities and is coordinated with Federal and State response activities.
- Develops an agency-wide technical training program for the formal qualification of a broad range of technical positions within the NRC staff.
- Provides technical training, as needed, to maintain requisite level of knowledge by NRC technical personnel, such as regional and headquarters-based inspectors, reactor operator license examiners, Operations Center duty officers, and other NRC technical staff. In addition, provides training to certain other Federal, State, and foreign government employees, license examiners, reviewers and researchers.
- Provides continuous manning and operational readiness of the Operations Center and provides centralized comprehensive screening of reactor events, non-reactor events, and other information reported to the Operations Center for emergency response and other prompt reaction to events data.
- Develops the agency program and procedures for evaluation of reactor plant performance indicators to provide data to regional and headquarters management.
- Develops and directs the NRC program for conduct of diagnostic evaluation of licensee performance and directs the diagnostic evaluation teams.
- Develops policy, program requirements, and procedures for NRC investigations of significant operational events.
- Provides management and technical support for the CRGR and ensures that its functions are implemented in a thorough and timely manner.
- Identifies needed operational data and related technical information to support operational safety data analyses activities and develops agency-wide operational data reporting, organization, and retrieval methods and systems.
- Develops a coordinated system for feedback of operational safety information to other NRC organizational components, licensees, and non-licensee organizations, as appropriate, and prepares the Abnormal Occurrence Report to Congress.
- Provides coordination on behalf of the agency with respect to generic operational safety data information and systems with industry, foreign governments and other agencies involved with the collection, analysis, and feedback of operational data.
- Performs independent in-depth analysis or evaluation of any operational data, event, or situation, as warranted.

Committee to Review Generic Requirements.

All generic requirements proposed by the NRC staff related to one or more classes of reactors, including backfit requirements, must be reviewed by the Committee to Review Generic Requirements (CRGR). The Committee seeks to eliminate unnecessary demands on licensees and NRC resources by ensuring that the need for a new requirement can be demonstrated by those proposing it. (See the *1983 NRC Annual Report*, pp. 1-3, for a full description of CRGR's structure and review process.) Through its review, the CRGR seeks assurance that a proposed requirement (1) is necessary for the public health and safety, or (2) is likely to result in a net safety improvement, and (3) is likely to have an impact on the public, industry, and government which is consistent with and justified by the urgency of the need for the safety improvement to be realized.

Following its review, the CRGR recommends to the Executive Director for Operations that the proposed requirements be approved, disapproved, modified, or conditioned in some way. The EDO considers CRGR recommendations, as well as those of cognizant NRC offices, in deciding whether a requirement shall be imposed. From its inception in November 1981 through September 1987, the CRGR has held 121 meetings and considered a total of 190 separate issues. In fiscal year 1987, the CRGR held 25 meetings and considered 42 issues.

Some of the backfit issues completed by CRGR in fiscal year 1987 include the following:

(1) Endorsed proposed final resolution of boiling water reactor (BWR) pipe cracking problem to be imposed via Generic Letter.

(2) Endorsed issuance of Bulletin on licensee programs for inspection and monitoring of pipe wall thinning.

(3) Endorsed proposed final resolution of an Unresolved Safety Issue, Station Blackout (USI A-44), to be imposed via GDC-17 Amendment and implemented via new Regulatory Guide.

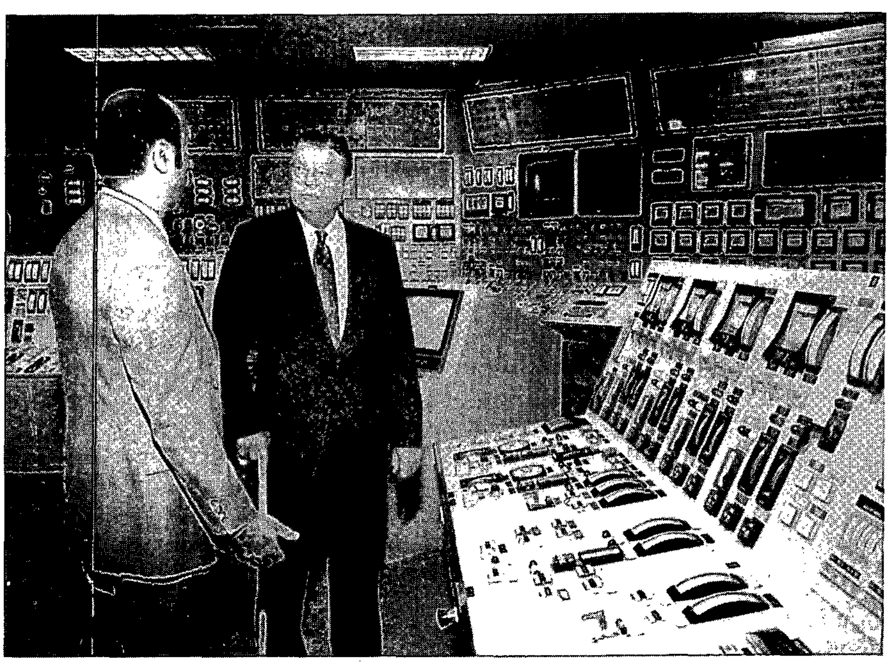
(4) Endorsed issuance of Generic Information Request Letter on residual heat removal with partially drained reactor vessel.

For fiscal year 1987, the CRGR endorsed 12 generic backfits in the form of seven Rules, one Regulatory Guide, three Generic Letters, and one Bulletin.

Annual Report to the Commission.

In April 1987, AEOD submitted an annual report to the Commission (AEOD/S701) for calendar year 1986. The report describes operating experience at nuclear power plants and other licensed facilities, summarizes AEOD activities, and provides a status report of AEOD recommendations. The report contains a number of significant observations on nuclear power plant operational experience, including the following:

- Based on a review of 21 significant events occurring at nuclear power plants between January 1985 and July 1986, AEOD identified some common characteristics of the events. Twenty of the events involved hardware problems (design problems, common mode failures, system interaction problems, and generic problems) and human factors problems related to procedural and training deficiencies. To reduce the occurrence of such



Responsibility for the NRC Technical Training Center (TTC) in Chattanooga, Tenn., was transferred to the NRC's Office of Analysis and Evaluation of Operational Data (AEOD) in the 1987 reorganization. On July 16, Chairman Lando W. Zech, Jr., visited the TTC, where he was briefed on a reactor simulator which had been leased from the Westinghouse Electric Co. for training in PWR technology. Chairman Zech is shown here with PWR Branch Chief Steve Showe in the simulator control room.

events, AEOD recommended efforts to reduce scrams or trips and the identification and correction of the root causes of safety system failures.

- AEOD has continued to identify common mode failure mechanisms through in-depth studies of operational events and has alerted NRC program offices and industry groups to their existence, potential significance, and the need for corrective action.
- AEOD's case studies in 1986 addressing failures of inverters, motor operated valves, and electronic components in instrumentation and control systems revealed recurring failures of this equipment at operating nuclear plants and found that such failures seemed to reflect incorrect or incomplete root cause determinations or a lack of effective corrective actions.
- Analysis of the 1986 Licensee Event Reporting (LER) indicates that the largest percentage of the reports were associated with scrams and Engineered Safety Features (ESF) actuations and the second largest with a condition prohibited by technical specifications or a shutdown required by technical specifications. AEOD found that the scram frequency was noticeably improved (i.e., reduced) for 1986 as compared to 1985 and that the number of plants exhibiting relatively high scram rates has decreased significantly since 1984. The LER description of each scram indicated that hardware failures dominated in 1986 as they had in previous years. The data also indicated that the rate of ESF actuations was decreasing for the first time since such events become reportable.
- Analysis of component failure data from the Nuclear Plant Reliability Data System (NPRDS) by AEOD's Trends and Patterns Analysis Branch indicated that proper maintenance and the use of appropriate materials were the dominant considerations in avoiding problems with the main feedwater (MFW) components. Although the MFW is not a safety system, upgrades of the MFW and supporting systems to make the MWF system more reliable will reduce reactor scrams and unnecessary demands on safety systems.

Analyses of Operational Experience

Domestic. The Licensee Event Report (LER) rule (10 CFR 50.73) became effective on January 1, 1984 (see the *1985 NRC Annual Report,* p. 61). The rule requires that the licensee event reports describe in a reasonably complete and detailed manner all actuations of engineered safety features (ESF), including scrams (reactor shutdown), all losses of safety function at a system level, all significant systems interactions, all technical specification violations, and all significant internal and external threats to plant safety.

The LERs provide the NRC with operational data with which to judge the safety of nuclear plants and potential problems at nuclear plants. To effectively manage and utilize the large quantities of LER reports, AEOD contracts with the Nuclear Operations Analysis Center (NOAC) at Oak Ridge, Tenn., which operates and maintains the Sequence Coding and Search System (SCSS), a computerized storage and retrieval system for LER data. SCSS encodes all the relevant technical information provided by the licensee in the LER with enough "tags" to assure ready retrieval of individual items. During fiscal year 1987, about 3,200 LERs were added to the system. The latest increase brought the number of LERs added to the data base (since 1980) to more than 26,100. During the report period, SCSS was made directly accessible to about 10 additional users, making a total of more than 65 authorized users of the data base.

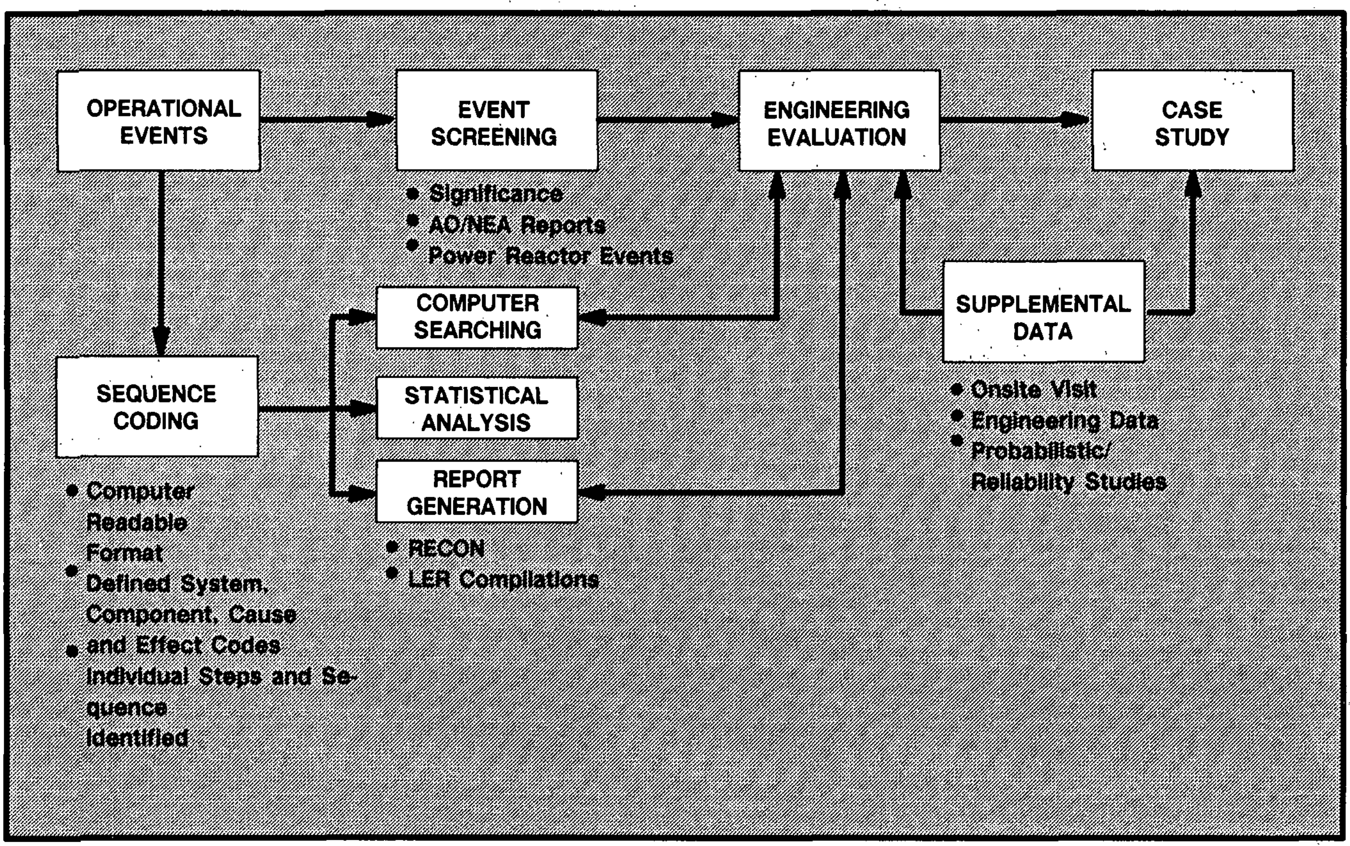
Trends and patterns analyses are performed on the LER data to detect anomalous or deteriorating trends in the operation of the plants and reliability of the plant's safety equipment. The program is designed to detect, through statistical and engineering analysis, those trends or patterns in incidents of low individual significance that may indicate an unrecognized safety concern. Several trends and patterns analysis reports on operational experience are summarized below. During fiscal year 1987, the NRC continued a trends and patterns analysis of component level data using data from INPO's Nuclear Plant Reliability Data System. Based on the more detailed data provided under the LER rule, the 1987 program included studies focused on reactor trips, ESF actuations, system unavailability, and technical specification violations.

INPO's maintenance of the NPRDS, the industry's component failure data base, is a voluntary initiative. The Commission has requested that a continuing NPRDS evaluation program be carried out by the NRC staff. An evaluation report on NPRDS progress (February 1987, SECY-86-35) noted substantial improvement in the percentage of NPRDS-reportable failures submitted to the data base. As a result, the NRC staff believes the increased use of NPRDS as a source of operating experience data is warranted.

Foreign. During 1987, the NRC continued efforts to increase the number and usefulness of foreign experience reports that are received. Such reports supplement U.S. experience, particularly with regard to the effect of different safety equipment configurations, of redundancy, and of operator actions and degree of involvement required during normal or off-normal plant operations. With the help of the Nuclear Operations Analysis Center, the NRC continues to systematically screen and assess selected foreign information for its applicability to the U.S. program and to abstract it for computerized data filing. This file now contains information on more than 8,300 foreign events.

NRC also continued its participation in the exchange of operational event information with other countries through activities involving the Nuclear Energy Agency (NEA), the International Atomic Energy Agency (IAEA), and various bilateral agreements. In September 1987, the NRC participated in the annual IAEA/NEA meetings. A number of

AEOD Analysis and Evaluation Program



significant technical papers and events were identified there which were relevant to U.S. reactor operations. The NRC will continue to take an active part in efforts to improve the International Reporting System, in effect since the late 1970s.

Engineering Analyses of Operational Experience

AEOD special studies issued during fiscal year 1987 included the following:

Air Systems Problems At U.S. Light Water Reactors (LWRs). This study provides a comprehensive review and evaluation of the potential safety implications associated with air systems problems at U.S. LWRs. The study analyzes operating data, focusing upon degraded air systems, and the vulnerability of safety-related equipment to common mode failures associated with air systems. Several recommendations are presented to reduce risk, enhance safety, and improve plant performance.

Air systems are not safety-grade systems at most operating plants. As a result, plant accident analyses assume that safety-related equipment dependent upon air systems will either "fail safe" upon loss of air or perform their intended function with the assistance of backup accumulators. The report highlights 29 failures of safety-related systems that resulted from degraded or malfunctioning air systems, thereby contradicting prevailing assumptions. Some of the systems which were significantly degraded or failed were decay heat removal, auxiliary feedwater, BWR scram, main steam isolation, salt water cooling, emergency diesel generator, containment isolation, and the fuel pool seal system. These are viewed as important precursor events.

The report addresses specific deficiencies deriving from the following observations: (1) the air quality capability of the instrument air system filters and dryers does not always match the design requirements of the equipment using the air, resulting in mismatched equipment; (2) maintenance of instrument air systems is not always performed in accordance with the manufacturer's recommendations; (3) air quality is not usually monitored periodically; (4) plant personnel frequently do not understand the potential consequences of degraded air systems; (5) operators are not well trained to respond to losses of the instrument air, and emergency operating procedures are frequently inadequate; (6) at many plants the response of key equipment to a loss of instrument air has not been verified to be consistent with the FSAR; (7) safety-related backup accumulators do not necessarily undergo surveillance testing or monitoring to confirm their readiness; and (8) the size and the seismic capability of safety-related backup accumulators at several plants are inadequate.

The study concludes that additional actions are necessary to assure that air systems are maintained and operated at levels which will enable plant equipment to function as designed and not be subject to unanalyzed failure modes. To date, such failures have not occurred in connection with a limiting transient or accident. The recommendations include steps (1) to ensure that air system quality meets the requirements specified by the manufacturers of the plant's air-operated equipment; (2) to ensure adequate operator response by formulating and implementing anticipated transient and system recovery procedures for loss-of-air events; (3) to ensure, by improved training, that plant operations and maintenance personnel are sensitized to the importance of air systems and the vulnerability of safey-related equipment served by the air systems to common mode failures; (4) to confirm the adequacy and reliability of safety-related backup accumulators; and (5) to verify equipment response to gradual losses of air to ensure that such losses do not result in events which fall outside licensing analyses.

Loss of Decay Heat Removal Function at Pressurized Water Reactors With Partially Drained Reactor Coolant Systems. On April 10, 1987, the residual heat removal (RHR) pumps at Diablo Canyon Unit 2 (Cal.) were tripped as the consequence of vortexing/cavitation, and the reactor coolant system was partially drained. As a result, the plant lost its ability to remove decay heat for 85 minutes. During this period, the reactor coolant system (RCS) heated up and bulk boiling was present in the RCS.

AEOD reviewed this event and issued a special report, S702, which noted that the loss of the decay heat removal function (DHR) is one of 37 such events that have been reported at U.S. pressurized water reactors (PWRs) over a 10 year period. These events have the potential for leading to more serious events.

The report notes aspects of partially drained (mid-loop) operation which contribute to risk, especially operation with an open equipment hatch. In addition, the report presents results of recent probabilistic risk assessments which address risks associated with shutdowns at PWRs. The report concluded that there is adequate justification for NRC to send a generic communication to PWR licensees regarding corrective actions to minimize the likelihood of loss of DHR during periods of high risk. Subsequently, this was done.

Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems. This analysis focused on four events in which overheating of solid state electronic components led to problems in safety-related instrumentation and control (I&C) systems. The problems involved plant transients, control system malfunctions, protection system channel inoperabilities, safety system inadvertent actuations and failures, and annunciation and indication system errors. The review of the four events raised concerns of decreased reliability of solid state components and increased susceptibility for common cause failure of redundant I&C system channels attributable to overheating of heat sensitive electronic components. These concerns are generic to all operating nuclear units that utilize solid state electronic components. The analysis pointed out the need for plant operators to be aware of and trained in the consequences of the overheating problem.

The review of the four events found that, in general, it was not easy to identify elevated room ambient temperatures of instrument cabinet internal temperatures as the root cause for the failure of electronic components. Licensees experienced several failures and many corrective actions over an extended period before finally identifying overheating of components as the underlying reason for many of the failures. Technical specifications regarding area ventilation cooling systems and instrumentation systems were also reviewed and were found inadequate with respect to the temperature rating of electronic components inside the instrument cabinets.

In addition, a review of the staff's proposed resolution of the Unresolved Safety Issue (USI A-44) regarding design adequacy and capability of instrumentation and control system equipment needed to function in environmental conditions associated with a station blackout found that additional actions were required. Specifically, plant specific evaluations are needed with regard to the actual temperature and condition of heat sensitive components inside instrument cabinets.

The report's recommendations address the following issues: (1) the establishment of procedures and training of operators to cope with loss of cooling to instrument cabinets; (2) the need to monitor actual conditions (specifically, temperature) in instrument cabinets; (3) the need for plant technical specification requirements governing the operability of control room cooling and ventilation systems which reflect actual temperature in the instrument cabinets; and (4) the need for specific consideration of this issue in the plant-specific evaluation and resolution of the station blackout issue (Unresolved Safety Issue A-44).

Operational Experience Involving Losses of Electrical Inverters. This study involved (1) a review of previous activities in this area by both the NRC and industry groups; (2) an analysis and evaluation of inverter loss events which occurred during 1982, 1983, and 1984; and (3) recommendations stemming from these events.

The major findings and conclusions of this study were: the number of events involving inverter losses per reactor-year shows little or no improvement in each calendar year; events involving inverters illustrate that a loss of an inverter often results in a loss of power for the associated bus; the dominant cause of the inverter loss events is component failure.

The review indicates that a major contributing factor in the component failures is incompatibility between actual plant service conditions and design service conditions. The study identifies three potential failure mechanisms for inverters—high ambient temperature and humidity within inverter enclosures, electrical interconnections and physical arrangements of components which form the inverter circuitry, and voltage spikes and perturbations. The second major cause of events involving inverter losses was personnel action. Principal contributing factors to such actions are inadequate maintenance and testing procedures; deficient practices; and inadequate planning, training, and verification for related maintenance and testing activities.

The study also concluded that two specific areas of circuitry design warrant further consideration. The first involves the RCS pressure instrumentation channels associated with PWR RHR system isolation valves. A loss of power to either of two instrumentation channels as a result of a single inverter loss causes a loss of shutdown cooling. The second involves the circuitry which monitors the position of circuit breakers for reactor coolant pump (RCP) motors in Westinghouse plants which use the Solid State Protection System (SSPS). Upon loss of power output from an inverter unit, this circuitry de-energizes, thus indicating to the SSPS that a circuit breaker for an RCP motor is open when it is not. Above a certain reactor power level, the SSPS causes a trip of the reactor with an attendant plant transient.

A final finding was that plant technical specifications of operating restrictions (e.g., action statements) for an inoperable inverter, or the unavailability of one of two input power sources for inverters with dual power inputs, are not consistent among comparable plant designs. At multiple unit sites, inconsistencies in the technical specifications between plant units can contribute to operating errors by plant personnel.

The study recommended issuance of an Information Notice addressing the inverter losses and other findings of the report; reassessment by NRR of the circuitry which monitors the position of the circuit breakers for reactor coolant pump motors; and review of the technical specifications related to inverters to ensure that operating restrictions for comparable plant designs are consistent.

A Review of Motor-Operated Valve Performance. The purpose of this study was to provide an overview of operating experience to identify failure modes and assess valve assembly performance. The study reviewed previous studies and operating experience from 1981 to the present and incorporated new data from a limited test program using signature tracing techniques on valves in operating plants. The study was performed in response to the June 9, 1985, event at the Davis-Besse (Ohio) facility in which two motor operated valves in the Auxiliary Feedwater System shut and failed to reopen in accordance with the actuation signal.

The events reviewed in the study included 565 LERs (1981 to the present) from the Sequence Coding and Search System (SCSS) and more than 600 events for 1984 and 1985 from the Nuclear Plant Reliability Data System (NPRDS). These data indicated that recent motor-operated valve events involve failures that are similar to those observed in earlier studies. Since no improvement in the rate of failure is apparent, the previous recommendations are still valid.

The study's most important conclusion is that current methods and procedures at many operating plants are not adequate to assure that motor-operated valve assemblies will operate when needed (e.g., under credible accident conditions). The limited NRC test program, using signature tracing equipment, identified several safety-related valves in operating plants that exhibited deficiencies which could prevent valve operation under accident conditions, even though the valve worked under test conditions. The most common deficiencies involved incorrect adjustments that were undetected by plant procedures intended to assure operability, such as surveillance testing (plant technical specifications and American Society of Mechanical Engineers (ASME) Code, Section XI inservice testing) or operator observations. AEOD suggested a concerted, high priority licensee effort to develop and implement improved guidance, procedures, and/or equipment to address all aspects of safety-related motor-operated valve assembly operability.

The study recommended (1) implementation of the recommendations presented in AEOD Case Study Report C203 (May, 1982) and AEOD Special Study Report S503 (September, 1985); (2) establishment by licensees of procedures and diagnostic capability to determine root causes of failure to establish programs that would provide assurance of motor-operated valve assembly performance and reliability under accident conditions; (3) development by licensees of a strong training program to ensure that appropriate information and instructions are disseminated to operating and maintenance personnel (this effort should receive site management support); and (4) extension of the scope of Bulletin 85-03 to cover all safety-related motor-operated valve assemblies required to be tested for operational readiness in accordance with 10 CFR 50.55a (g).

Trends and Patterns Analyses Of Operational Experience

Operational Experiences at Newly Licensed Nuclear Power Plants. In 1987, AEOD conducted a study of the operational experience of commercial power reactors during their first two years of operation. The goals of the study were (1) to characterize the trends in operational events experienced at newly licensed plants, (2) to identify correlations between plant attributes and performance, and (3) to provide feedback to facilitate improvement. The report documenting

Table 1. AEOD Reports Issued During FY 1987

Case and Special Studies

Designation	*Subject*	*Issued*
C603	A Review of Motor-Operated Valve Performance	12/86
C604	Effects of Ambient Temperature on Electronic Components in Safety-Related Instrumentation and Control Systems	12/86
C605	Operational Experience Involving Losses of Electrical Inverters	12/86
C701	Air Systems Problems at U.S. Light Water Reactors	3/87
P701	Trends and Patterns Program Report—Operational Experience Feedback on Main Feedwater Flow Control and Main Feedwater Flow Bypass Valves and Valve Operators (This contains proprietary information and is not publicly available.)	3/87
S701	AEOD Annual Report for 1986	4/87
S702	Loss of Decay Heat Removal Function at Pressurized Water Reactors with Partially Drained Reactor Coolant Systems	5/87

Engineering Evaluation	*Subject*	*Issued*
E611	Deficiencies in Seismic Anchorage for Electrical and Control Panels	10/86
E612	Emergency Diesel Generator Component Failures Due to Vibration	12/86
E613	Localized Rod Cluster Control Assembly Wear at PWR Plants	12/86
E701	Potential Containment Airlock Window Failure Due to Radiation	1/87
E702	MOV Failure Due to Hydraulic Lockup from Excessive Grease in Spring Pack	3/87
E703	Loss of Off-site Power Due to Unneeded Actuation of Startup Transformer Protection Differential Relay	3/87
E704	Discharge of Primary Coolant Outside of Containment at PWRs While on RHR Cooling	3/87

Engineering Evaluation	*Subject*	*Issued*
E705	RWCU System Automatic Isolation and Safety Considerations	3/87
E706	Inadequate Mechanical Blocking of Valves	3/87
E707	Design and Construction Problems at Operating Nuclear Plants	3/87
E708	Depressurization of Reactor Coolant Systems at PWRs	8/87
E709	Auxiliary Feedwater Pump Trips Caused By Low Suction Pressure	8/87

Technical Review	*Subject*	*Issued*
T608	Hydrogen Fire and Failure of Detection System	11/86
T609	Foreign Material and Debris in Safety-Related Fluid Systems	12/86
T610	ADS/RCIC System Interaction Events at River Bend Unit 1	12/86
T611	Denied Access Due to Negative Room Pressure	12/86
T612	Degradation of Safety Systems Due to Component Misalignment and/or Mispositioned Control/Selector Switches	12/86
T701	Compression Fitting Failures	1/87
T702	Leaking Pulsation Dampener Leads to Loss of Charging System	3/87
T703	Potential for Loss of Emergency Feedwater Due to Pump Runout During Certain Transients	3/87
T704	Pressurizer Code Safety Valve Reliability	5/87
T705	Occurrence of Events Involving Wrong Unit/Wrong Train/Wrong Component—Update through 1986	5/87
T706	Recent Events Involving Turbine Runbacks at PWRs	6/87
T707	Undetected Loss of Reactor Water	8/87
T708	Problems with High Pressure Safety Injection Systems in Westinghouse PWRs	8/87

the study, NUREG-1275, was subjected to the formal AEOD peer review process that included the staffs of the plants that were analyzed, industry organizations, and all major NRC staff offices.

The study concluded that it was possible to achieve significant improvements in early plant performance, learning curves, and early commercial operation. The study disputed the assumption that new plants must experience a high frequency of unplanned events during their first two years of operation. Some power reactor licensees have already recognized the need for preventive programs and have developed programs that, when implemented fully, could result in improved performance in many respects.

The analyses also showed that, without early effective corrective action, the root causes of a high event frequency will likely persist during early commercial operation. At this stage, the relatively high challenge frequency coupled with the potential of undetected systems problems might present significant difficulty to a relatively inexperienced operating crew. It was determined that a root cause corrective action program, in response to events, is a necessary factor in achieving good performance.

Since October 1986, both the NRC staff and the industry have given increased attention to newly licensed plants. The new plant study reinforced the need for these efforts. Following publication of NUREG-1275, the staff met with the Commission (August 4, 1987) to discuss new plant performance. As a follow-up, the staff transmitted at the Commission's request a copy of NUREG-1275 to newly licensed nuclear power plants, to plants still under construction, and to plants undergoing a prolonged shutdown, and requested that management review the applicable improvement lessons identified in NUREG-1275.

Accident Sequence Precursor (ASP) Program. This AEOD program, performed at the Oak Ridge National Laboratory (ORNL), involves the evaluation of operational data from the perspective of risk of reactor core damage. The ASP program is intended to systematically determine and document potential safety significant events experienced by LWR power plants. The program reviews operational events (LERs) from light water reactors to identify, categorize, and evaluate precursors to potential severe core-damage accidents. Accident sequences considered in the study are those associated with inadequate core cooling. Accident sequence precursors can be infrequent initiating events or equipment failures which, when coupled with one or more postulated events, could result in a plant condition with inadequate core cooling. The precursor events give an indication of the kinds of scenarios to which nuclear plants are now, or have in the past, been vulnerable. Precursor events generally involve one or more of the following:

(1) Total failures of safety systems.

(2) Degradation of two or more safety systems.

(3) Initiator events with plant safety system response off-normal.

Event significance is appraised by quantifying an event tree upon which event particulars are mapped. For example, given a stuck-open power operated relief valve in a PWR, the event tree model indicates all foreseen paths or scenarios that the equipment and operator response could take in reaching a final state of either safe shutdown or core damage.

The ASP data represent the most significant initiators and system failure occurrences at the operating U.S. commercial nuclear power plants. By the end of 1986, the ASP data base contained 99 plants. These plants had achieved a cumulative operating experience of approximately 844 reactor-years, with 367 precursors identified, documented, and evaluated.

On the basis of the precursor analyses, there appears to be a downward trend in the number of serious precursor events for 1984-1986 data as compared to 1969-1979 data. Two reports on this program were prepared in 1987:

(1) "Precursors to Potential Severe Core Damage Accidents: 1984, A Status Report", NUREG/CR-4674, Volumes 3 and 4, May 1987.

(2) "Precursors to Potential Severe Core Damage Accidents: 1986, A Status Report", NUREG/CR-4674, Volumes 5 and 6, late summer 1987.

Reviews of Main Feedwater Control and Control Bypass Valve Experience. In 1987, the Idaho National Engineering Laboratory (INEL), under NRC contract, completed analyses of data from the NPRDS for main feedwater (MFW) flow control valves and flow control bypass valves in U.S. commercial PWRs during 1984 and most of 1985. The purpose was to identify the lessons from operating experience and recommend appropriate measures to resolve safety concerns. Major components within the MFW system were the first ones selected for analysis because they support primary coolant heat removal and are important in accident sequences where the initiator does not affect availability of the power conversion system. MFW valve component failures have been a significant cause of unplanned reactor trips which result in demands on safety systems. The statistical trends and patterns analysis indicated that the particular plant in which a component was installed had a greater influence on the performance of the components than attributes of the component.

The follow-up engineering evaluation focused on the identification of plant-specific practices and conditions that were responsible for some of the problems, and of practices to remedy these problems and prevent their recurrence. The evaluation indicated that the main source of variation in MFW control and bypass valve failure rates was the difference in maintenance practices among units and stations. The evaluation also showed that proper maintenance and the use of appropriate subcomponents are dominant in avoiding problems with the MFW components. Although the MFW system is not a safety system, upgrades of that system and supporting systems—such as the control air and oil systems

Table 2. Non-Reactor Evaluations

Designation Issued	*Subject*
N701 1/87	Diagnostic Misadministrations Involving the Administration of Millicurie Amounts of Iodine-131
N702 3/87	Medical Misadministrations Reported to NRC for the Period January 1986 through December 1986
N703 3/87	Report on 1986 Non-reactor Events

to make the MFW system more reliable—reduce reactor scrams and unnecessary demands on safety systems. The evaluation contains proprietary information and has been issued as an internal report.

Analyses of Non-Reactor Operational Experience

In addition to the screening and analysis of reactor operating experience, the AEOD reviews the non-reactor operational experience associated with the activities and facilities licensed by NMSS (see Chapter 5) and by the Agreement States (see Chapter 9). AEOD also conducts studies from a human factors perspective on both reactor and non-reactor, as well as medical misadministration, data files. From the events screened during fiscal year 1987, about 150 non-reactor events and 400 medical misadministrations were entered into the files.

During fiscal year 1987, the AEOD issued two survey reports which contain a review of all 1986 reports. In addition, the staff issued a report on diagnostic misadministrations involving iodine-131. Although this latter report was issued as an engineering evaluation (i.e., without peer review), it provided the basis for proposed substantial changes to the NRC regulations and hence is discussed in this section.

Medical Misadministrations Reported to NRC for the Period January 1986 through December 1986. A total of eight therapy and 438 diagnostic misadministrations were reported in 1986. Four of the therapy misadministrations involved teletherapy, two involved brachytherapy, and two involved radiopharmaceutical therapy. Of the 438 diagnostic misadministrations, five involved the administration of therapy range dosages of iodine to patients. The findings contained in the report indicated that:

- A number of teletherapy and brachytherapy misadministrations reported to NRC for 1986 could likely have been prevented by quality assurance procedures directed to verifying dose calculations, type of treatment, and patient identification.
- Most of the diagnostic misadministrations reported to NRC for 1986 involved either the administration of the wrong radiopharmaceutical or the administration of a radiopharmaceutical to the wrong patient.

The number, type and cause of diagnostic misadministrations are about the same as reported for 1985. The causes reported by licensees are generally the same as those reported in the past, that is, simple errors associated with (1) preparation of radiopharmaceuticals, (2) processing nuclear medicine requisitions, and (3) patient identification. In addition, the primary cause of misadministrations involving the administration of millicurie amounts of iodine to patients was the failure of licensees to exercise adequate control over the administration.

Report on 1986 Non-reactor Events. The report shows that, as in prior years, most 1986 non-reactor events concern incidents of modest overexposure, lost or abandoned sources, or leaking sources.

A significant event that occurred at a non-reactor facility in 1986 was the rupturing of a large cylinder of uranium hexafluoride, releasing most of its contents. As the result of that event, more rigorous reporting of minor events at uranium hexafluoride plants resulted in an apparent increase in the numbers of events at fuel cycle plants.

Diagnostic Misadministrations Involving the Misadministration of Millicurie Amounts of Iodine-131. This engineering evaluation documents AEOD's review and evaluation of 14 diagnostic misadministrations involving the administration of a therapy range of iodine-131 to patients. The misadministrations were reported between January 1982 and

June 1986. Some of the findings of the engineering evaluation are as follows:

- The direct cause of ten of the 14 reported iodine misadministrations (71 percent) was ascribed to either the physician's order being misinterpreted by or miscommunicated to the technologist (seven cases), or the technologist's not knowing the correct dosage to administer for thyroid scan procedures that involved scanning the chest area (three cases).
- Causal factors associated with occurrence of the misadministrations were:
  (1) Use of verbal orders for nuclear medicine studies,
  (2) Use of similar terms by referring physicians and licensees to refer to different procedures,
  (3) Lack of technologist training,
  (4) Lack of procedures, and
  (5) Failure of technologist to follow procedures.
- The underlying cause of 11 of 14 (79 percent) of the misadministrations appears to have been a lack of licensee control over the administration of millicurie amounts of iodine-131 to patients. These 11 misadministrations could likely have been prevented, despite the errors that led to the them, if the prescription for the iodine-131 dosage had been verified for each patient before the iodine-131 was administered.

Two suggestions were made: communicate the contents of the report to affected licensees; and assess proposed regulatory changes to 10 CFR Part 35 to determine whether quality assurance procedures should be required for this type of diagnostic study.

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, P.O. 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, or at Local Public Document Rooms throughout the country (see Appendix 3.))

There were three abnormal occurrence (AO) reports issued in 1987 ((NUREG-0090, Vol. 9, No. 2 (April-June, 1986); Vol. 9, No. 3 (July-September, 1986); and Vol. 9, No. 4 (October-December, 1986)). Vol. 10, No. 1 (January-March, 1987) was issued in October 1987 and is included here since all of the AOs described in that report occurred during fiscal year 1987. The four reports covered 10 AOs at nuclear power plants, 19 AOs at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and five AOs at Agreement State licensees. The reports also contained updated information for certain AOs which had been previously reported.

The AOs reported in the four quarterly reports are listed in Table 3 and are briefly described below. Some of the events resulted in escalated enforcement actions, including civil penalties, by the NRC. (See Chapter 1 for a listing of all civil penalties imposed by the Office of Enforcement during the report period, with capsule descriptions of the reasons therefor.)

Nuclear Power Reactors

Out of Sequence Control Rod Withdrawal. On March 18, 1986, during a startup of Peach Bottom Unit 3 (Pa.), personnel errors by four licensed operators resulted in a control rod being withdrawn out-of-sequence without being detected by the operators. The next operating shift detected the error and manually scrammed the unit.

Boiling Water Reactor Emergency Core Cooling System Design Deficiency. On May 19, 1986, the Boston Edison Company notified the NRC that a significant design deficiency in the residual heat removal system minimum flow protection logic at the Pilgrim (Mass.) nuclear power plant had been discovered. Subsequently, it was found that some other GE-designed BWRs, i.e., Dresden 2 and 3 (Ill.) and Quad Cities 1 and 2 (Ill.) also contained the same design deficiency. A similar deficiency was also discovered in some PWRs. (See the update to AO 86-9 in NUREG-0090, Vol. 9, No. 3.)

Differential Pressure Switch Problem in Safety Systems at LaSalle Facility. On June 1, 1986, LaSalle Unit 2 (Ill.) experienced a feedwater transient that resulted in low water level in the reactor vessel. The level reached a point where an automatic reactor scram would be expected; however, no such scram occurred. Subsequent investigation found that the problem was caused primarily by inadequate calibration of mechanical differential pressure switches supplied by SOR, Incorporated (formerly the Static "O" Ring Pressure Switch Company). Similar switches are installed in safety systems at many nuclear power plants.

Abnormal Cooldown and Depressurization Transient at Catawba Unit 2. On June 27, 1986, while Duke Power Company was conducting a startup test at Catawba Unit 2 (S.C.) from remotely located control panels, the reactor underwent an unexpected depressurization and cooldown. There were no actual consequences to public health or safety. However, if the decay heat load of the reactor core had been greater and if the use of the remote shutdown panels had actually been required during a plant emergency, a more severe transient could have occurred.

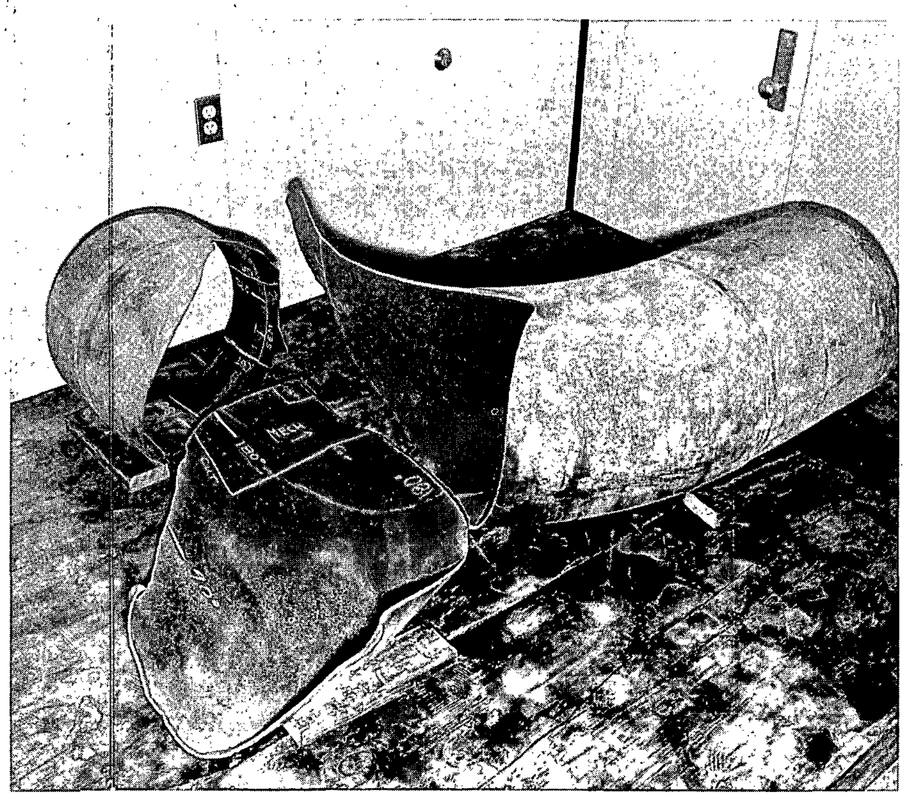
Significant Safeguards Deficiencies at Wolf Creek and Fort St. Vrain. On July 7, 1986, NRC Region IV (Dallas) issued enforcement letters containing Severity Level II violations to the licensees of two nuclear power stations for serious deficiencies in plant physical barriers. In the more serious case, the Wolf Creek (Kans.) site, it was determined that multiple uncontrolled access paths existed from the Owner Controlled Area (OCA) into the Protected Area (PA) and in two instances into Vital Areas (VAs). At the Fort St. Vrain (Colo.) site, NRC inspectors identified paths from the OCA to the PA and VA. Although each access had a barrier installed, these were judged to be inadequate and not capable of preventing an intruder from easily defeating them.

Significant Deficiencies in Access Controls at River Bend Station. On August 7, 1986, NRC Region IV (Dallas) issued an enforcement letter to the licensee for the River Bend (La.) nuclear power plant containing a Severity Level II violation regarding serious deficiencies in controlling the access of personnel to vital areas. Conditions existed whereby an intruder could have obtained unauthorized and undetected access into vital areas from either the protected area or other vital areas.

Loss of Low Pressure Service Water Systems at Oconee. On October 1, 1986, while Oconee Unit 2 (S.C.) was in a refueling outage, the Unit 2 load shed test was twice performed. During both tests, the low pressure service water system pump suction was lost. Investigation showed that, because of a design deficiency, the condenser circulating water system (which performs various safety-related functions) was degraded. A similar design deficiency existed on Units 1 and 3, which were operating at the time. These units were taken to cold shutdown until the problem was corrected.

Degraded Safety Systems Due to Incorrect Switch Settings on Rotork Motor Operators at Catawba and McGuire Nuclear Stations. On October 23, 1986, the licensee discovered that many valves in safety systems were degraded at the Catawba (S.C.) facility. On October 28, the licensee found a similar situation at its McGuire (N.C.) nuclear plant. The problem was caused by the licensee using improper torque switch settings on the valves' Rotork motor operators. This could result in the valves not performing as designed (e.g., activator motors switching off before the associated valves completed their travel).

Secondary System Pipe Break Resulting in Death of Four Persons at Surry Unit 2. On December 9, 1986, with both Surry Units 1 and 2 (Va.) at 100 percent power, Unit 2 tripped because of a low-low level in the "C" steam generator, followed by a rupture of an 18-inch suction line to the "A" train main feedwater pump. The reactor was taken to a cold shutdown condition with no release of radioactivity. However, eight individuals in the vicinity of the pipe rupture were injured by the release of steam and water. Four of the individuals subsequently died.



Four persons working near this 18-inch main feed pipe at the Surry Unit 2 power plant at Gravel Neck, Va., were killed and four others were injured when the pipe ruptured on December 9, 1986. The Surry reactor had been operating at 100 percent power when it was automatically shut down; the pipe ruptured shortly thereafter.

NRC Order Suspends Power Operations of Peach Bottom Facility Due to Inattentiveness of the Control Room Staff. On March 31, 1987, the NRC issued an Order Suspending Power Operation and Order to Show Cause (Effective Immediately) to the Philadelphia Electric Company. The Order directed the licensee to place Peach Bottom Unit 3 (Pa.), operating at about 100 percent power at the time, in cold shutdown (Unit 2 was already in cold shutdown for refueling) and maintain both units in cold shutdown pending further Order.

The Order was based on the fact that, at times during various shifts, one or more of the Peach Bottom operations control room staff—including licensed operators, senior licensed operators, and shift supervisors—had over a period of at least five months occasionally slept or otherwise been inattentive to specified duties. In addition, it was determined that plant management either knew of and condoned this inattentiveness, or should have known, and either took no action or inadequate action to correct the situation. Prior NRC inspections had identified other instances of inattention to duty or failure to adhere to procedures on the part of licensed operators in the control room at Peach Bottom.

Other NRC Licensees

Willful Failure to Report a Diagnostic Medical Misadministration. On May 8, 1985, a patient at Mercy Hospital, Wilkes-Barre, Pa., received an injection of a radiopharmaceutical (a diagnostic dose of technetium-99m) intended for another patient. The misadministration was willfully not reported to the NRC as required by 10 CFR 35.43.

Therapeutic Medical Misadministration. On April 9, 1986, at Maryview Hospital, Portsmouth, Va., a patient received a therapy dose in a chemical form other than that intended. This resulted in an unintended dose of several hundred rads to the patient's bone marrow. This could result in an increased chance of the patient's contracting leukemia.

Willful Failure to Report Diagnostic Medical Misadministrations. On April 22, 1986, the NRC issued an Order, effective immediately, removing a physician from the position of Radiation Safety Officer (RSO) and Authorized User at Bloomington Hospital, Bloomington, Ind. The physician had willfully not reported five diagnostic misadministrations; in addition, the physician obstructed the NRC inspection and misled the inspectors.

Diagnostic Medical Misadministration. On May 7, 1986, an outpatient of the Robert Packer Hospital and Guthrie Clinic in Sayre, Pa., received 10 millicuries of iodine-131, rather than the prescribed radiopharmaceutical for a bone scan, technetium-99m. This action resulted in a considerable dose to the thyroid, which could result in reduced thyroid function.

Diagnostic Medical Misadministration. On June 17, 1986, at the Tripler Army Medical Center, Haw., a patient received a dose of 3.09 millicures of I-131 instead of a prescribed dose of 50 microcuries for a thyroid imaging procedure. The radiation exposure received by the patient from the 3.09 millicure I-131 dose is estimated to be 2,472 rads to the thyroid, 0.43 rad to the ovaries, and 1.45 rads to the whole body. The dose to the thyroid could result in an impairment of function.

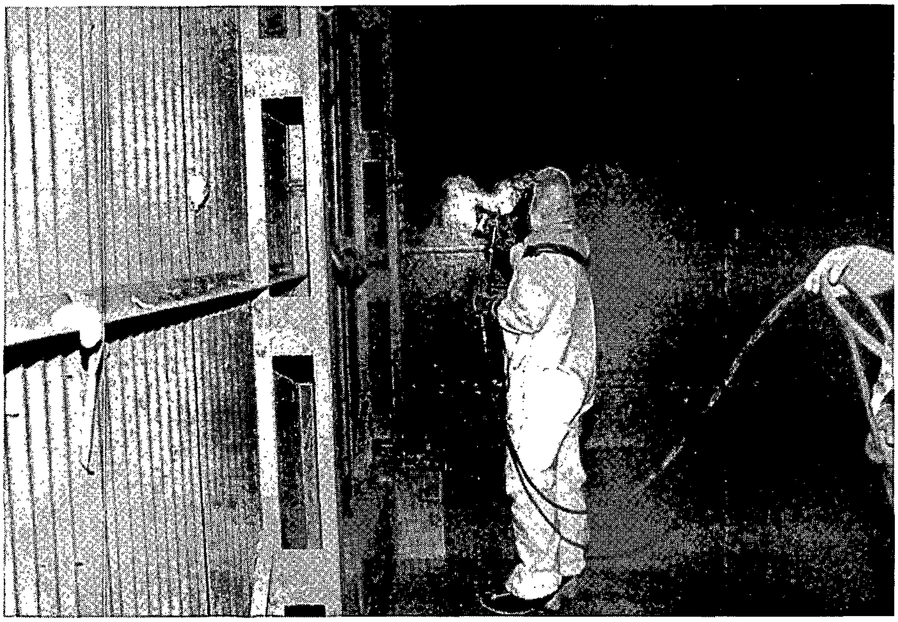
Release of Americium-241 Inside a Waste Storage Building at Wright-Patterson Air Force Base. On September 18 and October 6, 1986, a drum containing radioactive waste was opened to inspect its contents at Wright-Patterson Air Force Base, near Dayton, Ohio. Both openings resulted in a substantial release of radioactive americium-241 inside the waste storage building, which significantly contaminated the building. The cost to date of this incident is approximately $500,000. Additional costs will be incurred for further decontamination of the building or for dismantling and disposal of the building as radioactive waste.

Therapeutic Medical Misadministration. On October 6-8, 1986, a patient at the Cleveland Clinic Foundation, Cleveland, Ohio, received a series of therapeutic radiation exposures which resulted in a radiation dose of approximately 2,000 rads (head to waist), instead of an intended dose of 1,200 rads.

Suspension of License for Servicing Teletherapy and Radiography Units. On October 10, 1986, the NRC issued an Order suspending certain NRC-licensed service activities of Advanced Medical Systems, Inc., Geneva, Ohio. This action was taken after the NRC determined that the firm had been using untrained and unqualified employees to service cobalt-60 teletherapy units.

Diagnostic Medical Misadministration. On October 21, 1986, a patient at St. Luke's Hospital, Racine, Wis., received a whole body iodine-131 diagnostic scan while the intended procedure was to be a thyroid scan. The whole body scan involved 1.53 millicuries of iodine-131, about 30 times the normal dosage for a thyroid scan. The patient may experience reduced thyroid function.

Diagnostic Medical Misadministration. On November 18, 1986, a patient at Toledo Hospital, Toledo, Ohio, received a misadministration of a radiopharmaceutical when the wrong radioactive material was administered. The patient's physician prescribed a bone scan, which normally involves about 20 millicuries of technetium-99m MDP. Instead, the patient received about 20 millicuries of iodine-131. As a result, the patient's thyroid received a dose of several thousand rads which is expected to significantly reduce the thyroid's function.



Workers at the Wright-Patterson Air Force Base near Dayton, Ohio, are shown here spray-painting the interior walls of a waste-storage shed to seal in residual americium-241 contamination. The radioactive material had been released inside the building when a drum of radioactive waste was mistakenly opened on two occasions in late 1986. Nearly two months elapsed before the Air Force notified the NRC of what it described as a "minor spill." Remaining wastes in the building were repacked and, later, the building was dismantled.

Immediately Effective Order Modifying License and Order to Show Cause Issued to an Industrial Radiography Company. On December 30, 1986, an Order was issued to Met-Chem Testing Laboratories of Utah, Inc., located in Salt Lake City, that removed a senior management employee from any assignment or position influencing or involving the performance or supervision of any NRC-licensed activities. This action was taken after the individual admitted he had typed a letter and forged the signature of a radiographer for the purpose of explaining away an overexposure indicated on the radiographer's film badge.

Diagnostic Medical Misadministration. In a January 6, 1987 letter, Allegheny Valley Hospital, Natrona Heights, Pa., notified NRC Region I (Philadephia) that on November 21, 1986, a patient received an intravenous dose of 100 millicuries of technetium-99m rather than the prescribed dose of 20 millicuries. Estimated doses to various organs of the patient were: stomach wall, 25 rads; thyroid, 13 rads; intestinal wall, 6-7 rads; and bladder wall, 5 rads. These doses are about five times those which would have been expected had the prescribed doses been administered. No significant health effects are expected by the licensee.

Diagnostic Medical Misadministration. On January 21, 1987, NRC Region IV (Dallas) was notified by St. Anthony Hospital, Oklahoma City, Okla., that on January 12, 1987, a 15-year-old female was administered 400 microcuries of I-131 rather than the prescribed dose of 400 microcuries of I-123, resulting in a thyroid dose of about 1,490 rads. This may result in a small increased risk of reduction in thyroid function, and a small increased risk of latent thyroid cancer.

Diagnostic Medical Misadministration. In a letter dated March 2, 1987, the NRC received written notification that on February 19, 1987, a patient referred to the Nuclear Medicine Department of the University of Massachusetts Medical Center in Worchester, Mass., received a 5.5 millicurie dose of iodine-131 rather than the prescribed 5.0 microcuries. Based upon the 24-hour uptake and the measured effective half-life, the licensee estimated that the radiation dose to the patient's thyroid was 730 rads and the total body dose was 1.7 rads. The effect on the thyroid, if any, would be of no importance because prior to the event, the patient was scheduled for a thyroidectomy to be performed in March.

Significant Breakdown in Management Oversight and Control of Radiation Safety Program at Two of a Licensee's Irradiator Facilities. On March 17, 1987, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $10,000 (later reduced to $7,500) to Radiation Sterilizers, Inc. of Menlo Park, Cal. The violations were found at the licensee's irradiator facilities in Schaumburg, Ill., and Westerville, Ohio. Some of the violations related to unsafe practices which could have resulted in serious overexposures of licensee personnel. The base civil penalty for the violations would be $5,000. However, the penalty was escalated because of the licensee's prior knowledge of the problems, the licensee's failure to take prompt and effective corrective measures for previously identified violations, and the duration of some of the violations (over several months).

Diagnostic Medical Misadministration. On April 27, 1987, NRC Region IV (Dallas) was notified by Veterans Administration Medical Center, Boise, Idaho, that on April 1, 1987, 400 microcuries of I-131 were administered to an adult male for a total body scan; on April 6, 1987, it was

Table 3. Abnormal Occurrence Reports Issued During FY 1987

Occurrences at Nuclear Power Plants

Designation (AO#)	*Subject*	*NUREG-0090 Issue*
86-8	Out of Sequence Control Rod Withdrawal	Vol. 9, No. 2 January 1987
86-9	Boiling Water Reactor Emergency Core Cooling System Design Deficiency	"
86-15	Differential Pressure Switch Problem in Safety Systems at LaSalle Facility	Vol. 9, No. 3 April 1987
86-16	Abnormal Cooldown and Depressurization Transient at Catawba Unit 2	"
86-17	Significant Safeguards Deficiencies at Wolf Creek and Fort St. Vrain	"
86-18	Significant Deficiencies in Access Controls at River Bend Station	"
86-20	Loss of Low Pressure Service Water Systems at Oconee	Vol. 9, No. 4
86-21	Degraded Safety Systems Due to Incorrect Torque Switch Settings on Rotork Motor Operators at Catawba and McGuire Nuclear Stations	"
86-22	Secondary System Pipe Break Resulting in Death of Four Persons at Surry Unit 2	"
87-1	NRC Order Suspends Power Operations of Peach Bottom Facility Due to Inattentiveness of the Control Room Staff	Vol. 10, No. 1 October 1987

Occurrences at Other NRC Licensees (Industrial Radiographers, Medical Institutions, etc.)

Designation (AO#)	*Subject*	*NUREG-0090 Issue*
86-10	Willful Failure to Report a Diagnostic Medical Misadministration	Vol. 9, No. 2 January 1987
86-11	Therapeutic Medical Misadministration	"
86-12	Willful Failure to Report Diagnostic Medical Misadministrations	"
86-13	Diagnostic Medical Misadministration	"
86-14	Diagnostic Medical Misadministration	"
86-19	Therapeutic Medical Misadministration	Vol. 9, No. 3 April 1987
86-23	Release of americium-241 Inside a Waste Storage Building at Wright-Patterson Air Force Base	Vol. 9, No. 4 July 1987
86-24	Therapeutic Medical Misadministration	"

Table 3. Abnormal Occurrence Reports Issued During FY 1987 (Cont'd)

Occurrences at Other NRC Licensees (Industrial Radiographers, Medical Institutions, etc.) Designation (AO#)	Subject	*NUREG-0090* Issue
86-25	Suspension of License for Servicing Teletherapy and Radiography Units	Vol. 9, No. 4 July 1987
86-26	Diagnostic Medical Misadministration	"
86-27	Diagnostic Medical Misadministration	"
86-28	Immediately Effective Order Modifying License and Order to Show Cause Issued to an Industrial Radiography Company	"
87-2	Diagnostic Medical Misadministration	Vol. 10, No. 1 October 1987
87-3	Diagnostic Medical Misadministration	"
87-4	Diagnostic Medical Misadministration	"
87-5	Significant Breakdown in Management Oversight and Control of Radiation Safety Program at Two of a Licensee's Irradiator Facilities	"
87-6	Diagnostic Medical Misadministration	"
86-11	Therapeutic Medical Misadministration	"
87-7	Significant Breakdown in Management Oversight and Control of Radiation Safety Program at an Industrial Radiography Licensee	"
87-8	Significant Breakdown of Management Controls for Radiographic Operations	"

Occurrences at Agreement State Licensees Designation (AO#)	Subject	Issue
AS86-5	Uncontrolled Release of Krypton-85 to an Unrestricted Area	Vol. 9, No. 2 January 1987
AS86-6	Contaminated Radiopharmaceutical Used in Diagnostic Administrations	"
AS86-7	Therapeutic Medical Misadministration	Vol. 9, No. 3 April 1987
AS87-1	Breakdown in Management and Procedural Controls at an Industrial Radiography Licensee	Vol. 10, No. 1 October 1987
AS87-2	Breakdown in Management and Procedural Controls at an Industrial Radiography Licensee	"

discovered that a bone scan using technetium-99m was the desired treatment plan. The licensee calculated that the patient received a whole body and thyroid dose of about 0.47 and 400 rads, respectively. The physician-user evaluated the exposure and concluded that the irradiation posed a small, but still significant, risk of reduction in thyroid function.

Significant Breakdown in Management Oversight and Control of Radiation Safety Program at an Industrial Radiography Licensee. On April 1, 1987, the NRC issued a Demand for Information and Notice of Violation and Proposed Imposition of Civil Penalties to Grede Foundries, Inc., Milwaukee, Wis. This action was taken after an October 1986 inspection showed a significant breakdown in the licensee's oversight and control of its radiation safety program. The inspection showed not only that the Radiation Safety Officer was unfamiliar with NRC requirements for radiographic training, but also that an unqualified/untrained radiographer made 43 radiographic exposures on August 6, 7, and 8, 1986, in violation of NRC requirements and contrary to the conditions of Grede's license. In addition, the individual made the exposures with the knowledge of an authorized radiographer, who in turn entered the information into a log and signed off on it as though he had made the exposures himself.

Significant Breakdown of Management Controls for Radiographic Operations. On April 10, 1987, the NRC issued an Order Temporarily Suspending License (Effective Immediately) and Order to Show Cause why the license to A-1 Inspection, Incorporated, of Evanston, Wyo., should not be revoked. The Order was based on NRC inspections which identified two instances in which the licensee had permitted unauthorized individuals to conduct radiography. In one instance, the licensee stated to an NRC inspector that he had not employed such individuals to conduct radiography while later he admitted to an investigator that he had. These actions indicated a disregard for requirements and lack of reasonable assurance that the licensee could be trusted in the future.

Agreement State Licensees

Uncontrolled Release of Krypton-85 to an Unrestricted Area. On May 8, 1985, during routine operation of a Trio-Tech "Tracer-Flow" system at Micro-Rel Division, Medtronic, Incorporated, of Tempe, Ariz., a malfunction occurred which caused approximately 11.2 curies of radioactive krypton-85 to be vented into the atmosphere.

Contaminated Radiopharmaceutical Used in Diagnostic Administration. On May 9, 1985, a breakthrough of molybdenum-99 (a radioactive contaminant) occurred in a molybdenum-99/technetium-99m generator at Scripps Memorial Hospital, Encinitas, Cal. The breakthrough went unrecognized, and the contaminated technetium-99m radiopharmaceutical was administered to four patients scheduled for diagnostic medical tests. Therefore, these patients received exposures higher than necessary.

Therapeutic Medical Misadministration. On September 5, 1986, the Iowa Radiological Health Section, Bureau of Environmental Health, was notified of a therapeutic medical misadministration received by a patient at the University of Iowa Hospitals and Clinics, Iowa City, Iowa. The patient's bronchial tumor was being treated by an iridium-192 source placed in the bronchus. While sedated and asleep, the patient apparently pulled the tube containing the source out of the bronchus, and the tube came to rest on his chest. The patient received an estimated 1,500 rads to the chest in an area 3.4 cm long and 2 mm wide.

Breakdown in Management and Procedural Controls at an Industrial Radiography Licensee. On February 17, 1987, the Arizona Radiation Regulatory Agency issued an order to U.S. Testing Company, Unitech Services Group, San Leandro, Cal., to cease all radiographic operations within the state of Arizona. The order was based on the findings of inspections performed on February 6 and 7, 1987, to investigate the circumstances associated with two of the licensee's employees (a radiographer and an assistant radiographer) receiving radiation exposures in excess of regulatory limits while performing radiographic operations at the Navajo Generating Station, Page, Ariz. The licensee had not properly trained the radiographers.

Breakdown in Management and Procedural Controls at an Industrial Radiography Licensee. On February 27, 1987, an Emergency Order suspending all radiographic operations was issued by an inspector for the California Department of Industrial Relations to Continental Testing and Inspection (CTI), Signal Hill, Cal. During a routine compliance inspection of CTI's licensed radiographic operations, it was determined that individuals acting as radiographers may have lacked the required training and experience, since substantiating records were not available for inspection.

PERFORMANCE INDICATORS

In October 1986, the staff presented SECY-86-317, Performance Indicators, for Commission approval. The program proposed seven performance indicators for monitoring on a quarterly basis. (For background on the performance indicators program, see the 1986 NRC Annual Report, pp. 140 and 141.) In December 1986, the Commission approved the program with some changes, including the removal of the enforcement action index from the set of indicators. Subsequently, staff added a new indicator, collective radiation exposure, to the program.

Definitions of the seven performance indicators currently in the program are as follows:

PERFORMANCE INDICATORS

- **Automatic scrams while critical**
- **Safety system actuations**
- **Significant events**
- **Safety system failures**
- **Forced outage rate**
- **Equipment forced outage per 1000 critical hours**
- **Collective radiation exposure**

(1) *Automatic Scrams While Critical:* This is identical to the indicator, "unplanned automatic scrams while critical," used by the Institute of Nuclear Power Operations (INPO). In addition, the number of automatic scrams from above 15 percent power per 1,000 critical hours and the number of automatic scrams while critical below 15 percent power are monitored.

(2) *Safety System Actuations:* This is identical to the indicator, "unplanned safety system actuations," used by INPO, and includes actuations of emergency core cooling system (actual and inadvertent) and emergency a.c. power system (actual).

(3) *Significant Events:* These events are identified by the detailed screening of operating experience by NRR and AEOD, and include degradation of important safety equipment; unexpected plant response to a transient or a major transient; discovery of a major condition not considered in the plant safety analysis; or degradation of fuel integrity, primary coolant pressure boundary or important associated structures.

(4) *Safety System Failures:* These include any event or condition that alone could prevent the fulfillment of the safety function of structures or systems. Twenty-four systems or subsystems are monitored for this indicator.

(5) *Forced Outage Rate:* This indicator is identical to the one used by INPO and the NRC Grey Book (NUREG-0020) and refers to the number of forced outage hours divided by the sum of forced outage hours and service hours.

(6) *Equipment Forced Outages Per 1,000 Critical Hours:* This is the inverse of the mean time between forced outages caused by equipment failures. The mean time is equal to the number of hours the reactor is critical in a period divided by the number of forced outages caused by equipment failures in that period.

(7) *Collective Radiation Exposure:* This is identical to the one used by INPO. It is the total dose at the station. The station total is divided by the number of units at the site contributing to exposure to obtain unit values.

Quarterly data for all indicators except collective radiation exposure are collected from NRC sources, primarily from the immediate notifications (10 CFR 50.72 reports), licensee event reports (10 CFR 50.73 reports), and monthly operating data (NUREG-0020). The collective radiation exposure data are not available to the NRC on a quarterly basis. A coordination plan with INPO is being developed that will enable NRC to obtain exposure data along with data for other common indicators from INPO on a quarterly basis.

The performance indicator data in the form of tables and charts are presented in quarterly reports to the senior NRC management and Commission. There are three types of charts for each performance indicator on a plant-specific basis: (1) a bar chart showing the number of standard deviations by which the moving average for the latest two quarters varies from the plant's moving average for the previous four quarters; (2) a bar chart showing the number of standard deviations by which the plant's moving average for the latest four quarters varies from the industry mean; (3) detailed bar charts of the quarterly data, including operating history and industry mean values.

There are several new developments currently under way for improving the performance indicator program, including the refinement of data presentation and display methods and alternate methods for the statistical treatment of data. Other activities include development of risk-based and programmatic indicators, such as indicators of safety system unavailability and maintenance. Identification of the causes of events which cut across many programmatic areas are already developed and are being implemented.

The performance indicator program provides an objective view of operational performance and enhances NRC's ability to more promptly recognize poor and/or declining safety performance of operating plants. However, it is only a tool and is used in conjunction with other tools, such as the results of routine and special inspections and SALP, for providing input to NRC management decisions regarding the need to adjust plant-specific regulatory programs.

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 includes the investigation of significant operational events involving 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 the lessons of the events to the NRC, the industry, and the public.

With its focus on the causes of operating events and the identification of associated corrective actions, the IIP process contributes to nuclear safety by providing for a complete technical and regulatory understanding of significant events. The IIP generates two investigatory responses based on the safety significance of the operational events. Both are provided by an NRC team put together to determine the circumstances and causes of an operational event. For an event of potentially major significance, an Incident Investigation Team (IIT) is established by the EDO. The investigation of less significant operational events is conducted by Augmented Inspection Teams (AITs), which consist of regional-directed teams complemented by headquarters personnel and, in some cases, by personnel from other Regions. Of the more than 3,000 reportable events which have occurred during fiscal year 1987, no event was judged to have a high enough level of safety significance to warrant an IIT investigation. AITs dispatched during fiscal year 1987 are shown in Table 4.

IIT Manual. AEOD developed an Incident Investigation Manual to provide procedures and guidelines for the conduct of investigative activities by the IITs. The procedures and guidelines reflect the experience gained from previous IITs and other pertinent investigations. The Manual addresses the specific points and concerns identified in the Commission Paper establishing the IIT (SECY-85-208) and includes guidance on the following activities: activating an IIT, conduct of investigation, interviews, treatment of quarantined equipment, records and documentation control, and report preparation. AEOD distributed the Manual in August 1986 to all utilities through their respective Owner's Groups for review and comment and modified it to address the industry comments. AEOD also conducted a workshop in each Region starting in January 1987 to acquaint utilities with the IIT and help senior plant management and corporate managers become better prepared should an incident at one of their facilities trigger the establishment of an IIT. The Manual has been revised to incorporate comments received during the regional workshops and will be published as a NUREG document in early fiscal year 1988.

IIT Training Program. The purpose of this program is to provide IIT candidates with comprehensive guidance and methodology for 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. AEOD also gained valuable insights from the experience of team leaders and members of the three IITs that had already conducted incident investigations. The training course consists of an intensive two-week curriculum that includes an overview of the IIT, perspectives drawn from previous IITs, IIT investigation guidelines, and analytical techniques. The second IIT training course is scheduled for completion in early fiscal year 1988.

DIAGNOSTIC EVALUATION PROGRAM

The Diagnostic Evaluation Program (DEP), established by the EDO and approved by the Commission in 1987, provides an assessment of licensee performance at selected reactor facilities. Authority for staffing, maintaining, and implementing the DEP was given to a new organizational unit within AEOD which developed initial procedures, guidelines, and methodologies for performing diagnostic evaluations. The DEP evaluates the degree of involvement of licensee management and staff in ensuring safe plant operations, the effectiveness of their actions, the need for improvement in facility safety programs, and the root causes of performance problems with an adverse effect on plant safety. The DEP supplements the licensee assessment information provided by the Systematic Assessment of Licensee Performance (SALP) Program, the Performance Indicator (PI) Program, and the routine and special inspections performed by the NRC Headquarters and Regional Offices, and it helps NRC senior management make more informed decisions concerning the need for NRC and licensee actions to improve plant safety performance.

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 experienced technical staff members from the AEOD, experienced NRC technical staff members from other headquarters offices, experienced regional and resident inspectors, and contractors, if appropriate. Team members are selected in all cases so as 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 review in a number of functional areas important to plant safety such as maintenance, surveillance testing, corrective actions, safety evaluation, management involvement, conduct of operations, safeguards, plant modifications and design changes, radiation protection quality assurance, and architect engineer/contractor control.

Table 4. Augmented Inspection Teams Dispatched in FY 1987.

Event Date	*Plant*	*AIT Report*	*Report Date*	*Description*
12/03/86	Hatch 1&2	50-321/86041 50-366/86041	01/06/87	Leak from spent fuel pool to environment during refueling.
12/09/86	Surry 2	50-280/86042 50-281/86042	02/10/87	MFW pump suction line rupture.
03/11/87	Turkey Point 4	50-251/87016	05/15/87	Boric acid buildup on RV head.
04/11/87	Diablo Canyon 2	50-323/87018	06/18/87	Loss of decay heat removal.
06/17/87	Perry 1	50-440/87014	07/30/87	MSIV solenoids not powered independently.
07/02/87 07/28/87	McGuire 1&2	50-369/87022 50-370/87022	08/31/87	Failure of Westinghouse reactor trip breaker.
07/14/87	Palisades	50-255/87019	10/08/87	Loss of off-site power.
07/15/87	North Anna 1	50-338/87024 50-339/87024	08/28/87	Steam generator tube rupture.
08/03/87	Arkansas 1	In Draft		Containment high temperatures.
08/07/87	Dresden 3			Feedwater oscillations result in reactor trip.
09/06/87	Davis Besse	50-346/87025	10/01/87	Scram with multiple equipment failures.
09/11/87	Oyster Creek	50-219/87029	09/28/87	Violation of safety limit.
10/03/87	Fort St. Vrain			Loss of cooling due to fire.

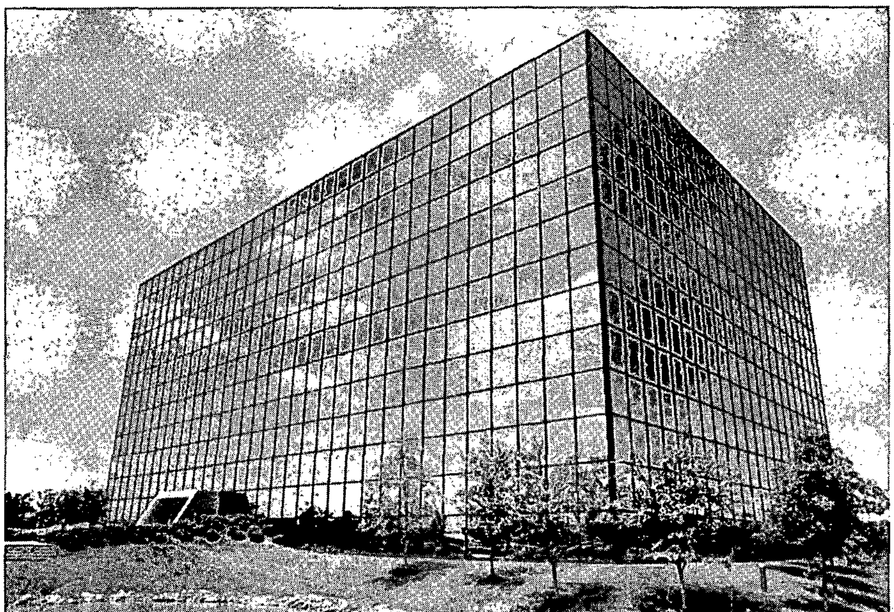
DET at Dresden Units 2 and 3. NRC senior managers determined in June 1987 that the first DET should be established for the Dresden (Ill.) nuclear power plant to provide necessary information regarding its overall regulatory performance. The perception was that Dresden's performance was fluctuating between average and below average. In addition, previous improvement programs did not appear to have achieved long-term results. The team was instructed to focus its attention on personnel attitudes toward safety, management involvement in station operations, and the effect of recent improvement programs on station performance and personnel attitudes.

The Dresden DET spent two weeks on-site during August 1987 to fully evaluate Dresden's performance and found major weaknesses still existed in maintenance, inservice testing, communications, and operator training. The team concluded that improvement programs with additional management involvement and resources were warranted. The team also found that excessive operator overtime was occurring on a regular basis without corrective actions by management. (This was identified as an immediate safety concern and prompt corrective actions were taken.) The team found too that Dresden's performance was improving slowly and that plant personnel had a positive attitude toward plant safety. The team could not conclude that the performance initiatives would be long-lasting, however, because of Dresden's previous cyclical performance history, and the fact that the licensee improvement program failed to address several important areas.

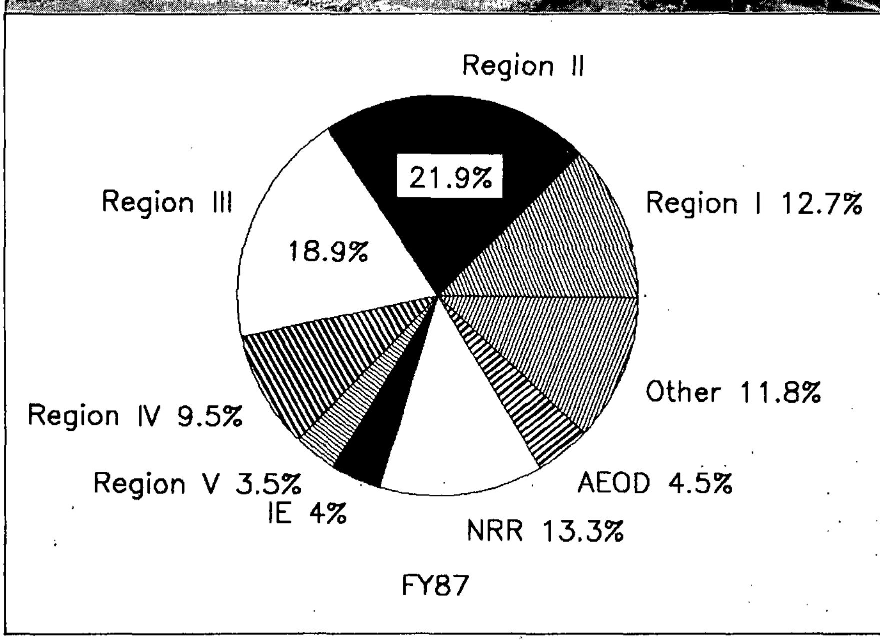
The team concluded that greater management attention, involvement, commitment, and resources would be required to sustain the improving trend necessary to meet the Commonwealth Edison Company (CECo) corporate objective of achieving an overall SALP rating at Dresden of 1.5 by 1990. Following a presentation of the team's findings, CECo management agreed that resource limitations had been a factor in Dresden's past performance but gave their firm commitment to provide the necessary resources to strengthen and expand Dresden's improvement programs.

TECHNICAL TRAINING PROGRAMS

The NRC Technical Training Center (TTC), located in Chattanooga, Tenn., provides training in reactor technology and specialized technical areas to resident inspectors, headquarters and region-based inspectors, operator license examiners, Operations Center duty officers, other NRC tech-



The NRC Technical Training Center (TNC), located in the Osborne Office Center in Chattanooga, Tenn. (top photo), offers training in reactor technology and associated technical areas for NRC personnel and those of other Federal, State and foreign government agencies. TTC student distribution for FY 1987, by percentage, is shown in the graph.



nical staff, and other Federal, State, and foreign government employees. Reactor technology courses are normally conducted by the TTC staff while specialized technical training courses are normally conducted through commercial contracts or inter-agency agreements.

Separate, parallel curricula exist for each of the four major U.S. reactor vendor designs—General Electric (GE), Westinghouse, Babcock and Wilcox (B&W), and Combustion Engineering (CE). Each of these curricula involves both classroom and full scope reactor simulator training courses and includes initial and refresher training and training for managers. Concepts covered within each of these curricula include system design, design problems, system interrelationships, operational problems, technical specifications, transient and accident analysis, and technical issues. Reactor technology training is also provided in the high temperature gas reactor design. Essentially generic reactor technology training is provided on a more basic level in the form of reactor concepts courses for NRC non-technical staff and news media seminars in various geographical locations in support of the public affairs function. The specialized technical curriculum consists of courses in several broad areas which include generic, engineering support, health physics, and safeguards.

In fiscal year 1987, the TTC presented or coordinated attendance at 115 different courses for a total of 1,162 students, during 162 course weeks of effort. Reactor technology training represented 65 percent of the course week total and 74 percent of the student total.

Great progress was made in the development of TTC reactor simulator capabilities. During the fiscal year, a contract was successfully negotiated with the Westinghouse Electric Company which resulted in the relocation of the Westinghouse SNUPPS simulator from Zion, Ill., to the TTC facility in Chattanooga, Tenn.

The NRC is now leasing two full scope reactor simulators (General Electric and Westinghouse designs) with options to buy. Boiling water reactor (BWR) simulator time, in the amount of 2,000 hours per year, and Westinghouse PWR simulator time, in the amount of 4,000 hours per year, are available to the NRC staff. During 1987, the TTC made a feasibility study of the potential relocation of another full scope reactor simulator (B&W design) to the TTC facility and initiated a competitive procurement action. B&W simulator training is presently conducted using the TVA Bellefonte simulator in Scottsboro, Ala., while Combustion Engineering simulator training uses the Combustion Engineering Calvert Cliffs simulator in Windsor, Conn.

In addition to the Westinghouse SNUPPS simulator acquisition, other training aids were obtained or enhanced during the year. An engineering model of the Yellow Creek plant (a cancelled Tennessee Valley Authority (TVA) nuclear project) was obtained. This scale model shows details of the Combustion Engineering plant design in a fashion similar to the Hartsville model (cancelled TVA nuclear project) of the BWR design. Both engineering models are currently being leased from TVA for a nominal amount. A B&W design nuclear steam supply system model was obtained from Consumers Power Company (cancelled Midland nuclear project). A Westinghouse design dummy fuel assembly was obtained from Public Service of Indiana (cancelled Marble Hill nuclear project). The existing BWR jet pump and hydraulic control unit (obtained earlier through the GE simulator contract) were mounted in the correct orientation in the major BWR classroom. Three-dimensional color plots of several major systems and buildings were obtained and mounted under plastic for use as training aids (River Bend BWR/6 design and Three Mile Island PWR B&W design). Some modifications were made to the BWR simulator including the addition of a screen plotter for the Emergency Response Information System (GE Safety Parameter Display System) and the creation of TTC custom designed display control system dynamic displays of the BWR Electro Hydraulic Control System, Recirculation Flow Control System, and Power/Flow map. Additional modifications involving containment hardware displays were in progress as the fiscal year ended.

The TTC revised the Power Plant Engineering Course and associated manual to address lessons learned during the initial course presentation. The TTC also coordinated the complete overhaul of the Fundamentals of Inspection Course manual. This represented an effort which spanned all of the Regions and several of the program offices. The results of this major project will be consistent, standardized courses in support of inspection personnel in both the Regions and program offices. In addition, a number of reactor technology curriculum modifications were made during the year.

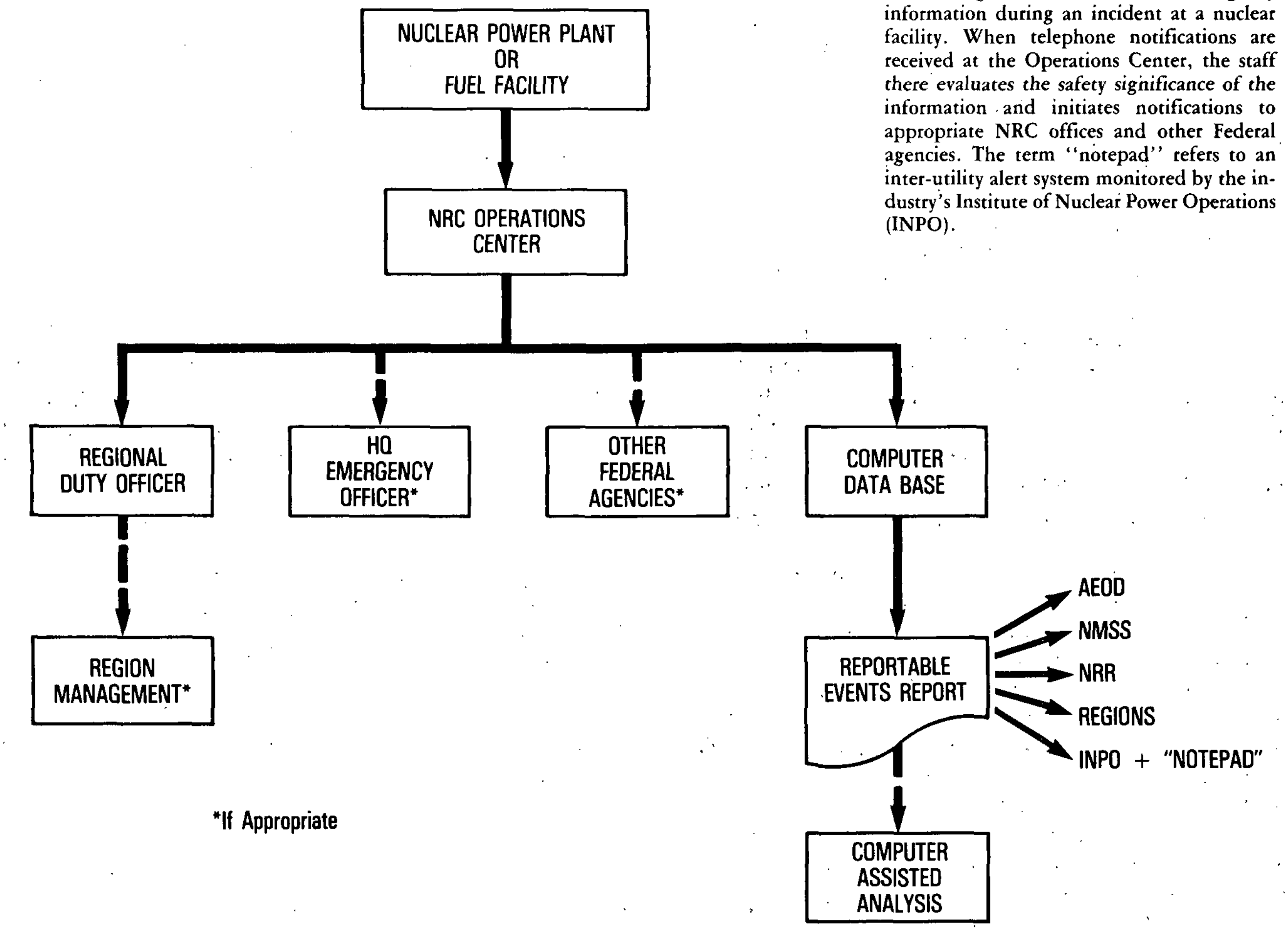
INCIDENT RESPONSE

Events Analysis. The Nuclear Regulatory Commission 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 plants. During the first nine months of 1987, there were 190 incidents (six alerts and 184 unusual events) reported to the Operations Center under the NRC emergency classification system.

The staff at the Operations Center evaluates telephone notifications 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 region 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 monitor the event to ensure that appropriate actions are being taken to protect the health and safety of the public. The NRC recognizes that the agency's role is secondary to those of the licensee and off-site organizations, whose immediate responses are defined ahead of time in their own emergency planning.

Each of the 4,000 events reported each year to the Operations Center by a licensee or Regional Office is evaluated to determine whether there are any generic implications for other 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 system interaction and raises questions as to plant safety, an augmented inspection team or an incident investigation team may be formed. Events that may be significant from a generic standpoint receive additional in-depth evaluation and, if appropriate, the NRC issues an Information Notice or Bulletin to potentially affected licensees and construction permit holders.

Operations Center. Considerable resources are needed to maintain a prompt incident response capability, which 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 1987, the Operations Center was involved in several real events which, while not requiring complete activation, necessitated the use of the Operations Center's capabilities. The Center was staffed to monitor the steam generator tube rupture event at North Anna nuclear plant in Virginia and to follow the loss of off-site power event at Calvert Cliffs Nuclear Power Plant in Maryland. The tele-



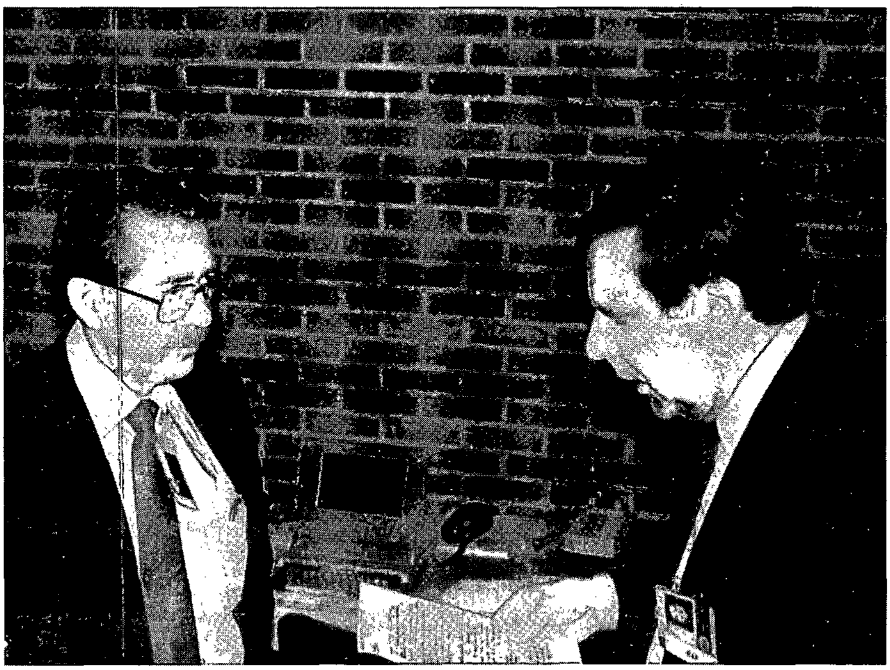
This diagram shows the flow of emergency information during an incident at a nuclear facility. When telephone notifications are received at the Operations Center, the staff there evaluates the safety significance of the information and initiates notifications to appropriate NRC offices and other Federal agencies. The term "notepad" refers to an inter-utility alert system monitored by the industry's Institute of Nuclear Power Operations (INPO).

communications capability of the Operations Center was used by NRC management in teleconference discussions of a number of events that were significant but did not warrant staffing of the Operations Center.

During 1987, 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 included a full-scale exercise at the Hope Creek (N.J.) nuclear plant, the Federal Field Exercise at the Zion (Ill.) facility, and two computer-generated reactor accident simulations. An exercise scenario involving a radioactive materials accident was also run. All of these exercises were supported through the Operations Center. Throughout the year, tours of the Operations Center were frequently provided for representatives of other NRC offices, industry, State and local governments, and foreign countries. The tours included detailed descriptions of the NRC response role and typical activities within the 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 work with the headquarters program. The extent of Regional Office response to an incident would be 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. Once a site team of 12 to 18 specialists led by the Regional Administrator arrives at the site (two to six hours after being dispatched) and is fully briefed, the Chairman of the NRC or his designate would consider transferring appropriate responsibility and authority to the Regional Administrator.

Each Region has its own supplement to the NRC Incident Response Plan providing specific implementation details. During 1987, Headquarters and the Regions worked together to develop standardized portions of the regional supplements and upgrade the agency-wide response capability. Regional response capabilities are assessed annually, and the Regions participate in several exercises each year.



More than 300 NRC personnel took part in the second Federal Field Exercise, conducted at the Zion (Ill.) nuclear power plant from June 23 to 25, 1987. In all, some 1,200 persons, including 175 domestic and foreign visitors, participated in the exercise. Shown here are (left) A. Bert Davis, NRC Region III Administrator and Director of Site Operations, and Clemens J. Heltemes, Jr., Deputy Director of the NRC Office for Analysis and Evaluation of Operational Data.

Federal Response Capability: Federal Field Exercise-2 (FFE-2). Over 300 NRC personnel participated in the second FFE-2 involving the Zion (Ill.) nuclear power plant. This special exercise involved a total of 1,200 individuals, including 175 domestic and international visitors, and was conducted over a three-day period from June 23-25, 1987. In addition to the NRC, participants included representatives of CECo, Illinois and Wisconsin agencies, the two local counties, 12 other Federal agencies, and private organizations. The extensive Federal participation in the exercise derived mainly from a need to test the Federal Plan and Federal agency procedures for responding to a large radiological emergency. In general, the results indicated that the NRC's capabilities to respond to a nuclear power plant accident have improved substantially since the initial Federal Field Exercise in 1984. The improvements are largely attributed to the new NRC Operations Center, more detailed regional and headquarters procedures, and better trained staff. The FFE-2 also showed that the Federal agencies can provide significant support to State and local authorities under emergency conditions. However, there still needs to be improvement in the managing of Federal capabilities and in the way decisions are made, as well as in the coordination of response actions among State, Federal and licensee organizations.

Emergency Response Data System. Development work on the Emergency Response Data System (ERDS) for use during emergencies at commercial nuclear power plants continued during 1987. The ERDS concept provides for licensee-activated automatic transmission of pre-selected plant data from the licensee to a computer at the NRC Operations Center. The design phase of ERDS development included surveys of existing electronic data systems at operating and nearly completed nuclear power plants and determinations of hardware and software requirements at licensee facilities. Survey visits to licensees and preliminary designs were completed during 1987, and the first plant connections are expected in 1988.

Emergency Response Training. The Incident Response Branch (IRB) published a five-volume set of response training manuals (NUREG 1210) in 1987 that collectively deal with the NRC response to severe accidents, licensee and State response, and public protective actions. The training manuals, together with NRC's standard procedures for the assessment of reactor accident consequences, will form the basis of standard agency wide response training. General response training sessions based on the training manuals were offered to all response personnel prior to the dry run for the Federal Field Exercise-II and the actual exercise. Such sessions covered the roles of the NRC, other Federal agencies, the licensee, and State and local authorities, as well as general topics concerning the Operations Center.

The IRB continued its work on developing technical tools to be used for assessment of reactor accidents and their consequences. The tools include a system to review plant data, methods to project source terms based on plant conditions, and upgraded dose projection models.

Nuclear Materials Regulation

Chapter 5

The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) administers the regulation of nuclear materials, as distinct from nuclear reactor facilities (discussed in Chapters 2 and 3). NMSS conducts this regulation under three broad programs: fuel cycle and material safety, discussed in this chapter; materials and facilities safeguards, discussed in Chapter 6; and waste management activities, discussed in Chapter 7.

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; (5) production and use of reactor-produced radioisotopes ("byproduct material"); and (6) transportation of nuclear materials.

Highlights of actions taken during fiscal year 1987 include:

- Completion of nearly 130 licensing activities dealing with fuel cycle plants and facilities.
- Completion of 200 fuel facility inspections, 2,400 material licensee inspections, and over 1,500 transportation inspections.
- Completion of 12 team assessments at major fuel facilities.
- Completion of over 5,600 licensing actions on applications for new byproduct materials licenses and amendments and renewals of existing licenses. Over 5,200 of these actions were completed by the five Regional Offices; the remainder were completed at Headquarters.
- Completion of 84 design certification reviews for transportation packages.

FUEL CYCLE ACTIONS

Post-Accident Activities at Sequoyah Fuels

In January of 1986, an accident took place at the Sequoyah Fuels Corporation (SFC) facility in Oklahoma involving a massive release of uranium hexafluoride (UF-6) and resulting in one fatality and several injuries to workers. (See *1986 NRC Annual Report*, pp. 87 and 88 for background.) On October 16, 1986, the Commission voted to allow restart of the SFC nuclear fuel facility following improvements in equipment, retraining of personnel, establishment of quality assurance programs, and rewriting of procedures. The facility resumed operations on December 11, 1986. No major problems were experienced during restart and subsequent operations. Restart activities were monitored 24-hours-a-day by a third party oversight group and by the NRC. Current NRC oversight consists of regular inspections. The third party oversight team continues its monitoring eight hours a day, seven days a week.

As a result of this accident, the NRC decided to conduct operational safety assessments at each of the major fuel cycle facilities licensed to operate in the United States. These assessments were conducted using a multi-disciplinary team approach, involving NRC personnel from the responsible Region as well as headquarters personnel. Team personnel included not only those normally inspecting radiation and criticality safety and emergency preparedness programs at fuel facilities, but also experts from reactor inspection groups with fire protection and operations engineering expertise. Assessment of chemical hazard protection was included, with assistance provided by chemical safety experts from Oak Ridge National Laboratory. In addition, personnel from the Environmental Protection Agency, Occupational Safety and Health Administration, and Federal Emergency Management Agency participated in some of the assessments.

The most important safety issues among those identified as requiring increased attention on the part of licensees were in the areas of fire protection, chemical hazards identification and mitigation, management controls and quality assurance, safety-related instrumentation and maintenance, and emergency preparedness. The staff concluded that the deficiencies identified are amenable to correction through regional case-by-case follow-up, coupled with amendments to facility licenses, as indicated, when they come up for renewal.

"Orders Modifying License" were issued to all fuel cycle facility licensees who use 48- and 30-inch diameter cylinders, following notification of the NRC by the Department of Energy (DOE) of cracks observed in and around the threads of certain one-inch valves used in these cylinders. The licensees were ordered to inspect specified valves for defects and, where found, to discontinue use of the valves.

Wright-Patterson Air Force Base

An incident at the Wright-Patterson Air Force Base (WPAFB) near Dayton, Ohio, on September 18, 1986, resulted in extensive americium-241 contamination in a radioactive waste storage building. Personnel had opened a waste storage drum to identify the contents; it was later ascertained that about two curies of unencapsulated americium-241 was in the drum. Extensive cleanup of the building was carried out. Personnel changes at WPAFB have been made by the Air Force and measures have been taken by the Air Force Radioisotope Committee to upgrade the handling and regulation of licensed materials. (See discussion under "Abnormal Occurrences," in Chapter 4.)

Workers prepare to enter a building in a secured area at Wright-Patterson Air Force Base in Dayton, Ohio, to clean up americium-241 contamination from the accidental release of September 18, 1986. Air, soil, and water samples by the licensee showed no spread of contamination outside the building, and, other than for a minor initial contamination of one individual, there was no evidence of any uptake of radioactive material. The building was later disassembled and sent to a waste burial site. (See discussion in Chapter 4.)

Materials Safety Regulation Review

In May 1986, the NRC contracted with five experts to obtain their insights on the agency's current fuel cycle and materials regulatory program. The NRC's main interest in seeking this review was to obtain fresh perspectives and recommendations on ways to improve the efficiency and effectiveness of the materials regulatory program in health and safety and environmental protection.

In October 1986, the agency received the "Materials Safety Regulation Review Study Group Report," from Dr. Clifford V. Smith, Jr., the Group Chairman. The report contained 22 recommendations for improving the quality of NRC's fuel cycle and materials licensing and inspection programs. The NRC requested public comments on the recommendations at the time it published the report in the Federal Register, on December 17, 1986. The staff received 207 public comments in 49 letters and summarized the comments in a Commission paper (SECY 87-94) in April 1987. Staff analysis of the recommendations and public comments was completed in July and submitted to the Commission for approval (SECY 87-189). The Commission approved the staff's analysis and recommended action plan. At the close of the report period, efforts were continuing to assure timely scheduling and implementation of items remaining open from the action plan.

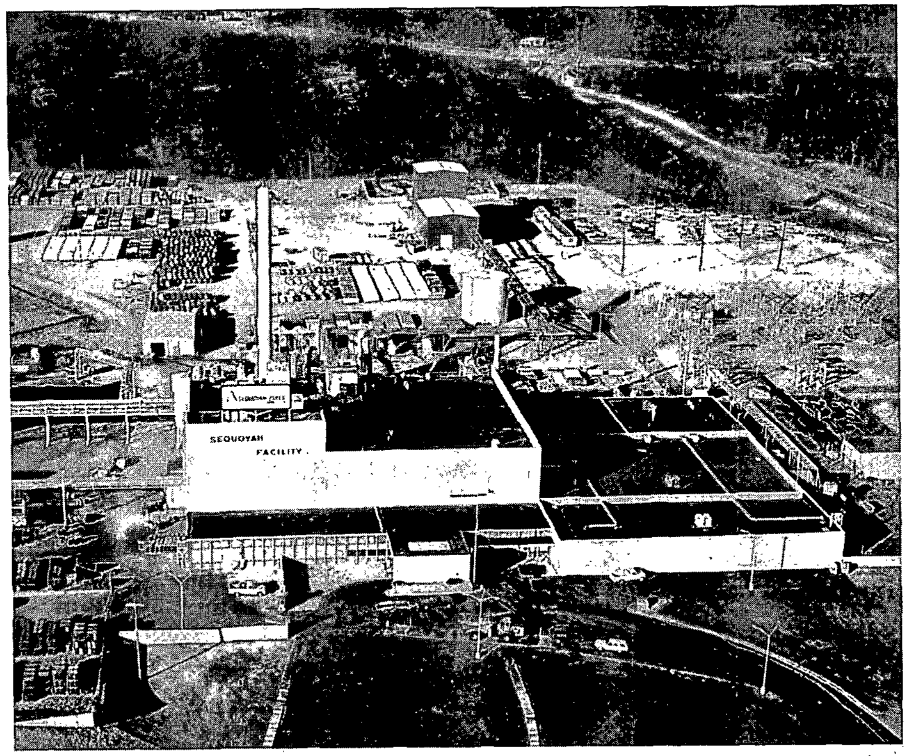
From these efforts, several regulatory improvements have been or will be achieved. Rulemaking efforts, such as the Emergency Preparedness Rule and the Part 20 rule changes, will continue; chemical and other non-radiological hazards will be addressed more completely; staff will expand the use of multi-disciplinary team assessments that evaluate licensee management controls, fire protection, radiation safety, nuclear criticality safety, emergency preparedness, and safety-related instrumentation and maintenance; and more formal staff training programs will be identified and developed.

Hearings on Fuel Cycle Facilities

Sequoyah Fuels, UF6-to-UF4 Production Plant. Kerr-McGee had applied for permission to construct and operate a UF6-to-UF4 production plant, using depleted uranium, at the Sequoyah Fuels Corporation (SFC) site prior to the accident at that UF-6 production facility in Gore, Okla. (see discussion above). On January 12-15, 1987, the hearing on this application was held, with testimony presented by SFC representatives and the intervening parties. On March 4, 1987, a decision was rendered by the presiding officer authorizing the Director of NMSS to issue a license amendment to SFC permitting operation of the UF6-to-UF4 facility. The amendment was issued March 25, 1987.

West Chicago: Kerr-McGee Rare Earths Facility. At the direction of the Licensing Board, the staff issued a Draft Supplement to the Final Environment Statement on the West Chicago, Ill., facility. The deadline for comments on the draft was October 1, 1987, by which time 12 comment packages had been received. The comments will be considered and resolved and the Final Supplement should be issued by summer 1988. The board may resume a portion of the hearing based on the Draft Supplement. At issue are decommissioning and the on-site stabilization of radioactive wastes. (See the 1986 NRC Annual Report, p. 88, for background on this and the following case.)

Kress Creek. At issue in this matter is the cleanup of contaminated areas in and around Kress Creek, near the Kerr-



On December 11, 1986, the Sequoyah Fuels Corporation's UF-6 to UF-4 facility at Gore, Okla., resumed operation, following a January 1986 release of uranium hexafluoride (UF-6) that killed a worker and injured several others. NRC observers were on hand to monitor the restart of the plant; regular inspection of its operation continues.

McGee West Chicago, Ill., facility. The staff filed a motion to terminate the proceeding before an Appeal Board, which the board denied. The denial was based on the classification of the contaminated material as source material (as distinct from byproduct material). Staff then petitioned for Commission review of the Appeals Board decision. Characterization of the material as byproduct material or as source material was pending Commission determination at the close of the report period. Once a decision is made, the proceeding will either be terminated, with jurisdiction going to the State of Illinois, or will proceed as scheduled on the original appeal.

Sequoyah Fuels Comprehensive Waste Disposal Plan. Sequoyah Fuels submitted a new waste disposal plan which eliminated an on-site burial request. SFC filed a motion to terminate the proceeding on the basis that they are no longer seeking on-site burial. Early in fiscal year 1988, the hearing was terminated.

Incinerator Licensing

Babcock & Wilcox. Following staff reviews in 1986 and an informal hearing in October 1986, a license amendment was granted authorizing operation of the high-force compactor portion of Babcock and Wilcox's Volume Reduction Services Facility in Parks Township, Pa. Difficulties in obtaining State permits have delayed start-up of the compactor. The Administrative Judge on the case ordered the staff not to authorize operation of the incinerator portion of the facility until several conditions related to construction, operation, and environmental monitoring have been met. Resolution of these matters was still pending at the close of the report period.

Battelle Columbus. The staff issued a license amendment to Battelle Columbus Laboratories in March 1987, authorizing operation of an incinerator for a five-year demonstration of low-level radioactive waste volume reduction. Construction of the incinerator facility is unlikely, however, since Battelle has decided to discontinue nuclear activities at its Columbus and West Jefferson, Ohio, locations.

Interim Spent Fuel Storage

The Nuclear Waste Policy Act of 1982 (NWPA) established the requirement that utilities take primary responsibility for interim storage of their spent fuel until a Federal repository or monitored retrievable storage (MRS) installation is available; such a facility is, by current estimates, a decade or more away. Although some contingency storage is available from DOE, Federal interim storage is intended only as a last resort under NWPA criteria and NRC implementing regulations (10 CFR Part 53). Thus, utilities are continuing to develop plans for providing additional storage capacity as they approach 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, rod consolidation is being considered by some utilities as a means of increasing pool capacity. On-site dry storage of aged spent fuel in modular units is also being closely studied as a means of meeting storage needs.

Following the 1986 issuance of 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 and Light Company (CP&L) for its H. B. Robinson (S.C.) nuclear power plant, the NRC staff continued to monitor developments closely as the facilities were constructed and storage cask and canisters were fabricated. Design changes resulted in additional technical reviews and license amendments.

In September 1987, the NRC staff issued a letter of approval with related safety evaluation of the topical report submitted by Westinghouse for its Model MC-10 dry spent fuel storage cask design. The Westinghouse MC-10 cask is a forged steel cylinder approximately 16 feet in length and eight feet in diameter. The steel wall is about 10 inches thick and the solid neutron shield is three inches thick. The cask is designed to hold 24 PWR fuel assemblies, decayed at least ten years. When loaded the cask weighs about 113 tons.

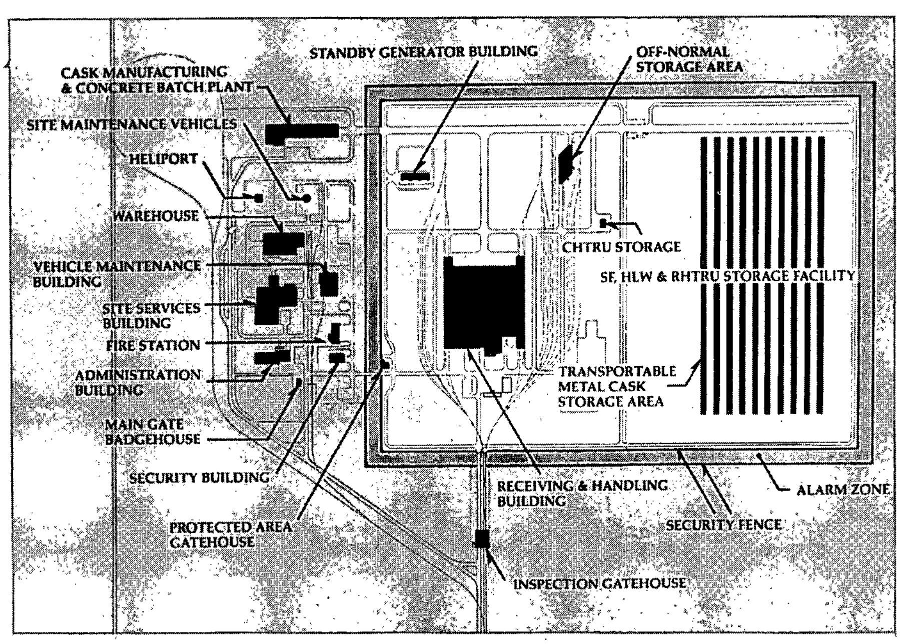
The NRC staff has under review three topical reports for dry storage casks of varying designs submitted by Nuclear Assurance Corporation (NAC), Transnuclear, and Combustion Engineering, and one topical report for a modular dry storage vault submitted by FW Energy Application, Inc. If found acceptable, these topical reports may be referenced by a utility in a license application or in an amendment to an existing Part 72 license to expedite the review of a proposed dry storage system, or proposed modification to an existing system.

In order to further streamline the licensing process for use of spent fuel dry storage casks at reactor sites, the NRC staff has initiated rulemaking through 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 for a site-independent rule for at-reactor-site dry cask storage have been prepared to provide for storage cask certification and use by reactor operators with a general license. The proposed rule is expected in 1988, and a final rule in 1989.

Monitored Retrievable Storage

The Department of Energy (DOE) had planned to submit a proposal for MRS to the Congress in February 1986, as directed by the Nuclear Waste Policy Act. As conceived by DOE, the MRS would be a large, hot-cell complex in which spent fuel and, perhaps, solidified high-level waste would be packaged for disposal and then temporarily stored in large concrete casks. In this mode, the MRS would be an integral component of the DOE's high-level waste disposal system.

On March 31, 1987, DOE submitted its proposal for monitored retrievable storage to the Congress, following the Supreme Court's denial of the petition of the State of Tennessee to grant certiorari review. The State of Tennessee had filed a motion in the U.S. District Court for the Middle District of Tennessee requesting an injunction to prevent DOE from submitting its proposal. The court had granted the injunction and DOE had successfully appealed the decision.



One architect's concept for a planned Monitored Retrievable Storage (MRS) facility is shown. Proposals by the Department of Energy for MRS have been submitted to the Congress, where action was pending at the end of the report period.

The newly submitted proposal is virtually identical to the proposal reviewed by the Commission staff in 1986, except for slight modifications make it consistent with schedule and cost updates presented in an amended DOE Mission Plan. None of the changes affected the Commission's comments on the original proposal.

At the close of the report period, the DOE was awaiting Congressional action on its proposal.

Licensing of Uranium Enrichment Facilities

Although no applications for a uranium enrichment facility have been received to date, the Commission staff is continuing work on the development of a licensing perspective. Several meetings have been held with representatives of URENCO, Ltd., a group of companies in the Federal Republic of Germany, the Netherlands, and the United Kingdom, established in 1971 to pursue uranium enrichment. URENCO has three operating centrifuge plants, one in each of the participating countries. If URENCO can enter into a partnership agreement with U.S. firms, an application for a plant almost identical to their newest plant in Gronau, in the Federal Republic of Germany, may be forthcoming.

Decommissioning and Decontamination

West Valley Demonstration Project. Throughout 1987, the Commission continued its safety oversight role for the West Valley (N.Y.) Demonstration Project (WVDP). The primary purpose of the project is to demonstrate solidification and preparation of high-level radioactive waste for disposal in a Federal repository. The current schedule for the WVDP calls for "supernatant treatment" to begin in 1988, and a verification process to start in early 1990 and continue into 1991.

The Safety Analysis Report (SAR) for WVDP is being prepared by the Department of Energy in separate sections, keyed to the development of discrete systems within the project. The NRC staff reviews and issues a Safety Evaluation Report (SER) on each SAR section, reporting the staff's conclusions about the public health and safety implications of DOE's plans. During 1987, the staff issued SERs on (1) the principal design criteria and management organization, (2) the cement solidification system, and (3) the supernatant treatment system. The staff also received an SAR on the verification process from DOE in 1987. On the basis of studies conducted thus far, it is the staff's conclusion that the DOE's proposed processes will not adversely affect the public's health or safety.

Remedial Actions at Contaminated Sites. This project began in 1976 with a request from the Government Accounting Office for NRC assurance that no radiation safety problems existed at fuel cycle sites previously operated under license by the Atomic Energy Commission. The final site examined under the remedial action program was released in March 1987, and the final report covering these sites was scheduled to be published in late 1987.

MATERIALS LICENSING AND INSPECTION

The NRC currently administers approximately 8,200 licenses for the possession and use of nuclear materials in applications other than the generation of electricity or operation of a research reactor. The program is designed to ensure that activities involving such uses of radionuclides do not endanger the public health and safety. With the exception of certain distribution licenses and sealed source and device design reviews, all materials licenses are now administered by the NRC Regional Offices.

The NRC completed over 5,600 licensing actions during fiscal year 1987. Of these, 700 were on new license applications, 3,800 concerned amendments, 1,000 were license renewals, and 100 were sealed source and device reviews. There are about 15,000 additional licenses administered by the 29 Agreement States. NRC's Regional Offices completed nearly 2,400 inspections of materials facilities during fiscal year 1987. Table 1 shows the regional distribution of NMSS inspections and the number of violations identified. Table 2 shows the number of byproduct material licenses by type of use.

Oversight Program

Headquarters and regional staff continued to refine the National Program Review regimen, which was developed to assure the technical adequacy, timeliness, and consistency of the decentralized licensing program. This oversight process includes day-to-day information exchanges between headquarters and regional staff, monthly conference calls, annual management seminars, reviewer workshops, and visits to each Region.

Under the 1987 reorganization of the NRC (see Chapter 1), NMSS became responsible for overseeing materials and fuel cycle inspection activities, in addition to the decentralized licensing program. To meet this new charge, work began during the period to integrate the Office's regional oversight activities with those of the former Office of Inspection and Enforcement.

Table 1. Distribution of Byproduct Material Licenses by Type of Use

Academic	
Broad Academic	85
Medical	
Medical Institutions and Private Practice	2,104
Eye Applicators	53
Mobile Nuclear Medicine	21
Teletherapy	268
Veterinary	8
In Vitro Testing Laboratories	91
Nuclear Pharmacies	41
Medical Product Distribution	22
	2,608
Commercial/Industrial	
Well-Logging	140
Field Studies	2
Gauges and Measuring Systems	2,989
Commercial Manufacturing and Distribution	182
Nuclear Laundries	4
Leak Testing and Instrument Calibration	160
Waste Disposal	16
General License Distribution	77
Exempt Distribution	159
Radiography	285
Irradiators	240
Research and Development	690
Civil Defense	36
	4,980

*Counts active NRC licenses as of October 1987. Excludes source and special nuclear materials licenses.

Consolidation of Military Licenses

For several years, the NRC staff has been considering the possibility of consolidating licenses covering military activities which involve radioactive materials. The United States Air Force and the United States Navy expressed an interest in obtaining consolidated licenses for their activities. The NRC issued a Master Materials License to the Air Force's radioisotope program in June 1985. The consolidation was completed in October 1986, and Region IV (Dallas) was given the lead responsibility for the Air Force license. The NRC issued a Master Materials License to the Navy's radioisotope program in March 1987. The consolidation was completed in December 1987 and Region II (Atlanta) was given the lead responsibility for the Navy license. Consolidation of the Air Force and Navy licenses replaced approximately 275 individual NRC licenses. It is anticipated that substantial administrative resources and paperwork will be saved by this consolidation.

Medical and Academic Uses

An estimated 10 million clinical procedures are performed each year using radioactive materials for the diagnosis or treatment of patients. Many of these procedures involve NRC-licensed materials and are conducted in hospitals or in physicians' offices. NRC-licensed materials are also used in universities, colleges, and other academic institutions in certain laboratory courses and in research programs.

Medical User's Qualifications. In May 1985, the NRC staff held a public meeting of the Advisory Committee on the Medical Uses of Isotopes (ACMUI, see Appendix 2) to consider NRC's training and experience criteria for the qualification of physicians using radiopharmaceuticals for diagnostic imaging procedures. In response to concerns over misadministrations of radiation, the NRC staff will request public comment on the appropriate training and experience criteria for all individuals who participate in the medical use of radioactive byproduct material.

Table 2. NMSS Inspections in FY 1987

	Inspections	*No. of Violations*
Materials Facilities		
Region I	837	634
Region II	365	315
Region III	881	734
Region IV	177	226
Region V	103	144
	2,363	2,053
Fuel Facilities		
Region I	15	18
Region II	105	49
Region III	13	2
Region IV	33	18
Region V	32	5
	198	92
Transportation		
Region I	302	59
Region II	357	58
Region III	585	75
Region IV	147	39
Region V	130	8
	1,521	239

Part 35 Revision. The NMSS staff led the Task Force that prepared a revision of 10 CFR Part 35, "Medical Use of Byproduct Material," also noted in last year's annual report (p. 93). The purpose for the revision was primarily to consolidate requirements that were spread throughout a variety of regulatory instruments, including regulations, regulatory guides, and license conditions. Under the revision, licensees can make minor changes in their radiation safety procedures that are not potentially important to safety without NRC review and approval. However, these changes will require approval by the licensee's Radiation Safety Officer, and, if at a hospital, by its Radiation Safety Committee. The revision became effective April 1, 1987. The headquarters staff has conducted training sessions for regional staff, and participated in scientific and clinical meetings sponsored by professional organizations, in order to answer industry questions about the revision.

Quality Assurance in Radiation Therapy. In response to the incidence of misadministration in radiation therapy, the NRC is preparing rules to require radiation therapy licensees to implement quality assurance programs with certain specified features. The staff will work with other government agencies and professional organizations in developing this rule.

Advisory Committee on the Medical Uses of Isotopes

The Advisory Committee on the Medical Uses of Isotopes (ACMUI) was established in July 1958. The ACMUI, composed of qualified physicians and scientists, considers technical medical questions referred to it by the NRC staff and renders expert opinions regarding the medical use of byproduct material. The ACMUI also advises the NRC staff, as required, on matters of policy. Members of the committee are listed in Appendix 2.

Industrial Uses

Reactor-produced radionuclides are used extensively throughout the United States in both civilian and military industrial applications in such areas as industrial radiography, manufacture of gauging devices, gas chromatography, and well-logging; they are also used by the general public in various consumer products such as household and industrial smoke detectors. The NRC's evaluation, licensing, and inspection program is designed to assure that these activities pose no undue risk to the public health and safety.

Industrial Radiography. This is a form of non-destructive testing carried out by commercial firms licensed by the NRC to use radioactive byproduct material in instruments which examine the structure of materials by means of radiation. Testing may be done at field sites or temporary job sites using portable devices containing radiation sources of up to 200 curies of iridium-192 or up to 100 curies of cobalt-60. At the close of the report period, the NRC had issued a total of 352 radiography licenses; of these, 97 were for operations in fixed locations and 255 for use on temporary job sites.

During 1987, the NRC expended considerable effort on a rulemaking dealing with radiographic equipment. In doing so, the staff provided performance criteria for improved reliability and safety in the design of the equipment; required the use of audible-alarming dosimeters; and upgraded incident reporting requirements. Also during the year, an industrial radiography device and source cross reference table was completed to provide NRC Regional Offices and the Agreement States with a reference list of approved combinations of radiography sources and devices.

The NRC regional experience in the inspection of radiography field sites indicates that, during 1987, the 25 percent goal for temporary job site inspections was generally achieved. Revised NRC Inspection Manual Chapter 2800 requires the NRC Regions to furnish inspectors to accompany licensee auditors as part of the total effort in field inspections of radiographers. This method of inspection has the added value of providing the opportunity for an NRC appraisal of the quality of licensee audit programs. As a result of this vigorous inspection program at sites under NRC and Agreement State jurisdiction, numerous violations—involving overexposures, ineffective management control, failure to follow operating and emergency procedures, failure to properly train, examine, and certify radiographic personnel, and failure to maintain required records—were identified.

General Licenses. There are two types of NRC licenses for byproduct, source, and special nuclear materials: specific and general. Specific licenses are documents issued only to individually named persons or organizations, following application and NRC review. General licenses take effect without the issuance of license documents to particular persons. However, the manufacturer of products to be distributed to these "general licensees" must apply to the NRC for a specific license. Before issuing this type of specific license for distribution, the NRC conducts a thorough safety analysis of the product. If it meets the criteria for a general license and the regulations contained in 10 CFR 32, 40, and 70, then the applicant is granted a specific license for distribution of the product to general licensees.

An estimated 200,000 devices are used throughout the country under the general license provisions. The bulk of these are relatively low-hazard devices, such as the tritium exit signs used in office buildings and aircraft. History has shown that the more hazardous devices, the gauges which contain radioactive sources, have been able to come through the trials of explosion, fire, and even of being run over by heavy earth-moving equipment, with source intact.

In 1984, the NRC undertook an evaluation of the adequacy of existing policy pertaining to the distribution of gauges containing byproduct, source, and special nuclear materials under a general license. The study combined the efforts of NRC Headquarters and Regional Offices, and of the Agreement States. Findings indicated extensive lack of compliance with 10 CFR 31.5 requirements by general-licensed gauge-users. Information Notices were sent to the manufacturers, distributors, and the general licensees, summarizing the NRC's findings and stressing the importance of complying with all regulatory requirements.

The results of the 1984 investigation of general-licensed gauge-users prompted an additional study to determine if similar problems existed with industrial devices other than gauges, used under the general license. The findings here were similar to the 1984 study, and included inadequate accountability and improper redistribution of devices. Users of the devices are often unaware of the regulations concerning transfer, disposal, and record-keeping, and labels on the devices often become illegible because of corrosion and wear. For these reasons, the devices become susceptible to loss, improper transfer, and improper disposal. The NRC is using the rulemaking process to clarify and modify current general licensing policy. To help alleviate some of the accountability problems and to keep users up-to-date on the regulations, the NRC is developing a computerized national registry to track all devices and users of the devices in the United States. This registry would allow the NRC to send periodic notices to the users.

Source/Device Registration. The NRC and the Agreement States maintain a sealed source/device registration program which helps to expedite the licensing review process when new requests for sources or devices are received. During the report period, 140 safety evaluations were completed for radioactive sources and containment devices. The computerized registry system for approved sealed sources and devices is updated twice a year, producing 300 reports to NRC Regional Offices, Agreement States, the Center for Devices and Radiological Health (CDRH), and the Atomic Energy Control Board of Canada. During the report period, approximately 50 special reports were produced for the NRC and other government users. To augment the registration program, two comprehensive regulatory guides (10.10, 10.11) were developed and distributed. Rulemaking to clearly define the radiation safety information on sources and devices that is necessary for safety review and to set forth the responsibilities of the registrant became effective on August 24, 1987.

NMSS staff assisted State, Local and Indian Tribes Programs staff with sealed source/device registration audits of two Agreement States. The NRC is working with the Center for Devices and Radiological Health (CDRH) and the Conference of Radiation Control Program Directors, Inc., to incorporate the CDRH "Radioactive Materials Reference Manual" into the NRC's computerized registry. This will be done as a service to the Agreement States, to improve management of source/device designs which contain naturally occurring and accelerator-produced radioactive materials.

Irradiated Gemstones. The NRC staff verified that two U.S. university reactor licensees were engaged in irradiating topaz to enhance the appearance and market appeal of gemstones. One of the licensees was distributing these gems within the U.S., and the other was exporting them. The staff has also received numerous reports of extensive imports of radioactive topaz from foreign suppliers. The NRC received applications from two companies seeking authorization to begin such distribution.

The issue gave rise to significant and complex regulatory questions, involving considerations such as de minimis quantities and the need to evaluate the risks to the public from distribution of consumer products with low levels of radioactivity. Several regulatory alternatives were weighed in an effort to find the proper balance between avoiding unjustified exposures to the public through the use of radioisotopes in consumer products and allowing a distribution of products for which there is some demand and which appear to have acceptably small radiological conseqences. The Commission expected to resolve this issue in December 1987.

New Uses. The NRC and the Agreement States worked jointly to resolve technical and procedural licensing issues for new uses of californium-252, cobalt-60, cesium-137, and iridium-192. The californium-252 could possibly be used to detect explosives in baggage and cargo prior to loading on an aircraft. Iridium-192 and cobalt-60 are now used in a computerized industrial tomography unit as a new type of non-destructive testing.

The NRC is proceeding to establish regulations and guidance for megacurie quantities of cobalt-60 and cesium-137 to be used in process irradiators.

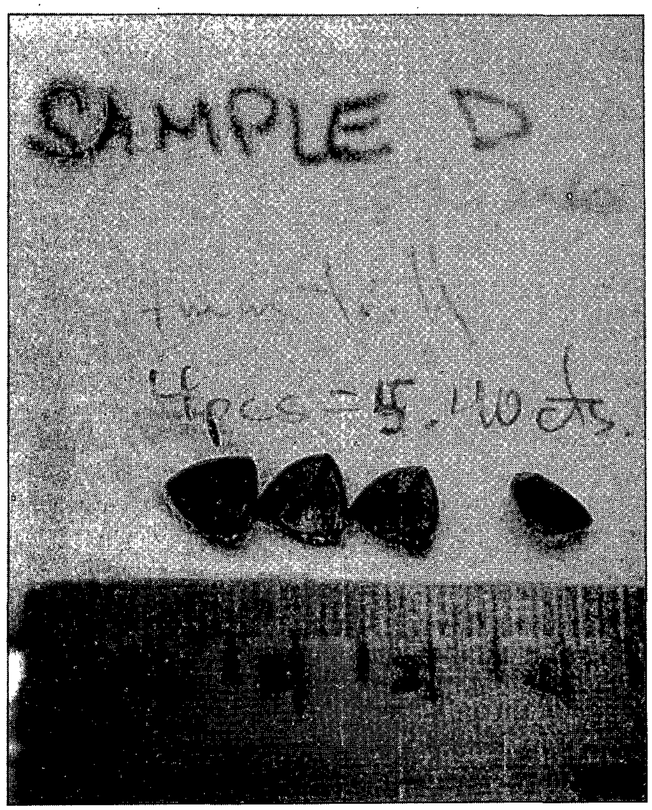
TRANSPORTATION OF RADIOACTIVE MATERIALS

The Federal Government regulates safety in the transportation of radioactive materials primarily through the NRC and the Department of Transportation (DOT). These two agencies have delineated their respective regulatory responsibilities in this area through a Memorandum of Understanding. Shipments that occur within the United States also come under regulation by the States in certain circumstances. 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.

The NRC staff worked on several tasks during fiscal year 1987 addressing transportation safety issues. Brief descriptions of some of these efforts follow.

Defense High-Level Waste (DHLW)

GA Technologies Inc. submitted to the NRC, on behalf of the DOE, a safety analysis report for Model No. DHLW (Defense High-level Waste) cask. This truck cask has a



These samples of imported topaz gems were reported to have been irradiated in a foreign reactor to produce an attractive blue color. Measurements by NRC inspectors indicated negligible levels of radioactivity in the gems.

capacity for one canister of classified DHLW. The cask is a thick-walled stainless steel cylindrical container with a bolted closure. It has an overall length of 162 inches and increases in diameter from 39 inches to 49 inches at the closure end. The cask is provided with upper and lower circumferential aluminum honeycomb impact limiters. The cask has a removable stainless steel clad-depleted uranium shield liner. The gross weight of the loaded cask is about 49,000 pounds. DOE plans to use these casks to transport DHLW from waste-generating sites to a federal repository.

IAEA Regulations

The NRC began work in 1987 to revise its transportation regulations in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," to make them compatible with the 1985 edition of the IAEA transportation regulations. In combination with a parallel effort by the U.S. Department of Transportation, this activity will produce United States transportation rules which are consistent with those of the international community. 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 its safety program.

Spent Fuel Shipments

All reactor spent fuel in storage at the former fuel reprocessing plant at West Valley, N.Y., has been returned to the reactor sites where the fuel was generated, with the exception of approximately 27 metric tons under title of DOE. DOE plans to ship this fuel to its Idaho Nuclear Engineering Laboratory for research and development purposes. Other spent fuel shipping projects included the rail transport of fuel from the Cooper Nuclear Station in Nebraska and from the Monticello Nuclear Station in Minnesota to the General Electric Spent Fuel Storage Operation near Morris, Ill. Agreement by General Electric to receive and store approximately 1,000 fuel assemblies from each of these reactors was the result of fuel supply contracts held by the utilities since the beginning of reactor operations. Receipt of this fuel will essentially fill the Morris pool under its present storage configuration.

NRC/DOE Activities Under The Transportation Procedural Agreement

The NRC/DOE Transportation Procedural Agreement published in the Federal Register (48 FR 51875) on November 14, 1983, remains in force. This agreement focuses on the important task of exchanging information and identifying transportation packaging issues at the earliest opportunity, to assist in DOE's new cask development program. In a meeting of the technical staff on June 26, 1987, with representatives of DOT also participating, members reported on the NRC-sponsored research activities and discussed package certification issues. The DOE provided information on their contracting plans and the schedule for developing the new generation of shipping casks and the complete transportation system. The meetings included extensive discussion of methods of assuring that major public concerns are identified and addressed in the DOE development program. Future meetings will focus on various aspects of the development program for new transportation casks.

Highlights of Transportation Safety Efforts

The NRC concluded a major study of the safety provided by its design regulations for packages used to transport large quantities of radioactive material. This study, performed for the NRC by the Lawrence Livermore National Laboratory, evaluated how well packages designed to meet NRC performance criteria will withstand the forces generated in severe accidents. The study considered data from severe non-nuclear accidents that have actually occurred, supplemented by data from various package test programs. Comparing the forces from resulting from severe accidents with with those the casks are designed to withstand gives a measure of the degree of protection afforded by casks in conformity with regulatory requirements. Accidents which produced forces in excess of hose the casks are designed to withstand were studied in more detail to assess the potential for release of radioactive material from the cask. Also, the probability of such an accident actually occurring was evaluated and the resulting risk to the public health and safety was compared with the risks previously calculated in the "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes" (NUREG-0170). The study's final report, "Shipping Container Response to sEvere Highway and Railway Accidents," was published in February 1987 as NUREG/CR-4829. The NRC staff subsequently published a summary brochure entitled "Transporting Spent Fuel-Protection Provided Against Severe Highway and Railroad Accidents" (NUREG/BR-0111, March 1987).

Transportation Inspection and Enforcement

The NRC continued its transportation-related safety inspection program. The total effort involved over 1,000 individual inspections covering byproduct, source, and special nuclear material licensees; fuel cycle facilities; and shippers of spend reactor fuel.

Safeguards

Chapter 6

In accordance with provisions of the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, the NRC regulates safeguards for licensed nuclear materials, facilities, and activities to assure protection of the public health and safety and the common defense and security. In this regulatory context, "safeguards" denotes measures which are taken to deter, prevent, or respond to the unauthorized possession or use of significant quantities of special nuclear material through theft or diversion, and to protect against radiological sabotage of nuclear facilities. In general, safeguards for licensed nuclear fuel facilities and non-power reactors emphasize protection against theft or diversion of special nuclear material (SNM), while safeguards for power reactors stress protection against radiological sabotage. (SNM and strategic special nuclear material, or SSNM, are technical designations of certain types, quantities, and/or isotopic compositions, defined by formula, of various nuclear materials. In general, SSNM is high-enriched uranium or plutonium.)

During fiscal year 1987, NRC safeguards requirements were applied to 109 power reactors, 54 non-power reactors, and 28 fuel cycle facilities. They were also applied to 120 shipments of spent fuel, 19 shipments of SNM involving more than one but less than five kilograms of high-enriched uranium, and one shipment of SNM involving five or more kilograms of high-enriched uranium.

STATUS OF SAFEGUARDS IN 1987

Reactor Safeguards

Power Reactors. During fiscal year 1987, NRC safeguards regulations covered 109 licensed power reactors. In response to revised NRC regulations, all power reactor licensees submitted amended security plans to provide more safety-conscious safeguards systems, while maintain current levels of protection. The revised requirements resulted from a Commission review of the potential impact of safeguards requirements on plant safety. Licensees also amended their security plans in response to revised NRC regulations clarifying requirements for entry searches at power reactor facilities. Protection of licensed power reactors was also enhanced by the issuance in March 1987 of a new NRC regulation (10 CFR 73.57) requiring fingerprinting and FBI criminal history checks of persons having unescorted access to nuclear power plants or to Safeguards Information.

Non-power Reactors. In fiscal year 1987, 54 licensed non-power reactors were subject to NRC safeguards regulations. Following publication in fiscal year 1987 of the NRC final rule requiring non-power reactors to be converted from the use of high-enriched uranium to low enriched uranium (LEU) or to as low an enrichment as possible, affected licensees have begun the task of actual conversion. The full conversion process will typically take two-to-three years and fall into three distinct phases: an initial study of necessary core changes and safety considerations, fabrication and delivery of the LEU fuel, and loading of the new fuel and removal of the old fuel off-site. Conversion was made contingent on funds being provided by the Federal government. At the end of fiscal year 1987, 10 facilities had received funding from the Department of Energy to begin the conversion process.

Regulatory Effectiveness Reviews at Power Reactors. The NRC staff, assisted by U.S. *Army Special Forces personnel*, continued the Regulatory Effectiveness Review (RER) program, evaluating the practical effectiveness of safeguards for vital equipment at licensed reactors. RERs are conducted to assure that safeguards programs, as implemented by licensees, are effective against the design basis threats defined in 10 CFR 73.1. During fiscal year 1987, reviews were conducted at 16 power reactors. RERs have led to the identification of both strengths and weaknesses in licensees' programs. Commonly noted strengths include effective routine access control features and good rapport and coordination with local law enforcement agencies. The most common problem areas identified in RER reports concern vital area barriers and intrusion detection and alarm assessment systems. Problems and issues raised in RER reports are resolved through voluntary actions of licensees or through licensing, inspection, enforcement, or rulemaking, as appropriate.

Reactor Safeguards Inspections. Safeguards inspectors from NRC's five Regional Offices conducted hundreds of inspections at licensed reactors throughout the United States during fiscal year 1987. Resident inspectors at operating power reactors also contributed to the safeguards inspections program at their respective sites. Enforcement actions resulting from NRC inspections are described in Chapter 1.

By law, the NRC regulates safeguards adopted for the protection of nuclear materials from theft or willful damage and the protection of nuclear facilities from sabotage. Contingency plans for the protection of facilities, as well as nuclear material transport routes, are carefully and regularly coordinated with local and State police.

Fitness For Duty at Power Reactors. In parallel with reactor safeguards programs designed to assure the trustworthiness and reliability of persons having access to nuclear power plants, the Commission continued efforts to assure that all nuclear power plant personnel with access to vital areas at operating plants are fit for duty. Following publication of the Commission policy statement on fitness for duty (51 FR 27921), the NRC staff began reviewing and evaluating a number of licensee fitness-for-duty programs. Information developed during these reviews and other data concerning the effectiveness of the industry programs will be considered by the Commission in deciding whether further regulatory action is needed.

Fuel Cycle Facilities

During fiscal year 1987, the number of licensed fuel facilities subject to NRC safeguards requirements was 28, of which 10 are major fuel fabrication facilities. The activities at these 28 facilities include full-scale reactor fuel production, pilot plant operations, decommissioning efforts, and the storage of sealed items. Seventeen of the facilities maintained both physical security and material control and accounting systems. Four of these 17 facilities had actual holdings of formula quantities of SSNM requiring the implementation of extensive physical security and material accountability measures.

In August 1987, the staff issued license conditions to the four licensees holding formula quantities of SSNM, requiring incorporation of three near term improvements—recommended by the NRC/DOE Comparability Review Team and approved by the Commission—into their physical security plans. The new measures require 100 percent search of personnel and hand-carried packages entering the protected area, night-qualification in all assigned weapons for security force personnel, and the use of armed guards at material access portals during operation. Three additional improvements, recommended by the Review Team and endorsed by the Commission, will be implemented through rulemaking. These include a requirement for security system performance evaluation, through response force tactical exercises; a change to the design basis threat for theft, to include the use of land vehicles by adversaries to commit a theft; and two protected area barriers.

Other major activities in the area of safeguards for fuel facilities included completion of reviews of two licensee Fundamental Nuclear Material Control (FNMC) plans required to implement the material control and accounting requirements contained in the new 10 CFR 74.31. The review of the remaining four FNMC plans is continuing. In all, the NRC received and completed actions on approximately 160 safeguards licensing matters associated with fuel facilities in fiscal year 1987.

Inspection at Fuel Cycle Facilities During fiscal year 1987, material control and accounting inspections were conducted at the 10 major fuel fabrication facilities, with physical security inspections at six of the 10, including the four that possess formula quantities of SNM. Two new inspection procedures, involving International Atomic Energy Agency safeguards requirements at U.S. commercial low enriched fuel facilities and power reactors, were prepared.

Transportation

Spent Fuel Shipments. During the year, NRC approved 38 transportation routes with respect to acceptable protection against radiological sabotage. One hundred twenty spent fuel shipments went over these routes. To keep the public informed about spent fuel shipment routes, NRC publishes a document entitled "Public Information Circular for Shipments of Irradiated Reactor Fuel" (NUREG-0725), containing information on approved routes.

SSNM Shipments. One export shipment, involving five or more kilograms of high-enriched uranium, was made during fiscal year 1987. There were also 10 exports, two foreign shipments which transited the United States, and seven domestic shipments—each involving less than five but more than one kilogram of high-enriched uranium—during the fiscal year.

Shipment Route Surveys. In fiscal year 1987, NRC regional personnel worked with local law enforcement agencies to conduct field surveys of routes proposed for shipments of spent fuel or SSNM. Thirty-eight routes were analyzed through 38 States, involving over 3,000 miles of route travel. The NRC brochure entitled "Information Package on Spent Nuclear Fuel Shipments for Law Enforcement Agencies" (NUREG/BR0020) was distributed to local officials and agencies during these surveys.

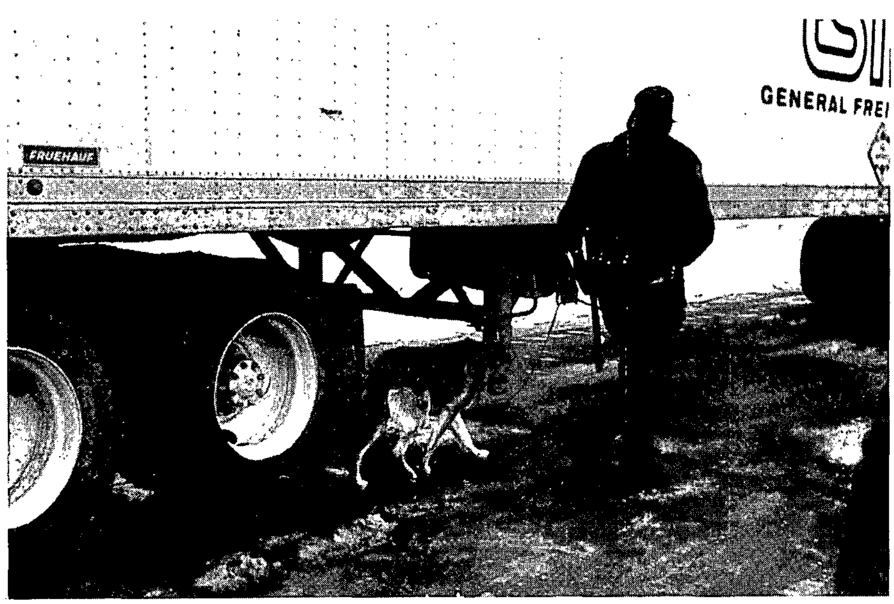
Tracking International Shipments of SNM. NRC regulations requiring licensees to comply with the provisions of the Convention on the Physical Protection of Nuclear Materials became effective on March 26, 1987. Licensees shipping the materials defined under the Convention began making notifications thereafter. The information was forwarded from the NRC to the Department of State for appropriate international notifications. Through September 30, 1987, there have been approximately 163 actions. It is expected that this figure will continue to increase to approximately 1,000 actions per year.

Transport Inspection and Enforcement. The NRC continued to inspect selected domestic shipments and the domestic segments of import and export shipments of spent fuel. No significant problems were identified from inspections carried out during the report period.

Incident Response Planning And Threat Assessment

The NRC staff assesses threats to NRC licensed facilities, materials, and activities, and prepares NRC incident response plans for responding to actual thefts of nuclear material or radiological sabotage of nuclear facilities or activities. A continuing working relationship is maintained with other Federal agencies concerned with threat-related matters. Particular attention is paid to foreign terrorist groups, their activities, and their relationship with possible state-sponsored activities. Based on NRC review and interaction with other agencies, no significant change in the threat environment addressed by current NRC safeguards regulations was discerned. Both the domestic and foreign threat environments are reviewed on a continuing basis to assure the adequacy of current NRC domestic safeguards regulations. The Commission, as part of its reconsideration of the design basis threats, continued to solicit other agency views of the domestic threat environment as it relates to the protection of domestic nuclear facilities.

Two techniques are employed in assessing reported threats to NRC licensees. Internally, the NRC Information Assessment Team, composed of headquarters and regional personnel, promptly assesses all reported threats and recommends appropriate response actions to NRC management. Additionally, an interagency team, the Communicated Threat Credibility Assessment Team, which is jointly funded by the NRC and the Department of Energy (DOE), conducts analyses of written or recorded threats.



Security canine teams are used in the safeguards effort. Here, a security guard at a nuclear plant, together with his trained assistant, search a vehicle for explosives prior to authorizing entry to the plant site.

Incident response plans detail the NRC response to reported acts of theft or radiological sabotage involving licensed materials or facilities. In 1987, the plans, including team organizations and procedures, were reviewed and revised as a result of a reassignment of safeguards responsibilities within NRC Headquarters.

The staff continued its analysis of safeguards events in order to identify trends, patterns, and anomalies. The "Safeguards Summary Event List" (NUREG-0525), a compilation of safeguards events, was updated in July 1987 (Rev. 13). This list contains information about safeguards-related events that involve licensed nuclear material or facilities.

NRC/IAEA Interaction. During 1987, the International Atomic Energy Agency (IAEA) continued to carry out routine inspections of the Westinghouse low-enriched uranium (LEU) fuel fabrication plant in Columbia, S.C., the Salem Unit 1 power reactor in New Jersey, and the Turkey Point Unit 4 power reactor in Florida. The NRC also continued to submit accounting data on a monthly basis for these facilities, as well as for the LEU fuel fabrication plants of Babcock & Wilcox at Lynchburg, Va., of Advance Nuclear Fuel Corporation at Richland, Wash., of Combustion Engineering Corp. in Connecticut, and of General Electric at Wilmington, N.C.

In May 1987, representatives of the NRC and the IAEA met in Washington, D.C., to discuss IAEA safeguards implementation issues in the U.S. Also, in May 1987, the IAEA notified the U.S. of their intention to select the General Electric (GE) low-enriched uranium fuel fabrication facility at Wilmington, N.C., for application of IAEA safeguards. Design information for that facility was obtained, and appropriate negotiations between the U.S. and the IAEA are underway. At the close of the report period, the anticipated date for implementation of IAEA safeguards at GE was January 1988.

SAFEGUARDS REGULATORY ACTIVITIES AND ISSUES

Safeguards Events Reporting

The NRC issued revised reporting requirements for safeguards events in fiscal year 1987. Safeguards events include actual or attempted theft of special nuclear material (SNM); actual or attempted acts or events which interrupt normal operations at power reactors because of an unauthorized use of or tampering with machinery, components or controls; certain threats made against facilities possessing SNM; and safeguards system failures having an impact on the effectiveness of the system. The purpose of the revision is to simplify and clarify previous requirements in this area.

Fingerprint Rule for Power Reactors

A new regulation, Requirements for Criminal History Checks, was published in final form on March 2, 1987. This regulation was developed to implement Public Law 99-399, the Omnibus Diplomatic Security and Anti-Terrorism Act of 1986. The Act requires that each individual granted access to Safeguards Information or unescorted access to a nuclear power plant be fingerprinted and a criminal history records check made by the Federal Bureau of Investigation. The Commission's rule provides for control of the data to prevent misuse, to limit re-dissemination, and to restrict the use of certain arrest information.

Material Control and Accounting for Fuel Facilities

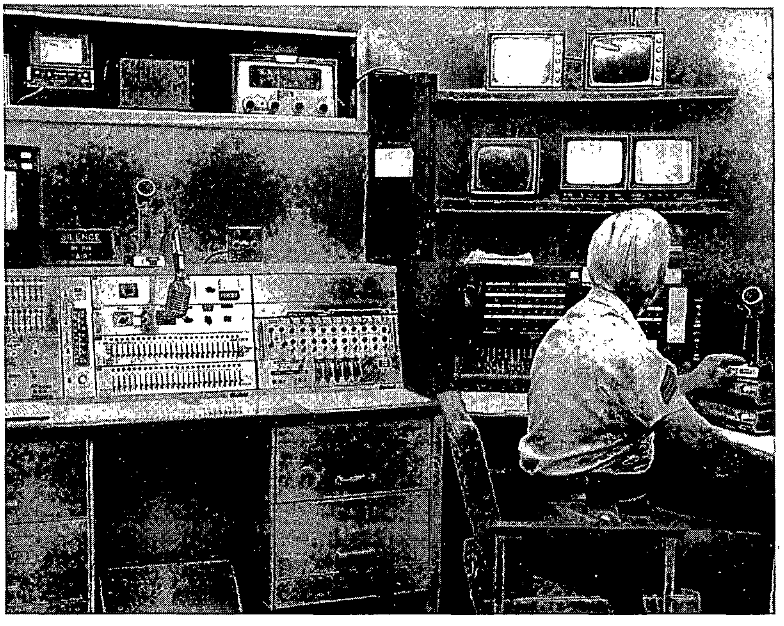
A final rule revising material control and accounting (MC&A) requirements for fuel cycle facilities licensed to possess and use formula quantities of strategic special nuclear material was issued on March 30, 1987. The final rule takes into account public comments on the draft rule and information obtained through site-specific value-impact analyses. The rule shifts the emphasis of MC&A away from periodic physical inventories and toward the use of monitoring information for safeguards. Timely detection of anomalies potentially indicative of material losses and enhanced loss-localization capabilities are the principal benefits to be realized. Fundamental Nuclear Material Control Plans describing how the affected licensees will implement the new requirements are to be submitted to the NRC for review and approval. Reactors, waste disposal operations, and irradiated fuel reprocessing plants (if any should be licensed) remain subject to the current MC&A requirements in 10 CFR Part 70.

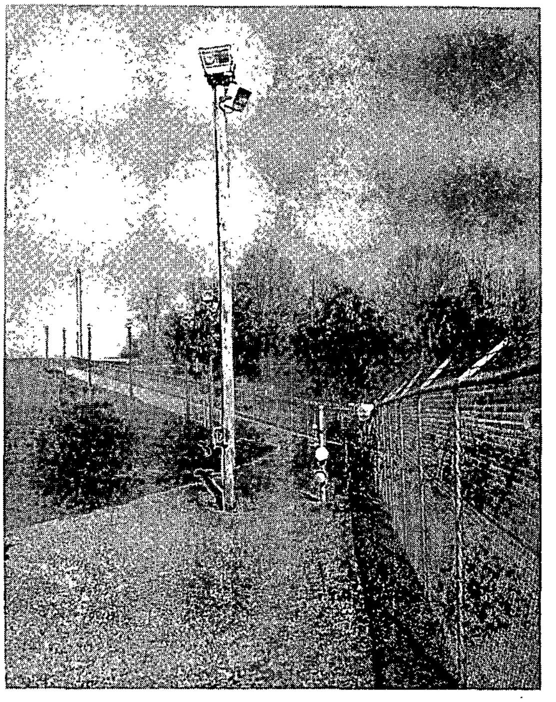
Transportation

The Convention on the Physical Protection of Nuclear Materials. The United States is a signatory of this Convention which provides for the establishment and maintenance of adequate physical security for international shipments of significant quantities of source or special nuclear material. A final rule to bring NRC regulations into accord with the Convention became effective on March 26, 1987, following ratification of the Convention by 20 other countries.

SAFEGUARDS TECHNICAL ASSISTANCE

Approximately $2.8 million was spent in fiscal year 1987 on safeguards technical assistance contractual projects. Some of these projects are described briefly below.





Guard forces and alarm systems are the main elements in any plant's security. Criteria for their use in sensitive areas of nuclear power plants are spelled out in NRC requirements. The photo at top left shows a security guard at a secure entrance/checkpoint, and the plant's central security control room. TV monitors in the control room afford continuous surveillance of key passages and doors throughout the plant. Top right is a protected area barrier, including fencing, microwave intrusion detection systems, lighting and clear zones. Lower left is an "intruder" thwarted in an attempt to enter the plant; intrusion detection systems are tested at irregular intervals. Lower right is an individual challenging the perimeter intrusion detection system in adverse weather.

- Communicated Threat Credibility Assessment. This project, jointly funded with DOE, is a continuing effort to maintain and refine a capability to perform credibility assessments of nuclear extortion threats and to provide advice to the NRC, DOE, FBI and other Federal, State and local agencies. The assessment methodology evaluates technical, behavioral, and operational factors associated with threat messages. The project supports NRC's mandated responsibility for contingency planning, and NRC's responses to threats, theft, and radiological sabotage.
- Nuclear Materials and Safeguards System. This project, also jointly funded with DOE, continues the operation and maintenance of the Nuclear Materials Management and Safeguards System (NMMSS). Basically, this is an accounting system for all licensed SNM in the U.S., including both U.S. and foreign origin materials. Material is tracked from facility to facility on a continuing basis from original refinement to eventual disposal. Export/import transactions are also tracked and selected data, based on NMMSS output, is furnished to the IAEA in fulfillment of the U.S.'s international obligations.
- Facility Systems Analysis Support. This project provides facility systems analysis reports which are used by the Regulatory Effectiveness Review teams to assist in analyzing and identifying components of power reactors that need to be protected as vital equipment. The reports are used both in developing data sheets and during plant walk-throughs to ensure that all important equipment is examined.

Waste Management

Chapter 7

The NRC's regulation of nuclear waste is managed and coordinated by the Office of Nuclear Material Safety and Safeguards (NMSS). The activities of this office include the regulation of all commercial high-level and low-level radioactive waste and uranium recovery activities. Specifically, the functions of NMSS include:

- Developing the criteria and the framework for high-level waste (HLW) regulation, including the technical bases for the licensing of high-level waste repositories.
- Providing program management for NRC's responsibilities under the Nuclear Waste Policy Act of 1982 (NWPA).
- Developing rules and guidance to assure a consistent national program for the regulation and licensing of low-level waste disposal facilities.
- Developing guidance and providing technical assistance to States and compacts to ensure the goals of the Low-Level Radioactive Waste Policy Amendments Act (LLRWPAA) of 1985 are met.
- Providing national program management for licensing and regulating uranium recovery facilities and associated mill tailings.
- Reviewing and concurring in significant Department of Energy (DOE) decisions related to inactive mill tailings sites and the licensing of stabilized tailings piles for monitoring and maintenance programs.

During the report period, fiscal year 1987, the Division of Waste Management was divided into two separate Divisions to better meet the programmatic needs of the agency: the Division of High-Level Waste Management (HLWM), and the Division of Low-level Waste Management and Decommissioning (LLWM).

Highlights of High-Level Waste Program

In fiscal year 1987, NRC staff continued its work to assure that the milestones of the NWPA can be met. It is the NRC's policy that, absent any unresolved safety issues, the NRC will support DOE schedules for meeting NWPA requirements, as set forth in the DOE final Mission Plan and final Project Decision Schedule. During the year, the NRC defined its position on the implementation of DOE's Mission Plan and issued comments to DOE on the Draft Mission Plan Amendments. Other significant accomplishments include the issuance of three staff technical positions providing guidance to DOE on a variety of issues, and the publication of an advance notice of proposed rulemaking indicating the Commission's intent to modify the definition of "high-level radioactive waste" in its regulations so as to bring it into conformity with the NWPA definition. The NRC has also conducted meetings and workshops with DOE to resolve pre-licensing issues related to DOE's development of Site Characterization Plans (SCPs), and to the initial development of the Licensing Support System to streamline the licensing process. HLWM staff played a key role in the the development of a negotiated rulemaking on the submittal and management of records and documents related to the licensing of a high-level radioactive waste repository.

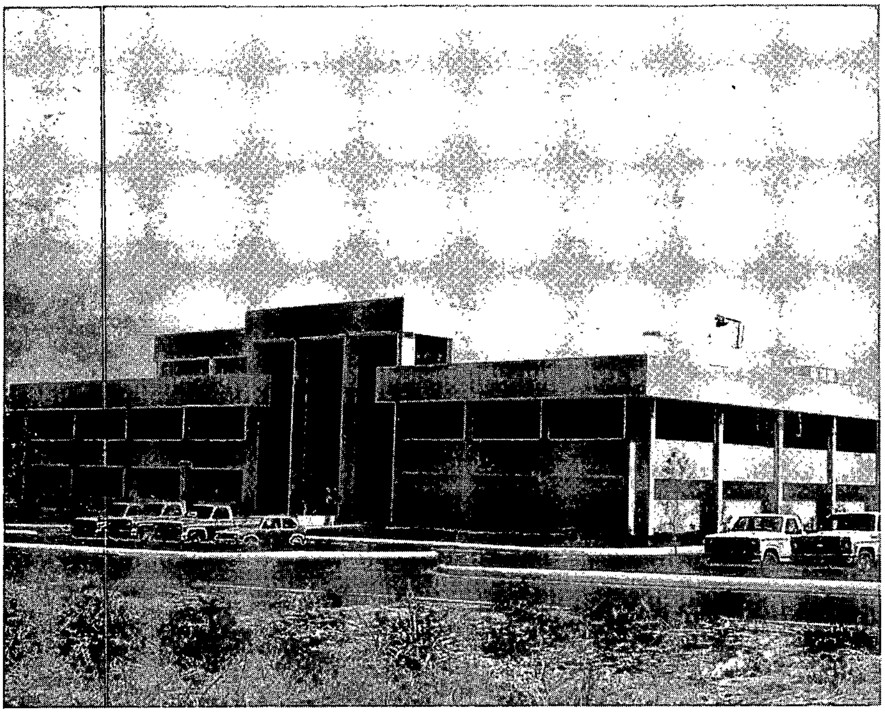
The staff has continued to devote significant effort throughout the year to its meetings and workshops with DOE, and with the States and Indian Tribes, in an effort to identify and resolve potential licensing issues as early as possible.

During the report period, NRC staff received and reviewed nine proposals to operate the Center for Nuclear Waste Regulatory Analysis (CNWRA), a Federally funded research and development center whose major purpose is to provide technical assistance for the NRC's high-level waste program. The contract was awarded to the Southwest Research Institute of San Antonio, Tex., in October 1987 and the Center is scheduled to begin operation in fiscal year 1988.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development

Three rulemaking actions were taken during the report period. An Advanced Notice of Proposed Rulemaking (ANPR) to redefine "high-level waste" in light of the NWPA definition was published in February 1987. The staff also initiated action to amend Parts 60 and 51 to conform National Environmental Policy Act-related requirements to NWPA requirements concerning NRC adoption of DOE's Environmental Impact Statement for the geologic repository. A proposed rule is expected to be published during fiscal year 1988.



The NRC's new Center for Nuclear Waste Regulatory Analysis occupies the third floor of the Southwest Research Institute's Electromagnetics Building in San Antonio, Tex.

During this period, the staff continued its efforts in the development of the negotiated rulemaking on the submission and management of records and documents related to the licensing of a geologic repository for the disposal of High-Level Radioactive Waste. Section 114(d) of the NWPA provides three years, with a possible extension of 12 months, for the NRC to reach a decision on a construction authorization for a high-level waste repository. Ready access to all pertinent information must be assured if the Commission is to make its decision within this timeframe. DOE has already committed itself to developing an electronic information management system which would be used to facilitate the licensing process.

The Commission has decided to use negotiated rulemaking to develop proposed changes to its 10 CFR Part 2 domestic licensing procedures providing for the use of an electronic information management system for the high-level waste repository licensing proceeding. Negotiated rulemaking offers an opportunity for comprehensive treatment of the issues and for creative solutions, because participants with ideas on how to solve the problem are brought together to discuss and react to each others' concerns and positions directly.

The NRC contracted with the Conservation Foundation to handle the convening and facilitating stages of the negotiation. In September 1987, the NRC held the first meeting of the HLW Licensing Support System Advisory Committee in Washington, D.C. The proposed rule is expected to be published in late fiscal year 1988, and the final rule in early fiscal year 1989.

For much of fiscal year 1987, the staff pursued an ongoing rulemaking to conform 10 CFR Part 60 to the requirements of the EPA high-level waste standards, but this rulemaking was suspended when a Federal Court ruled EPA's standards invalid.

Regulatory Guidance

NRC's regulatory guidance in the area of high-level waste is directed mainly at apprising DOE of acceptable methods, tests, and design characteristics for meeting performance objectives and siting and design criteria of Part 60. In conjunction with its regulatory guidance, the NRC staff is also developing its own tools and methodologies for evaluating DOE's assessments of repository performance.

The NRC staff continued to develop Generic Technical Positions (GTPs) and other guidance documents during this reporting period. The following GTPs were published in final form during fiscal year 1987:

- Final GTP on Sorption
- Final GTP on Qualification of Existing Data for High-Level Nuclear Waste Repositories
- Final GTP on Peer Review

In addition, public comments were received on the following draft GTPs and the staff is developing them in final form:

- Draft GTP on Interpretation and Identification of the Disturbed Zone

- Draft GTP on Groundwater Travel Time
- Draft GTP on Items and Activities in the High-level Waste Geologic Repository Program Subject to 10 CFR Part 60 Quality Assurance Requirements

Further, the staff managed the development of contractor documents (NUREG/CRs) that support NRC regulatory guidance in geochemistry, geology/geophysics, hydrology, performance assessment, quality assurance, geotechnical engineering/design, and waste package engineering.

Site Investigations

Section 112(b) of the NWPA requires DOE to recommend three sites to the President for characterization as the first repository, and to publish Environmental Assessments (EAs) for each of at least five nominated sites from which the recommended ones are to be chosen. In May 1986, DOE published final EAs for each of the five sites nominated as suitable for site characterization, and also recommended three sites for characterization. The President approved the DOE-recommended sites for characterization. The three sites are the Yucca Mountain site in Nevada, the Deaf Smith County site in Texas, and the Hanford site in Washington.

Section 113(b) of the NWPA requires that, for each of the sites to be characterized, DOE must issue for NRC and State/Tribal comment a Site Characterization Plan (SCP), with a description of a proposed waste form and packaging and a conceptual repository design. On August 26, 1987, DOE revised its schedule for the issuance of the SCPs. Whereas the previous schedule called for a sequential release of the SCPs, all three SCPs are now scheduled to be released simultaneously—as Consultation Draft SCPs in January 1988, and as SCPs in early 1989. NRC activity has included reviewing available data and information on the sites from investigations to date, reviewing design documents and preliminary plans for site characterization, and working toward resolution of the significant concerns in open, documented technical meetings before the SCP's are issued. These meetings have speeded progress in resolving the technical issues identified by NRC staff.

Quality Assurance in Site Characterization

During the year, the staff continued to provide guidance to DOE on an acceptable quality assurance (QA) program for the site characterization phase of the geologic repository project. The rule, 10 CFR Part 60, requires that the information used to support DOE's repository license application be subject to the QA program set forth in 10 CFR Part 50, Appendix B, "as applicable and appropriately supplemented." The Appendix B criteria for construction and operation of a nuclear power reactor required some modification for use in the research and development and exploration work, which is a large part of repository site characterization.

The staff continued development of quality assurance guidance for the repository program to help assure that the power reactor quality assurance criteria in Appendix B are appropriately utilized in the geologic repository program. The staff issued two final Generic Technical Positions entitled "Peer Review for High-Level Nuclear Waste Repositories" and "Qualification of Existing Data for High-Level Nuclear Waste Repositories." These documents contain staff positions that represent acceptable approaches for meeting the quality assurance regulations in 10 CFR Part 60. Both GTPs were issued as drafts for public comment. Prior to issuing the GTPs as final documents, the staff met

In the foreground is the NRC's on-site representative, Paul Prestholt, accompanied by H. L. McKague, a representative of the DOE's Lawrence Livermore National Laboratory and a consultant to the NRC, as they examine calcite/silica deposits in trench #14 at the proposed Yucca Mountain (Nev.) site for a high-level nuclear waste repository. Safety standards for the repository, which is envisioned as a 1,500 acre grid of tunnels deep inside the mountain, were developed by the NRC and the Environmental Protection Agency. They require that the waste be prevented from contaminating the environment for up to 10,000 years.

with the Department of Energy, the affected States and Tribes, and industry representatives to review the proposed resolution of public comments. Agreement was reached among all the parties on the disposition of the public comments. The staff also published a draft final GTP addressing the items and activities covered by the 10 CFR Part 60 quality assurance program (the Q-list). A public meeting was held to discuss the proposed final positions and the resolution of the public comments. The final version is expected to be issued in early 1988.

During 1987, the staff continued its review of the DOE Quality Assurance plans and procedures. Quality assurance documents used by DOE Headquarters and each of the three project offices was reviewed for comment by the staff.

The staff conducted its first audit of the DOE high-level waste repository program in 1987. Early in the year, DOE identified several areas that were ready for NRC audit, and the staff selected the mineralogy/petrology studies at the Los Alamos National Laboratory (LANL). LANL is conducting laboratory investigations of the geochemistry aspects of the Yucca Mountain site in Nevada. The audit consisted of a team of quality assurance and technical personnel who examined both the documentation related to the work being performed and the quality of the technical work at LANL. Several findings, and deficiencies, were identified by the audit team in programmatic QA. The staff will be working with DOE and LANL personnel to resolve these matters and to assure that the program is fully in place for site characterization activities. The staff will continue to monitor and audit the DOE program as it is upgraded to meet the Commission's regulations.

DOE Mission Plan and Project Decision Schedule

Section 301 of the NWPA requires that DOE submit to Congress a Mission Plan, delineating how the activities required by the NWPA will be implemented. Section 114(e) of the NWPA requires DOE to prepare and update, in cooperation with affected Federal agencies, a Project Decision Schedule (PDS) for those activities. Any Federal agency that determines that it cannot comply with a deadline in the PDS must prepare a written explanation of the reason it cannot and submit the explanation to the DOE and to the Congress.

The final version of the original Mission Plan was submitted to Congress on July 9, 1985. On January 28, 1987, the DOE issued a Draft Mission Plan Amendment which proposed a five-year extension in the schedule for receipt of spent fuel at a repository, from 1998 until 2003, and a postponement of site-specific work at a second repository until the mid- to late 1990's. The Draft Mission Plan Amendment also indicated DOE's intention to submit a proposal to Congress for a Monitored Retrievable Storage (MRS) facility to be constructed as an integral part of the civilian radioactive waste management system. The MRS proposal was submitted in March 1987. On April 7, 1987, the NRC provided comments to DOE on the Draft Mission Plan Amendment. DOE submitted the Mission Plan Amendment to Congress on June 9, 1987 and requested Congressional approval.

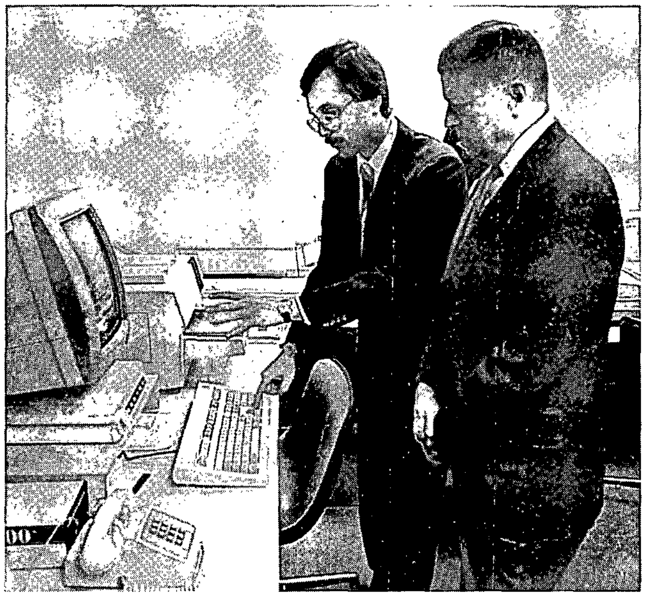
After receiving comments from the NRC and other interested parties on a draft Project Decision Schedule (PDS), DOE issued the final PDS on April 10, 1986. On April 3, 1987, DOE informed NRC that a revision to the PDS had been initiated. DOE will await Congressional approval of the revised schedule and other changes in the Mission Plan Amendment before providing the revised PDS for review and comment.

State and Tribal Interactions

The NWPA contains provisions for State and Tribal participation in the repository program. It includes specific provisions for DOE's interaction with the States and Tribes, and requires both NRC and DOE to provide "timely and complete" information to States and Tribes on all repository-related "determinations or plans." In addition, NRC regulations (10 CFR Part 60, Subpart C) specify a variety of mechanisms by which States and Tribes may participate in NRC's NWPA activities. It is NRC's policy to maintain close communication with the States and Tribes so that licensing issues—as well as required activities and lead times for State/Tribal participation—are identified early.

One of the key events in the agency's State and Tribal interaction was the Commission's June 16 meeting with State and Tribal representatives on the DOE program for the first repository and the proposed Monitored Retrievable Storage (MRS) facility. Officials from Nevada, Texas, Utah, and Washington spoke on the repository program, along with representatives of the Nez Perce, Umatilla, and Yakima Tribes, located near the proposed Basalt Waste Isolation Project (BWIP) at Hanford, Wash. Also attending was a spokesman for the State of Tennessee, which contains the site selected in DOE's April 1987 proposal to Congress for an MRS. NRC staff also hosted these officials, as well as representatives of potential host States for the DOE second repository program, at an annual meeting in Washington (June 30) to present the status of NRC staff efforts to implement the NWPA and to discuss State and Tribal concerns. NRC officials made follow-up visits to the Nez Perce, Umatilla, and Yakima Indian reservations in September to discuss high-level waste program issues; and representatives of the National Congress of American Indians briefed Commission staff and NRC senior management on Tribal concerns in October. NRC officials also made presentations to the National Association of Regulatory Utility Commissioners, to the State Liaison Officers, and to interested State legislators and staff in the National Conference of State Legislatures and its High-Level Waste Working Group.

State and Tribal officials participated in meetings between NRC and DOE on generic and site-specific technical issues. These included two DOE briefings in the Washington, D.C., area on the DOE Issues Hierarchy for the repository program; meetings in Las Vegas and Houston, respectively, on proposed exploratory shaft facilities for the Yucca Mountain, Nev., and Deaf Smith, Tex., candidate sites; a meeting in Las Vegas on seismo-tectonic issues at the Yucca Mountain site, and a meeting in Richland, Wash., on pre-exploratory shaft geohydrologic testing at the BWIP candidate site. State and Tribal officials also attended NRC staff meetings in Washington, D.C., to address public comments on draft Generic Technical Positions on peer review and the qualification of existing data on DOE's candidate repository sites, on the "Q-List" of items and activities subject to NRC quality assurance requirements for repository licensing, and on the repository design basis accident dose limit.



An NRC Senior Information Systems Analyst, Avi Bender (left), demonstrates the on-line full-text search and retrieval capability—part of a high-level waste R&D project—to NRC Chairman Lando W. Zech, Jr. Using this system, NRC documentation related to high-level radioactive waste can be accessed from anywhere in the country.

Other Activities

In October 1986, the Commission approved the establishment of an NRC-sponsored Federally Funded Research and Development Center (FFRDC) to provide long-term technical assistance and research related to NRC's regulatory program under the NWPA. An FFRDC is being proposed as a solution to the problems of contractor conflict-of-interest (with DOE and other parties to the high-level waste licensing proceeding) and to provide long-term continuity in NWPA-related technical assistance and research. The FFRDC will provide support to NRC in the following areas: (1) waste systems engineering and integration and overall program activities; (2) long-term performance of a geologic setting; (3) long-term performance of an engineered barrier system; (4) transportation, special projects, and analytical studies; and (5) monitored retrievable storage (MRS) and repository design, construction and operation. A competitive solicitation for proposals to operate the Center and provide the necessary resources was published. NRC staff received and reviewed nine proposals. The \$42.5-million, five-year contract was awarded to the Southwest Research Institute of San Antonio, Tex., in October 1987. Unlike most Federal contracts, the five-year contract can be renewed without competition if NRC is satisfied with the contractor's performance. The Center began operations in late 1987.

The NRC has completed the first phase of a pilot project which has resulted in an operational full text search and retrieval system. The system enables users to quickly search for and retrieve licensing documents. The initial test database is now being expanded as documents are captured and stored on a daily basis.

The second phase of the pilot project is examining advanced text and image applications using optical disk technology. This effort will demonstrate the feasibility of distributing a document database on an optical disk, which would then allow users to conduct full text search and to retrieve the original image of the document.

Information and experience gained from the pilot project is being shared with the Department of Energy, and with the HLW Licensing Support System Advisory Committee, which is now actively involved in defining the requirements of the Licensing Support System (LSS). The NRC portion of the data base, currently being stored in electronic full text, will become part of the future licensing database under the LSS.

The staff is also demonstrating a high-level waste issue tracking and management system, the Open Item Management System (OIMS), for identifying, tracking, and resolving potential high-level waste licensing issues. NRC is developing OIMS to ensure that potential licensing issues are identified and resolved beginning as early as possible in the site selection process.

LOW-LEVEL WASTE PROGRAM

The NRC continues to meet the statutory mandates of the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA). As directed by the LLRWPAA, NRC established licensing review procedures for new low-level waste disposal facilities. In January 1987, NRC issued a Standard Format and Content Guide (NUREG-1199) for a license application and a Standard Review Plan (NUREG-1200) for the staff's review of a license application. In April 1987, NRC issued an Environmental Standard Review Plan (NUREG-1300). The LLRWPAA also directs NRC to publish technical guidance regarding licensing of disposal facilities that use methods of disposal which are alternatives to shallow-land burial. In addition to updating existing guidance documents to include alternatives, the staff published the technical position, "Licensing of Alternative Methods of Disposal of Low-Level Radioactive Waste" (NUREG-1241).

With regard to uranium recovery activities, the staff continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978. The staff also continued work in conforming 10 CFR Part 40 regulations to the final EPA standards for mill tailings. In July 1986, the staff published a proposed rule addressing ground-water protection; the comment period closed in November 1986, and on July 20, 1987, the final rule was forwarded to the Commission for approval. On October 16, 1987, the Commission approved, by affirmation, publication of the final ground-water protection rule. Publication in November 1987 is anticipated.

New Organizational Structure

In April 1987, a separate Division of Low-Level Waste Management and Decommissioning was created in the Office of Nuclear Material Safety and Safeguards to allow greater management attention to low-level waste management, decommissioning, and uranium recovery activities.

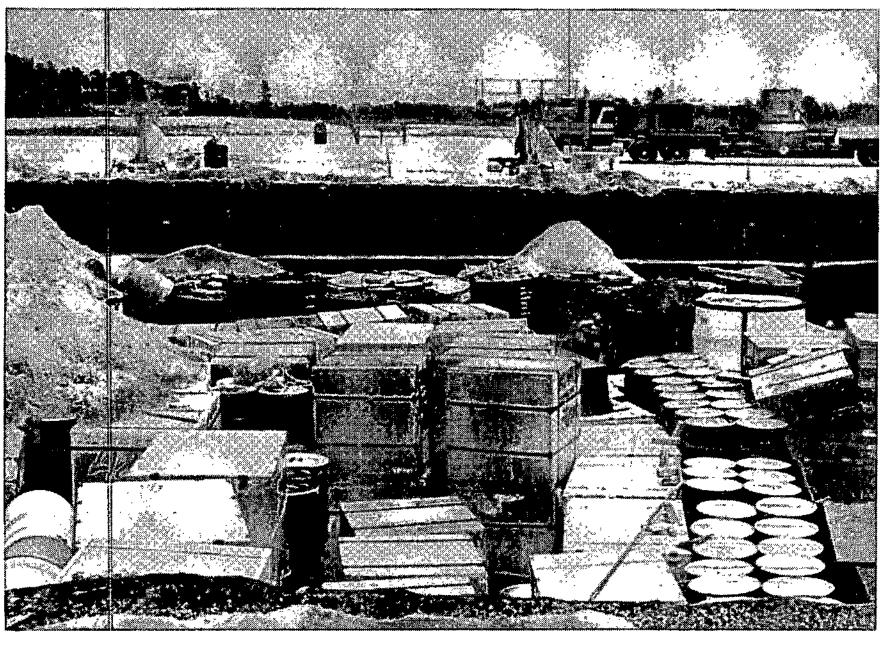
Regulation and Guidance

Throughout fiscal year 1987, NRC staff continued efforts to develop regulations and to provide guidance that will assist States and compacts in developing the low-level waste disposal capacity required by the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA).

Section 6 of the LLRWPAA authorizes the NRC to grant emergency access to any non-Federal low-level waste disposal facility, if necessary, to eliminate an immediate and serious threat to the public health and safety or to the common defense and security. Under Section 5(e)(2)(A)(ii) of the Act, if certain prescribed actions have not been taken by a State, generators within that State may be denied access to the existing low-level waste disposal facilities, beginning on January 1, 1987. The NRC may be requested to grant emergency access any time after that date.

The NRC announced in the *Federal Register* on January 15, 1987 (52 FR 1634), that it intended to develop criteria and procedures to evaluate requests for emergency access to non-Federal, low-level radioactive waste disposal facilities. The NRC plans to issue a proposed rule for public comment in late 1987, and to issue the final rule in late 1988.

In addition to the guidance described under "highlights" above, the NRC published, "Review Process for Low-Level Radioactive Waste Disposal License Application Under Low-Level Radioactive Waste Policy Amendments Act"



This is a low-level nuclear waste disposal site at Barnwell, S.C. The trench is nearly full and will soon be covered and sealed.

(NUREG-1274). The staff is developing draft regulatory guides on low-level waste classification and manifest reporting, acceptable waste forms, and site selection. The staff is also developing technical positions on site closure and environmental monitoring. These guides and positions will be available soon for public comment or as final products. Work is also under way in the areas of decommissioning wastes and the performance of concrete and steel as engineered barriers.

Assistance to States and State Compacts

The NRC is continuing an active outreach program as a means of providing guidance to States and compacts regarding regulation of new LLW disposal sites. Typical of such efforts were comments provided to California on their designated licensee's site characterization plan; guidance given to Nevada on resource requirements for license renewal; technical assistance provided to Washington for their license renewal at Hanford; and consultation provided to Michigan on class C and greater-than-class-C wastes.

Work with Other Federal Agencies

The NRC and EPA staffs continued to work on resolving the mixed low-level radioactive and hazardous waste issue to remove uncertainty regarding the applicability of the Resource Conservation and Recovery Act upon NRC-regulated activities. After briefly considering the option of Congressional action to resolve the problem, the staffs focussed their efforts on an administrative approach, and issued a series of NRC/EPA joint guidance documents on the identification of mixed LLW, siting of a mixed LLW facility, and land disposal technology. Both agencies are continuing to simplify the dual regulatory process by developing procedures for dual license and permit issuance, inspection, and enforcement.

The NRC staff consults with the DOE staff on this subject in three areas: (1) coordinating management of the national low-level commercial waste program, on such efforts as identifying alternative methods and developing data bases; (2) reviewing the closure and disposition of waste at West Valley, N.Y., under the West Valley Demonstration Project Act (see Chapter 5); and (3) reviewing DOE's policy and plans on greater-than-class-C waste disposal.

Status of Current Facilities

During fiscal year 1987, NRC staff continued working on the renewal of the special nuclear material licenses at two disposal facilities: Barnwell, S.C., and Hanford, Wash. Both sites are licensed, for source and byproduct material, by the Agreement State in which they are located. The Beatty, Nev., waste disposal site is licensed entirely by the State of Nevada. The NRC staff has provided assistance to the State of Nevada to renew the Beatty license and to develop an adequate closure plan.

The non-operating disposal facility at Sheffield, Ill., came under the jurisdiction of the State of Illinois when the State attained Agreement State status on June 1, 1987. Subsequently, staff transferred licensing files and records to Illinois, and the Commission terminated the ongoing licensing proceedings.

URANIUM RECOVERY AND MILL TAILINGS

The NRC licenses and regulates uranium mills, "heap leaching" facilities, ore-buying stations, commercial in-situ solution milling operations, and uranium extraction R&D projects. The NRC also evaluates and concurs in the Department of Energy's (DOE) Remedial Action Plans for the cleanup of inactive uranium mill tailings sites and contaminated vicinity properties. The NRC Uranium Recovery Field Office (URFO), located in Denver, Colo., enhances the agency's ability to carry out this regulatory role by virtue of its proximity to the uranium industry and the affected States.

Regulatory Development

The Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA), which was enacted to prevent or minimize environmental hazards from active and inactive mill operations, requires the Environmental Protection Agency (EPA) to develop radiation standards for mill tailings and the NRC to develop regulations for uranium recovery operations consistent with the EPA standards. The NRC promulgated its regulatory requirements for uranium mill tailings in 1980, but was embargoed by Congress from spending funds to implement its requirements until 1983, by which time EPA was mandated to promulgate its final standards. The final EPA standards were issued in October 1983. The NRC is currently completing a two-step process to conform its regulations to these standards.

The first step, completed in October 1985, modified NRC's regulations pertaining to radiological protection and long-term stabilization of mill tailings to conform to the EPA standards. The second step, remaining to be completed by NRC staff, is incorporation of the EPA ground-water standards. A proposed rule addressing ground-water protection was published July 8, 1986. The final rule, approved by the Commission, was expected to be published in November 1987.

The NRC staff has continued work on regulatory guides dealing with such topics as long-term stabilization and erosion protection for mill tailings piles, bioassay at uranium mills, meteorological measurement programs at uranium facilities, and tailings-pile cover material.

Licensing and Inspection Activities

During fiscal year 1987, the Uranium Recovery Field Office (URFO) performed 27 inspections of uranium recovery facilities. The Office issued a new commercial in-situ license for an Everest Minerals site in Wyoming. In other regulatory actions, the URFO staff completed 3 license renewals, 35 major license amendments, and 58 minor amendments to licenses.

Of the 39 NRC-licensed uranium recovery facilities, 21 are uranium mills, 4 are either heap leach or other byproduct recovery operations, 10 are research and development solution mining operations, and 4 are commercial in-situ facilities.

Only seven of the licensed facilities were in operation at the end of fiscal year 1987: four uranium mills, two research-and-development solution mining facilities, and one commercial scale solution mining facility. Given the economic state of the uranium industry, little licensing of new facilities is expected. 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 remediation for ground-water contamination.

Technical Assistance to Agreement States

Section 274 of the Atomic Energy Act of 1954, as amended, authorizes the Commission to enter into agreements with the Governor of any State to relinquish to that State the Commission's authority with respect to source materials and byproduct materials associated with uranium recovery facilities. The NRC currently has such agreements with three States: Colorado, Texas, and Washington.

The NRC conducts periodic reviews of the Agreement States' licensing and inspection programs to determine their compatibility with the NRC's programs, and provides training and technical assistance to the Agreement States to help them fulfill their regulatory responsibilities. During fiscal year 1987, the NRC reviewed the uranium recovery licensing programs of Colorado and Washington, examining the States' programs for mills, commercial solution mining facilities, and research-and-development solution mining facilities. The NRC provided technical assistance to the Agreement States, on both generic issues and site-specific licensing issues, and conducted two generic and 11 site-specific reviews.

Remedial Action at Inactive Sites

The NRC has continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive mill tailings sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act (UMTRCA). The NRC is a cooperating agency and is required by UMTRCA to concur in remedial actions planned by DOE for inactive mill tailings properties. The NRC reviewed and commented on modifications to cooperative agreements with North Dakota, Colorado, Wyoming, Pennsylvania, and Texas. Comparative Analysis of Disposal Site Alternatives Report (CASDAR) and NEPA documents reviewed by NRC included those for Ambrosia Lake, N.M.; Green River and Mexican Hat, Utah; Slick Rock, Rifle, and Grand Junction, Colo.; Belfield/Bowman, N.D.; Falls City, Tex.; Lowman, Idaho; Spook and Riverton, Wyo.; and Tuba City, Ariz. Remedial action plans and modifications reviewed by NRC in fiscal year 1987 included Slick Rock and Rifle, Colo.; Lakeview, Ore.; Shiprock and Ambrosia Lake, N.M.; Salt Lake City, Green River, and Mexican Hat, Utah; and Tuba City, Ariz. Conditional concurrences in the selection of remedial action were provided for the Durango and Riverton, Colo., sites. The Canonsburg, Pa., final certification of remedial action, and the Shiprock, N.M., draft certification, were reviewed and commented on by NRC. Construction design reviews and site inspections included Canonsburg, Lakeview, Shiprock, Salt Lake City, Tuba City, Grand Junction, Rifle, Riverton, and the Burrell, Pa., Vicinity Property.

Generic efforts included: revisions of the Memorandum of Understanding with DOE, investigation of an infiltration model to be used in the UMTRA project, investigation of the feasibility of co-disposal of UMTRAP and active site uranium milling waste, and revision of the soil cleanup verification methods.

The NRC also reviewed DOE generic documents, including the Environmental Health and Safety Plan and the Vicinity Property Management and Implementation Manual. The NRC has continued to review Vicinity Property Radiological and Engineering Assessments (REA) where DOE has proposed the use of supplemental standards. The NRC has concurred in the use of supplemental standards at vicinity properties associated with the Lakeview, Riverton, Durango, Salt Lake City, and Canonsburg sites.

Communicating with Government and the Public

Chapter 8

As part of NRC's extensive reorganization, implemented in April 1987 (see Chapter 1), the Office of Governmental and Public Affairs (GPA) was created. This office consolidates the former Offices of Congressional Affairs, Public Affairs, International Programs, and State Programs. The new office, which reports directly to the Commission, is responsible for the agency's cooperation and communication with persons, organizations, and institutions outside the agency. Specifically, GPA provides liaison with the Congress, information to the general public and news media, participation in the international nuclear community, and liaison with State and local governments and Indian Tribes. This chapter covers GPA activities in the first year of operation for the office.

PUBLIC COMMUNICATION

Public Information

GPA provided information on the activities and policies of the NRC by talking with reporters at public meetings of the NRC Commissioners and NRC staff, arranging interviews and press briefings, and responding to numerous telephone inquiries. The office also issued more than 300 public announcements on Commission programs and actions, including such matters as proposed fines against licensees, regulation changes, and public hearings. While the primary audience for the public announcements is the news media, they are also received by the scientific community, the industry, and members of the general public.

Partners in Education. About 110 NRC employees participated in an NRC program coordinated by GPA that furnishes volunteer services to the public school system in Maryland's Montgomery County. In keeping with the spirit of the National Partnership in Education Program initiated by the President in 1983, the volunteers lectured in the classroom, tutored students, served as mentors during career-awareness field trips, and assisted as judges at science fairs. The NRC volunteers have backgrounds in law, engineering, mathematics, physical sciences, accounting, biology, and health sciences.

Media Seminars. For the seventh consecutive year, the NRC, through GPA, continued its one-day education workshop program for reporters and editors on the fundamentals of nuclear power and the risks of exposure to radiation.

Headquarters Public Document Room

Persons interested in detailed information about commercial nuclear facilities have found the NRC's principal Public Document Room (PDR) a rich source of useful materials. The PDR is tentatively scheduled to relocate to 2120 L Street, N.W., Washington, D.C., on or about June 1, 1988. This specialized documentation center houses significant documents on nuclear regulation which have been made available to the public. 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: NRC reports; transcripts and summaries of meetings; licenses and their 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 accession listings and an on-line bibliographic data base available for staff and public use.

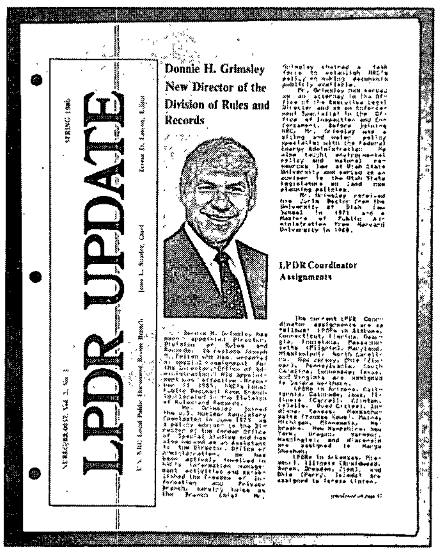
The Headquarters PDR contains about 1.5 million documents, and the collection is enlarged by an average of 274 new items every day. During an average month, the PDR serves about 1,120 users. The staff retrieves an average of 3,245 files of documents or microfiche per month for researchers on-site and provides about 2,100 documents in response to letters and telephone requests. The public purchased 3.9 million pages of documents and about 7,750 microfiche cards in fiscal year 1987. During an average month, there were about 2,500 user sessions on the PDR's on-line computer data base.

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 Root, Washington, D.C. 20555. 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 for users in how to search the PDR automated bibliographic retrieval system (an on-line card catalogue).

NRC Public Document Rooms, located in Washington and at 94 locations across the United States, provide access to the entire range of publicly available NRC documents. In 1987, arrangements neared completion for a major demonstration project, involving a vast expansion of accessible documentation through LPDR computer installations, as described on the facing page. The nerve-center for the project is in the offices of the Local Public Document Room Branch in the NRC Office of Administration and Resources Management (ARM) in Bethesda, Md. Shown at left are Margaret Sheehan, LPDR Coordinator, at the microfiche reader-printer, with Vivian Reid, Branch Secretary. The project is directed by Jona Souder (inset), LPDR Branch Chief.

Each quarter, the LPDR Branch publishes a newsletter to keep the hundreds of librarians at LPDRs across the country up-to-date on program developments. Teresa Linton, below, is editor of the newsletter, a sample of which is shown to the left.

Representative of an LPDR is the one housed in the Wharton County Junior College Library, shown bottom right, in Wharton, Tex., which contains about 80 feet of "hardcopy" documents, plus microfiche files, related to the South Texas nuclear power plant. Library Director Patsy Norton, bottom left, maintains the LPDR collection.

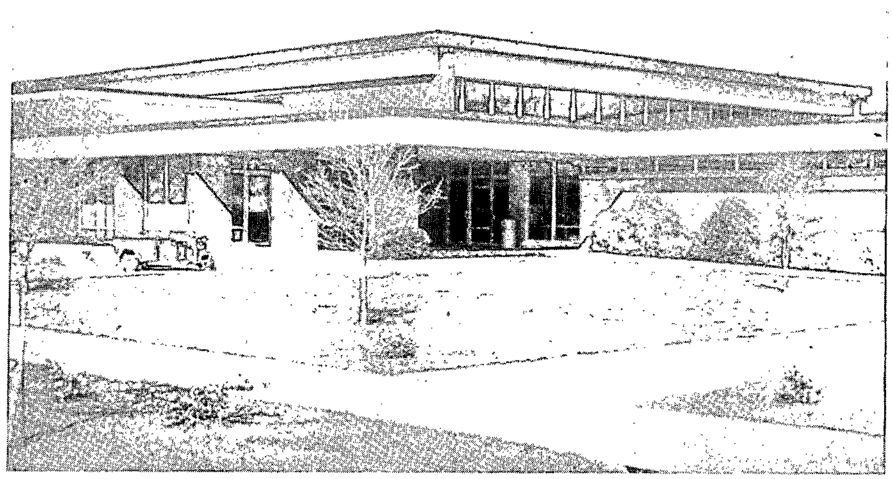


SPRING 1986

LPDR UPDATE

Donnie H. Grimsley New Director of the Division of Rules and Records

LPDR Coordinator Assignments



Local Public Document Rooms

The NRC's Office of Administration and Resources Management (ARM) staff neared completion in 1987 of a demonstration project that will expand public access to the agency's bibliographic files. By year's end, ARM personnel in the Local Public Document Room (LPDR) Branch projected that by February 1988 a total of six LPDR libraries across the country would have on-line computer-terminal access to the NRC document control system, newly designated NUDOCS, and thus to all publicly available records from 1981 forward. This will be the first time the public will be able to access this data base from remote locations. The six LPDR libraries selected for the demonstration project are Louisiana State University, Baton Rouge, La. (LPDR for the River Bend nuclear power plant); California Polytechnic State University, San Luis Obispo, Cal. (LPDR for the Diablo Canyon nuclear power plant); State Library of Pennsylvania, Harrisburg, Pa. (LPDR for the Three Mile Island and Peach Bottom nuclear power plants); Monroe County Library System, Monroe, Mich. (LPDR for the Fermi nuclear power plant); University of North Carolina at Charlotte, N.C. (LPDR for the McGuire nuclear power plant); and the White Plains Public Library, White Plains, N.Y. (LPDR for the Indian Point nuclear power plant). Selection of these six was based on geographic distribution and LPDR usage factors, as well as to achieve a balance of academic and public library facilities in the sample. Budget restraints precluded a larger trial program.

Each of the six LPDRs will be provided with a computer terminal and printer, a telecommunications hook-up, and a microfiche file of post-1981 records related to commercial nuclear power plants, fuel-cycle facilities, and waste disposal facilities. Other material received or generated by the NRC in its regulatory role will also be accessible.

LPDR librarians involved in the project were brought to Washington for several days' training and were prepared to begin operations in February 1988 at the six locations mentioned. They will be able to assist members of the public in accessing the system and searching for desired information by several different methods, including subject searches employing Boolean logic. The demonstration project will be evaluated after six months to assess the prospects for expansion to others of the 94 LPDRs in the NRC program. (See Appendix 3 for a roster of all LPDRs.)

Since its inception in 1971 under the Atomic Energy Commission, the Local Public Document Room Program has grown in both size and importance. The 94 LPDRs of 1987 include 76 in communities near nuclear power plant sites (seven of these document rooms also serve fuel or waste disposal facilities); six LPDRs are dedicated to fuel-cycle or waste facilities; and 12 "mini-LPDRs" maintain limited data collections for a limited time, usually in support of the NRC hearings process. In the 16 years of its existence, the LPDR system has become the principal mechanism for providing citizens located near nuclear installations access to NRC safety-related documents, serving as the primary source of information for the news media, intervenors, and local groups representing a wide diversity of opinion. At the close of the report period, the NUDOCS data base included more than a million publicly available records, such as inspection reports, emergency plans, safety analyses reports, licensee event reports, and environmental records. Reports on enforcement and antitrust matters are also available, where appropriate. Since 1981, the NRC has provided financial support to most of the libraries maintaining full-service LPDR collections at power reactor sites. In 1981, 28 libraries received NRC funds totaling approximately $76,000. By 1987, support has grown to more than $230,000 for 66 libraries for the maintenance of more than 4,000 linear feet of LPDR records and for assistance to patrons in locating information on nuclear facilities in the area. Local librarians and their patrons may use a toll-free telephone number, 1-800-638-8081, for assistance and advice from NRC Headquarters on collection content, search strategies, and the use of reference tools and indices.

Commission History Program

The Commission History Program studies the origins and evolution of regulatory policies and programs. The History Office has begun preparing a sequel to its book, Controlling the Atom: The Beginning of Nuclear Regulation, 1946-1962, published in 1984 by the University of California Press. The new volume will cover the period from 1963 into the early 1970s, a time of vital change and controversy over the commercial development and regulation of nuclear power. Like the first volume, it is intended to serve as a reference for general readers as well as the agency staff.

CONGRESSIONAL OVERSIGHT

Public concern over nuclear issues is reflected in the significant number of Congressional hearings during the report period involving the NRC. During fiscal year 1987, the NRC participated in 24 hearings, listed below.

COOPERATION WITH THE STATES

The NRC's contacts with regional, State and local agencies, and Indian Tribes, for purposes other than inspection and enforcement or emergency planning are administered through State, local and Indian Tribe Programs (SLITP) of GPA. These include the State Agreements Program and various liaison and cooperative programs that are administered in accordance with policies and procedures established by Headquarters and implemented primarily by the Regions.

Table 1. Congressional Hearings Involving the NRC—FY 1987

Date	*Committee*	*Subject*
10/1/86	Committee on Energy & Commerce, Subcommittee on Oversight & Investigations (House)	TVA Management
11/18/86	Committee on Energy & Commerce, Subcommittee on Energy Conservation & Power (House)	Status of Emergency Planning for Seabrook
2/5/87	Committee on Interior & Insular Affairs Subcommittee on Energy & the Environment (House)	FY 1988 NRC Budget Request
2/18/87	Committee on Environment & Public Works (Senate)	NRC FY 1988 Budget Request
3/17/87	Committee on Science, Space & Technology, Subcommittee on Energy Research and Development (House)	DOE's Nuclear Fission Budget Request for FY 1988
3/19/87	Committee on Appropriations Subcommittee on Energy and Water Development (House)	NRC FY 1988 Budget Request
3/27/87	Committee on Interior & Insular Affairs Subcommittee on Energy & the Environment (House) Committee on Energy and Commerce Subcommittee on Energy & Power	Price-Anderson
4/9/87	Committee on Governmental Affairs (Senate)	Inspector General Bill
4/21/87	Committee on Energy and Commerce Subcommittee on Energy and Power (House)	NRC FY 1988 Authorization
4/23/87	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	Decommissioning
4/28/87	Committee on Interior & Insular Affairs Subcommittee on Energy and the Environment (House)	Emergency Preparedness Rule Change
4/28/87	Committee on Energy & Natural Resources (Senate)	High-Level Waste
4/30/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Price-Anderson
5/6/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation Topics (Senate)	Oversight/Emergency Preparedness & Related
5/7/87	Committee on Interior & Insular Affairs Subcommittee on Energy & the Environment (Senate)	NRC Legislative Proposals & Arizona/South Dakota LLW Compact

Table 1. Congressional Hearings Involving the NRC—FY 1987 (Continued)

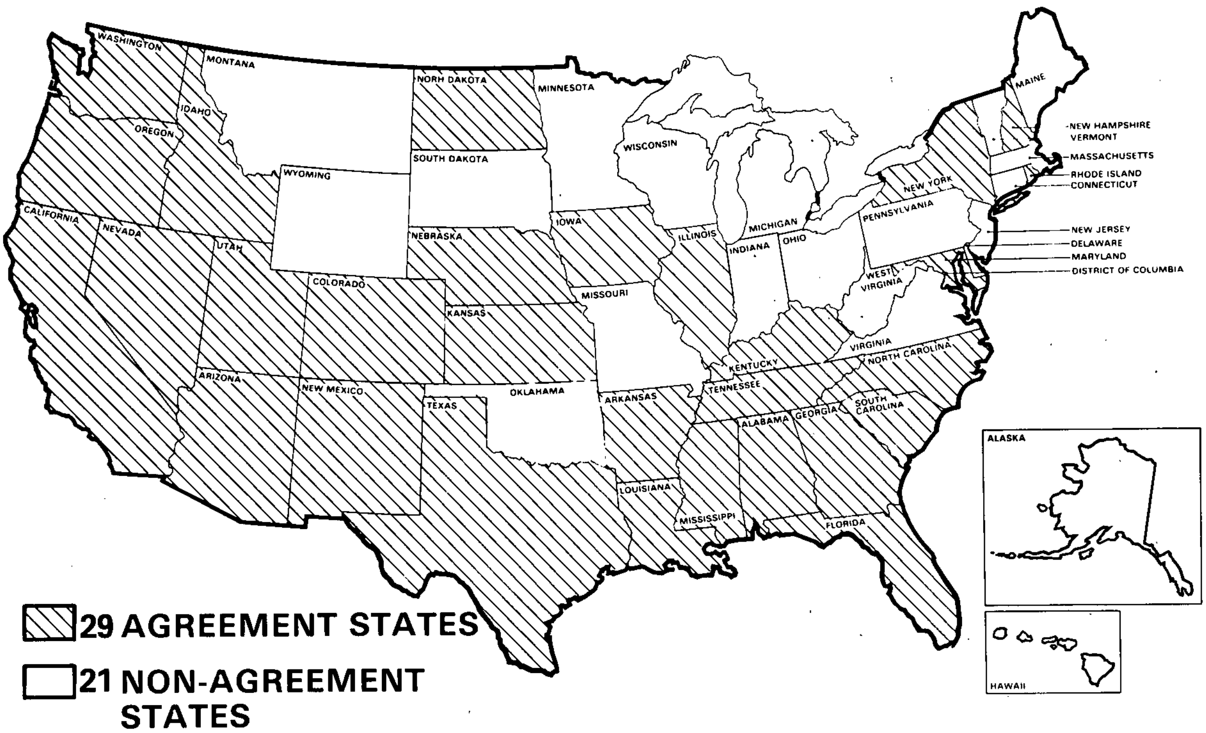
Date	*Committee*	*Subject*
5/12/87	Committee on Commerce, Science, & Transportation Subcommittee on Surface Transportation (Senate)	Reauthorization of Hazardous Materials Transportation Act
5/14/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Ex Parte Procedures in Shoreham Proceeding
6/2/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	High-Level Waste
6/3/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Hearing to Receive Testimony from Comm. Thomas M. Roberts
6/3/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Waste Transportation
6/11/87	Committee on Interior and Insular Affairs Subcommittee on General Oversight & Investigations (House)	Effectiveness/Performance of OI & OIA
6/18/87	Committee on Environment & Public Works Subcommittee on Nuclear Regulation (Senate)	Monitored Retrievable Storage
6/29/87	Committee on Energy & Natural Resources (Senate)	HLW Geologic Sites — 1st Repository
7/9/87	Committee on Environment & Public Works (Senate)	Nomination Hearing of Dr. Kenneth C. Rogers

State Agreements Program

By formal agreement with the NRC, a total of 29 States have assumed regulatory responsibility over byproduct and source materials and small quantities of special nuclear material. The latest (29th) agreement, with Illinois, became effective on June 1, 1987. Negotiations for an Agreement with the State of Maine are under way. At the end of fiscal year 1987, there were about 15,000 radioactive material licenses in these Agreement States; they represent about 65 percent of all the radioactive materials licenses in the United States. (See map of Agreement States in this chapter.)

Review of State Regulatory Program. The NRC is required by the Atomic Energy Act of 1954 to periodically review Agreement State radiation control programs and confirm that they are adequate to protect public health and safety and are compatible with NRC programs. The reviews follow the guidelines contained in a Commission Policy Statement which underwent minor revisions and updates. The revised Policy Statement was published in the Federal Register on June 4, 1987. Any problems identified in these reviews are brought to the attention of State authorities with recommendations for corrective action. Twenty-one routine program reviews and two follow-up reviews were conducted in 1987. As part of the program review, the NRC technical staff accompanied State inspectors to State-licensed facilities to evaluate inspector performance and reviewed selected license and compliance casework in detail. Follow-up reviews of the status of previously identified program deficiencies were conducted in four States in 1987. An orientation meeting was held with the newest Agreement State, Illinois, and special meetings were held with two States where significant changes in State management occurred.

AGREEMENT STATE PROGRAM



NRC Technical Assistance to States. The NRC provided technical assistance to Agreement States during 1987 in the areas of licensing, inspection, enforcement, and proposed statutes and regulations. Technical assistance can range from responding to telephone requests for technical information to assisting in State reviews of license applications and State inspections. Agreement States are expected to maintain a core staff knowledgeable in materials radiation safety and regulation and can also utilize in-State technical resources, such as advisory committees and consultants. Special or unusual radiation applications, however, may present radiation safety problems that need specialized expertise or knowledge. For States evaluating such problems, the availability of NRC expertise is a valuable technical resource. Examples in 1987 of NRC technical assistance would include assistance to California in the evaluation of an airport explosives detector which uses Cf-252, a neutron source, as well as work done with several States in evaluating the special radiation hazards associated with site-specific therapy, a nuclear medicine procedure which requires the use of large quantities of unsealed radioisotopes.

Training Offered by NRC. State radiation control personnel regularly attend NRC-sponsored courses to improve their technical and administrative skills and, thus, their ability to maintain high quality regulatory programs. In 1987, the NRC sponsored 11 short-term training courses, attended by 206 State personnel. Courses included health physics, industrial radiography safety, nuclear medicine procedures, introduction to licensing practices, inspection procedures, well logging, transportation of low-level radioactive waste, and other nuclear materials. A combined workshop and training course was also held with NMSS and was attended by regional and headquarters NRC staff and State staffs. On-the-job training in licensing and compliance is also given to individual staff members either in the States or through visits to NRC regional and headquarters offices.

Annual Agreement States Meeting. The annual meeting of Agreement State radiation control program directors was held in October 1987 in Louisville, Ky. The site was chosen to commemorate the 25th anniversary of the first State Agreement, made with Kentucky. Chairman Zech participated in the commemorative ceremonies which included a proclamation by Governor Martha Layne Collins. Technical issues covered in the meeting included low-level waste disposal, materials licensing and compliance, and new regulatory developments in industrial radiography and nuclear medicine safety.

Regulation of Low-Level Waste. The NRC continues to provide technical assistance to States in their programs for regulating low-level radioactive waste. Assistance was given to North Carolina, New York, and Nebraska on the promulgation of low-level regulations compatible with NRC. In addition, the NRC provided technical assistance to the States of California, Pennsylvania, Michigan, Nebraska, Kansas, and Arkansas in establishing their low-level regulatory programs and meeting the requirements under the Low-Level Radioactive Waste Policy Amendments Act of 1985. South Carolina, Washington, and Nevada continue to participate in the NRC review of several topical reports on high integrity containers waste solidification processes and computer codes used in implementing 10 CFR Part 61.

Regulation of Uranium Milling. The NRC continues to assist Agreement States in their programs for regulating uranium milling. This assistance has included guidance on surety arrangements and on the Environmental Protection Agency requirements. Direct technical assistance on specific cases to the States of Colorado, Texas, and Washington has also been arranged. Five representatives—from Texas, Washington, Utah, New Mexico, and Wyoming—participated in a Workshop on Reclamation of Uranium Mill Tailings in June 1987.

Special Projects. State Agreements program staff continued to closely study radioactive steel contamination incidents. A report of their studies was published as the lead article in the October 1986 issue of Health Physics. The staff also developed a hazard scrap warning poster (NUREG/BR-0108) to alert steel scrap and mill workers to the possibility of radioactive sources' becoming inadvertently mixed with scrap metal. The poster was patterned after one published by the Canadian Atomic Energy Control Board. Distribution was made to 4,000 U.S. steel scrap dealers and mills using mailing lists provided by the Occupational Safety and Health Administration and the Institute of Scrap Iron and Steel.

State, Local, and Indian Tribe Liaison Activities

NRC's Strategic Plan calls for the agency to assume a more proactive, as distinct from a reactive, role that includes outreach activities, to increase cooperation and communication between NRC and State and local governments and agencies and Indian Tribe representatives to promote increased awareness and understanding of activities related to nuclear safety.

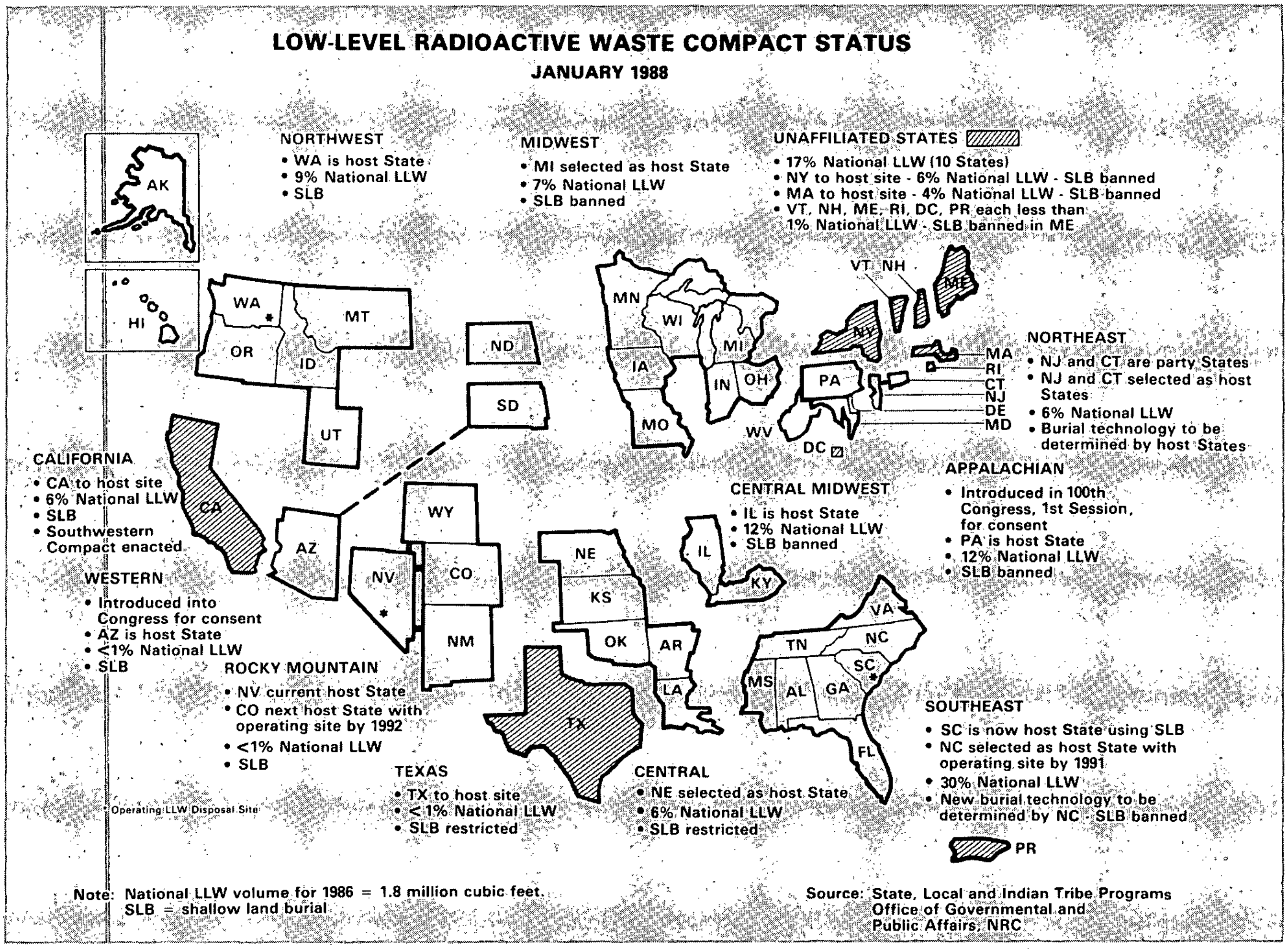
Cooperative Instruments with States. Certain State officials have long felt a strong need to better understand risks to public health and safety from incidents at nuclear power reactors, and to assure that all reasonable steps are being taken to prevent an incident or otherwise to reduce such risks. These feelings were reinforced by the accident at Three Mile Island in 1979, and they are often accentuated locally when problems occur at nuclear power plants. The accident at Chernobyl has and will continue to fuel the need for deeper understanding of these risks. Moreover, some State governments do not want to depend solely on NRC for information on reactor status, and Governors and other State officials are seeking ways in which they can routinely be apprised of the current status of specific NRC-licensed facilities that have a potential for affecting the health and safety of their citizens.

In recent years, States have become more interested in negotiating memoranda of understanding and letters of agreement with NRC to allow them to become more involved in the regulation of nuclear facilities within their borders and in adjacent States. During the past year, NRC has negotiated a broad memorandum-of-understanding with the State of Pennsylvania, and a more specific subagreement which allows the State to inspect low-level waste packaging and transport activities on the premises of NRC-licensed facilities.

Low-Level Radioactive Waste Compacts. The Low-Level Radioactive Waste Policy Amendments Act of 1985, enacted January 15, 1986, ensures that currently operating disposal facilities will remain available until the end of 1992, subject to specified volume limitations and other requirements; establishes a system of incentives and penalties to promote steady progress toward new facility development; and, under Title II, grants consent to seven interstate low-level waste (LLW) disposal compacts, covering 37 States (see "Waste Compact Status" map in this chapter). At the close of the report period, Congress was considering the Appalachian Compact and the Western Compact. The Act also directs NRC to provide additional guidance to the States to ensure that they have enough regulatory information to meet the milestones established by the Act. Some of the information States need includes guidance on waste disposal methods that can be used as an alternative to shallow land burial, on the licensing of facilities, and on determining what waste is below regulatory concern. In addition, NRC is working with the Environmental Protection Agency to provide guidance to the States for the disposal of mixed waste (LLW mixed with chemically hazardous waste). NRC also assists the States in the review of compacts and enabling legislation and provides States with training and technical assistance.

State Liaison Officers. The NRC is continuing its program of communicating directly with the Governor-appointed State Liaison Officers (SLOs) from all 50 States and Puerto Rico. The State Liaison Officer program was created to provide a direct communication channel between the States and NRC. The SLO is intended to be the key person in the State to keep the Governor informed of nuclear regulatory or emergency matters of interest, to keep other State officials informed of such matters, and to respond to periodic inquiries from the NRC.

In September 1987, the NRC hosted a national meeting for all the State Liaison Officers in Bethesda, Md. The



meeting featured presentations by invited State speakers and NRC officials, with discussion formats encouraging an open exchange for all in attendance. Discussions focused on issues such as emergency preparedness, including the background and purpose of NRC's proposed rule on off-site emergency planning; State views on coordination of Federal, State and local emergency planning procedures; State and NRC activities regarding regulation of nuclear facilities; national low-level waste trends; individual States' experiences in the low-level waste compacting process; and NRC's high-level waste program. Also discussed at the meeting were issues such as power plant aging/plant life extension, economic incentives for utilities, the Chernobyl implications report, and personal perspectives offered by Harold Denton, Director of GPA, deriving from his visit to the Soviet Union. The meetings offered an excellent opportunity for State officials to exchange information, on a broad spectrum of issues, with NRC staff and among themselves.

Outreach Activities. In line with the agency's commitment to enhance its relationship with States and their organizations, NRC sponsored an audio-conference—in cooperation with the National Conference of State Legislatures (NCSL)—for State legislators and their staff, in May 1987. Representatives of the Division of Low-Level Waste Management and Decommissioning, Office of Nuclear Materials Safety and Safeguards (NMSS), briefed the representatives from eight States on issues such as low-level waste management options, financial liabilities and sureties, alternative disposal technologies, Class C waste, and "Below Regulatory Concern (BRC)" waste. Feedback from the participants indicated the audio-conference was extremely successful, and because it proved to be an effective and low-cost method of communicating with the States, it will be used more frequently in the future. In November 1986, NCSL also arranged a trip to the Barnwell, S.C., repository for State legislators, where NMSS staff discussed NRC's LLW regulations.

NRC Regional Offices I (Philadelphia) and III (Chicago) held workshops in their respective Regions during the year, focusing on emergency preparedness issues. Representatives of the States, utilities, and other Federal agencies participated in the workshops, which covered clarification of roles and interactions among the various organizations responsible for implementing emergency response procedures during an event at a nuclear facility.

Liaison With American Indian Tribes

The President encouraged Federal agencies, in his January 24, 1983 Indian Policy Statement, to interact with Indian Tribes on a government-to-government basis. Under the Nuclear Waste Policy Act of 1982, three Indian Tribes have been deemed "affected" parties, because of the proposed location of a high-level radioactive waste repository at Hanford, Wash. The three affected Tribes are: (1) the Yakima Indian Nation, Washington; (2) the Confederated Tribes of the Umatilla Indian Reservation, Oregon; and (3) the Nez Perce Tribe, Idaho. The NRC, in its role as the licensing agency, has met frequently throughout the year with the three Tribes in a number of forums, including a Commission meeting (June 16, 1987, in Washington, D.C.), the 2nd Annual NMSS meeting with States and Tribes (June 30, Washington, D.C.), Licensing Support System (LSS) Advisory Committee meetings (August 5-6 and September 15-16, Washington, D.C.), National Congress of American Indians' (NCAI) National Indian Nuclear Waste Policy Committee meetings (March 10, Washington, D.C. and September 21-24, Tampa, Fla.), and on the reservations as well (September 9-11). With regard to the LSS, it should be noted that NCAI is a member of the first tier of the Advisory Committee, which has voting membership concerning proposals for consensus on the proposed rule applying to the submission and management of records and documents related to the high-level waste repository licensing proceeding. NCAI represents all Tribes affected by the siting of a second repository and by the transportation of high-level radioactive waste.

During a visit to Canada in May 1987, NRC Chairman Lando W. Zech, Jr. (left) and Director of Governmental and Public Affairs Harold R. Denton (right) were briefed on the design and operation of Canada's CANDU heavy water nuclear power reactor by Ken Elston, Operations Manager of the Bruce Nuclear Power Development on the southeast shore of Lake Huron in Ontario. The visit also included a visit to the Pickering Generating Station, north of Toronto.

INTERNATIONAL ACTIVITIES

The objectives of the NRC's international activities are to improve world-wide cooperation for nuclear safety and to assist the U.S. Government's effort to deter further proliferation of nuclear explosives capability in the world, especially such as might result from U.S. nuclear exports. The NRC coordinates its international activities through the international programs of GPA, and other NRC offices participate in these activities by contributing technical expertise and conducting research, both at home and overseas.

Highlights of Fiscal Year 1987

- Participated in the anniversary session of the International Atomic Energy Agency's (IAEA) Thirtieth General Conference held in Vienna, Austria, September 19-23.
- Renewed bilateral nuclear safety cooperation arrangements with The Netherlands and Switzerland.
- Continued to expand its network of mutually beneficial agreements on nuclear safety research, initiating the International Piping Integrity Research Group (IPIRG) program to investigate the behavior of degraded piping under service conditions.
- Arranged visits by representatives from foreign governments and from public and private organizations overseas for discussions of nuclear safety issues.
- Undertook the first bilateral interchange with the U.S.S.R. on nuclear safety since 1978, with Commissioner Bernthal leading a U.S. delegation to the Soviet Union in March 1987, and the U.S.S.R. reciprocating with a team in October 1987.
- Sent experts to Mexico, Egypt, South Korea, West Germany, and Yugoslavia in support of the technical assistance programs of the IAEA to provide safety advice in their nuclear programs.
- Partially sponsored the IAEA Operational Safety Review Team (OSART) mission to the Calvert Cliffs (Md.) nuclear power plant.

- Participated in OSART missions in Mexico, West Germany, and South Korea.
- Hosted 65 visitors from 18 countries and international organizations to observe the activities associated with the Zion Federal Field Exercise (Ill.).
- Issued 129 export licenses and 25 amendments to existing licenses.
- Worked closely with the Executive Branch and the IAEA in strengthening international safeguards and physical security. The NRC sent experts to Japan, France, West Germany, the United Kingdom, the European Community, and Australia for discussion in these areas.

International Cooperation

Bilateral Information Exchange Arrangements. The NRC participates in a wide-ranging, mutually beneficial program of information exchange and cooperative safety and research activities with its counterparts in the international community. Since 1974, when it instituted the program, the NRC has conducted most of its technical information exchanges through a series of general safety cooperation arrangements formally concluded with the regulatory authorities of Belgium, Brazil, China, Denmark, Egypt, Finland, France, West Germany, Greece, Israel, Italy, Japan, South Korea, Mexico, the Netherlands, the Philippines, Spain, Sweden, Switzerland, the United Kingdom, Yugoslavia, and Taiwan. These 22 arrangements involve, as full and active exchange partners, all the countries except India which are operating light-water reactors of U.S. origin, all countries with reactors of U.S. design under construction, and several countries which at some time in the past have seriously considered making a commitment to U.S. nuclear technology. (With respect to India, the NRC has exchanged letters agreeing to share information on accidents and incidents at nuclear facilities, rather than on the full spectrum of safety activities.)

The primary objective of these arrangements is to establish a formal channel for communication with foreign nuclear regulatory organizations to assure 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, and they serve as the vehicle for the NRC to provide assistance in improving nuclear health and safety practices to developing countries operating power reactors supplied by the United States. The bilateral arrangements are effective for five years as written but contain provisions for renewal by mutual agreement.

In 1987, the NRC's arrangements with the Netherlands and Switzerland were renewed. Negotiations on the renewals of existing arrangements with Belgium and Italy were concluded, with the texts awaiting signature at the close of the report period, and discussions were under way with South Korea and Mexico.

Bilateral and Multilateral Safety Research Agreements. The NRC is currently involved in about 55 agreements for research cooperation, with 17 countries, in ongoing nuclear safety research projects both in the U.S. and overseas. These research projects cover a wide range of activities, including the direct contribution of data and analyses needed to confirm and assess computer codes used in the NRC licensing and regulatory process. (See the 1985 NRC Annual Report, pp. 139 and 140.)

In a major venture in January 1987, NRC's Office of Nuclear Reactor Regulation (NRR) initiated the International Piping Integrity Research Group (IPIRG) program to develop data needed to verify engineering methods for assessing the integrity of nuclear power plant piping. The program—composed of five tasks that will require a minimum of three years to complete—is designed to build on the NRC Degraded Piping Program, and will include sufficient research and testing to achieve a reasonably complete understanding of the behavior of degraded piping under service conditions. It is currently being funded by Canada, France, Japan, Sweden, Switzerland, the United Kingdom, the United States (by NRC and the Electric Power Research Institute), and Taiwan. Should the IPIRG program results justify simplifying piping and piping support, significant savings in the construction of new power plants could be realized.

Activities with the U.S.S.R. In March 1987, a U.S. Government delegation led by Commissioner Frederick M. Bernthal visited the Soviet Union for discussions and activities related to nuclear safety. This was the first bilateral interchange with the U.S.S.R. on nuclear safety since 1978. The U.S. delegation was made up of representatives from the U.S. Nuclear Regulatory Commission, Department of Energy, and National Institutes of Health. Useful discussions on numerous topics within the four broad technical areas listed below were undertaken with a view to identifying areas for possible future cooperation. The broad technical areas were:

- Nuclear safety regulation, policy, and practices.
- Aspects of safe power plant operation.
- Safety research.
- Health care and environmental protection.

While in the U.S.S.R., the U.S. delegation visited the Kurchatov Institute of Atomic Energy, the Ministry of Health of the Ukrainian S.S.R., the Izhora Nuclear Components Production Plant, the Leningrad Division of the

Scientific Research and Design Institute, the All-Union Scientific Institute for Operation of Atomic Power Plants, and the Beloyarsk, Chernobyl, and Zaporozhiye atomic power stations. The visits were valuable in updating the U.S. on the status of safety standards and practice in the Soviet Union.

A reciprocal visit to the U.S. by a Soviet delegation took place in October 1987, after the close of the report period. The U.S.S.R. delegation was headed by Alexander L. Lapshin, Deputy Minister, Ministry of Atomic Power, and included other representatives from the State Committee for Supervision of Nuclear Power Safety, State Committee for Utilization of Atomic Energy, Kurchatov Institute of Atomic Energy, the All-Union Nuclear Power Plant Research Institute, and the Atomenergoproject. In discussions in Washington, each side described its activities relating to nuclear plant safety and outlined approaches for improving these areas in the future. The U.S.S.R. delegation visited a number of nuclear facilities in the U.S., including the Electric Power Research Institute; Bechtel Group, Inc.; the Institute for Nuclear Power Operations; the Westinghouse Electric Corporation; Brookhaven National Laboratory; the U.S. NRC Region II Office (Atlanta); and the LaSalle (Ill.), McGuire (N.C.), and Three Mile Island (Pa.) nuclear power plants. Additionally, Dr. Ponomarev-Stepnoi of the Soviet delegation presented a paper at the annual NRC Water Reactor Safety Meeting on "Improvement of Safety of Nuclear Power Plants in the U.S.S.R."

The Soviet side presented a proposal for future U.S.-U.S.S.R. cooperation on a wide range of nuclear safety issues. The U.S. welcomed this proposal and will review and consider it in detail during the coming months in preparation for the 7th meeting of the U.S.-U.S.S.R. Joint Committee on Peaceful Uses of Atomic Energy, tentatively scheduled for March 1988 in Washington, D.C.

Regulatory Exchange with Japan. In fiscal year 1987, the NRC hosted the Third Regular Meeting with its Japanese regulatory counterpart, the Agency of Natural Resources and Energy (ANRE). Coordinated licensing review of advanced light water reactors, measures to counter intergranular attack (IGA) in steam generator tubes, a review of plant maintenance programs, and the regulatory implications of the Chernobyl accident were among the wide range of regulatory and other technical issues discussed at this meeting. Such regularized exchanges have become useful forums for the sharing of regulatory experiences and perspectives and help lay the foundation for other in-depth exchanges throughout the year on individual technical issues. Preparations are under way to prepare for the Fourth Regular Meeting, scheduled for May 1988 in Tokyo.

International Emergency Preparedness Cooperation. Considerable international interest was focused upon the Federal Field Exercise (FFE) conducted at the Zion (Ill.) nuclear power plant in June 1987. (See Chapter 2 for description of the exercise.) Sixty-five visitors from 18 countries and international organizations observed the activities associated with the FFE. Represented were Federal-level regulatory authorities, as well as municipal government officials with a broad spectrum of responsibilities related to emergency planning and response coordination.

At the time of the FFE, the NRC took the opportunity to test communications and notification procedures with its regulatory counterparts in Canada and Mexico. The NRC also worked closely with the Department of State during the FFE to test procedures for the notification of the IAEA

NRC Commissioner Frederick M. Bernthal headed a U.S. delegation to Moscow in March 1987 to discuss nuclear safety matters and to visit Soviet nuclear facilities. In this photo of a Moscow meeting are, seated at table (left to right), Harold R. Denton, Director of the NRC's Office of Governmental and Public Affairs; Commissioner Bernthal; and Andronik M. Petrosyants, Chairman, U.S.S.R. State Committee for the Utilization of Atomic Energy, and head of the Soviet delegation. The occasion represented the first bilateral exchange of safety views between the two countries since 1978.

In October 1987, a Soviet delegation repaid the U.S. visit to the U.S.S.R. in March and, on an extended trip through the U.S., visited several nuclear power plants, a national laboratory, and NRC Headquarters and Regional Offices. Shown here, during a visit to the LaSalle nuclear power plant in Illinois are (left to right) Evgenie P. Larin, All-Union Nuclear Power Plants Research Institute; Anatoli Beliaev, State Committee for the Supervision of Nuclear Power Safety; Alexander L. Lapshin, Deputy Minister for Nuclear Power and head of the U.S.S.R. delegation; Cordell Reed, Senior Vice President for Nuclear Operations, Commonwealth Edison Co.; and Mikhail V. Nikitin, Protocol Office and interpreter for the delegation.

as required under the international safety convention on early notification of a nuclear accident, which was signed by the U.S. in September 1986 and submitted to Congress for ratification.

International Exchange of Information on Nuclear Waste Management. The NRC has been exploring the possibility of convening, sometime in 1988, a meeting of regulators from countries with active programs for disposing of high-level nuclear waste in geologic repositories. The objectives of such a meeting would be to: (1) identify generic regulatory and licensing issues for each country, and how they are being handled; (2) establish, if feasible, an international consensus on dealing with such issues; and (3) identify further licensing issues and concerns that could benefit from international collaboration. A key goal would be to identify to the extent possible the principal similarities and dissimilarities between the approaches adopted by various countries and to gain a better understanding of the reasons for any differences. NRC's Office of Nuclear Material Safety and Safeguards in conjunction with the Office of Nuclear Regulatory Research would have prime responsibility for this activity.

Technical Safety Cooperation. In 1987, the NRC held policy and technical meetings with over 150 visitors from foreign countries and organizations. GPA coordinated these visits in advance with the NRC staff to assure extensive and detailed discussion of topics of mutual interest and to promote a two-way flow of information. Responses to more than 100 requests for technical and safety information were provided during the report period.

Foreign Assignees to the NRC Staff. The NRC work/training assignee program continues to be of strong interest to foreign regulatory organizations. Eleven countries sent 25 staff members to participate in the program. While licensing activities related to specific engineering and scientific disciplines attract a number of participants, an increasing number of requests have been accommodated in activities related to operating data analysis, systems interaction, probabilistic safety/risk analysis, and emergency planning and response.

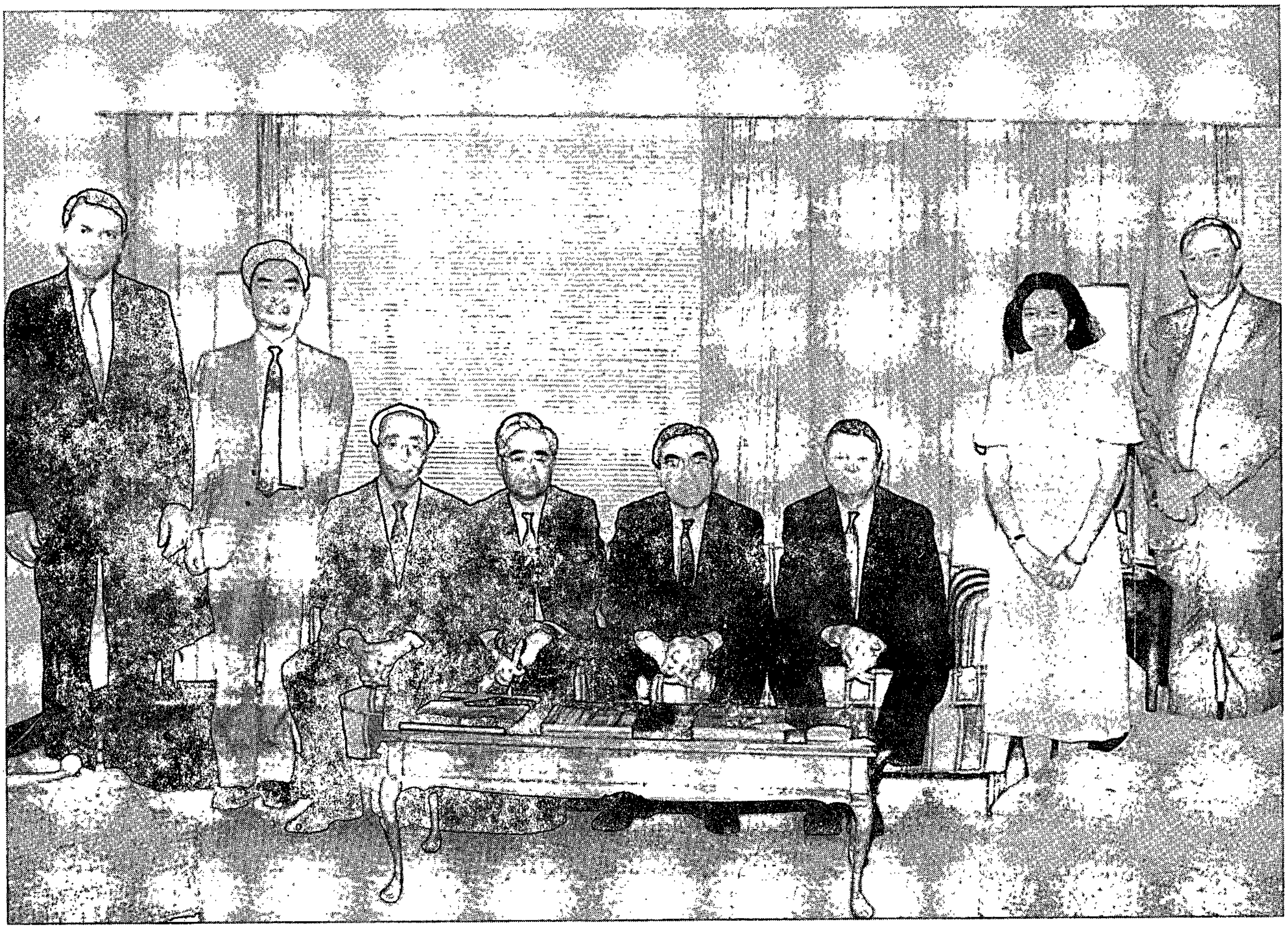
Participation In International Organizations and Conferences

IAEA General Conference. The NRC Chairman, Lando W. Zech, Jr., Office of Governmental and Public Affairs Director, Harold R. Denton, and GPA's Director of International Programs, James R. Shea participated in the anniversary session of the IAEA's Thirtieth General Conference held in Vienna, Austria, from September 19-23. Both the Chairman and Mr. Denton chaired special IAEA sessions on nuclear safety issues during the General Conference. The controversy over South Africa's membership, expected to arise at this year's meeting, was postponed until next year. Chairman Zech and Mr. Denton had leadership roles in the Scientific Session of the General Conference and had appointments with Deputy Director General for Nuclear Energy and Safety Konstantinov and Deputy Director General for Safeguards Jennekens. Chairman Zech took the opportunity of the trip to renew nuclear safety arrangements with The Netherlands and Switzerland. He was also honored at a dinner attended by NRC staffers currently working at the IAEA.

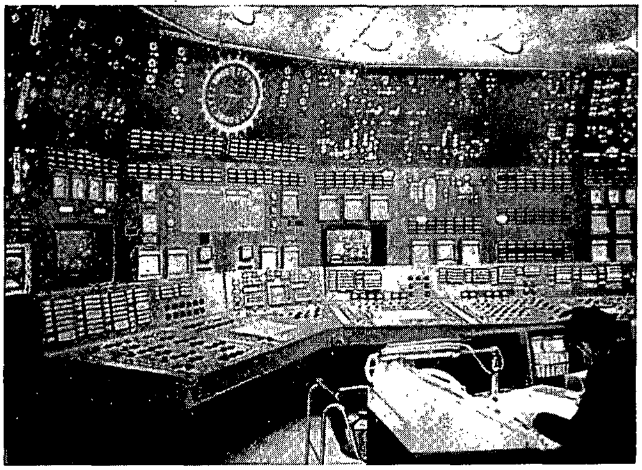
Mr. Denton participated in the International Conference on Nuclear Power Performance and Safety held the week of September 28, 1987. E. Jordan, Director of the Office of the Analysis and Evaluation of Operational Data (AEOD), E. Branagan of NRR, and two contractors (one of whom presented a paper on NUREG-1150) also participated in the conference.

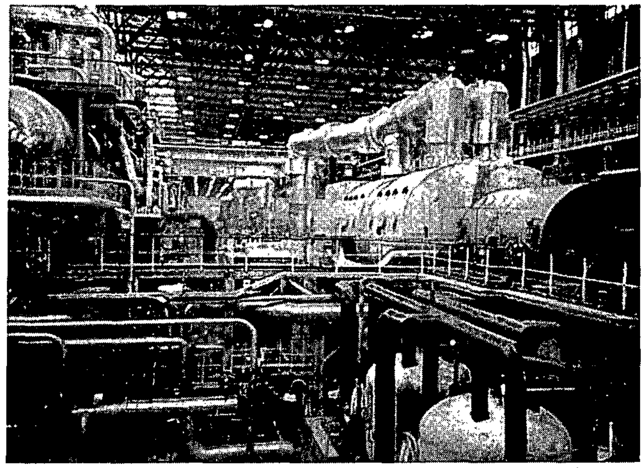
OSARTS and Other IAEA Safety Activities. The NRC helped sponsor the IAEA Operational Safety Review Team (OSART) mission to the Calvert Cliffs (Md.) nuclear power plant in August. In fiscal year 1987, the NRC continued to support the OSART program at the IAEA by sending employees on OSART missions to Mexico, West Germany, and South Korea. Currently, the NRC is paying for a cost-free expert in the Nuclear Safety Division at the IAEA. The NRC is also actively participating in the IAEA's work revising the Nuclear Safety Standards (NUSS) safety guides and several other documents as a result of the post-Chernobyl evaluation of the IAEA's nuclear safety program.

Technical Safety Assistance. During fiscal year 1987, 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 nuclear power program. A Region I (Philadelphia) inspector twice visited Mexico for pre-service inspections of the Laguna Verde reactor. An inspector from Region IV (Dallas) visited Mexico for radiation protection inspection at that facility. And an inspector from Region II (Atlanta) went to Mexico to participate in an IAEA Radiation Protection Advisory Team (RAPAT) mission as an Emergency Preparedness expert. One NRC staff member went to Egypt to lecture on fire protection. Another went to Yugoslavia to lecture on severe accidents and emergency planning at a three-week IAEA course on "Safety Reliability in Nuclear Power Plant Operation," held at the Josef Stefan Institute.



In June 1987, the NRC hosted the third annual meeting with its Japanese regulatory counterpart, the Agency of Natural Resources and Energy (ANRE). The primary conferees shown here are (left to right) James R. Shea, NRC Director of International Programs; Tsutomu Inoue, President, Japan Power Engineering and Inspection Corp. (JAPEIC); Messrs. Matsue, Hatano, and Togo, all of JAPEIC; NRC Chairman Lando W. Zech, Jr.; Janice Dunn-Lee of the NRC Office of Governmental and Public Affairs (GPA); and Harold R. Denton, Director of GPA.





A U.S. Government interagency team headed by NRC Commissioner Frederick M. Bernthal visited regulatory counterpart agencies in Hungary during 1987 and touring nuclear facilities such as the Pak Nuclear Station, shown here. At left is the plant's control room and at right the plant's turbine hall.

Activities in the OECD/NEA. The NRC remained actively involved in reactor safety, radiation protection, and waste management programs of the 24-nation Organization for Economic Cooperation and Development (OECD), through the Nuclear Energy Agency (NEA). Work was completed on a report on the Chernobyl accident and the safety of reactors in the OECD area, and various working groups carried out associated studies related to better understanding and management of severe accidents. Radiation protection experts also discussed differences among countries in setting contamination levels and imposing requirements to protect the public from foodstuffs and other sources of radioactivity after an accident. As a separate matter, NRC waste management specialists participated in NEA activities on assessing the performance of waste storage facilities. The OECD/NEA includes the industrialized countries of Western Europe plus Australia, Canada, Japan, and the United States.

Export-Import and Non-Proliferation Actions

NRC Export License Summary. The NRC has responsibility under the Atomic Energy Act, as amended, for the licensing of the export of nuclear-related materials and equipment. This export authority extends to production and utilization facilities, to 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 required.

In 1987, the NRC issued 129 export licenses and 25 amendments to existing licenses. Of these, 84 were "major" licenses or significant amendments in four categories: (1) special nuclear material, (2) source material, (3) nuclear reactor materials (graphite and deuterium), and (4) major reactor components. The majority of these major cases involved routine exports of low-enriched uranium intended for use in commercial light-water power reactors. Two licenses involved exports of high-enriched uranium to the French HFR-Grenoble research reactor. A total of 11 nations received shipments of special nuclear material under major export licenses during the year. As in the previous year, several major export licenses were issued for shipment of source material to the European Community for enrichment and subsequent power reactor use. The remaining 70 licenses and amendments included 14 for exports of small quantities of special nuclear materials: 11 for source material, 10 for byproduct material, 10 for components and materials, and 25 for miscellaneous license amendments, such as extensions of expiration dates.

Almost all imports of nuclear-related materials may take place under the NRC's general import licensing authority. However, on December 31, 1986, the NRC amended its import regulations to require specific licensing for all imports of South African-origin uranium. This action was taken to implement the provisions of the Comprehensive Anti-Apartheid Act of 1986, which prohibits the import into the U.S. of South African-origin uranium ore or oxide. Subsequently, in a decision published on September 21, 1987, the Commission concluded that the Anti-Apartheid Act bars the import of uranium ore and oxide, but that importation of other forms of uranium and uranium, which is substantially transformed prior to importation into the U.S., is not barred. This conclusion is identical to that adopted by the Department of the Treasury. In accordance with this decision, the NRC issued four new import licenses and amended 11 existing import licenses in 1987 to permit the import of uranium of South African origin.

NRC Consultations with the Executive Branch on Nuclear-Related Export Matters. In addition to its own licensing actions, the NRC consults with the Executive Branch on other types of transactions with potential proliferation implications. These transactions include exports of nuclear-related items licensed by the Department of Commerce, nuclear technology transfers, subsequent arrangements, and agreements for cooperation. The significant number of transactions involve the nuclear-related export cases licensed by the Department of Commerce. The NRC was consulted by Commerce on over 200 of these cases during fiscal year 1987. The NRC also reviewed 25 Executive Branch requests for subsequent arrangements. These arrangements describe further actions that an importing country wishes to take with previously exported U.S.-origin nuclear material and equipment.

In addition, the NRC reviewed the proposed U.S.-Japan Agreement for Peaceful Nuclear Cooperation, and provided its final views to the President in September 1987. At the close of the fiscal year, the Agreement awaited final Congressional review. Also during fiscal year 1987, the NRC reviewed the implementing arrangements of the proposed U.S.-China Agreement for Peaceful Nuclear Cooperation and provided its final views to the Executive Branch.

SNEC—Interagency Review of Nuclear Exports. The NRC continued active participation in the Subgroup on Nuclear Export Coordination (SNEC), an interagency body that oversees the U.S. nuclear export control system. The SNEC serves an important role in assuring that U.S. agencies with different perspectives and expertise—technical, economic, and foreign policy—all contribute to the decision-making process of specific export cases and to the formulation and implementation of U.S. non-proliferation policy as it relates to nuclear export control. SNEC reviewed over 200 cases in 1987, primarily intended for export to sensitive destinations such as Argentina, Brazil, India, Israel, South Africa, Iraq, and Pakistan. The cases involved are primarily Commerce-licensed requests for commodities controlled for nuclear non-proliferation reasons, called Nuclear Referral List (NRL) items. The list currently contains 65 commodities. In 1987, SNEC provided its final revisions to the Commerce Department in a diligent effort to simplify and clarify the NRL items.

International Safeguards and Physical Security. In all pending export cases to be reviewed by the NRC, the staff reviews the implementation of the IAEA safeguards and physical security arrangements to be applied to the exports 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 not be used for proscribed purposes, such as the making of nuclear explosives.

The NRC participates in U.S. Government efforts to assist the IAEA in improving its safeguards system. The U.S. Program of Technical Assistance to IAEA Safeguards (POTAS) and the U.S. Action Plan Working Group (APWG) are the primary programs in this area. Through the activities of these groups, the U.S. is able to participate in joint projects with other countries and the IAEA itself in support of the international safeguards system. Under the auspices of the APWG, the NRC participated in bilateral and multilateral discussions on safeguards experience with Japan, France, West Germany, the United Kingdom, and the European Community in 1987.

The NRC has substantial responsibility for implementing the U.S./IAEA Agreement to apply international safeguards to selected U.S. nuclear facilities. The NRC participates in negotiation of the arrangements for applying international safeguards on facilities it licenses. The NRC also assists the IAEA in scheduling and organizing its inspection activities at NRC-licensed plants and accompanies the inspectors during inspections. In 1987, three NRC-licensed facilities were subject to the application of international safeguards—two power reactors, Salem Unit 1 in New Jersey and Turkey Point Unit 4 in Florida, and the Westinghouse low-enriched uranium fuel fabrication plant in South Carolina. Four other NRC-licensed low-enriched uranium fuel fabrication plants are subject to limited international safeguards. They must report regularly to the IAEA on the amount of nuclear material in their inventory, and any changes in the amount since the previous report. These facilities are operated by Combustion Engineering, Exxon, General Electric, and Babcock & Wilcox.

In support of its review of physical security arrangements of U.S.-controlled materials in other countries, the NRC participates in Department of Energy-sponsored trips to the importing countries to discuss their physical security programs. In this regard, a U.S. delegation visited France, Japan, and Australia during 1987.

In August 1986, the Congress passed the Omnibus Diplomatic Security and Anti-Terrorism Act. Title VI, Section 604 of that Act provides that the Departments of Energy and State, the Arms Control and Disarmament Agency, and the NRC review and submit written reports by February 1987 on the "adequacy of the physical security standards currently applicable with respect to the shipment and storage (outside the United States) of plutonium, and uranium enriched to more than 20 percent. . . which is subject to United States prior consent rights, with special attention to protection against risks of seizure or other terrorist acts." The NRC submitted its report to Congress in February, with a favorable finding.

Nuclear Regulatory Research

Chapter 9

Activities of the Office of Nuclear Regulatory Research (RES) provide an essential contribution to the regulatory process. 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. (See Chapter 1 on NRC reorganization and reassignment of tasks in 1987.) Besides conducting research needed to support the NRC mission, RES has responsibilities related to implementation of Commission policies on safety goals and severe accident regulation, to the resolution of generic safety issues, and in 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," on the following page.) Regulations issued by NRC in 1987 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 1987.

This chapter summarizes RES activities during fiscal year 1987 under the following headings: Integrity of Reactor Components, Preventing Damage to Reactor Cores, Reactor Containment Performance and Public Protection from Radiation, Confirming Safety of Nuclear Waste Disposal, and Resolving Safety Issues and Developing Regulations.

Integrity of Reactor Components

The research program dealing with the integrity of reactor components examines reactor plant systems and related components to see that they perform as designed and that their functional integrity and operability are maintained over the life of the plant. Reactor safety depends on maintaining the integrity of the reactor system pressure boundary, i.e., maintaining it free from damage and leak-tight. Failure to maintain pressure boundary integrity could compromise the 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

Vessel Aging and Pressurized Thermal Shock Studies. Under certain postulated accident conditions—such as small-break loss-of-coolant accidents, main steam line breaks, steam generator overfilling conditions, and associated instrument and component failures—a pressurized water reactor (PWR) pressure vessel could be subjected to severe differential cooling rates, coupled with a continuing high pressure. This combination of thermal stresses and internal pressure, called pressurized thermal shock (PTS), could pose a serious challenge to the integrity of some older pressure vessels that have developed a significant degree of embrittlement because of neutron irradiation.

NRC-sponsored research has been conducted primarily by Oak Ridge National Laboratory (ORNL) under the Heavy Section Steel Technology (HSST) program, with supporting activities conducted by Materials Engineering Associates (MEA), Inc., and by the National Bureau of Standards (NBS) at Gaithersburg, Md. These activities have developed data that were instrumental in the early recognition and rapid resolution of the PTS problem. The resolution took the form of an embrittlement screening criterion to be applied to operating reactor vessels. The criterion represents an embrittlement level beyond which the reactor cannot be permitted to operate without the specific approval of the NRC. In 1985, an amendment to 10 CFR Part 50 (U 50.61) established the screening criterion, and, in 1987, the regulatory guide on performing PTS analyses was issued.

Although the rule amendment and the associated regulatory guide provide reasonable assurance that potential PTS accidents will not lead to failure of PWR pressure vessels, the actual margin against failure is clouded by uncertainty deriving from the assumptions which have to be adopted to resolve the problem. Consequently, research has continued on several fronts to validate the rule and regulatory guide analysis, and to quantify the inherent margin against failure. Topics that continue to be investigated include the effects of different materials (particularly the low upper-shelf welds), the effects of warm prestressing (WPS), the extension of the American Society

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 NRCstaff 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 is required by the Administrative Procedure Act. Following the public comment period, the regulations are revised, as appropriate, to reflect the comments received. Once adopted by the NRC, they are published in the *Federal Register* in final form, with the date they became effective. After that publication, rules are codified and included annually in 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 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 its expected effectiveness and impact.

of Mechanical Engineers (ASME) crack arrest toughness curves to higher values, and the effects of stainless steel cladding.

One way the NRC assesses the fracture behavior of pressure vessels subjected to PTS loadings is by intermediate-scale pressure vessel experiments. These experiments are performed by subjecting an intentionally flawed pressure vessel, whose wall thickness is nearly that of an actual PWR pressure vessel, to combined pressures and temperatures that simulate a postulated PTS accident. Pressure, temperature, and flaw depth are carefully monitored so that the crack driving force and crack behavior (propagation or arrest) can be evaluated as a function of time. The first of these experiments, performed in 1984, evaluated the fracture behavior of a current-practice high toughness pressure vessel steel. That experiment confirmed that the NRC's PTS analysis procedures were indeed conservative for this material. The first experiment also showed the beneficial effects of the WPS phenomenon, which, in some cases, would prevent the initiation of cracking and, in other cases, would limit the extent of cracking of a flawed pressure vessel subjected to a PTS loading.

The second experiment was performed in 1987 and evaluated a material that simulated a low upper-shelf weld material. This experiment with stress and toughness states representative of reactor pressure vessels demonstrated for the first time the arrest of a brittle fracture with an immediate tearing instability, and brittle fracture following WPS. The principal conclusions from the experiment are that: (1) low upper-shelf material can exhibit very high crack arrest toughness—an important concept in evaluating crack stability during a PTS accident scenario; (2) WPS inhibits brittle fracture to some degree even when crack driving forces are increasing with time, although the benefits of WPS are diminished by ductile tearing; and (3) a simple theoretical analysis of WPS represented fracture conditions reasonably well, but calculations of ductile tearing based on state-of-the-art fracture analysis concepts did not consistently predict the observed fracture behavior.

As noted above, the WPS phenomenon can prevent the initiation of cracking or limit the extent of cracking if it has been initiated. Although this concept has not been incorporated explicitly in the PTS analysis methodology, it should increase the inherent margin of safety in the analysis. Further, if WPS can be verified and modeled, its potential beneficial effects could be quantified and included explicitly in PTS analyses. To that end, the NRC has supported research into the WPS phenomenon as part of the PTS experiments and as a separate task in the MEA program. As discussed, the PTS experiments have shown that WPS effects exist and that they have a beneficial effect, although it is not as large for crack growth from ductile tearing as it is for crack growth attributable to brittle fracture. The MEA work has sought to model the WPS phenomenon and to experimentally validate that model. That work continued during 1987, evaluating existing models and showing that they could in factpredict WPS effects. The MEA work will be completed in 1988 and a final report issued at that time.

The idea that a crack, extending rapidly through the pressure vessel wall with an increasing crack driving force, might slow and eventually stop seems contrary to common sense. However, as a hypothetical crack would propagate from the inner surface to the outer surface of the reactor pressure vessel, the materials show an increasing resistance to crack propagation, because of the increasing material temperature and less severe radiation embrittlement. Recognizing these facts has led to the inclusion of crack arrest concepts in the PTS analysis methodology.

To make practical use of the crack arrest concept requires crack arrest toughness values for the material that are well in excess of values in the ASME Code. Therefore, the NRC began a study to validate the existing ASME Code curves, to extend the range of those curves, and to provide data to develop improved analytical models for a better understanding of the fracture process and of the margins against failure provided by current analysis criteria. The work is being performed by NBS under subcontract to ORNL. The test specimens are 1 meter wide, 10 meters long, and either 0.1 or 0.15 meter thick. These large specimens are needed to prevent premature crack arrest and artificially low values of crack arrest by means of stress waves reflected from the ends of the specimen. The work was started in 1984 and the 11th experiment was performed in September 1987. Approximately 10 more experiments are planned. The results have shown that the existing ASME curves are a lower boundary over their range. The higher crack arrest values obtained from the experiments agree with crack arrest values from the PTS experiments and from other work from Japan. These results and results from Europe and Asia were discussed during the third NRC-sponsored Crack Arrest Workshop held at NBS in May 1987.

Work on the effect of cladding on crack initiation, crack propagation, and crack arrest is not complete. However, work previously completed at MEA and ORNL gave preliminary indications that reactor pressure vessel cladding has no significant mitigating effects. Although the potential for deleterious effects had not been completed, no significant effects were expected. The work in 1987 continued to examine the effects of cladding in separate studies, considering each variable independently. Irradiated specimens were tested by MEA and, in some areas, their results seem to contradict earlier observations by ORNL. The discrepancies will be resolved in 1988 as the work is completed.

Radiation Embrittlement. Normal operation of reactors produces excess neutrons that strike the reactor pressure vessel walls, causing the steel of these walls to lose its fracture toughness. The degree of toughness loss depends on several factors, including the chemical composition of the steel. Thisproblem has been studied for many years, and the research, particularly the NRC-sponsored research at the Naval Research Laboratory (several years ago) and at MEA, has identified certain alloying and residual chemical elements that contribute to the radiation embrittlement problem. Based on this work, chemical composition standards for reactor pressure vessel steels have been developed to effectively minimize the problem. All the newer reactor pressure vessels were fabricated using materials made to these standards.

To evaluate the effects of radiation embrittlement, the NRC has sponsored several irradiation and testing efforts where test specimens of specific materials are irradiated to fluences corresponding to the projected end-of-life fluence for a typical reactor pressure vessel, and then the specimens are tested. The results are contrasted to results from unirradiated samples to determine the degree of irradiation damage. The fourth irradiation series was completed and analyzed in 1986. This study confirmed that control of chemical constituents such as copper, nickel, and phosphorus results in materials that are reasonably resistant to radiation embrittlement. The fifth irradiation series was initiated in 1985. Testing was initiated in 1986 and completed in 1987. This series is designed to validate the ASME Code's trend properties for the irradiation-induced changes in fracture properties used to evaluate pressure vessel integrity under both normal and accident conditions. The sixth irradiation series also was initiated in 1985 and uses the same material as the fifth series. The sixth series examines the effects of irradiation on crack arrest properties, again in order to confirm the ASME Code curves. The testing of the sixth irradiation series specimens began during 1987 and will be completed in 1988.

Studies are in progress to determine the ameliorating effects of annealing on pressure vessel steels that were embrittled by irradiation and the trends ofradiation-induced re-embrittlement. These studies will establish the merits of annealing temperature options. Also being evaluated are the effects of material composition, fluence, flux, and irradiation temperature. A significant effort is being applied to identifying and understanding radiation damage mechanisms to help in the prediction of irradiation effects on specific steels. Tests on the effects of irradiation on a decommissioned pressure vessel from the Gundremmingen reactor in the Federal Republic of Germany are in progress to aid in confirming experimental results.

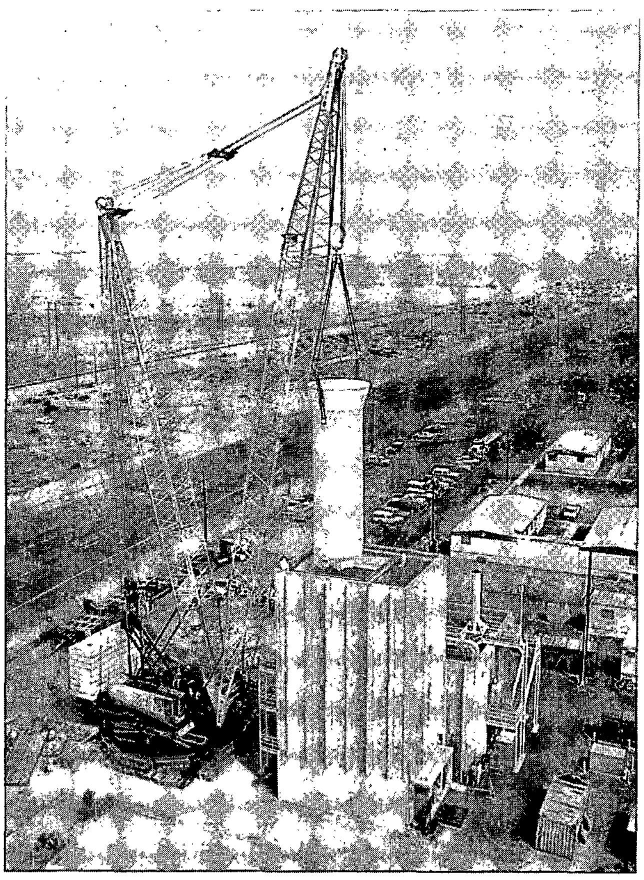
Surveillance Dosimetry. An important aspect of the surveillance program to determine the degree of embrittlement in the pressure vessel of an operating nuclear power plant is the prediction of the amount of neutron radiation exposure (neutron fluence) of the vessel. Fluence determinations are made by 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. It is necessary that these predictions be reasonably accurate to ensure that the plant is operating in conformance with NRC safety regulations. Experimental aspects of the program are described in the 1986 NRC Annual Report, pp. 162 and 163.

A draft regulatory guide that identifies methods and assumptions for establishing pressure vessel fluence has been prepared and is being reviewed and evaluated prior to publication for comment. The guide makes use of the developments generated by the surveillance dosimetry program.

Steam Generator Integrity

The Steam Generator Group Project at Battelle-Pacific Northwest Laboratories (PNL) has used a retired-from-service steam generator from an actual PWR facility as a test bed for measuring the effectiveness of eddy current (EC) inspection techniques to detect and size 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.

In prior years, to establish EC inspection performance, four "round robin" studies were conducted on a subset of 320 tubes. The tubes were determined to have a high probability of containing defects, based on two initial EC inspections of approximately 3,000 tubes in the generator. Following the round robin examinations, more than 550 tube segments were removed from the generator to validate the in situ EC results. Specimens were removed from all levels of the generator, but emphasis was given to specimens where an EC indication had been reported. Specimens without indications were also removed to assess the reliability of EC inspections in establishing tubing condition at locations where defects were both expected and not expected.



An NRC-sponsored research project to determine the effectiveness of eddy-current inspection techniques in detecting flaws in steam generator tubing, completed in 1987, used a "retired" generator in the testing. In the photo, the generator is being removed from the specially constructed examination facility. It was subsequently loaded on a trailer, visible at left, and taken to a burial site at Hanford, Wash.

During fiscal year 1987, detailed examination of the removed tube segments and correlation with the EC data were completed. In agreement with the EC inspection results, most defects were observed in the sludge pile region near the top of the tube sheet. Pitting/wastage defects were the predominant types observed in this region, and the most severely degraded specimens were from the hot leg side. Comparison of metallographic results and EC estimates of maximum defect depth showed a relatively large degree of scatter, and EC generally tended to underestimate the defect depth. The complex defect morphology coupled with the analyst's interpretation of the resulting complex EC signals were the primary causes of the observed variability in defect sizing.

Specimens removed from the generator with pitting/wastage defects along with tubes containing laboratory-produced stress corrosion cracks were burst-tested in 1987 to validate empirical models of remaining tube integrity. Results indicated that these models adequately predict the failure pressure of inservice flawed tubing. Nearly all the specimens tested failed at levels several times the maximum pressure attainable during an accident involving a main steam line break. This was because of the short axial extent of these defects, and it underscores the importance of knowing the length, as well as depth, of defects to arrive at a proper flaw evaluation.

Analyses and simulations were performed to evaluate and compare candidate sampling plans for inservice inspection. The measured EC performance from the round robin inspections was used to guide the selection of input parameters for this work. Results of these analyses indicated that small-scale sampling plans were not effective for detecting and plugging defective tubes. Calculations indicated that even 100 percent inspection may not be effective if the number of defective tubes is large. However, it was determined that a 40 percent systematic sequential inspection plan could perform with nearly the same effectiveness as 100 percent inspection.

Piping Integrity

Environmentally Assisted Cracking. A very significant problem encountered in boiling water reactors (BWRs) has been the intergranular stress corrosion cracking of austenitic stainless steel piping at weldments. This condition has been responsible for over 400 pipe-cracking incidents throughout the world, over the last 10 years. Because these problems have resulted in extended and unscheduled outages—with extensive inspections, repairs and replacements, and significant occupation exposures—the NRC and the electric utility industry have devoted much research to their resolution.

NRC research in this area is focused on developing the capability to predict stress corrosion cracking in BWRs and to verify the acceptability of proposed fixes. (For background on proposed solutions to the problem, see the *1986 NRC Annual Report*, pp. 163 and 164).

The stress corrosion susceptibility of Types 316 NG and 347 stainless steelwas investigated over a wide range of environmental and mechanical loading conditions using both constant extension rate and fracture-mechanics crack-growth-rate tests. A phenomenological model was developed to aid in the understanding and interpretation of the data. These tests have shown that Type 316 NG stainless steel is extremely resistant to intergranular stress corrosion cracking but becomes susceptible to transgranular stress corrosion cracking at 289xC in water with the dissolved-oxygen levels (0.25 ppm O_3) characteristic of conventional BWR water chemistries. The presence of very low levels (25-50 ppb) of sulfate in the water can significantly increase susceptibility to transgranular stress corrosion cracking.

The stress corrosion cracking susceptibility of Type 347 stainless steel was similar to that of Type 316 NG stainless steel. Failures in weldment specimens always occurred away from the weldment, and there was no evidence of "knife-line attack" adjacent to the fusion zone. A long-term (6,000 h) fracture-mechanics crack-growth-rate test to compare the behavior of Type 347 Mod SS with that of a lightly sensitized Type 304 SS control specimen was completed. No crack growth was observed in the Type 347 Mod SS specimen under any of these loading conditions.

An extensive program has been carried out to demonstrate the strong interactions among dissolved oxygen and various impurities, as well as the effects of individual impurity species on stress corrosion of sensitized Type 304 SS in low-oxygen, high-temperature water. The results of this work indicate that stress corrosion of the sensitized steel appears to be controlled by the rate of cathodic reduction of dissolved oxygen and/or oxyanion impurity species that have a central atom that can assume different oxidation states. These results imply that cations, which can also undergo cathodic reduction reactions, may contribute in a similar manner. To test this hypothesis, experiments were performed in high-temperature, low-oxygen water containing salts that can undergo reduction in water, as well as with cations that have a single oxidation state. The data provide the basis for affirming the benefits of good water quality and the role of different impurities in stress corrosion cracking of sensitized austenitic stainless steels. By removing species from the water that provide a cathodic reduction partial process, which couples directly with the anodic dissolution process at the crack tip, crack growth can be suppressed or halted.

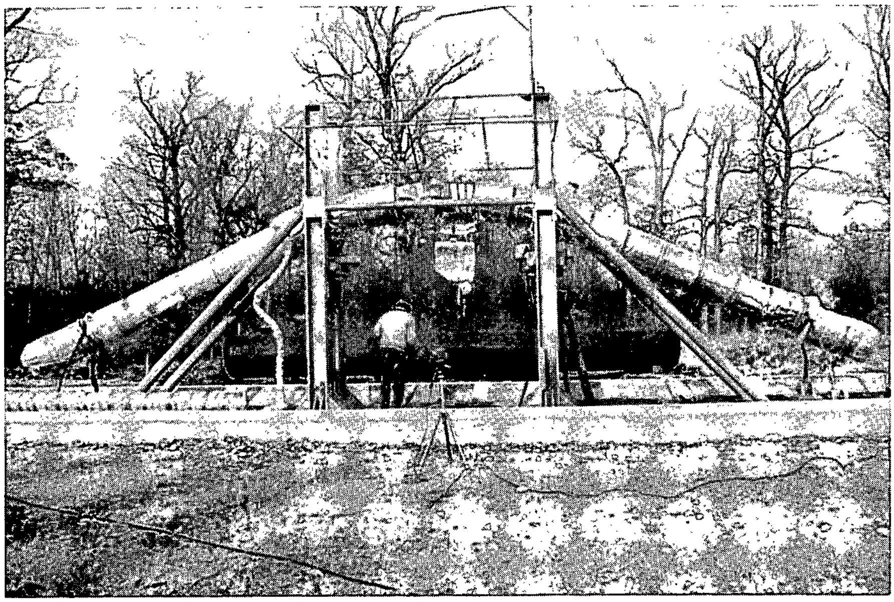
In addition to ionic impurities, organic chemicals used in power plants frequently enter the coolant and conceivably can contribute to stress corrosion cracking of system materials. Cleanup systems are typically designed to remove ionic species by ionic exchange, detritus by filtration, and nonvolatiles by evaporation residue; but organic substances can pass through the system. To determine whether some of these substances contribute to stress corrosion of sensitized Type 304 SS, tests were performed in 289xC water containing 0.2 ppm dissolved oxygen and 1.0 ppm of several organic acids. Results showed that carbonic, carboxylic (acetic, formic, lactic, and oxalic), and short-chain aliphatic (propionic and butric) acids do not have a deleterious effect on stress corrosion cracking under simulated normal BWR water chemistry.

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, using standard industrial practice, in such a manner 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.

A thermal aging program was initiated in 1982 to evaluate the long-term effects on degradation of toughness in cast stainless steel as a function of time of exposure and temperature of the material composition. Through 1987, 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. During 1987, a heat treatment was evaluated for recovery of toughness loss.

Work is under way to assess the degradation of material properties as a result of aging at the Shippingport (Pa.) reactor. Components (piping, pumps, and valves) have been identified, sectioned into smaller pieces, and shipped to Argonne National Laboratory for evaluation. Test samples are being prepared from the aged material to evaluate tensile strength, Charpy-test impact energy, fracture toughness, corrosion fatigue, stress corrosion cracking, and microstructural changes.

Piping Fracture Mechanics. The NRC's piping fracture mechanics research covers a broad range of topics in this general area, with three laboratories contributing to the effort during the report period—David Taylor Naval Ship Research and Development Center (DTNSRDC), in Annapolis, Md.; Materials Engineering Associates, Inc. (MEA), in Lanham, Md.; and Battelle Columbus Laboratories (BCL), in Columbus, Ohio. The research has contributed to the development and validation of standards for evaluating flaws in nuclear power plant piping, and identifying areas where the intended margins are not being achieved; it has provided a forum for achieving an international consensus on how leak-before-break technology should be implemented.



Fracture tests on typical BWR pipe, repaired by a weld-overlay procedure, were performed in 1987 at Battelle's Columbus Division. The specimen shown is a 16-inch-diameter stainless steel pipe with a standard weld overlay in the center of the pipe. The overlay is about six inches long. The specimen was heated to 550° F, pressurized to a desired stress level, and subjected to slowly increasing bending loads until the intentional flaw grew through the wall thickness. Test results validate the weld-overlay design criteria and margin of safety.

The DTNSRDC program has focused on developing data to validate flaw evaluation procedures being implemented in Section XI of the ASME Code and to validate a widely accepted modification to the J-integral fracture toughness parameter. The pipe fracture experiments generally support the ASME flaw evaluation procedures for welds in carbon steel pipe. However, the work to validate the modification to the J-integral has shown that the modified parameter can produce artificially high estimates of a material's fracture toughness. The work is continuing and will define conditions where the modified parameter can be used. The discovery that the modified J-integral can produce non-conservative estimates of fracture toughness has had an adverse impact on piping fracture analyses, where the modified parameter has been widely used. Equally important, however, is the adverse impact this discovery has had on the procedures proposed by the ASME Section XI committee for evaluating low upper-shelf welds in reactor pressure vessels. Since the margin of safety incorporated in the Code does not compensate for the non-conservatism that could be introduced by the modified parameter, it appears that the proposed procedures must either be changed or their final acceptance postponed until the problems with the parameter are resolved. The DTNSRDC research will have the leading role in resolving these problems.

The MEA effort involves preparing a computerized data base of piping fracture toughness data and related material properties. The data base structure was completed during the previous report period. The work during the present report period focused on collecting data from other NRC contractors and including those data in the data base. At the end of the report period, the data base included data on 62 different piping materials and welds. The data base is available to users via a telephone link. However, to provide an alternative to accessing MEA's computer system, the data have been copied to floppy disks in a format acceptable to a commercially available data base management system. This approach allows each end-user to customize the data base to suit his needs. The data base provides material property data that can be used in evaluating leak-before-break applications where archival materials are not available for testing.

The Degraded Piping Program—Phase II, being conducted by BCL, continues to be the mainstay of the NRC's piping fracture mechanics effort. The research emphasizes full-scale pipe fracture experiments to validate specific aspects of the piping fracture mechanics technology. The program was initiated in 1984 and will conclude at the end of fiscal year 1988. At the end of this report period, 53 pipe fracture experiments have been completed, with an additional nine experiments planned for fiscal year 1988. The pipe fracture experiment results have provided a means to validate the flaw evaluation procedures being developed by the ASME Code. For example, during fiscal year 1987, work on welds in stainless steel pipe showed that the flaw evaluation procedures used for submerged arc welds are conservative and typically achieve actual margins of safety greatly in excess of the Code's intended minimum margin. However, the research also showed that the fracture toughness of the actual weld fusion line is on the order of one-half that of the weld metal or the heat affected zone material. This discovery is important in defining material properties to be used in evaluating leak-before-break analyses, as well as in making certain that the ASME Code is adequately conservative.

The NRC's work at DTNSRDC, as well as piping fracture work performed nationally and internationally, has tended to use small-diameter pipe (approximately 4 in.-

to-16 in. diameter). The BCL work has extended the range of diameters considered up to 42 in.-diameter pipe. The need for these larger diameter experiments was demonstrated in 1987 by the discovery that as diameter and wall thickness increase, the effective fracture toughness of welds decreases. This fact stems from the welding processes, where the first layers are deposited by a non-flux process that has a much higher fracture toughness than the flux weld processes used to complete the welds. For thicker pipe, the percentage of the total weld made by the the lower toughness process is greater than for thinner pipe. Consequently, the effective fracture toughness of the weld decreases.

The Degraded Piping Program results also have led to the validation of design criteria for the weld overlays used as repairs to stress corrosion cracks. The work has shown that, for flaws that have been repaired by a weld overlay, the margin against failure at least meets, and generally exceeds, the margins used in Section XI of the ASME Code. Further, very large deflections must be imposed to cause failure of the repaired pipe. It is unlikely that these large deflections would be physically possible inside the containment building, suggesting that the effective margin against failure may be even larger than determined in the experiments.

Prior to the Degraded Piping Program observations about dynamic strain aging, the NRC was working with national and international groups to establish the International Piping Integrity Research Group (IPIRG). The IPIRG is a consortium of government and industrial organizations formed to jointly fund research on the integrity of piping subjected to seismic and dynamic loading, as well as other piping integrity issues within the group's area of interest. In 1987, the IPIRG research, conducted by BCL, was started with the design of the test facilities and some initial work on validating leak rate estimation models. Further, early results on the fracture of stainless steel pipe under rapid loading rates show that the fracture resistance did in fact increase. However, material property tests on carbon steel piping material show that the dynamic strain aging phenomenon should be expected to reduce the fracture toughness of piping subjected to seismic loading.

The IPIRG program currently is examining one end of the spectrum of piping failure modes. Another program, discussed as part of the Seismic and Fire Protection Research Section of this chapter, is examining the other end of the spectrum—the failure of unflawed piping subjected to seismic loading. These two research efforts are being coordinated within the Division of Engineering, and future work will lead to a clear picture of how piping fails over the range of flaw sizes of interest to the NRC.

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 Technique for Ultrasonic Testing). The SAFT 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). The University of Michigan demonstrated the technology in the laboratory, and PNL has had the role of transferring the technology into a field-demonstrable real-time system. The SAFT-UT field system was assembled in 1985 and successfully demonstrated at a field site. The field system was made real-time in 1986 through the development of a real-time processor so that image analysis could be performed as the inspection is being conducted. Thus, decisions can be made on the presence, location, and size of flaws during the inspection. Also in 1986, a cooperative agreement was developed with Combustion Engineering for their technical and financial participation in the program for commercialization and field implementation of the technology. In 1987, the real-time SAFT processor was extended to provide real-time operation for thick section material, the tandem mode (for imaging the vertical extent of a flaw) was implemented on the real-time processor, and the tandem mode was modified for application to thick-section material. Work was performed in cooperation with Westinghouse and Consolidated Edison to aid in the resolution of an indication in the Indian Point Unit No. 2 pressure vessel with the SAFT technology. The SAFT technology was transferred to Sandia National Laboratories, and the technology has been pulled together into a package for easy transfer to the nuclear industry.

Inservice Inspection System Qualification. Research work, 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. Research has been conducted and criteria developed for proper qualification of the ISI process. Subjects of greatest relevance for qualification are the education, experience, and examination requirements of inspection personnel; procedure requirements; equipment performance measurements and tolerances; and evaluation and requirements for actual performance testing of the total personnel-procedure-equipment aggregation, using actual components and realistic flaws, as a prerequisite to conducting an inspection on reactor components. Based on the research conducted at PNL, criteria and requirements were prepared and reviewed by the NRC and the industry in 1985. These criteria formed the basis for NRC cooperation with ASME in developing and implementing ISI system qualification requirements in the ASME Code. Accordingly, in 1985 and 1986, the NRC worked with designated Code committees to develop documents for incorporating into the Code the recommended qualification requirements.

In 1987, two mandatory appendices to Section XI were being assessed by the appropriate committees for acceptance into the Code. Other work in progress at PNL is concerned with assessing the overall effectiveness of current Code requirements for ISI to ensure operational safety of the reactors. A technical basis is being developed upon which to base new criteria for overcoming identified shortcomings.

Continuous Monitoring for Crack Growth and Leak Detection. Research has been under way at PNL 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. The test, which produced crack growth under simulated reactor operating conditions, was conducted over a one-year period in the Federal Republic of Germany. Test results established that continuous monitoring of reactors for crack growth detection is feasible. Also in 1986, an agreement was concluded with the Tennessee Valley Authority (TVA) for their technical and financial contribution to the research program to allow for the final field validation and commercial implementation of the technology. AE monitoring of TVA's Watts Bar Unit 1 during cold hydro and hot functional testing has already provided valuable input to the AE technology. Plans are to monitor Watts Bar Unit 1 during power operation, which should occur in fiscal years 1989 or 1990.

In 1987, activities focused on technology transfer by developing an ASTM standard for continuous AE monitoring of pressure boundaries (E 1139), which has been approved, and by preparing a non-mandatory appendix to ASME Section XI Code for continuous monitoring of reactor pressure boundaries during operation, which is now in the approval process. Also, two successful applications of technology developed under this program were accomplished in the nuclear area. One was to monitor the High Flux Isotope Reactor vessel at ORNL during a critical hydrostatic pressure test to verify that cracking did not occur in irradiation-embrittled sections of the vessel. The other was to monitor vitrified high-level waste during cooling to determine if and when cracking of the glass matrix occurred. Efforts continue to identify a circumstance where AE monitoring can be applied to reactor piping over a short period (one year) to demonstrate AE detection of crack growth under actual reactor operating conditions. The availability and proper use of this technology will mean that reactors can be continuously monitored and that any cracks that develop can be detected and evaluated. In this way, proper and timely action can be taken to avoid extensive crack growth or component failure.

AGING OF REACTOR COMPONENTS

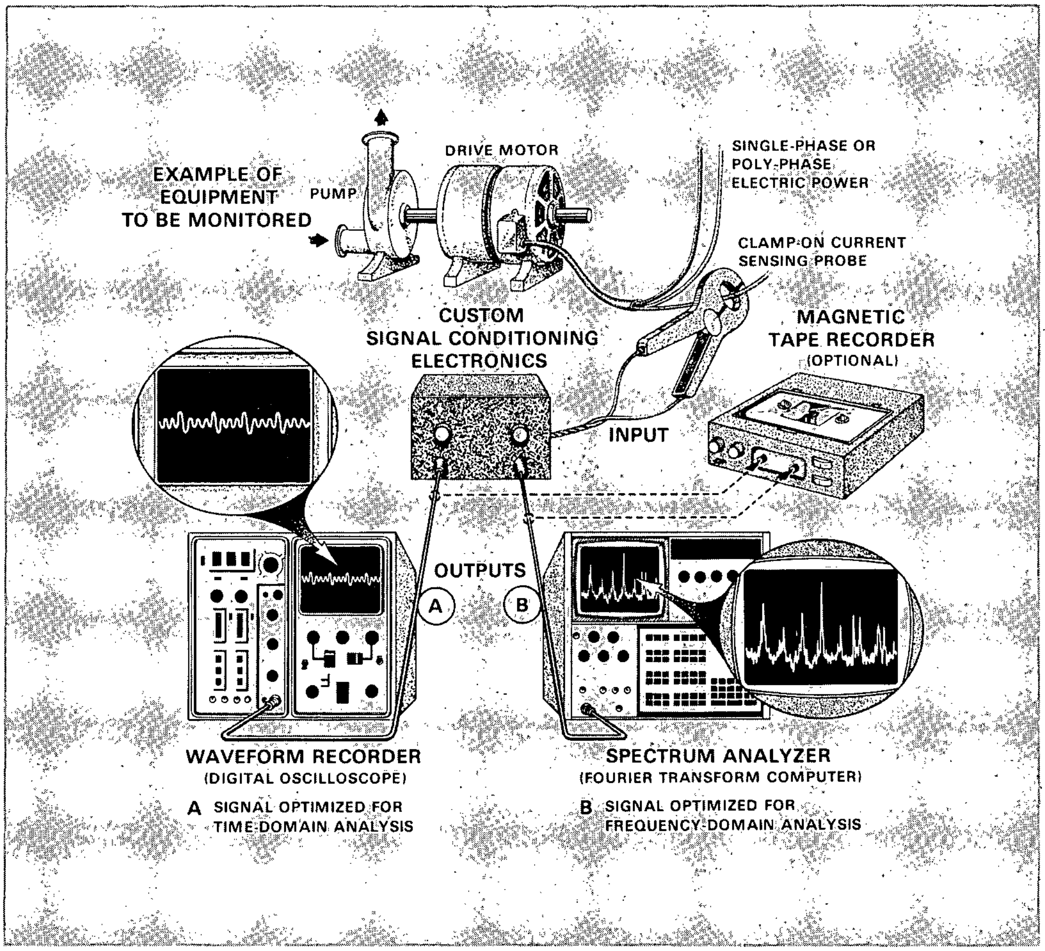
Aging Research

Research into aging phenomena in power reactors, though well under way in the area of integrity of primary system components, is just beginning in other areas. The importance of aging research lies in ensuring the continuing reliability of redundant systems or components that serve safety needs. For example, if a common mode failure should develop as a result of aging, the vital systems needed in response to a fault or accident might not be available. The purpose of aging research is to uncover such phenomena so that necessary corrective actions can be anticipated. In the coming year, aging research on the primary system will focus on the effects of radiation on reactor vessel toughness, validation of piping performance under dynamic and seismic loadings, and evaluation of advanced non-destructive examination tests. In other areas, aging studies of components and systems needed for operation and safe shutdown of nuclear power plants will continue. The thrust of the research will be shifting from screening studies (Phase I) to more in-depth studies (Phase II) of selected components. This work will include tests of naturally aged components, including components removed from the Shippingport reactor, tests of samples with simulated degradation, and verification of improved inspection, surveillance, and monitoring methods proposed or implemented at operating nuclear power plants.

Electric and Mechanical Components. The purpose of this research is to identify and resolve safety issues related to the effects of plant aging on components.

Research studies were completed in 1987 on specific safety-related equipment in order to (1) identify failure mechanisms resulting from aging and service wear; (2) recommend maintenance, inspection, surveillance, testing, and condition monitoring to ensure operational readiness; and (3) establish degradation patterns for use in detecting incipient failures.

The evaluation of motor current as a diagnostic technique for condition monitoring of motor-operated gate valves was demonstrated during the year in tests at the Duke Power Company Oconee plant (S.C.). This surveillance technique has been shown to provide a valuable non-intrusive method



The diagram illustrates the condition-monitoring system, using electric motor signature analysis, as demonstrated at the Duke Power Company's Oconee plant (S.C.).

of identifying valve degradation from aging, improper setting of limit switches, and maintenance deficiencies in lubricating and repacking the valve stem. A diagram of the system, including a small portable computer used for analyzing the motor current spectrum, is shown.

A survey and evaluation of power-operated relief valve (PORV) and block valve operating experience in nuclear plants (NUREG/CR-4692) was completed. The study provides data to support the resolution of Generic Issue 70 on PORV and block valve reliability.

An assessment of aging and service wear of auxiliary feedwater pumps (NUREG/CR-4597) was completed. Potential damage to auxiliary feedwater pumps from low-flow testing operation was identified as a possible contributor to aging degradation.

One of the primary concerns in nuclear plant licensing extension is the aging of safety-related electric cables located in containment. The long-term aging of cables, condition monitoring for detecting aging degradation, and requalification for extended use beyond 40 years are being evaluated by tests of 12 of the principal safety-related cable types in the radiation and LOCA test facilities at Sandia National Laboratories. This work is being coordinated with research efforts by the University of Connecticut for the Electric Power Research Institute (EPRI) in which cables are being monitored in a radiation and thermal environment in containments at several nuclear power plants.

The service water system was chosen as one of the reactor systems for study because it is the final link in the heat transfer chain between the reactor core and the ultimate heat sink. The focus of the investigation was on documenting from existing operational records the principal mechanisms of aging degradation of the system and on determining the adequacy of the current inservice surveillance and maintenance methods. The investigation revealed that the most prevalent degradation was related to corrosion of the piping, pumps, and valves forming the water passage from a natural source through the system.

The high-pressure safety injection system's function is to cool the reactor core in the event of a small-break loss-of-coolant accident (LOCA) and to prevent uncovering of the core. The investigation centered on determining the aging mechanisms of components in this system. The study, which examined several plant operational data bases, determined that pumps, nozzles (thermal sleeves), valves, and valve operators were the components most susceptible to malfunction because of aging degradation, while design deficiencies and procedure or personnel error have been cited as the dominant identifiable root causes of system failure. Subsequent planned research will examine some advanced methods of surveillance and maintenance of the system, as well as recommend thresholds of acceptable performance and system reliability for license renewal consideration.

Based on information derived from operating experience records, nuclear industry reports, and manufacturers' supplied information, initial research studies were completed, and reports were issued on electric motors (NUREG/CR-4156) and battery chargers and inverters (NUREG/CR-4564). Predominant electric motor failure modes are associated with the stator insulation system and the bearings. The failure mechanisms for stator insulation included loose laminations, shorted windings, overheating, and corrosion of electrical connections. The battery charger and inverter capacitors, transformers, inductors, and silicon-controlled rectifiers are the components most susceptible to aging. The research report concluded that overheating and electrical transients are two major causes of charger and inverter failures. Based on these research results, a national standard, IEEE Std. 650-1979, is being revised to reflect the research findings.

Batteries are installed at nuclear facilities to provide power to critical functions in the event of loss of all a.c. power. Batteries provide power for equipment vital to safety shutdown, control, and monitoring of plant parameters. Redundant batteries are installed to ensure that at least one safety-related train of equipment is available given an assumed single failure in any safety system, including the Class 1E d.c. power system. Therefore, reliable operation of the batteries is necessary to ensure safety of a nuclear facility. An initial study to evaluate aging effects on safety-related batteries in nuclear power plants has been completed. The study identifies materials used in battery construction, stressors, and aging mechanisms; presents operating and testing experience; analyzes battery-failure events reported in various data bases; and evaluates recommended maintenance practices.

Results of the evaluation of stressors indicate that the single most important aging-related stress mechanism for batteries is thermally induced oxidation of the grids and top conductors that are usually made of a lead-calcium alloy. Oxidation of the grids causes the plates (including grids) to swell, causing poor contact between the grid and the active material in the plate, and results in decreased capacity of the battery. Plate growth ultimately results in stressing the containers and covers, causing cracks to develop in the containers, and subsequent loss of electrolyte.

Evaluation of operating experience combined with testing of naturally aged batteries shows that cracking of the containers and oxidation (flaking) of the lead are significant problems. Seismic testing of five models of naturally aged batteries has identified oxidation of the lead, deterioration of separators, and cracking of containers as problems with batteries. These problems occur more frequently in old batteries near their end of life.

Emergency diesel generators (EDGs) used in nuclear power plants are exposed to aging stressors from the environment and from operational practices. A review of over 2,000 failures associated with EDG systems revealed that roughly half the failures appear to be caused by some form of aging degradation. Examination of the various failure causes revealed that fasteners vibrating loose and metal fatigue are major contributors to EDG system failures. Components of the instrumentation and control system (including governor, alarm system, and control air system), the cooling system pumps and piping, the fuel system injector pumps and engine piping, and the turbocharger are particularly sensitive to vibration-induced failure.

Some conclusions drawn from the survey performed on EDG systems are:

(1) Monthly testing of the EDGs should not be used to gather statistical information on the EDG start and run reliability, but only to obtain data that indicate engine operability and status; testing requirements should be modified to minimize the stresses caused by fast start times.

(2) Preventive maintenance programs for EDG systems should be improved to include governor maintenance and instrument and gage calibration and to help mitigate vibration stressors.

(3) Major engine overhauls that are based entirely upon inspection needs are not recommended.

The Shippingport (Pa.) nuclear power plant, now undergoing decommissioning, is a major source of naturally aged equipment for the nuclear plant aging research program component and system evaluations. As the first U.S. large-scale, central-station nuclear plant, Shippingport station is similar to current commercial PWRs in design and operation. Its 25-year service (1957-1982) exceeds the operating times of most currently operating nuclear plants. Also, because of substantial modifications during the mid-1960s and 1970s, Shippingport offers unique examples of identical or similar equipment used side-by-side, but representing different vintages and degrees of aging. A comprehensive *in situ* evaluation of 46 Shippingport station electric components and circuits representing more than 1,600 individual measurements of insulation resistance, circuit resistance, capacitance, inductance, and impedance has been conducted. In addition, the ferrite content of the cast austenitic stainless steel primary system main valves and coolant pump volutes has been measured *in situ* to identify candidate materials for NRC-sponsored thermal embrittlement studies. These *in situ* measurements indicated that nine of the 24 cast primary system components had sufficiently high ferrite levels to make them of interest for detailed materials studies. More than 100 Shippingport components and samples have been selected for evaluation and testing through site visits by NRC and contractor experts representing a range of disciplines and interests. These items include battery chargers, inverters, relays, breakers, switches, power and control cables, electrical penetrations, check valves, solenoid valves, and motor-operated valves. Samples of piping from various plant systems also have been acquired for radiological characterization studies, and samples from the primary system check valves, main stop valves, and main coolant pumps will be used for materials degradation studies.

Residual Life Assessment of Major Components and Structures. The capability to predict the residual operational lives of major LWR components and structures can be important in resolving technical issues associated with plant aging and license renewal. It is the objective of the residual life assessment program to integrate technical information on component and structural aging effects and to develop residual life assessment models and predictive procedures to be used in the regulatory process in regard to plant aging issues. These models and procedures must consider the effects of original design, component operation, component inservice inspection and testing programs, maintenance procedures, and the results obtained by the surveillance and monitoring of appropriate performance indicators.

To date, the major LWR components and structures whose aging will affect plant safety have been identified. The analysis of these components is proceeding to determine the various component age degradation sites, life-limiting processes, and potential aging failure modes.

Decommissioning

The NRC continues to develop an information base for decommissioning LWRs and other nuclear facilities. Reports on decommissioning cost estimate updates, as well as progress reports on activities and information being obtained from actual or related decommissionings of LWR activities, are in preparation.

In 1987, a draft of the final rule amendments, the supplementary information thereto, and the regulatory analysis were prepared and concurred in by cognizant NRC offices. At the Commission's request, a briefing on the status of the decommissioning rule was held, and at the end of the fiscal year the Commission comments made at the briefing were being incorporated into the rule package.

Effective amendments to 10 CFR Parts 30, 40, 50, 70, and 72 were published in January 1987, setting forth requirements that licensees notify NRC in the event they are involved in bankruptcy proceedings. The purpose of this rulemaking is to have rules in place that require prompt notification to the NRC of licensee bankruptcy, alerting NRC in a timely manner. The NRC can then take necessary actions to deal with potential hazards to public health and safety that may be posed by a licensee that does not have the financial resources to properly handle licensed radioactive material or to clean up possible contamination.

Spent Fuel Storage

Comment letters were received from 195 States, organizations, and individuals on the proposed revision to Part 72 that incorporates (1) the effect of experience in using Part 72 in licensing independent spent fuel storage installations, and (2) the rule changes needed to extend the rule provisions to cover the licensing criteria for both short and long-term storage of spent fuel and high-level radioactive waste in monitored retrievable storage facilities to be constructed and operated by the Department of Energy. These letters were categorized and analyzed, and based on this analysis the final rule amendments were being reviewed for concurrence to send to the Commission for action at the end of the fiscal year.

Chemical Decontamination

The NRC continued to develop an information base for reducing occupational doses in nuclear power plants and for assessing the impact of decontaminations on nuclear plant solidification systems. Measurements were made of recontamination rates following chemical decontaminations at the Hatch Unit 2 (Ga.), Pilgrim (Mass.), Peach Bottom Unit 2 (Pa.), and Limerick Unit 1 (Pa.) nuclear power plants. A report analyzing these results and similar earlier measurements conducted at other nuclear power plants was being prepared at the end of the fiscal year. NUREG/CR-3444, published in 1987, describes the impact of LWR decontaminations on solidification and waste disposal.

REACTOR EQUIPMENT QUALIFICATION

Survival of Electric Equipment

Research was completed during the report period on the survival of safety-related electric equipment when exposed to a hydrogen burn environment resulting from hydrogen deflagration, such as might occur in a LOCA core melt accident in PWRs with dry containment buildings (NUREG/CR-4763). The results of this research support resolution of Generic Issue 121 with a regulatory position on "Hydrogen Control for Large, Dry and Subatmospheric PWR Containments."

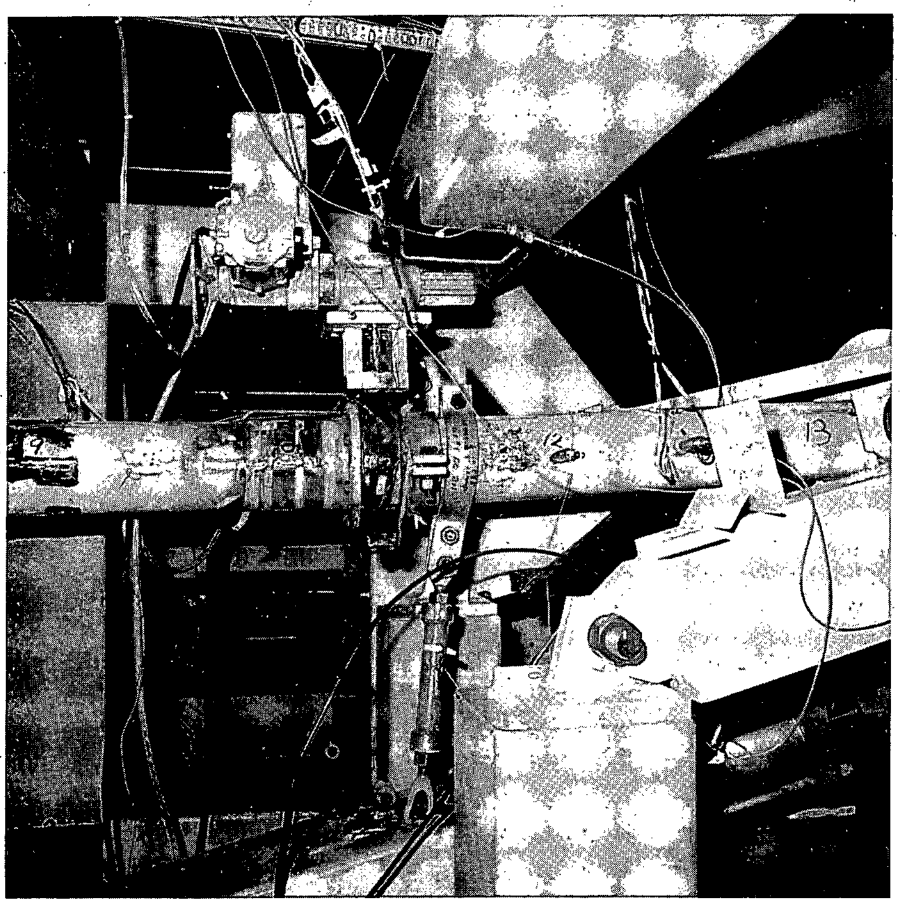
Equipment temperatures were calculated for typical PWR large, dry, and sub-atmospheric containments using multicompartment models of the TMI and the Surry nuclear plants and the HECTR hydrogen distribution and combustion code. A 75 percent metal-water reaction was assumed for the core melt accident, as postulated in the Hydrogen Control Rule, 544 of 10 CFR Part 50. Hydrogen concentrations in the detonable range were calculated to occur throughout containment for the smaller PWR subatmospheric containments in the absence of igniters. Detonable concentrations were calculated to be formed in all large, dry PWR containment subcompartments containing the LOCA system break in the absence of hydrogen igniters. Multiple hydrogen burns are possible in these subcompartments when igniters that can lead to equipment temperatures threatening their operation are present. The equipment temperatures from the analyses were verified by tests in the SCETCH facility at Sandia National Laboratories, which simulated the thermal and steam environment of a LOCA followed by a hydrogen burn.

Environmental Qualification of Mechanical Equipment

Experimental research conducted in 1987 demonstrated that typical dual-valve piping systems that penetrate the containment building will not experience failures when these piping systems are subjected to very large forces from containment wall deflection. If a LOCA occurs inside containment, the internal environment will cause the containment wall to deflect outward and large forces can be transmitted to internal and external valves and piping supports through the attached piping. These forces can cause the piping and supports to deform plastically and create stresses in the valves that may cause binding of movable parts and affect the operability of these safety components. The penetration structure welds may crack under these loads and result in a leakage path to the outside atmosphere. Although all these forces and stresses were simulated for three different, but typical, dual-valve piping systems, there was no evidence of system failures from loads resulting from severe accident environments. These tests have provided the NRC staff with baseline data for assessing containment leak integrity and for demonstrating dual-valve functionality under severe accident conditions.

Dynamic Qualification of Equipment

Dynamic tests completed on an aged gate valve and actuator, in a cooperative program with the Federal Republic of Germany, have demonstrated the seismic ruggedness of these types of components. The valve and actuator were manufactured in mid-1950 and had been in service in the

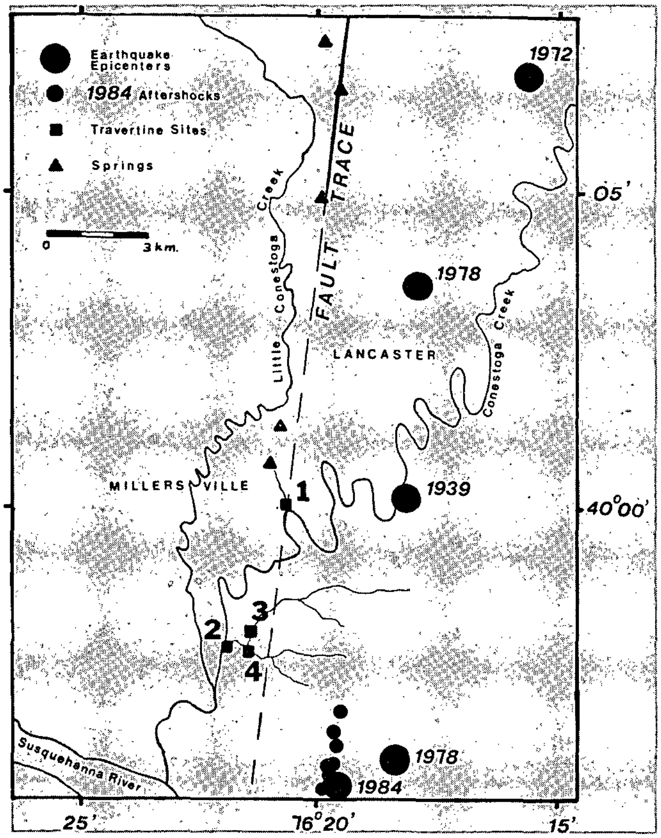


Shown in the photo is one of the eight-inch dual-valve piping systems that was tested as part of the research program to demonstrate the ability of such valves to function under severe accident conditions. When the containment wall displacement was simulated by moving the section at the right (marked 13), the piping between the valve and location 13 deformed, causing the pipe clamp to the right of the valve and attached to the bottom strut to slide left. The force transmitted to the strut caused its bottom-bearing rod end to fail. The valve leaked after the severe accident loading; however, another butterfly valve outside containment did not leak at any time. Thus, the capability of this dual-valve system to perform during and following a severe accident was successfully demonstrated.

Shippingport facility for approximately 25 years. Although these components were manufactured during the time when earthquake qualification standards did not exist, tests like these are confirming that classes of equipment in older plants have adequate margins to resist earth-quakes. The actuator did experience an operational anomaly during the combined seismic and hot flow tests, but the evidence points to causes unrelated to the seismic loads. This effort is also providing a data base for validating computer codes used for pipe design and for identifying multiload characteristics that may be required for qualifying specific valve designs.

Other research has resulted in the development of a methodology for seismically qualifying safety equipment by use of experience data. Therefore, new as well as old components that are similar to other previously qualified components can also be qualified by use of this methodology. This provides the staff with a tool for assessing the safety margins of some new and old components without requiring lengthy and expensive analyses or tests.

Another effort focused on developing a plan for determining whether isolation valves in certain high-energy piping systems will prevent leakage if the pipe breaks outside containment. The plan derives from the need to conduct tests on these valves to ensure that the method used to size actuators results in conservative estimates of valve opening and closing forces. Of particular importance is the need to understand the effect of corrosive surfaces that can impede either the opening or the closing of the valve disc. Research continues to focus on resolving this high-priority safety issue. All of the efforts described above are providing the NRC staff with a basis for evaluating the integrity of valves when subjected to various kinds of dynamic loads.



As part of a seismic research program funded by the NRC, Pennsylvania State University seismologists studied the Fruitville Fault near Lancaster, Pa., site of several recent earthquakes. The map shows travertine sites, springs, and epicenters along the southern trace of the fault.

SEISMIC AND FIRE PROTECTION RESEARCH

Earth Sciences

The primary goal of the NRC research program in the geological sciences is to be able to define the potential for earthquakes at nuclear power plant sites and in the regions around the sites and to determine the possible effects earthquakes would have on the plants and their safety systems.

A major focus of the NRC research programs in geology, seismology, and geophysics continues to be identifying and defining potential earthquake sources or source zones in the Eastern United States and using that information in assessing seismic hazards with respect to nuclear power plants. Many unknowns exist regarding these issues, including a strong basis for seismic zonation, source mechanisms, characteristics of ground motions, and site-specific response. The NRC is addressing these uncertainties through research that encompasses sustained seismic monitoring, neotectonic investigations, exploring the earth's crust at hypocentral depths, and conducting ground motion studies.

The backbone of the NRC program in the Eastern United States has been the seismic networks deployed throughout the Eastern and Central United States. The NRC is currently funding seismograph networks in the following regions: Northeastern United States, Virginia, Charleston, S.C., the Southern Appalachian region, the New Madrid (Mo.) region, and Ohio and Indiana, eastern Kansas, and Oklahoma. The NRC has negotiated an interagency agreement with the U.S. Geological Survey (USGS) to jointly support the establishment of the eastern portion of a national seismographic network.

Northeastern Neotectonics. As part of the seismic research program to improve NRC's ability to estimate seismic hazard in the Eastern United States, Columbia University and Pennsylvania State University have been investigating for the past several years seismically active regions in the northeast for evidence of Quaternary surface or near-surface tectonic deformation. Methods of accomplishing this have evolved from classical field geological techniques used in earlier studies of this region, to newer ones developed for neotectonic investigations in the New Madrid, Charleston, and California regions. These include search for paleo-seismic evidence of prehistoric earthquakes to extend the seismic record beyond the limited historic record, correlation Quaternary marine and fluvial terraces, and utilization of new, relatively shallow, high-resolution exploratory techniques, and equipment such as ground-penetrating radar.

Columbia University has completed its studies of surface structures in the epicentral area of the 1983 Goodnow earthquake in the Adirondack Mountains. No relationship between surface faults and the earthquake was found. Columbia researchers are now investigating seismically induced liquefaction features near Cape Ann, Mass., that are known to have occurred during the 1755 MMI VIII earthquake. They plan to use the data obtained to identify other such features that may have been caused by prehistoric earthquakes. They are also investigating the 125th Street fault in New York City for evidence of young displacements, the Lancaster, Pa., seismic zone, the Lower Hudson Valley-Eastern Newark Basin seismic zone, and the New Jersey Coastal Plain. The paleo-seismic investigations, by providing isotopic dates of large prehistoric earthquakes, have the potential for providing deterministic guidance for calculating return periods of large earthquakes in the Northeastern United States. This would be a major step in assessing seismic hazards in the Eastern United States.

A study performed by the Pennsylvania State University focused on the Lancaster, Pa., and Moodus, Conn., seismic zones. The Lancaster area reveals a north-south trending structural and seismic zone that cuts across the strike of the major Appalachian structures. The zone is favorably oriented to be activated by the prevailing compressional stress. The trend is outlined by epicenters, aftershocks, and focal plane of the 1984 earthquake, and geologic and geomorphic trends, which include diabase dikes, faults, springs, drainage, and lineaments.

At Moodus, no specific seismogenic structure has been found, but there are some characteristics that may be related to the seismicity. Lineaments at Moodus are more numerous and shorter than in adjacent areas, possibly because the crust is more broken up and hence more easily activated by local stresses. Stream-water samples from the seismic zone have higher than normal pH, which is consistent with influx of water from the subsurface along fractures. A recent deep borehole has found a water-filled zone at 3,200 feet, below the Honey Hill fault. Above this zone, stresses are higher than below. This may explain the shallowness of the hypocenters, which seem to be concentrated in the more highly stressed surface layer.

A third area of study was travertine deposits and their possible use as indicators of recent fault movement. It had been found previously in Virginia that travertines have formed on the downstream side of faults that cut limestone strata. Limestone crushed by the faults would be more easily dissolved and carried to the surface into streams, where travertine is deposited in locations where the saturated water is aerated, such as riffles and falls. In the present study, it has been found that near Lancaster very recent travertine deposition occurs downstream (either east or west) from the surface projection of the fault that is assumed to be associated with the seismicity. Thus, there is substantial evidence for the association of travertines with recent fault movements.

Charleston Studies. The NRC has funded over the past few years studies by the USGS and the University of South Carolina of soil deformed by liquefaction during the 1886 earthquake and of similar, but older, features (paleoliquefaction features) that were apparently formed by prehistoric earthquakes of about the same size. These investigations suggest recurrence intervals between 1,000 and 2,000 years for earthquakes of the same size as, or greater than, the Charleston event. To support the position that the Charleston seismic area is unique, or to demonstrate that such an assumption is not valid, the NRC has encouraged expanding the area of investigation to determine whether or not there are paleoliquefaction features elsewhere on the Atlantic Coastal Plain. The USGS has identified paleoliquefaction features beyond the immediate Charleston earthquake area and postulates that either a much larger earthquake occurred in the Charleston epicentral area, or earthquakes of similar size occurred prehistorically at other locations along the southern Atlantic coast.

In 1986, the NRC awarded a research contract to Ebasco Services Incorporated to look for paleoliquefaction structures throughout the Atlantic Coastal Plain. Ebasco researchers have applied the results of those studies to identifying other areas in the Atlantic Coastal Plain that have the potential for evidence of the occurrence of large prehistoric earthquakes. These sites are southeastern New Jersey-northeastern Delaware, the eastern part of the Central Virginia seismic zone, and northeastern North Carolina. Techniques of investigations and criteria developed at Charleston will be used to investigate these sites.

A study of the Charleston seismicity by Law Engineering Testing Company has employed two- and three-dimensional stress models to clarify causes of the seismicity and to complement the sparse stress data in this region. The stress models take topography, density, and plate boundary stresses into account to derive the stress distribution over the area. There are two major structures that should influence local stresses, namely, the Appalachians and the continental shelf edge. The models show that these features indeed generate large stresses. However, when ridge-push forces are taken into account, the stress near the Blue Ridge of the Appalachians is enhanced, whereas the shelf edge stress is largely cancelled. This corresponds to the observed seismicity, which is high in the Appalachians and low near the shelf edge. The Charleston area also emerges as a region of higher stress, while areas of minimal seismicity, such as eastern North Carolina and southwestern Georgia are characterized by low stress. It appears, therefore, that stress computations can be a valuable tool for analyzing the seismic potential of certain areas.

The University of South Carolina has performed shallow stratigraphic drilling in the seismic areas near Charleston and Bowman, S.C. The borehole data have shown the presence of a trough near the coast that may be bounded by faults. Seismicity seems to be concentrated at intersections between the trends. Relative travel time residuals from teleseismic data are being analyzed to gain additional information on the crustal structure in this area.

Virginia Piedmont. Virginia Polytechnic Institute has reprocessed and reinterpreted a seismic reflection profile acquired by the USGS near Route I-64 and shot new reflection profiles in the Virginia Piedmont, in the central Virginia seismic zone and in a non-seismic area along the Roanoke River. This has led to a reinterpretation of the basement structure under the Piedmont and Coastal Plain. The data confirm a previously suspected upwarp of the Moho (Mohorovicicdiscontinuity) underneath the Coastal Plain that coincides with a major gravity high. The Piedmont and western portion of the Coastal Plain are now interpreted to be underlain by Grenville basement. The eastern portion of the Coastal Plain may be underlain by a basement equivalent to the Carolina State Belt, the two basements being separated by an extension of the Taconic Suture.

Comparisons of crustal configuration between seismic and non-seismic areas show that the seismic areas have a greater number of subsurface reflectors. This may indicate that the crust is more sheared and segmented than the more massive crust in surrounding areas. It is also significant that the seismic zone overlies the shallow portion of the Moho. The new reflection profiles have provided excellent reflections from the Moho, but data processing has not yet been completed.

Studies of the crustal structure in southeastern Tennessee by Georgia Institute of Technology have located a sedimentary basin that is associated with the relatively high seismicity of that area. Magnetic and gravity data, travel time residuals, and refraction data were used to derive the interpreted crustal structure. The sedimentary basin parallels the New York-Alabama lineament, which probably represents an ancient strike-slip fault. Thickness and seismic velocity of the crust differ on the two sides of the lineament. Northwest of the basin a relict Precambrian rift is postulated, which is truncated by the New York-Alabama lineament. Epicenters relocated on the basis of new velocity information correlate well with the main NE and NW trend lineaments in the area. A two-dimensional stress model shows high compressive stress in the seismic area. The stress is derived from local topography and density anomalies in the crust.

Southern Illinois Earthquake. A magnitude-5 earthquake occurred at about 7:50 p.m. on June 10, 1987, near Lawrenceville, Ill. It was felt over 15 states and southern Canada.

Geologically, the earthquake occurred in the vicinity of the Wabash Valley Fault System near its intersection with the LaSalle Anticline. Historically, earthquakes of this size have occurred before in this area of southern Illinois. The NRC has funded geologic research of faults in this fault system, but no evidence of recent activity was found. The earthquake triggered instruments at several nuclear power plants in the region, including Dresden, at 300 km from the epicenter, Cook at 390 km, Quad Cities at 400 km, Palisades at 420 km, and Prairie Island at 770 km. It was felt at other nuclear plants, but instruments were not triggered. Immediately after the main shock, NRC contractors from Memphis State University, the University of Kentucky, and St. Louis University deployed a temporary network of seismographs. Excellent records of this event and aftershocks were obtained and are still being analyzed.

New Madrid/Anna, Ohio. Purdue University has analyzed seismicity, geologic and geophysical data, and borehole information of the midcontinent region between New Madrid, Mo., and Anna, Ohio. The improved data base that has been developed over the past 10 years has led to the conclusion that two hypotheses can explain most of the midcontinent seismicity. The dominant mechanism is reactivation of existing zones of crustal weakness that are favorably oriented with respect to the NE-SW direction of the maximum compressive stress in this region. Local basement inhomogeneities are a second mechanism that may explain seismic activity of low magnitude.

The Anna, Ohio seismic zone has been investigated in detail, and there is no evidence of a structural connection of the area with the NE extension of the New Madrid rift. The newer data do support an extension of the Grenville front that locally trends N-S and is found just east of the area. The seismicity may be related to intersecting trends in the basement or to lithologic differences (inhomogeneities) in the basement.

Meers Fault Studies. The initial NRC-funded investigations of the historically aseismic Meers Fault in Oklahoma have been completed. These investigations have shown, with several lines of evidence, that about 26 km of the fault have undergone recent displacement, the latest of which probably occurred 1,100 to 1,200 years ago. Cumulative displacements of up to five meters of reverse offset and a much larger left-lateral strike-slip offset were recorded.

A new contract was awarded to Geomatrix Consultants in May 1987. The purpose of the contract is to characterize completely the Meers Fault for seismic assessment and to determine if there are other such faults that may have been reactivated in the Quaternary within the Frontal Fault System.

Another fault in this system, which is east of the Meers Fault, is being investigated under an NRC grant to the University of Arkansas. Geologic and geomorphic evidence regarding this fault, the Washita Valley Fault, suggests Quaternary displacement. Like the Meers Fault, this fault is not known to be associated with historic seismicity.

These studies are extremely important not only to assess the seismic hazard posed by these faults but also to test the validity of an assumption used frequently in the licensing process, viz., that the lack of associated seismicity is an important criterion indicating that a fault is not capable within the meaning of Appendix A to 10 CFR Part 100.

Pacific Northwest. A major unknown concerning the Pacific Northwest is the nature of the Juan de Fuca subduction zone and its potential for generating a great earthquake. There is geological and geophysical evidence that subduction is taking place at a fairly rapid rate, but there have been no large-thrust earthquakes historically like those that characterize other subduction zones around the Pacific Ocean. The issue is whether subduction is occurring aseismically or seismically, but the historic record falls within the recurrence interval of large subduction zone earthquakes. The NRC is providing funding to the USGS to investigate paleo-seismic evidence that might have been induced by prehistoric large earthquakes. Evidence has been found in marsh and shallow marine deposits within bays and tidal estuaries along the coast of Washington and northern Oregon that suggest the occurrence of several great earthquakes during the Holocene, the last occurring about 400 years ago.

A second major issue in the Northwest is the nature of ground motion from a subduction zone earthquake. Along with the geologic investigations, the NRC is funding a USGS study in the Santiago, Chile region, the location of a magnitude 7.8 subduction zone earthquake in 1985. The study consists of analyzing all data from this event and its aftershocks to be able to determine the characteristics of strong subduction zone earthquake ground motion for use in nuclear licensing activities in the U.S. Pacific Northwest.

Soil Response to Earthquakes. A research program to validate dynamic stress models that would be capable of predicting soil settlement resulting from seismically induced liquefaction continues. The objective of the research, being conducted by the Army Corps of Engineers, is to evaluate various seismic settlement models identified in a previous phase of this project and reported in NUREG/CR-3880. During 1987, two two-dimensional plain strain centrifuge experiments simulating massive structures, such as nuclear power plants, were conducted at Cambridge University in England. The experiments were continued until liquefaction failure of the supporting soil was achieved. The data are being analyzed and compared to the predictions of the validated two-dimensional effective stress model TARA, developed during the course of this research program. In a related development, Japanese investigators have compared the results obtained from the TARA code with DIANA, an effective stress code developed by Dr. Zienkiewicz at the University of Swansea. While both solutions converged on the same answers on earthquake motions, pore pressures, and displacement, the TARA code used only a tiny fraction of the computer time taken by the DIANA code. It may be noted that the TARA code was developed and validated from data obtained from centrifuge testing using soil mechanics principles, while DIANA is a more theoretically exact and rigorous mathematical formulation, designed to predict the response of soil to earthquake motion. Research to expand the current research project to consider modeling three-dimensional effects and to update the two-dimensional TARA code is planned to start in fiscal year 1988.

Component Response to Earthquakes

Seismic Category I Structures Program. The static testing of two large reinforced concrete models representing a portion of a nuclear power plant building (i.e., shear wall and floor segment) was performed this year. This is part of a static and dynamic test series that began in 1985 and will conclude in 1988. The purpose of the test series is to investigate the large differences observed when analytical predictions of building response are compared with experimental data. Based on a preliminary evaluation, it appears that the 1987 static test data contradict previous dynamic test observations, i.e., an excellent comparison of analytically derived stiffness was obtained from the recent tests. Subsequent program activities will center on investigating the rationale for the differences obtained from the static and dynamic tests. The overall goal of this program is to assess (1) the ability of Category I structures other than the containment to sustain earthquake motions in excess of their original design bases, and (2) the effect that the changed building response has on the criteria used in the design of piping and equipment.

EPRI/NRC Piping and Fitting Dynamic Reliability Program. This cooperative EPRI/NRC research program was initiated in 1985 with three main objectives:

- To identify failure mechanisms and failure levels of piping components and systems under dynamic loadings.
- To provide a data base that will improve predictions of piping system response and failure resulting from high-level dynamic loads.
- To develop an improved and defensible set of piping design rules for inclusion into the ASME Code.

The majority of the experimental work was completed by the end of fiscal year 1987. A major milestone was reached in June 1987 when the Energy Technology Engineering Center (ETEC) completed the "System 1" series of design-level and high-level seismic input tests of a pressurized six-inch carbon steel piping system. The piping system was well instrumented, and the recorded response data will provide valuable benchmarks for future evaluation of linear and nonlinear piping analysis methods. Of immediate interest is that for the first time a failure of a pressurized prototypical piping system was achieved under very high seismic-like loads. An input scaled roughly 25-times higher than normal safe shutdown earthquake (SSE) design limits produced hanger and valve operator failures and ratcheting in elbows, but not leakage. The input was then scaled even higher and excessive ratcheting at an elbow resulted in rupture. ETEC also began construction of the stainless steel "System 2" in fiscal year 1987 and will complete testing by January 1988.

Over one-half of the 40 piping component tests were completed by ANCO Engineers by the end of fiscal year 1987. The rest will be completed by January 1988. Fatigue ratcheting specimen tests and water hammer tests are also currently under way.

The results of the pipe component and pipe system experiments have shown surprisingly consistent general trends:

(1) The results show that typical elastic piping design evaluations using the current ASME Code are very conservative for dynamic inertial loads. Margins to failure of 15 to 30 were usually observed.

(2) Dynamic failure is dependent upon cyclic effects, even at input levels of incredible earthquake size.

(3) Ratcheting and wall thinning led to the dynamic failure of pressurized piping.

(4) Cross-sectional collapse (as assumed by Equation 9 of the ASME Code) did not occur. It seems that dynamic load reversal prevents collapse.

(5) Failure locations were determined by loading and geometry and were independent of weldment locations.

(6) "Loss-of-flow" failures did not occur. Swelling occurred in the pressurized piping, and crimping was minimal in the unpressurized piping.

(7) Extensive testing at operating basis earthquake (OBE) and SSE levels produced no detectable permanent deformation or damage. Even at the five-SSE level, deformations were very localized and small.

Items (1) through (4) above indicate the need and justification for the future ASME Code criterion changes that will result from this program. Items (6) and (7) show the need to rethink piping "functionality" concerns.

General Electric is now heavily engaged with the tasks of identification, development, and evaluation of alternative piping design rules. These will produce proposed revisions to the ASME Boiler and Pressure Vessel Code for the dynamic load design criteria for Classes 1, 2, and 3 piping components. Several piping consultants review and support this effort, and both ASME and Pressure Vessel Research Committee (PVRC) standards groups are monitoring the progress. These proposed criterion revisions will be completed when the program ends in the spring of 1988.

Seismic Component Fragilities. Fragility data were developed for motor control centers, switchgears, panelboards, and d.c. power supplies. A new and more reliable single parameter fragility descriptor called the "average spectral acceleration" was developed to replace the "zero period acceleration" used previously. This information will be used in future seismic probabilistic risk assessments (PRAs) and margin studies to identify weaknesses and strengths in nuclear power plant seismic design and to assist in seismically related licensing decision-making.

A major additional task added to this program is related to establishing information to support the resolution of USI A-46 dealing with seismic qualfication of safety-related equipment. Electrical cabinet damping and amplification data have already been developed.

Validation of Methods. Probabilistic risk assessment methods for calculating seismic risks have been employed to clarify safety issues for nuclear power plants. The randomness of the seismic hazard, the uncertainties and variety of the data needed, and the inexactitude of the methodology raise questions of credibility with respect to the results of seismic PRAs. The objective of validation research is to obtain information that the NRC can use to develop criteria for judging predictions of the behavior of nuclear power plants subjected to large earthquakes and thereby improve the regulatory process. The predictive methods to be validated are used in both probabilistic and deterministic predictions.

Participation in cooperative research programs helps stretch available resources; the NRC is taking part in the three such efforts in this area:

(1) A soil-structure interaction (SSI) experiment at a site in Taiwan. This effort is in collaboration with the Electric Power Research Institute (EPRI) and the Taiwan Power Company. The objective of the experiment is to obtain data from a soft soil site that will test the fidelity of analytical predictions of SSI effects. Fourteen earthquakes have been recorded, three of which exceeded Richter magnitude 6.0. A workshop will be held in December 1987 at which the results of the experiment will be evaluated.

(2) The Phase II experiments being performed at the Heissdampfreaktor (HDR) facility in Kahl, Federal Republic of Germany, in collaboration with Kernforschungszentrum Karlsruhe (KfK). Results of the first series of tests are currently being evaluated. In those tests, the containment building was excited by a large eccentric-mass shaker, and the responses of a piping loop were recorded for different support conditions. Support conditions ranged from a stiff system, typical of early U.S. practice, to a very flexible system and included innovative systems intended to replace snubbers. Planning for the second series

of tests, to be run in the spring of 1988, in which the piping loop will be excited well into the inelastic range, were completed in 1987. The purpose of these experiments is to develop information about seismic margins.

(3) Tests of a 1/2.4 scale model of a PWR piping loop to be performed on the large shaker table in Tadotsu, Japan, in collaboration with the Japanese Ministry for International Trade and Industry (MITI). Final design of the experiment, to be carried out in April 1988, was completed in 1987. The experiment involves modification of the Japanese scale model, which will then be excited well into the inelastic range to develop information about seismic margins.

Seismic Design Margin Methods

Most seismic design experts agree that nuclear power plants are capable of withstanding earthquakes much larger than their original design basis without compromising their ability to safely shut down and remain in a safe shutdown condition. However, only recently, through the seismic design margins program, have the tools been available to effectively and efficiently quantify the inherent overall seismic capability of nuclear power plants and to provide results that can be used directly for licensing decisions. The successful completion of the Maine Yankee seismic margins review (NUREG/CR-4826) in March 1987 and the issuance of a safety evaluation report based on this review are major milestones in the seismic evaluation of nuclear power plants.

The Maine Yankee Atomic Power Station is a Combustion Engineering three-loop PWR located approximately four miles south of Wiscasset, Me. It started commercial operation in 1972. The design SSE has a horizontal acceleration of 0.1g.

The occurrence of two seismic events in the vicinity of the plant, one in 1979 and the other in 1982, prompted Maine Yankee to upgrade the capability of the plant to withstand a potential seismic event in excess of the original design basis event. Based on these upgrades and the inherent design capacity of the plant, Maine Yankee concluded that the plant structures, systems, and components had sufficient strength to withstand a seismic event of at least 0.2g with a Regulatory Guide 1.60 spectrum and still shut down without danger to the public health and safety. To assure the NRC that the plant could withstand earthquake motion greater than the design basis, the utility agreed to participate in the trial seismic margins review of the Maine Yankee plant. For this review, it was agreed that the seismic margin review earthquake level would be 0.3g with a 50th-percentile Newmark Hall Spectra defined in NUREG/CR-0098.

The Maine Yankee margins review followed the eight-step process outlined in the guidance of NUREG/CR-4482. The review involved the Maine Yankee Power Corporation, Yankee Atomic Electric, the NRC, Lawrence Livermore National Laboratory as project manager, and fragility and system analysis teams (EQE Incorporated and Energy Incorporated, respectively).

The Maine Yankee review demonstrated that the seismic design margins program methodology can be successfully implemented and be used to solve seismic licensing issues. Plant seismic vulnerabilities were found and upgraded as a result of the review. The importance of both peer review and utility cooperation was clearly shown.

In performing the review, some areas were revealed where the methodology's guidance was lacking. The resolutions of these procedural questions will improve the review guidelines. Areas of improvement include description of the seismic margins earthquake, the integration with non-seismic failures, use of equipment qualification data, HCLPF calculation techniques, and sampling.

The next major task of the seismic design margins program will be a trial plant review of a BWR. Because of the many advantages to be realized, this is planned as a cooperative effort between EPRI and the NRC. EPRI has extended the scope of the current seismic design margins methodology, adding guidance for new considerations such as soil failure and relay chatter. It has also made changes to make the methodology more utility oriented. An NRC panel of consultants has reviewed the EPRI methodology, and the NRC is currently in the process of issuing an endorsement of its use in the trial plant review. Negotiations are continuing with the plant owner of a candidate BWR.

Other smaller tasks now under way include a comparison of the two methods for calculating component HCLPFs: the Conservative Deterministic Failure Margins (CDFM) approach and the Fragility Analysis approach. Another study is looking at ways of extending seismic margin review guidance to better analyze plant damage states (providing radioactive release insights) and of improving the consideration of human factors and non-seismic failures.

It should be noted that the NRC Working Group on Seismic Design Margins, composed of representatives from both the Offices of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulation (NRR), oversees this program. This group is providing recommendations on the research program itself and how its products will integrate with other seismic policy action.

Preventing Damage To Reactor Cores

The program for preventing damage to reactor cores encompasses the operations of the reactor as a system, including control of power level, maintaining water in the reactor system, core cooling and heat removal, and maintaining proper coolant temperatures and pressures. It also includes consideration of operator actions as an integral part of the system. The research covers both normal and abnormal conditions, including accidents—such as a pipe break and loss-of-coolant accident—in which emergency systems are called upon to provide cooling water. A complete knowledge of the reactor's operating as a system makes it possible to define the conditions of operation that prevent core damage and hence maintain safety. The emphasis of this research is on prevention of severe core damage through understanding of both plant and human behavior during accidents. This information is used to ensure that plant equipment, operational procedures, and training are adequate to deal with operating events and to prevent serious accidents.

PLANT PERFORMANCE

The principal purpose of the thermal-hydraulic plant performance program is to improve the NRC's understanding of, and ability to predict, plant behavior during accidents and transients. This capability is needed to provide an assessment of the adequacy of LWR design and operations to ensure that transients will not lead to more serious accidents, and to modify NRC's regulations as required to ensure safe operation of nuclear power plants. Operating procedures and operator training are also assessed. The program continues to be based on both experiments and analysis methods. Experiments are needed to assess the ability of the codes to calculate complex plant transients. The codes are required because experimental data from scaled integral or separate-effect experiments are generally not directly applicable to the wide diversity of full-scale reactor designs. Current experimental facilities, e.g., Semiscale and LOFT, have been shut down or will reach the end of their experimental programs in the near future, as their limits of useful data are reached. Future facilities will center around Babcock & Wilcox (B&W) reactors and will include integral and separate-effect tests investigating the unique designs of B&W plants. In order to relax the present conservatisms in the Emergency Core Cooling System (ECCS) rule, the computer codes used for analyses must be carefully tested to determine their performance envelope over the range of plants and postulated loss-of-coolant accident (LOCA) scenarios. This testing is being achieved through the International Code Assessment Program and development of a Code Scaling Applicability and Uncertainty (CSAU) methodology. The CSAU methodology will also be applied to non-LOCA transients in B&W plants.

Multiloop Integral System Test (MIST) Program

The MIST facility is an integral facility designed to simulate thermal-hydraulic aspects of transients in B&W reactors of lowered-loop design. The facility is located in Alliance, Ohio. The experimental data are used to assess the capability of the NRC and industry thermal-hydraulic best-estimate codes in predicting B&W transients. In addition, the data obtained are expected to be sufficient to provide the small-break LOCA data base deemed necessary to satisfy NUREG-0737 (Clarification of TMI Action Plan Requirements) Item II.K.3.30, which required that small-break LOCA calculational models be compared to applicable data. This program is jointly sponsored by the NRC, B&W, the B&W Owners Group (B&WOG), and the Electric Power Research Institute (EPRI).

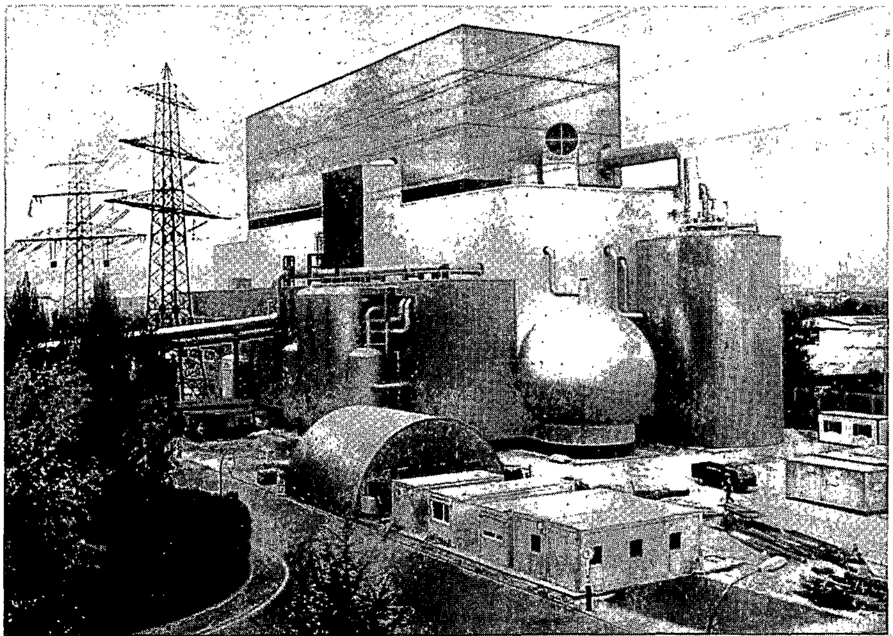
On September 3, 1987, the program successfully completed all testing under Phase III. Phase III testing had started in October 1985, with facility shakedown, followed by characterization testing. Composite testing commenced in June 1986, and since then B&W completed 50 tests, as delineated in the Phase III program. In addition, three tests were conducted in MIST for the Toledo Edison Company. These tests were funded independently by Toledo Edison to provide experimental data to benchmark a best-estimate computer code used by the utility to support a design change at the Davis-Besse (Ohio) nuclear power plant. Post-test analyses have also been made using NRC's best-estimate codes, RELAP5 and TRAC-PF1/MOD1. The remaining major activities under Phase III are data analysis, post-test analyses using the RELAP5 code, and report preparation. All these are scheduled for completion in February 1988.

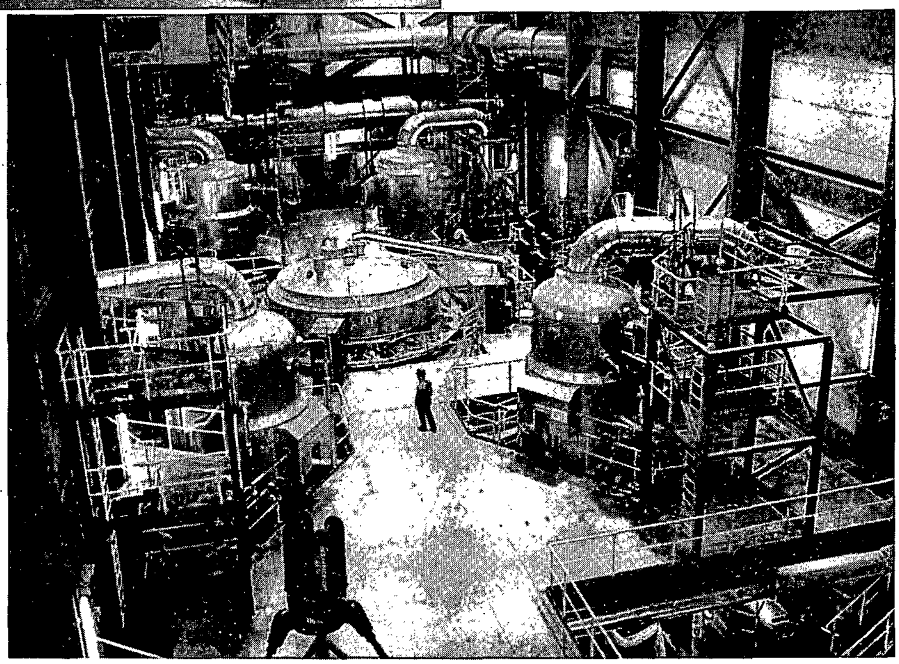
Facility modifications, such as installation of low-pressure injection system, reactor coolant pump seal leak sites, etc., to conduct Phase IV testing have been completed. Currently, the Phase IV program is jointly funded by NRC, B&W, and EPRI. The B&WOG has decided to participate in Phase IV and will be working with the rest of the participants in the near future to modify the contract. The last phase of the MIST project is scheduled to be completed in July 1988.

Basic Research. In support of the MIST program, adiabatic testing has been completed in the two- and four-inch visual air-water experimental loops that simulated the hot leg U-bend of a lowered-loop B&W plant to study the thermal-hydraulic phenomena associated with a small-break LOCA. The thermal-hydraulic phenomena studied were two-phase natural circulation, U-bend flow separation, and flow termination and flow resumption. Discussion of the data, as well as the conclusion reached on the thermal-hydraulic behavior of the hot leg U-bend during a small-break LOCA, has been published in NUREG reports. These data will provide a better understanding of the data from the large-scale MIST facility.

2D/3D

Many small-scale experiments have shown that ECCS rules prescribed in 10 CFR Part 50 have a large margin of conservatism. However, there was considerable uncertainty in ap-





In the photo above, the conduit from a coal-fired plant can be seen to the left, conveying steam into the Upper Plenum Test Facility (UPTF) at Mannheim, Federal Republic of Germany. The NRC provides advanced instrumentation and computer code analyses for experiments at this and other test facilities. At right, inside the Mannheim facility, the full-reactor-scale primary vessel and four steam generator simulators are used to conduct loss-of-coolant experiments. Steam and water flow conditions are monitored throughout the loops.

plying these small-scale test results to full-scale power reactors. It was therefore considered desirable to obtain large-scale test results so that the ECCS rule could be modified with more confidence, in order to relax the large conservatism believed inherent in the current rule. The 2D/3D International LOCA Research Program was initiated to fill this need. Under the 2D/3D program, the Japan Atomic Energy Research Institute (JAERI) constructed two large-scale test facilities called the Cylindrical Core Test Facility and the Slab Core Test Facility, and the Federal Republic of Germany constructed a full-scale test facility called the Upper Plenum Test Facility. The NRC provided these facilities with advanced two-phase flow instrumentation and computer code analyses using the Transient Reactor Analysis Code (TRAC).

ECCS Rule Revision

It is now known that the methods specified in the 10 CFR Part 50 Appendix K ECCS evaluation models, combined with other analysis methods currently in use, are highly conservative and that the actual cladding temperature that would occur during a LOCA would be much lower than those calculated using Appendix K methods. Therefore, on March 3, 1987, the NRC published in the *Federal Register* (52 FR 6334) proposed amendments to 10 CFR Part 50 and Appendix K. These proposed amendments were put forward because, since the promulgation of § 46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," of 10 CFR Part 50 and the acceptable and required features and models specified in

Appendix K to 10 CFR Part 50, considerable research has been performed that has greatly increased the understanding of ECCS performance during a LOCA. The revision to the ECCS rule would permit licensees to perform realistic evaluations of their ECCS based on the body of research currently available but accompanied by an evaluation of modeling uncertainties. The revised rule would also provide guidance with respect to the reporting of errors or changes appearing in evaluation models. The comment period for the draft rule expired on July 1, 1987. Twenty-six comment letters were received. In general, the comments were supportive of the rule and the approach taken by the Commission. A few relevant issues were brought up by rule commenters, and these will be addressed independently of this rulemaking. Therefore, the final amendment will be unchanged from the proposed rule. A Commission paper is currently being prepared, and publication of the rule is expected in April 1988.

ROSA-IV Liquid Holdup And Core Level Depression

A certain combination of reactor design characteristics may produce a core liquid level depression during a small-break LOCA and consequently a core heatup. Therefore, the liquid holdup and core level depression phenomena during a small-break LOCA have been studied in both the Semiscale tests and the ROSA-IV Large Scale Test Facility (LSTF) in Japan. To understand the physical mechanism behind these phenomena, several test results have been analyzed by NRC contractors during 1987. In addition, a four-loop Westinghouse calculational model has been developed for analyzing these phenomena. The primary purpose of this calculation is not to investigate the scaling to a full-scale reactor, but rather to uncover any scaling compromises attributable to the limitations of the LSTF, i.e., non-nuclear fuel and the lack of a baffle region.

Based on the experiments run to date, it appears that a 5 percent break produces the maximum core uncovery, with a duration sufficient to cause some core heatup. The differential pressure in the upflow portion of the hot leg and steam generator will be small at normal reactor decay power levels, so that the core uncovery will not be below the loop seal elevation. The magnitude of core heatup attributable to early core uncovery in all the experiments with realistic core decay power has been of such small magnitude as not to pose any threat to core integrity. Thus, previous regulatory concerns about the magnitude and duration of potential unexpected core breakup during a small-break LOCA have been resolved.

ROSA-IV Depressurization Process for Prevention of Direct Containment Heating

Three ROSA-IV test results have been used to gain a better understanding of the phenomena occurring during the depressurization process. Of concern is the need, after a possible vessel melt-through, to depressurize the reactor vessel and avoid the potential for direct containment heating. The tests are: (1) a 0.5 percent cold leg break test with no high-pressure injection flow, (2) a test simulating total loss of feedwater with three power-operated relief valves stuck open, and (3) a test simulating total loss of feedwater with one such valve stuck open.

The tests show that with existing power-operated relief valves (PORVs) and accumulator injection pressure setpoints, the core could start to heat up before the pressure can be lowered below the accumulator setpoint in a station blackout transient without auxiliary feedwater. At the time of cladding temperature rise, the pressure is still above the accumulator setpoint. Since the ROSA-IV facility is one of the largest system test facilities in the world and scaled well to simulate transients, it is important to be alert to the possibility of the occurrence of similar phenomena in reactor transients.

Tests show that higher accumulator injection pressure setpoints will help the quenching process since the accumulator water will be available for cooling of the core at the beginning of the core temperature rise. Tests also show that, as the number of PORV valves is increased, the pressure at the onset of the core heatup should drop further. It should be possible to lower the pressure during the front end of the transient by adding PORVs.

Technical Integration of Thermal Hydraulics

In response to an October 1985 request from the Executive Dirctor for Operations, the staff developed and published NUREG-1244, "Plan for Integrating Technical Activities Within the U.S. Nuclear Regulatory Commission and Its Contractors in the Area of Thermal Hydraulics," dated April 1987. The plan makes specific recommendations to improve and accelerate the integration of research results into the regulatory process, including establishment of a Regulatory Research Review Group (RRRG) and preparation of summary reports on completed research. The RRRG met frequently in 1987 to coordinate thermal-hydraulic needs and work.

The plan was further implemented by establishing a Thermal-Hydraulic Technical Support Center (TSC) at the Idaho National Engineering Laboratory (INEL). The principal purpose of the TSC is to ensure the continuing availability of the experience and depth of expertise needed to provide a response capability for priority issues or studies, as well as to perform ongoing work needed by the NRC. Priority studies would use staff from the other major program areas in the TSC, and other disciplines as needed, to resolve regulatory issues. In 1987, the priorities were (1) support for the revision of the ECCS rule and (2) development of methods by which the NRC staff can independently evaluate B&W Owners Group recommendations for safety improvements in B&W reactors.

One of the major TSC priority studies is the development of integrated methods by which to evaluate B&W plant safety and improvements resulting from the B&W Owners Group study. The effort entails the integration of methods of thermal-hydraulic transient analyses (computer codes), probabilistic risk assessment (PRA) of accident sequences, and human reliability evaluations of plant operator responses. Analyses have been completed for the response of plants for each major PWR vendor (Oconee (S.C.), Calvert Cliffs (Md.), and Robinson (S.C.) to three risk-significant transients (loss of feedwater, small-break LOCA, and steam generator overfill). Principal conclusions from the study were:

(1) Transients involving the steam generator proceed more rapidly at Oconee than at the other plants, which could lead to more severe consequences at Oconee.

(2) In the transients studied, there was no greater probability for operator non-response at Oconee than at the other two PWRs.

(3) There is a strong correlation between the correct and timely response of operators and performance-shaping factors affecting those tasks, e.g., training, procedures, and stress.

(4) Guidelines were proposed that could constitute a positive influence on performance-shaping factors and, thus, on plant safety.

Scaling Relationships for Future Integral Facilities

By the end of 1988, all major integral thermal-hydraulic test facilities in the United States are scheduled to be shut down. This move will affect the NRC's ability to provide timely resolution of future unforeseen safety issues with a high level of technical confidence. The NRC, therefore, evaluated available options for maintaining a continuing experimental capability for LWRs. To provide the most comprehensive and cost-effective set of options, a scaling study was begun in 1985 to evaluate capabilities and the costs associated with several alternative scaling approaches to test facility designs. The approaches used included those of current facilities and also newer approaches that could emphasize more realistic simulations of reactor operational transients. The scaling study results were published in NUREG/CR-4824, "Evaluation of Integral Continuing Experimental Capability (CEC) Concepts for Light Water Reactor Research-PWR Scaling Concepts," dated February 1987. The study concluded that the scaling concept generally used on existing facilities incorporating full height components and full system pressure provides the greatest transient fidelity. However, the study also concluded that the greatest remaining uncertainties involve multi-dimensional integral system response and, therefore, that CEC should be a nearly photographic reduction of a full-scale plant. A working concept was developed modeling a B&W plant; key features are full pressure, 150 ft_4 primary coolant volume, and length-to-diameter ratio of 2.0. The system has been modeled by the TRAC-PWR code and the calculated transient response of the facility is being evaluated. However, since a need for this facility commensurate with its estimated $15 million cost has not been established, the current work is being terminated and a re-evaluation scheduled for fiscal year 1990, by which time a need may be established for the advanced LWRs being proposed by industry.

Development and Assessment of Codes

The current versions of the computer codes used to simulate plant response to various transients and to assess procedures and operator training are TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1. In 1987, needed improvements were documented in a plan, which defines the need for each improvement; the models, correlations, or data bases that will be examined to effect the improvement; the degree of effort involved; and the extent to which the improvement will reduce uncertainty. During 1987, code improvement was carried out to provide improved simulation capability in these codes; with a focus on defining the accuracy of the TRAC-PWR code for large-break LOCAs, in connection with the forthcoming revision to 10 CFR Part 50 Appendix K, allowing realistic analysis of LOCAs.

A significant element of the NRC code improvement program is the International Code Assessment and Applications Program (ICAP), which was organized by the NRC to provide information for determining the scalability, applicability, and uncertainty of the transient codes. In addition to the NRC, ICAP enjoys the participation of the following nations and organizations: Belgium, Finland, France, Italy, Japan, Korea, Netherlands, Spain, Sweden, Switzerland, Taiwan, United Kingdom, the Federal Republic of Germany, and the CEC Joint Research Center Ispra Establishment. Information on code assessment also results from the 2D/3D, ROSA-IV, and MIST experimental programs. ICAP is a multi-year program that will continue through 1990. With the shutdown of all domestic experimental facilities except MIST, ICAP provides the NRC with most of its experimental data needs.

HUMAN PERFORMANCE

The human performance program is designed to improve NRC's understanding of the effect of human behavior on nuclear power plant risk. Operating experience has shown that a key element in preventing damage to the reactor core is the ability of plant operators to recognize conditions that could potentially lead to core damage and to respond to those conditions by taking appropriate remedial action. Research is being carried out to characterize human errors that have occurred at nuclear power plants and to evaluate the need for, and nature of, improved diagnostic tools or other operating aids to enhance the effectiveness of plant operators in responding to transients.

Human Factors Research

In response to recommendations made by the National Research Council in their report entitled "Revitalizing Nuclear Safety Research" (published in December 1986), RES initiated efforts to develop an expanded program in human factors research. The Reliability and Human Factors Branch in the Division of Reactor and Plant Systems was formed, as part of the April 1987 NRC reorganization, to take the lead in this area. Initial efforts in fiscal year 1987 included coordinating research needs with other NRC offices and developing a preliminary draft of a human factors research program plan. In December 1987, the National Academy of Sciences will also issue a report specifically on human factors research needs for U.S. nuclear power plants. Recommendations from this report will be factored into the final version of NRC's human factors research program plan to develop improved data and tools to support specific licensing actions and inspections of the quality of human performance at nuclear power plants and to support more general regulatory decision-making in the area of personnel utilization. The program involves six topic areas—including man-machine interface, procedures, organization and management, qualifications and training, maintenance, and the ongoing human reliability program discussed below.

Human Reliability Assessment

This continuing RES program provides the tools and data necessary for (1) performing assessments of human performance for use in plant PRA studies and (2) systematically applying the results of those studies in the resolution of generic issues and subsequent regulatory decision-making. Major products of 1987 research included (1) initiation of a Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) for processing, storing, and making available human error probability and hardware failure data to the PRA community; (2) implementation of a Maintenance Personnel Performance Simulation (MAPPS) computer code to analyze plant maintenance practices and predict benefits for various remedial actions; and (3) development of an artificial intelligence-based Cognitive Environment Simulation (CES) to model the decision-making behavior of plant personnel and a Cognitive Reliability Analysis Technique (CREATE) to develop quantitative probabilistic estimates of the cognitive decision-making process for use in PRA studies.

RELIABILITY OF REACTOR SYSTEMS

The program on reliability of reactor systems seeks to help reduce nuclear power plant risk through improved equipment reliability. Nuclear power plant engineered safety systems are designed with a high degree of reliability to prevent reactor core damage. That level of reliability may be degraded over the life of the plant by factors such as aging, poor quality control, or improper maintenance and testing. To ensure that critical safety systems and related components continue to provide an appropriate level of safety and reliability, research is being carried out to identify the principal causes of equipment and safety system malfunctions, to evaluate and document various programs that have been used in industry and elsewhere to improve reliability (including test and maintenance requirements), and to contribute to the development of performance indicators by which NRC can objectively monitor trends in licensee performance in maintaining the level of plant safety at acceptable levels or, as a corollary, to provide a sound basis for the Commission to take rapid and effective enforcement action whenever needed. To translate reliability methods into effective tools that facilitate the regulatory decision process, research will focus on methods that integrate dependent failure analysis, systems reliability, operational reliability, and operator reliability into probabilistic safety analysis. This research will provide more objective and better predictive plant performance indicators, which will serve as a tool to aid the Commission in decision-making about plant safety.

Performance Indicators

A research program has been undertaken to support NRC's development of plant performance indicators. In 1987, the Office for Analysis and Evaluation of Operational Data implemented an initial set of performance indicators for use by NRC. These performance indicators are logically related to safety in a qualitative way; but a reliability/risk-based method for quantitatively evaluating and integrating indicators was not yet available. RES is supporting this program by developing a method for reliability/risk-based evaluation of performance indicators. The results are intended to strengthen NRC's use of performance indicators (1) to improve Systematic Assessment of Licensee Performance (SALP) and (2) to identify trends of declining or improving performance between SALP reviews.

During 1987, a research project conducted by the Brookhaven National Laboratory and SAIC developed an improved method for monitoring trends in the availability of important safety systems. The NRC staff is now evaluating this method for possible future use.

ACCIDENT MANAGEMENT

Much of the work performed to develop the implementation of the Commission's severe accident policy has focused on severe accident phenomenology and methods by which to systematically discover severe accident vulnerabilities. An international consensus has already emerged that the cause and consequences of a severe core damage accident can be greatly influenced by the operator's actions. If operators are properly trained to diagnose severe accidents, to take beneficial actions when needed, and, most importantly, to refrain from specific actions that can have adverse effects, then the consequences of a severe accident can be significantly reduced. The NRC has initiated a substantial program to examine the efficacy of generic accident management strategies.

Individual Plant Examinations

During 1987, the accident management program developed five sets of guidelines and criteria for use in Individual Plant Examinations (IPEs), an integrated systematic approach (using either a method developed by the Industry Degraded Core Rulemaking (IDCOR) group or a more complete PRA method) to examine each nuclear power plant now operating and under construction for possible significant risk contributors that might otherwise be overlooked, the so-called "outliers." These guidelines and criteria apply to plant features and operator actions that were found to be important to either the prevention or mitigation of severe accidents. They incorporate the insights gained from industry-sponsored PRAs, NRC source term studies, and IDCOR reference plant analysis.

PLANT TECHNICAL SPECIFICATIONS

Technical specifications are design and procedural limits that entail explicit restrictions on the operation of nuclear power plants and the maintenance of safety systems in a pre-accident-ready condition. In response to an EDO task group's findings on enhancing the safety impact of technical specifications (NUREG-1024), RES established a broad-based research program (Procedures for Evaluating Technical Specifications) in 1984 to examine approaches for developing a methodology to permit setting limiting conditions for operations (LCOs) and surveillance requirements based on reliability and risk analysis principles. During 1987, progress was made in the program in several areas.

For the first time, a rigorous basis was established to examine the influence of different parameters affecting diesel generator availability and the criteria in Regulatory Guide 1.108. NUREG/CR-4810 (dated May 1987) details this effort and discusses other related issues—such as the separation of demand and standby time-related failures, testing strategies (e.g., sequential, staggered, and adaptive), and the impact of maintenance activities on diesel test intervals in developing risk-effective surveillance test intervals. Revising Regulatory Guide 1.108 based on the study results will allow diesel accident unavailability to be more effectively monitored and controlled. Also investigated were risk control and regulatory considerations related to allowed cumulative outage times (ACOTs). If, based on ACOTs, LCOs were established for components, licensees could be permitted greater flexibility in being able to accommodate widely varying component downtimes. Another area of research focused on the importance of uncertainties of risk analyses related to modifications of technical specifications. This analysis provided an understanding of the range of uncertainty and insights for reviewing risk-based submittals.

Current plans include developing guidance for licensees concerning submittals requesting extensions to, or modifications of, technical specifications and establishing criteria for NRC review of such submittals. This program maintains close coordination with industry through the LWR owners groups, Nuclear Utilities Management and Resources Council, the Institute of Nuclear Power Operations, and the Electric Power Research Institute.

Reactor Containment Performance and Public Protection from Radiation

To ensure that existing regulations related to severe accidents (i.e., siting, general design criteria, emergency planning) adequately protect the public, research is needed to confirm the technical bases upon which theregulations 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 (movement in air and water, uptake by plants and animals) is also an important consideration in protecting the public from radiation. With this kind of information, the Commission will be 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 to ascertain when and where improvements in the regulations are necessary.

SOURCE TERMS

A "source term" is defined as the quantity, timing, and characteristics of the release of radioactive material to the environment following a postulated severe reactor accident. Source term technology is employed for a variety of regulatory applications, including plant siting evaluation; emergency planning; evaluation of the performance of engineered safety features, such as containment isolation and containment spray additives; qualification of safety-related electrical equipment for performance under accident conditions; environmental impact statements; post-accident monitoring requirements; and criteria for re-entry of a plant after an accident. In addition, an understanding and quantitative assessment of source terms is necessary for conducting probabilistic risk assessments, which are emerging as a significant contributor to the regulatory decision process. New information and insights on radioactive source terms may have an impact on rules, guides, and other regulatory practices in the aforementioned areas through implementation of the NRC Severe Accident Policy Statement.

In light of the emerging severe accident technology and its expanding data base, the NRC undertook a reassessment of the technical bases for estimating source terms. The purpose of the reassessment was to evaluate the data base for validation of source term codes, to calculate source terms for selected plants and sequences, to conduct a broad-based peer review, and to appraise plant risk and the regulatory significance of the reassessed source terms. A major document describing the advances in source term technology and the staff's technology assessment was published in July 1986 and is entitled "Reassessment of the Technical Bases for Estimating Source Terms" (NUREG-0956).

A reviewed and tested analytical tool, the NRC's Source Term Code Package, emerged from this study; the code package is capable of dealing with plant-specific variations in a realistic way. The Source Term Code Package has been used in a major new risk study that was incorporated into the draft Reactor Risk Reference Document (NUREG-1150), published in February 1987. Notwithstanding recent advancements in source term technology, a number of large technical uncertainties remain. In order to adequately address the areas of uncertainty in draft NUREG-1150, these uncertainties were examined in a report entitled "Uncertainty Papers on Severe Accident Source Terms" (NUREG-1265), issued in May 1987.

Fission Product Behavior

The chemical forms of fission products affect the transport characteristics of fission products in the reactor coolant system and containment. They therefore influence the magnitude of source terms during severe reactor accidents. The chemical form of iodine in the reactor coolant system was identified as amajor uncertainty, in NUREG-0956. A sensitivity study assessing the impact of two forms of iodine emerging from the reactor coolant system—low-volatility cesium iodide and gaseous hydrogen iodide—on iodine source terms was conducted for that report. The study was performed for only two accident sequences each, for a PWR and a BWR. The study considered the effects of chemical form on fission product transport and retention within the containment, but it did not consider revaporization from reactor coolant system surfaces, which is discussed below. Results indicated that certain iodine chemical forms produce a less direct impact on severe accident source terms than expected.

Fission products deposited on the reactor coolant system structural surfaces may subsequently heat up these surfaces when they decay. The increase in surface temperature may result in the revaporization of the deposited fission products. The consequence may be an increase in the overall source term leaving the plant. One of the factors affecting the extent of fission product revaporization is fission product chemical form. The chemical form(s) of a specific fission product influences the volatility of that fission product and therefore its potential for revaporization. The phenomenon is particularly important for delayed containment failure accidents where the source terms are otherwise small and the quantity of the revaporized fission products may become significant. An estimate of the extent of fission product revaporization and its impact on severe accident risks was made for NUREG-1150. The analysis showed the issue of fission product revaporization to be risk significant for certain plants.

At present, research is being conducted to develop theoretically based fission product chemistry models to predict fission product chemical forms during transport in the reactor coolant system and the containment. The chemistry models will also be experimentally supported when the data become available. Experiments are under way to provide the necessary data.

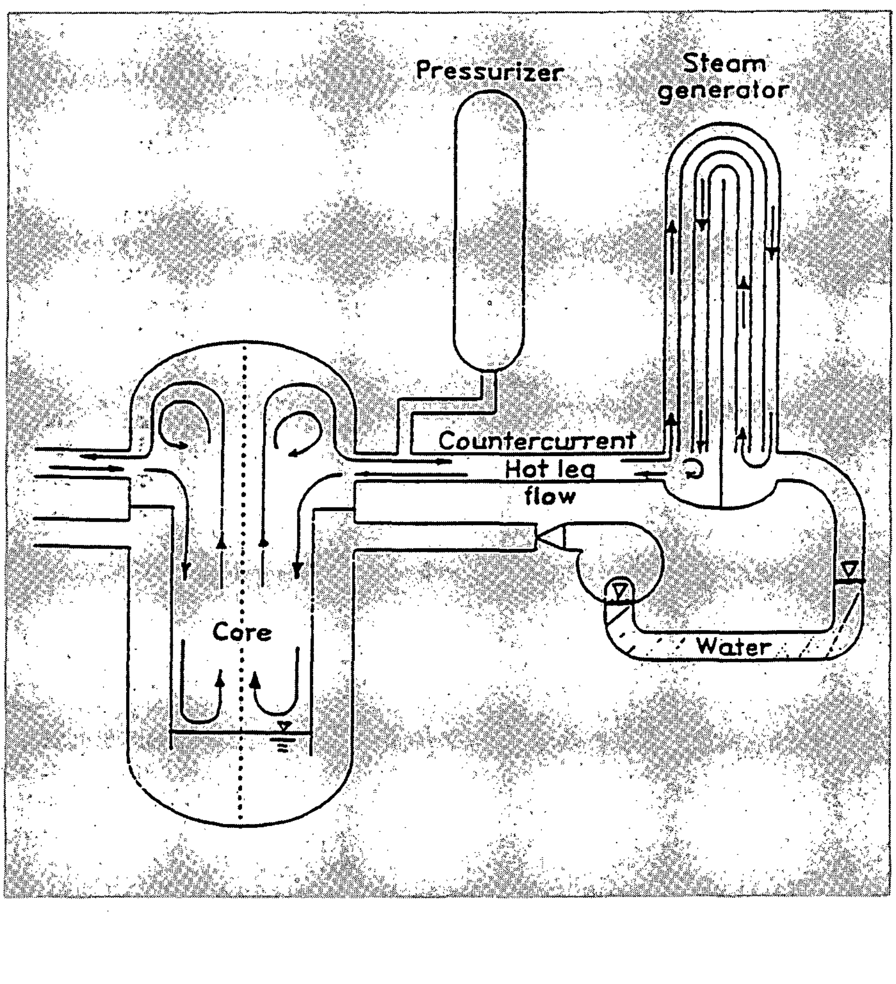
In addition to the above chemistry-related fission product work, the NRC is participating in an internationally sponsored project called LWR Aerosol Containment Experiments (LACE) to study the aerosol behavior of fission products within a containment and immediately after leaving a containment. The LACE program is being conducted by the Westinghouse Hanford Company. The six experiments now completed were performed to investigate inherent aerosol retention behavior in the containment or auxiliary buildings for postulated high-consequence accident conditions. These experiments will also provide a data base for validating containment aerosol and related thermal-hydraulic computer codes. Several NRC contractors are participating in the pre-test and post-test computer code calculations in support of the LACE experiments.

As part of the LA-5 and LA-6 vessel blowdown experiments, measurements were made of the localized deposition of particulates in the nearby ex-vessel area to determine if aerosols were plated out on the ground. A heavy condensed vapor atmospheric plume was observed under low wind speed, high humidity atmospheric conditions in LA-5. In LA-6 winds were much stronger and only a thin vapor cloud was observed. Tentative results indicate that as much as 20 percent of the discharged aerosol was deposited locally on the ground by the vapor cloud, with the remainder apparently transported aloft as the hot, wet plume evaporated. This indicates that localized deposition could be important in contaminating a site during a severe accident, but current off-site accident consequence models, which assume dry aerosol plumes, are not seriously in error in ignoring localized deposition.

Natural Circulation in Severe Accidents

Natural circulation in severe accidents is 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), as shown. This kind of multi-dimensional flow may exist during the core uncovery and core melt period of certain high-pressure severe accidents in a PWR. The flow serves as a means of transfering 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 to challenge structural integrity.

Based on the EPRI-sponsored experiments at a 1/7-scale Westinghouse test facility, the multi-dimensional natural circulation does indeed exist under certain simulated accident conditions. Data generated with the COMMIX code (valid for intact-core geometry and single-phase flow) was compared with the Westinghouse data, with good agreement found. To assess whether the natural circulation would also exist in a full-size reactor, COMMIX calculations for intact-core geometry were performed, and the results indicate that the natural circulation flow would also exist in a PWR. Since severe accidents involve core damage and core melt that is beyond the scope of COMMIX, MELPROG/TRAC calculations were performed for analyzing the in-vessel circulation in the Surry plant during a station blackout accident with the loss of auxiliary feedwater (the TMLB' accident). A comparative SCDAP/RELAP5 calculation was also performed for Surry using the counter-current flow information calculated with COMMIX. These preliminary calculations suggest that either the surge line or the hot leg connection at the vessel may fail by creep rupture at high temperature and pressure before the vessel lower



In this diagram, multi-dimensional natural circulation is depicted in the steam-filled space in the vessel, hot leg, and steam generator tubes of a pressurized water reactor.

head failure. As a result, high-pressure melt ejection may not occur during the TMLB' accident in a Westinghouse PWR. However, because uncertainties in these calculations are yet to be estimated or bounded, this conclusion is still preliminary in nature. Future work is needed to validate the MELPROG/TRAC code against data, to account for the fission product deposition heating of piping and structural surfaces (not modeled in the preliminary calculations), and to estimate or bound the uncertainties in the results.

REACTOR CONTAINMENT SAFETY

Structural Tests

Activity has continued on a set of programs whose objectives are to provide the data base required for the qualification of methods for predicting the response of LWR containment buildings during severe accidents (those beyond design basis events) and extreme earthquakes. This set of programs is examining the modes of containment failure that would result in the release of radioactive materials beyond the containment boundary. These modes include structural failure of the containment building, leakage through or past the penetrations (electrical or mechanical), failure of containment isolation systems, or failure of the basemat by the molten reactor core.

A 1/6-scale model of a reinforced concrete containment was tested to failure in July 1987. The containment was designed and built in conformance with the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers, just as are actual containments. The model was 22 feet in diameter and 37 feet in height and included representative features, such as four major penetrations (two airlocks and two equipment hatches) and several smaller penetrations that passed both separately and in clusters through the containment wall.

The containment had a design pressure of 46 psig. Pressure was increased in steps until failure occurred at 145 psig. At that point a major tear, 20 inches long, developed in the liner. Leakage through that tear overwhelmed the ability of the pressurization system. Additional minor tears were also present in the liner but the concrete outer structure, although visibly cracked, did not show great distress.

Post-test analyses will focus on the measurements of strain and displacement taken at each discrete pressure step to evaluate the accuracy of pre-test predictions made using different analytical techniques. Nine organizations, including three from the United States, three from the United Kingdom, and one each from France, Italy, and the Federal Repulic of Germany performed pre-test predictions and will participate in the post-test evaluation. Twelve hundred channels of data were recorded during the experiment; analysis of these data will permit assessment of the accuracy of the predictive methods.

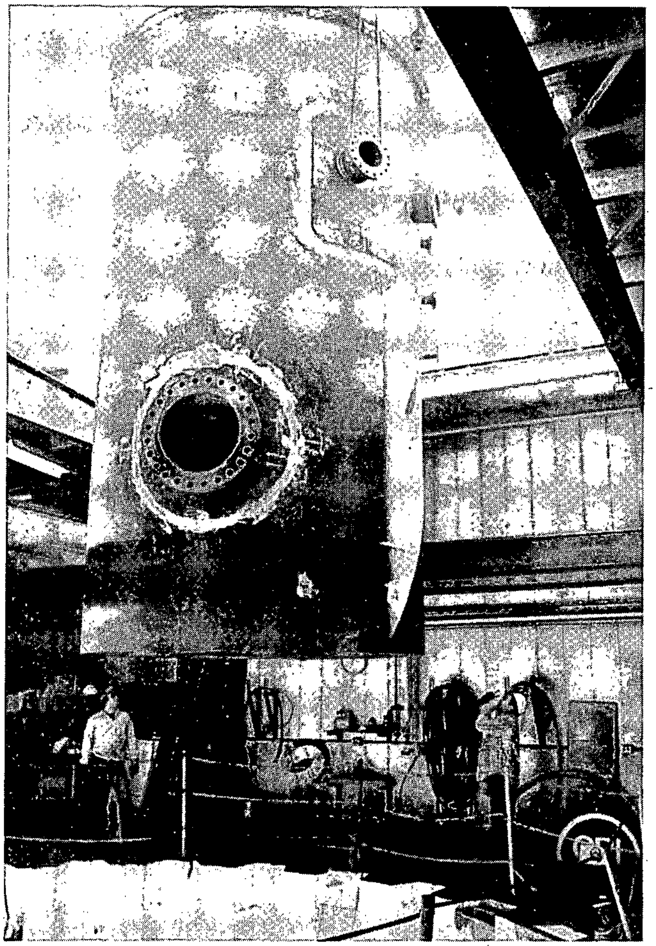
Personnel Airlock Test. A full-size personnel airlock, obtained from a cancelled nuclear power plant, was tested at Chicago Bridge & Iron Research and Development Center in Plainfield, Ill., under contract to Sandia National Laboratories. The work is part of the containment integrity research sponsored by the NRC. The objective of the tests was to obtain structural data on the behavior of an airlock, especially the sealing surfaces, under severe accident conditions. It was anticipated that leakage would not occur unless relative deformations between the sealing surfaces were developed and performance of the seal material was compromised. The sealing surfaces could separate because of a mismatch in the out-of-plane displacements of the door and bulkhead, which resist internal pressure through bending action. The performance of the seal material may be compromised in two ways: (1) a loss of resiliency associated with thermal or radiation aging, and (2) degradation associated with exposure to very high temperatures.

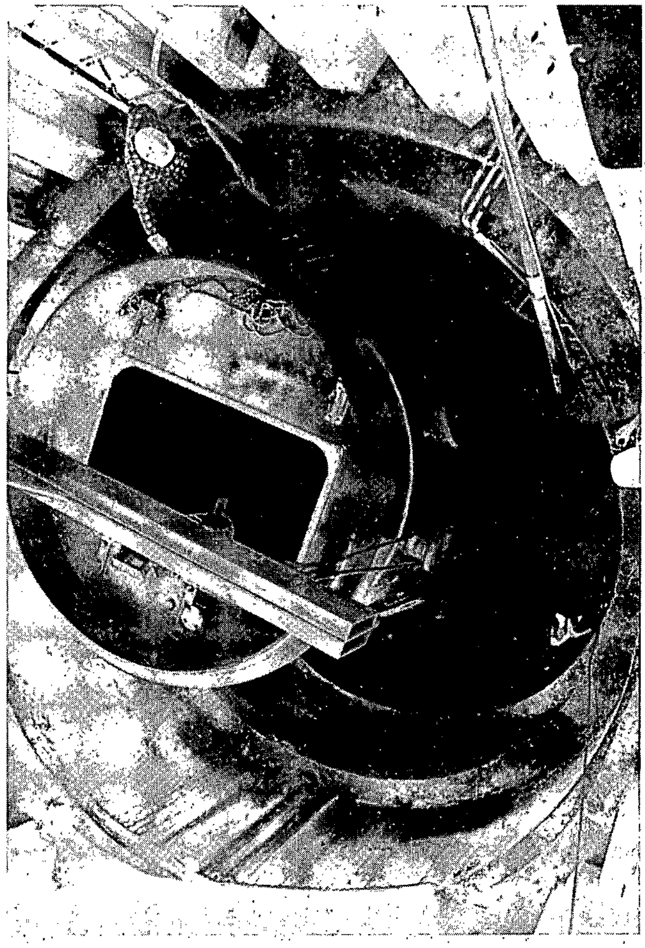
Two of the four tests to be performed were conducted on June 30 and July 2, 1987, with satisfactory results. In the first test, the inner and outer doors were without gaskets and pressurized to 69 psig (1.15 times the design) at room temperature. Leakage was measured at 45 scfm and 35 scfm on the inner and outer doors, respectively. In the second test, the inner and outer doors had aged gaskets installed and were pressurized to 69 psig (this pressure was held for one hour), and no leakage was measured. The next test scheduled for early fiscal year 1988 will involve higher pressure and temperature typical of severe accident conditions in BWR and PWR containments.

Core Melt Progression and Hydrogen Generation

In-vessel core melt progression research is concerned with the state of the reactor core from core uncovery up to reactor vessel melt-through in unrecovered accidents and up to the stabilization of core temperatures in accidents that are recovered by core reflooding, as at TMI-2. Sensitivity studies have shown that the uncertainties in the state of the core debris at vessel failure constitute the greatest uncertainties in the ex-vessel containment loads, including core-concrete interactions and direct containment heating. The state of the core in core melt progression is also the primary determinant of in-vessel hydrogen generation, fission product and aerosol generation and attenuation, explosive and nonexplosive rapid steam generation, and the potential for successful recovery actions in accident management.

The basic information source on in-vessel severe accident behavior has been the series of severe fuel damage tests performed in the Power Burst Facility (PBF) test reactor, which included extensive PIE (Post-Irradiation Examination). Tests in the National Research Universal (NRU) reactor in Canada provided full-length data on fuel damage during coolant boildown. FLHT-4 provided information on fission product release and deposition for PWR high-burnup fuel rods.





Part of NRC's containment integrity research program employs tests using a full-size personnel airlock, obtained from a cancelled nuclear power plant. The tests are conducted in Plainfield, Ill., by the Chicago Bridge and Iron Research and Development Center, under contract to Sandia National Laboratories. Shown at left, the airlock is ready to be lowered into the test cell, where, at right, it is positioned over the bottom test chamber.

FLHT-5 was also conducted with power compensation for the bundle heat losses at high temperature.

The BWR-DF4 test was performed in the Annular Core Research Reactor (ACRR) to investigate the effects of the BWR channel boxes and the BC control blades upon fuel damage, early core melt progression, hydrogen generation, and system chemistry.

In the MELPROG assessment and validation program, MELPROG calculations and comparisons with ACRR and PBF results are ongoing. Improvements and development to the MELPROG and SCDAP codes continue. BWR models of MELPROG and SCDAP have been developed. Analytical support to the KfK CORA experiments continued.

Core-Concrete Interactions

In those 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 may significantly impact containment loading, the modes of containment failure, and the radiological source terms. To characterize the threat to containment integrity and the nature of the ex-vessel releases, experiments are being performed, 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 heat and mass transfer, attack on structural concrete in the reactor cavity, and the influence of an overlying water layer. CORCON is incorporated into the NRC Source Term Code Package (available for licensing and regulatory applications) and has now been integrated into the CONTAIN and MELCOR codes. Improved models for the treatment of decay heat, time-dependent mass addition, and axial heat transfer to concrete have been developed. The code is being actively used in 17 research institutions throughout the world. Large-scale integral experiments with sustained induction heating were performed to study the effects of metallic zirconium present in molten stainless steel interacting with limestone and siliceous concrete. A summary review of available data on core debris-concrete interactions is being prepared in support of model validation.

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 were initiated, and data are being used to assess the VANESA code. VANESA has been linked to CORCON to form the CORCON/VANESA package.

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, core melt will slump and collect at the bottom of the reactor vessel. If molten core material attacks the bottom head of the reactor and 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 sprayed core debris can further oxidize in air or in steam to generate a large quantity of chemical energy and further pressurize the containment. Simple analyses indicate that even a large, dry PWR containment can be pressurized beyond its ultimate strength if a significant fraction of the core material participates in direct containment heating (DCH).

A program was developed at Sandia to investigate the debris dispersed 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. In fiscal year 1987, two experimental programs were in progress—the Surtsey direct containment heating test program at Sandia and the separate-effect test program at Brookhaven National Laboratory. Preliminary observations of the Surtsey facility test results confirm substantial pressurization and aerosol generation. Because of the complexity of the DCH problem coupled with the high cost of running large-scale tests in the Surtsey facility, separate-effect tests are being performed at Brookhaven to address core debris dispersal. Transparent plexiglass models, 1/42-scale, of Zion, Surry, and Watts Bar reactor cavities were constructed. Both water and Wood's metal were used to simulate core debris. Data are now being used to develop models for both lumped-parameter and finite-difference codes. An interim model has been used for DCH analysis in CONTAIN, and more detailed separate-effect calculations, using the KIVA code, are being explored at Sandia.

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 LWR plants. The HECTR lumped-parameter computer code was developed at Sandia National Laboratories and is used in the analysis of nuclear reactor accidents involving the transport and combustion of hydrogen. The assessment of HECTR is continuing, and it includes the use of the data from large-scale hydrogen transport experiments performed at the HDR facility in the Federal Republic of Germany.

A flame propagation model was incorporated into HECTR. The HMS-BURN code, a three-dimensional finite-element analysis tool (NUREG/CR-4020), developed at Los Alamos, is used to benchmark HECTR and to provide more detailed hydrogen transport and mixing calculations.

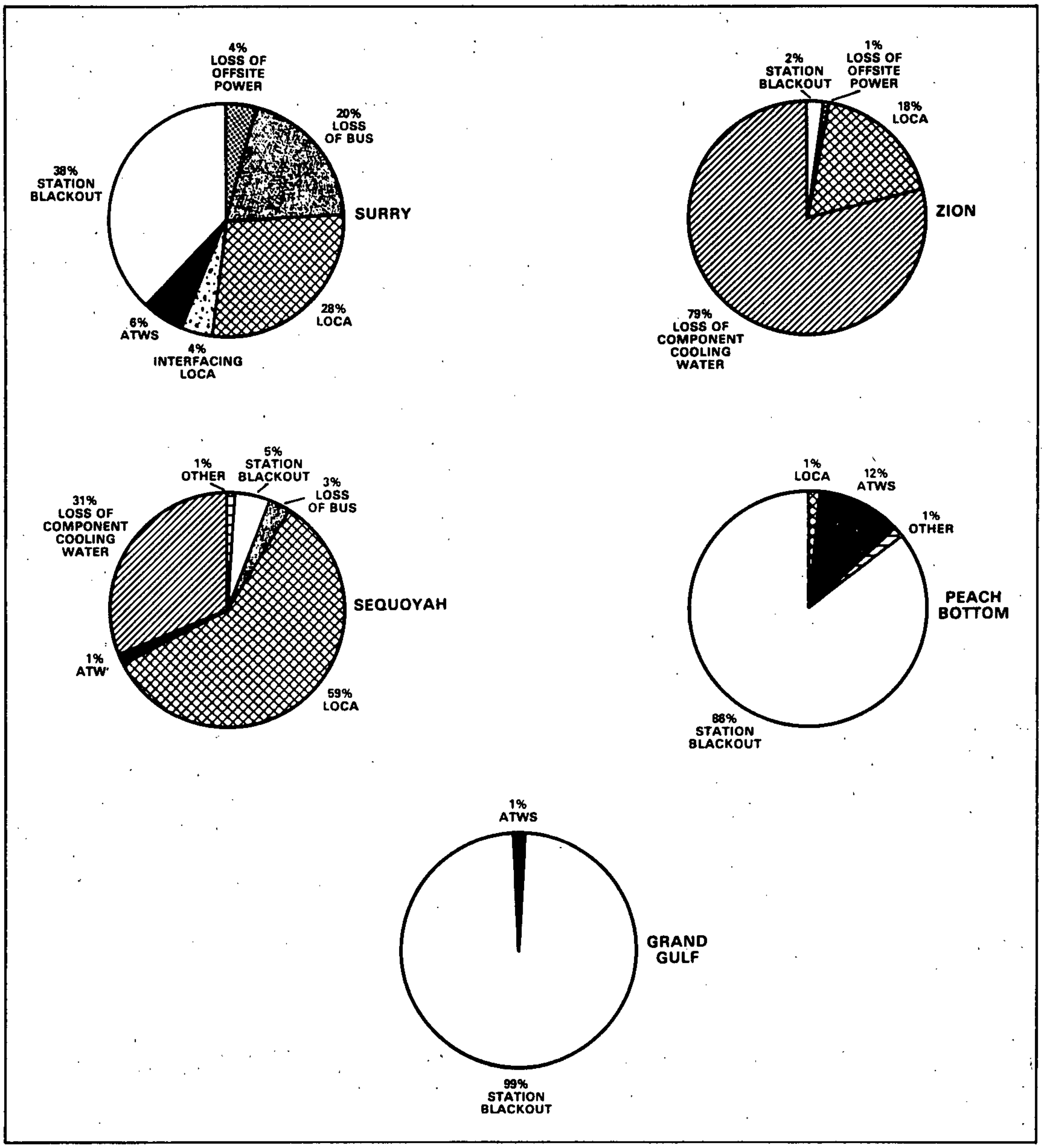
REACTOR ACCIDENT RISK ANALYSIS

Review of PRAs

Browns Ferry. A first draft of an industry-generated probabilistic risk assessment (PRA) of the Browns Ferry Unit 3 (Ala.) plant was submitted to the staff for review in 1987. The staff has conducted a limited review of the core damage accident analysis portion of this PRA and has developed a betteroverview of the plant's safety.

The review did not uncover any reason for any immediate regulatory action on this plant, nor did it identify any new significant generic safety issues. The draft PRA indicates that, for this plant, the potential accident sequences of most significance involve scram failure, rather than such events as station blackout or loss of main feedwater. In addition, it was found that the plant compressed air system (a support system) plays a major role in plant operation and provides a wide range of inter-system dependencies.

Diablo Canyon. In order to comply with a license condition, the licensee for Diablo Canyon (Cal.) has developed a Long-Term Seismic Program (LTSP) consisting of three phases. Phase I was the development of a detailed program plan to address the license condition. The plan was submitted for staff review on January 30, 1985. Phase II consisted of a study to refine the scope of work for Phase III and associated schedules. A report describing Phase II ac-



Risks from possible severe core damage accidents in five nuclear power plants have been studied by NRC researchers an, in February 1987, a draft report, "Reactor Risk Reference Document," was issued for public comment. The charts above reflect the relative contributions of various accident sequences to estimated total core damage frequency, for each of the five plants.

tivities and conclusions was submitted to the staff on January 30, 1986. Phase III work is currently continuing, and final reports are expected to be submitted in 1988. As a part of this LTSP, the licensee is performing a Level 1 PRA, including both internal and external events.

Risk-Based Accident Methodology

System Analysis and Risk Assessment System (SARA). The Committee to Review Generic Requirements has stated that there is a need for improved integration and accounting in generic issues evaluation; the purpose is to re-establish the plant risk base in light of previous regulatory activities, so that value-impact analyses will provide an accurate basis for proper resolution of generic issues. The System Analysis and Risk Assessment (SARA) program was initiated in 1985 to take account of the fact that the risk base has changed over time with the imposition of many generic safety issues, and that the cumulative backfitting burden on licensees is large. Since the requirement covers a broad spectrum of PRA activities, SARA is also intended to be a flexible tool to support the various levels of users who require risk and reliability information for decision-making and regulatory analyses.

Guidance for Inspections. The Plant Risk Status Information Management (PRISIM) system was initiated in 1983 to develop a method for presenting PRA information in a form that could be useful to NRC staff in setting priorities and planning activities. After considerable study of NRC's inspection program, a method was developed to the point where field tests were warranted. PRISIM systems, operable on an IBM personal computer, have been installed at three plant field offices and are in trial use.

In 1987, development was completed on a licensing version of PRISIM for NRR project managers. This version of PRISIM provides project managers with immediate details of system layouts, the risk importance of each system and its individual components, and the importance of technical specifications for any set of plant conditions. A demonstration system is now in use in the NRC Office of Nuclear Reactor Regulatoin, and further work will depend on the experience with this demonstration system.

Risk Methods Integration and Evaluation Program. The Risk Methods Integration and Evaluation Program (RMIEP) was started in 1984 to develop improved assessment methods supporting probabilistic risk assessments of nuclear power plants. Initial integrated logic models of LaSalle Unit 2 (Ill.) and the internal events screening analysis were completed in 1986. In 1987, the internal events analysis and the locations analysis for fire and internal floods were completed, completing the technical work for the project. It remains only to document the results in suitable form and evaluate the results for any insights into plant safety or PRA methodology. This effort will be completed early in 1988.

Completion and Review of Reactor Risk Reference Document

In February 1987, the NRC issued the draft version of NUREG-1150, "Reactor Risk Reference Document," as well as a series of supporting contractor reports, for public comment. The report assesses the risks from possible severe core damage accidents in five U.S. nuclear power plants. The five plants studied are Surry (Va.), Zion (Ill.), Sequoyah (Tenn.), Peach Bottom (Pa.), and Grand Gulf Miss.). The report discusses the implications of the five risk assessments on regulatory issues such as the technical basis for present emergency planning regulations and implementation of the Commission's Safety Goal and Severe Accident Policy Statements.

While the review process was under way, the NRC staff and supporting contractors have been updating the five risk analyses. These updates are intended to reflect the present plant design and operating characteristics, improve the methods used, and incorporate new experimental and calculational data on severe accidents resulting from the research programs of NRC and others. At the present time, the completion of this work, including consideration of public comments, is scheduled such that the final version of NUREG-1150 will be completed by July 31, 1988.

New Release Consequence Model

In coordination with the NRC staff work on draft NUREG-1150 discussed above, a new model for assessing the consequences of radioactive releases has been developed.

The model—MACCS—has the capability to treat radionuclide releases lasting for a short time or a prolonged period, including the effect of change in the wind direction at the reactor during the release, and to sample the variability of precipitation intensity from the reactor site's meteorological data.

MACCS incorporates newer or more realistic models for:

- Health effects projections developed for NRC after publication of WASH-1400 (1975) and BEIR-III (1980).
- Long-term (chronic) radiation exposure from continued use of contaminated environment.
- Emergency response and radiation protection measures.
- Economic impact estimates.

In October 1987, modifications to the MACCS model were suspended to permit use of this model in the final version of NUREG-1150. Public release and publication of documentation associated with the model is planned for January 1988. Publication of documentation of a more complete version of MACCS that would include the still ongoing revisions of health effect models is planned for September 1988.

SEVERE ACCIDENT POLICY IMPLEMENTATION

This program provides for the application of research results on severe reactor accident issues directly to the regulatory process. Modification of rules regarding siting, emergency planning, and containment design are examples of areas where the results of severe accident research may lead to changes in NRC regulations.

Emergency Preparedness

On April 20, 1987, the NRC published in the Federal Register (52 FR 12921) a proposed rule on emergency preparedness for fuel cycle and other radioactive material licensees. The rule would apply to about 30 large facilities.

The facilities that will be required to comply with this regulation are those for which a release large enough to require the support of off-site response organizations to protect the public was considered credible. The rule would require, among other things, prompt notification of off-site response organizations in case of a serious accident, procedures and equipment for coping with the emergency, and training and exercises for response personnel.

Seventeen public comments were received. A final rule is scheduled for publication in March 1988.

On March 6, 1987, a proposed rule dealing with nonparticipation of State or local governments in emergency planning was published in the *Federal Register* (52 FR 6981). An unprecedented number of public comment letters were received (approximately 38,000). All comment letters were evaluated by the staff and used in the development of a staff-proposed final rulemaking that was presented to the Commission on October 22, 1987, after the close of the report period.

Mark I Containment Improvement Program

Acting on insights gained from NUREG-1150, "Reactor Risk Reference Document," the staff has begun a program to assess the severe accident mitigation capability of the Mark I containment. The goal of the program is to assess the strengths and weaknesses of the Mark I in light of present-day knowledge concerning the likely dominant accident sequences and associated containment response characteristics. The staff will focus on two broad areas of concern.

First, phenomenological issues of reactor/containment behavior, such as core melt phenomena and liner melt-through, will be examined. After that, potential plant improvement issues—such as containment sprays, hydrogen control, core debris control, venting, reactor building fission product attenuation, and accident management—will be assessed.

In addressing these complex issues, the staff will conduct a dialogue with the research community, the nuclear industry, and interested members of the public, in order to gain a broad perspective of the phenomena and associated uncertainties, and will conduct a comprehensive view of potential means to improve containment response.

By late summer of 1988, the staff expects to have assessed and evaluated these areas sufficiently to be able to characterize them in terms of whether the issues involved are either resolved or unimportant to safety, are issues for which additional research will be necessary to draw definitive conclusions on their importance, or are topics for which an adequate basis exists to initiate a formal regulatory initiative.

RADIATION PROTECTION AND HEALTH EFFECTS

The NRC maintains a program of research and standards development in radiation protection intended to ensure continued protection of workers and the public from radiation and radioactive materials in connection with licensed activities. The program is currently focused on improvements in health physics measurement and the review 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 occupational dose and 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 the staff examines 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 in this critical area. Improved risk estimates are needed both for assessing severe accident consequences and for implementing agency safety goals.

Brookhaven ALARA Center

The Brookhaven National Laboratory (BNL) ALARA Center, funded by NRC, continued monitoring DOE and industry radiation dose reduction efforts and ALARA research in reducing radiation exposures in nuclear facilities to a level "as low as reasonably achievable" (ALARA). In 1987, BNL published Volume 3 of NUREG/CR-3469, which 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.

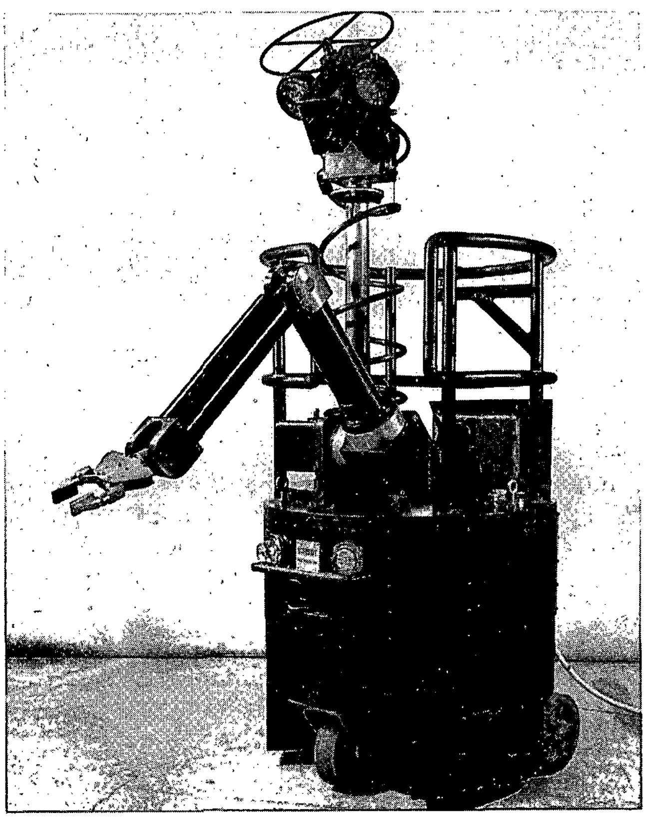
The center is recognized by the nuclear industry and others as a major source of information on new and effective dose reduction techniques and its publications are standard references for ALARA planning. The BNL staff is available through the center to the entire NRC organization and its licensees, providing information and advice on all aspects of radiation protection and dose reduction.

BNL reported in 1987 that dose reduction research has led to a clear reduction in occupational radiation exposure, observable in countries with dose reduction research programs, such as Japan, the Federal Republic of Germany, Canada, Sweden, France, and the United States.

Robotics

The Small Business Innovative Research contract for development of a surveillance robot was completed and a report published (NUREG/CR-4815) in March 1987. The report analyzes the costs and benefits of the demonstration testing of a surveillance robot at the Browns Ferry (Ala.) nuclear plant. It is the final report on the contract with Remote Technology Corporation (REMOTEC) in Oak Ridge to design, construct, and demonstrate a remote-controlled vehicle for inspection and surveillance in radiologically controlled areas of nuclear power plants.

The robotic system was operated for five months at the Browns Ferry (Ala.) plant by trained TVA personnel in real job situations. Data collected by the TVA evaluators indicate that the device can prevent over 100 person-rems of exposure per year at a nuclear power plant and permit more frequent and better inspections of safety-related components. Other benefits include cost savings in labor and protective clothing, more complete surveillance data on components during power operation, improved worker safety, and decreased liability of plant operators for worker injury claims. The cost-benefit analysis concluded that the initial cost of approximately $160,000 would be recovered in two years of operation; the system has an estimated useful life of 10 years.



This surveillance robot was built by Remote Technology Corporation of Oak Ridge, Tenn., under an NRC contract and was tested for five months at the Browns Ferry (Ala.) nuclear power plant. Results indicate that the robotic system affords more frequent and effective inspections of plant components, cost savings, and improved worker safety.

Investigation of Worker Registry

In response to a request by the National Cancer Institute, the NRC staff conducted an investigation of the alternatives available and the resources required to ensure the availability of appropriate occupational dose data for studies of possible health effects. The investigation included a meeting with 25 representatives of various government agencies and professional, labor, research, and standard-setting organizations, as well as the nuclear industry, in which the representatives discussed the creation of a registry of individual occupational exposure data. A paper is in preparation recommending that the NRC amend its regulations to require the reporting of occupational dose data that would be useful for various purposes, including studies of possible health effects. The ongoing major revision of 10 CFR Part 20 includes the needed changes in recording and reporting requirements.

Interpretation of Bioassay Measurements

A report (NUREG/CR-4884) that provides a practical and consistent method for estimating intakes from both in vivo and in vitro bioassay measurements was published during fiscal year 1987. Interpretations of bioassay measurements tend to show inconsistencies, particularly in the early phases after an accidental intake of radioactive material. The report proposes a consistent approach and gives instructions for the computation of intakes and committed organ dose equivalents. Tables for the interpretation of bioassay results are compiled for several hundred radionuclides. The values in the tables were determined by using retention models published by the International Commission on Radiological Protection (ICRP-79). A regulatory guide endorsing the methodology presented in the report will be issued in 1988.

Metabolism and Internal Dosimetry

A report (NUREG/CR-4915) of a one-year study of the severity and duration of renal injury produced in rats from exposure to low levels of uranyl fluoride was published in September 1987. Injury was apparent early in the dosing phase of the study, at a time when renal uranium levels were between 0.7-1.4 microgram-uranium-per-gram-kidney, and was most severe when the renal uranium burden was between 3.4-5.6 micrograms-uranium-per-gram. These levels are much lower than the nephrotoxic limit of three micrograms uranium per-gram-kidney used by the NRC in setting standards for exposure to soluble uranium compounds. Repair of the injury was rapid, with complete restoration within 35 days after the exposure.

The final report (NUREG/CR-4986) for a multi-year study of the metabolism of inhaled mixed (U, Pu) oxides was published in September 1987. Industrially collected aerosol materials were re-aerosolized in the laboratory to determine patterns of deposition, retention, and translocation in laboratory animals. Multiple species were used for inhalation exposures. A biokinetic model that used the measured physical/chemical characteristics of the particulates to describe the rate of dissolution of material deposited in the lung was developed.

Results of the studies showed that, for a given elemental (uranium or plutonium) component of the particulates, slight differences in the retention, distribution, and excretion of the element were accounted for by slight differences in the physical/chemical characteristics of the aerosol. A dose/response study in rats exposed to these elements showed development of pulmonary cancers, with no discernable difference ascribable to the composition of the aerosol. The incidence of pulmonary cancers in rats exposed to industrial materials containing plutonium were not different from rats exposed to laboratory-produced aerosols of plutonium oxide.

External Dosimetry

The NRC has published a report (NUREG/CR-4418) which describes the dose calculation for contamination of the skin; the calculation employs the computer code VARSKIN. The calculation method allows computation of the radiation dose rate at any desired depth beneath the skin from surface contamination, and it can be performed on an IBM PC or a compatible machine. The methods described in this report are considered acceptable for calculating skin dose from small, individual particles, as well as from distributed sources. This work contributes significantly to resolving the question of how best to deal with hot particle skin contamination events at nuclear power plants. Work is continuing to prepare an addendum to VARSKIN that will address problems involving other radionuclides.

Occupational Exposure Data System

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 REIRS (Radiation Exposure Information Reporting System). The system provides a permanent record of the data and permits expeditious analyses of the two kinds of reports required (annual statistical summaries and individual termination reports). Exposures received as a result of medical procedures are not required to be reported.

Summaries of the annual statistical reports for 1985 (compiled in 1987) disclosed that the seven categories of licensees required to report monitored about 213,000 persons, of whom about 52 percent received measurable doses. The workers received a collective dose of 47,000 person-rems or an average annual dose of 0.4 rem per worker among those receiving a measurable dose (0.2 rem per monitored person when the entire monitored population is considered). Of the persons monitored, 90 percent worked in nuclear power plants, and they incurred about 92 percent of the total annual collective dose. The average annual measurable dose received by individual nuclear power plant workers decreased to about 0.5 rem because the annual collective dose incurred by these workers decreased by 20 percent, falling to its lowest value in five years. Preliminary compilations of the exposure data reported by nuclear power plants for calendar year 1986 indicate that the collective dose remained at about 43,000 person-rems, even though seven new plants reported. The average measurable dose, however, decreased to 0.4 rem, which is less than 10 percent of the applicable dose limit.

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 360,000 persons, most of whom worked at nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories and to analyze the data for trends. The data also help ensure that transient workers moving from plant to plant do not receive doses in excess of regulatory limits. For example, analysis of the data reported for 67,400 persons terminating employment during 1984 revealed that 6,000 of them worked at two or more nuclear power facilities and that none of them received doses in excess of the regulatory limits as a result of their multiple employment.

Improvement of Health Effect Models

Considerable progress has been made toward the development of models for early health effects resulting from combined internal and external radiation in case of severe accidents.

Work on revising and updating NUREG/CR-4219, "Health Effects Model for Nuclear Power Plant Accident Consequence Analysis," continued in fiscal year 1987.

Changes to Radiation Protection Guidelines

Proposed Revision to 10 CFR Part 20. Staff work on a complete revision of the Commission's radiation protection regulation, 10 CFR Part 20, continued actively in 1987, following a review by the EDO of the alternative strategies for the revision. The revision is being carried out as a high-priority task by an inter-office working group under the direction of a special steering committee composed of division directors from RES, NMSS, NRR, and GPA, and a legal advisor from OGC. A final rule is expected to be transmitted to the Commission by mid-1988.

Personnel Dosimetry Processing Guidelines. In 1987, a final rule to improve personnel dosimetry processing has been completed. The rule requires licensees to use personnel dosimetry processors who have been accredited under the National Voluntary Laboratory Accreditation Program (NVLAP), which is operated by the National Bureau of Standards. The rule becomes effective on February 12, 1988, and is expected to improve the quality of dosimetry processing by requiring all processors to meet the guidelines of a national standard.

Testing of Extremity Dosimetry Standard. Ongoing testing of extremity dosimeter processors using current draft standards has begun. The results of the first test indicate that a significant number of processors did not meet the criteria of the draft standard. Ongoing work will include more testing and visits to processors to observe their procedures and quality control. Revised guidelines for accreditation of extremity dosimetry processors under the NVLAP program at NBS will be forthcoming.

Proposed Revision to Regulations Governing Radiographic Operations. A proposed rule that will incorporate the guidelines of a national standard on the design and construction of radiographic devices will be published for comment in the Federal Register by mid-1988. The purpose of the proposed rule is to reduce the number of radiographer overexposures and reduce the risk to the public from such devices. The proposed rule will also require the use of dosimeters with built-in alarm features by radiographers in the field.

Revision to Regulatory Guide 8.13. NRC's regulations at the present time do not specify a dose limit for the embryo/fetus. A proposed revision of 10 CFR Part 20 would limit the dose for the entire gestation period to 500 millirems.

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.

High-Level Waste

The NRC maintains active research programs in hydrology, geology, materials science, geochemistry, and several other disciplines related to the management of high-level waste (HLW). The research combines theoretical study with laboratory and field experiments to identify the physical processes that control and determine repository performance in the types of geologic media found at sites currently under consideration by DOE. 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 statutory mandates (10 CFR Part 60) and with the Environmental Protection Agency's HLW standard, as DOE goes about the task of providing a permanent high-level waste repository.

Geohydrology. Since transport by ground water is the most likely path by which radioactive nuclides from disposed waste can reach the environment, the NRC is actively studying the movement of ground water in the types of media being considered by DOE. Experimental sites have been located in fractured rock, both above and below the water table, and field testing is being conducted by In Situ, Inc., 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. The field study in saturated fractured rock was initiated in September 1985 to test the relationships between field measurements of parameters and model scales derived from earlier work. The importance of large natural anomalous hydrologic features, appropriateness of continuum-versus-discrete fracture models, measurement of effective porosity, theories of spatially projecting dispersivity measurements, and distinctions between matrix diffusion, dispersion, and sorption are among the subjects addressed in this study.

A cooperative agreement between NAGRA (Switzerland) and the NRC was negotiated during fiscal year 1987. The cooperative experiments and data analyses carried out under this agreement will augment the field testing program cited above. A similar study examining ground-water flow in unsaturated rock was begun at an unsaturated tuff site in Arizona in the spring of 1987. This work, being carried out by the University of Arizona, is assessing techniques and methodologies for fracture characterization, infiltration and percolation studies, rock and matrix permeability testing, vapor-phase flow and transport assessment, and numerical simulations of flow and transport in partially saturated media.

Waste Package Performance. Investigating the performance that can be expected from the waste form and waste package is essential to the NRC's ability to independently

As part of its waste management research activity, the NRC conducts theoretical studies as well as laboratory and field experiments, in a wide variety of disciplines associated with the disposal of high-level radioactive wastes. The goal is to provide technical bases for assessing the adequacy of DOE-proposed high-level waste disposal sites. Under one NRC-sponsored project, initiated in the spring of 1987, the University of Arizona is studying the unsaturated Apache Leap Tuff Site in that State. Shown are university researchers conducting airflow tests as part of a vapor-phase flow and transport assessment.

evaluate DOE's demonstration that the waste form and waste package comply with the containment and controlled release requirements of 10 CFR Part 60. During 1987, NRC-sponsored research into the mechanisms of local corrosion of carbon steel was completed. It resulted in a significant new understanding of localized corrosion in carbon steel, a matter of particular importance to geologic disposal of HLW. The next step is the integrated testing of HLW overpack materials in simulated repository environments, to begin in late 1987.

Under a cooperative research agreement with the NRC, the Japan Atomic Energy Research Institute (JAERI) is conducting a series of experiments of HLW waste package and HLW glass waste form performance in high radiation environments. This work complements the laboratory research studies on waste packages and waste forms being supported by the NRC.

Geochemistry. The NRC has an active research program in the vital field of geochemistry related to the management of HLW. Work at the University of California at Berkeley on the geochemistry of radioactive wastes in repository environments is continuing. In 1987, errors in the thermodynamics of alumina silicates (common to all clays) were investigated in the laboratory. In addition, the NRC has joined an international field study to examine actual movement of radionuclides in the environment. This is a continuation of the ore body work performed in Australia for the NRC and now under the sponsorship of the Nuclear Energy Agency.

Work on the use of ground-water dating techniques to help understand and model geohydraulic systems was completed during the report period. Results indicate that a combination of isotopic and geochemical techniques have the potential to provide an independent data base for ground-water flow model validation.

Rulemaking. In February 1987, the NRC published an advanced notice of proposed rulemaking on the definition of high-level radioactive wastes. The purpose of the notice was to solicit comments on the classification and management of waste above class "C" which are not now considered HLW. Comments were being analyzed at the close of the report period and a proposed rule was expected early in 1988.

Low-Level Waste

NRC research in support of licensing activities for low-level radioactive waste (LLW) disposal facilities is focused on (1) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial for LLW disposal, (2) evaluation of the overall performance of disposal systems, (3) water entry into disposal units, (4) performance of waste packages, (5) characterization of the LLW source term, and (6) mechanisms for transport of radionuclides from the disposal units. This research will be useful not only to the NRC licensing staff but also to States regulating LLW disposal. In order to make research results available to the States, research contractors have made presentations before State organizations and NRC-sponsored meetings with States. The DOE also sponsors an annual meeting at which DOE and NRC research results are presented; these meetings are well attended by State representatives.

Engineered Enhancements and Alternatives to Shallow Burial. There is great interest on the part of States and State compacts in alternatives to shallow land burial, as currently practiced. In 1987, work at the Idaho National Engineering Laboratory was concentrated on the reliability of the engineered components in the so-called alternatives to shallow land burial of LLW.

Contaminant Transport Modeling. An NRC-sponsored cooperative project between Atomic Energy of Canada Ltd. (AECL) and the Battelle Pacific Northwest Laboratories (PNL) used data collected from 40 years of LLW waste disposal at AECL's Chalk River facility to assess techniques for modeling LLW site performance. PNL approached the problem as though dealing with a pristine site, prior to waste disposal. This exercise confirmed the practicality and utility of modeling a site using a well-chosen data set collected during site characterization. This project is providing important insights into the design of data evaluation programs for future LLW disposal and the reliability of predictions based on site-characterization data.

LLW Waste Forms. In May 1983, the NRC issued a technical position paper that specified minimum performance standards for LLW waste forms. Current waste forms in commercial use are being tested at the Idaho National Engineering Laboratory and at the Brookhaven National Laboratory to ensure that leaching characteristics and compressive strength of the waste forms are consistent with the standards specified in the technical position. Various decontamination waste from actual power plants using commercial solidification processes such as Lomi, Candecon, NS-1, and Citrox are being investigated. In 1986, the Brookhaven National Laboratory began an NRC research project to study the use of concrete and high-density polyethylene for LLW containers and engineered barrier materials. Representative samples of each material are being subjected to the various environments expected in the waste forms and the surroundings, e.g., sulfates, acids, gamma fields, in order to study the failure and degradation mechanisms and, if possible, develop methods for predicting the performance of the materials over a period of 300-to-500 years.

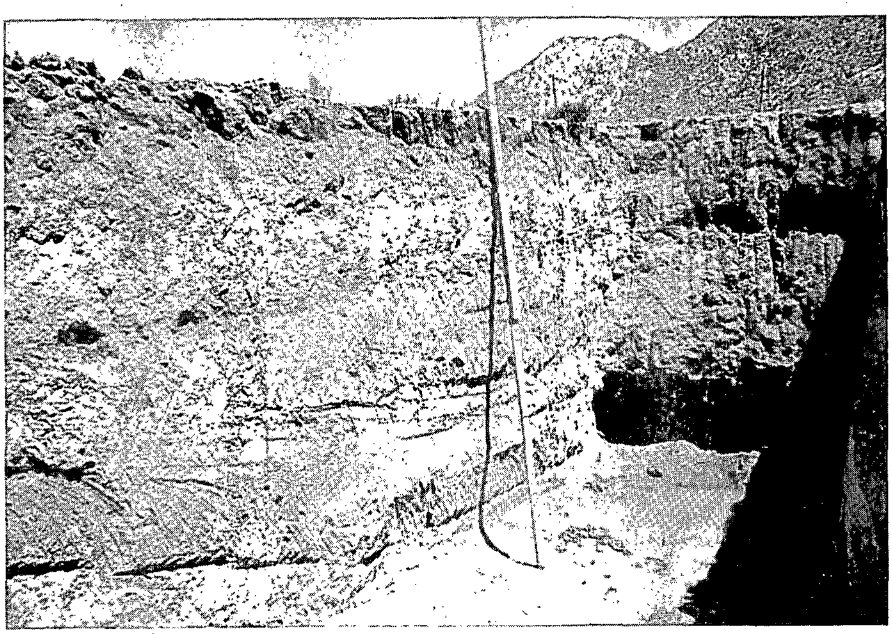
Infiltration of Water. The University of California at Los Angeles in cooperation with the University of Maryland is field testing, at Beltsville, Md., a system of enhanced runoff and bioengineering to control water-entry through trench covers. By artificially enhancing runoff and using vegetation to remove water through "evapotranspiration" (plant transpiration plus evaporation), water-entry through disposal unit covers can be reduced to a negligible level. Waste package degradation can thus be reduced and the performance of the waste disposal system improved. The results of this work will be applicable to any disposal scheme employing earthen covers.

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. This program, which includes NRC-sponsored research by PNL and the Massachusetts Institute of Technology, is intended to provide States and licensees with the ability to model realistically the expected performance of LLW disposal facilities.

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). Table 1 is a listing of former USIs for which a technical resolution has been achieved; and Table 2 sets forth the schedule for the resolution of USIs currently under review. These current issues are discussed in the summary that follows, with the exception of "PWR Steam Generator Tube Integrity" (Nos. A-3, A-4, A-5), whose resolution is virtually complete and which has been treated at length in previous NRC annual reports.



To better equip States and licensees for assessing potential low-level nuclear waste disposal sites, the NRC continued in 1987 to sponsor field tests of flow and transport in unsaturated soils. This trench, near Las Cruces, N.M., was heavily instrumented and, as the Jornado Test Facility, serves as the vehicle for many ongoing experiments dealing with unsaturated flow and transport through heterogeneous media.

SUMMARY OF STATUS

Systems Interactions

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 safety-significant, yet workable, scope for this issue.

The staff's proposed resolution requirements and supporting technical information were undergoing inter-office review at the close of the report period. The staff expects to issue the proposed resolution for public comment during the middle of fiscal year 1988, with final resolution near the middle of fiscal year 1989.

Seismic Design Criteria

Rapid advancements in state-of-the-art technology in seismic design over the past decade have made it possible and necessary to update NRC's acceptance criteria for seismic design of structures, systems, and components of nuclear plants. The Lawrence Livermore National Laboratory compared NRC Seismic Design Criteria with state-of-the-art knowledge and published the results in its "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria" (NUREG/CR-1161, dated May 1980). Based on these recommendations and results of a staff-sponsored workshop for soil-structure interaction, held in June 1986, the staff will propose modifications to related review criteria. Incorporation of the proposed changes is expected to eliminate some potential sources of unwarranted conservatism and result in seismic design criteria that reflect an up-to-date understanding of this technology.

The staff has prepared a proposed resolution for this issue for inter-office peer review. Review by the Committee to Review for Generic Requirements is scheduled for January 1988 and publication to obtain public comments scheduled for April 1988. Issuance of the final requirements, including resolution of public comments, is scheduled for fiscal year 1989.

Station Blackout

The loss of all alternating current (a.c.) electric power (from both off-site and on-site sources) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems that do not require a.c. power supplies and on the ability to restore a.c. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event that could result in unacceptable consequences, such as severe core damage.

The staff's proposed resolution of this issue, USI A-44, which includes a proposed rulemaking and a new regulatory guide, was issued for public comment on March 17, 1986. The public comment period ended on June 16, 1986. The final rule was reviewed by the Committee to Review Generic Requirements in May 1987 and was scheduled to be reviewed by the Commission in December 1987. The staff is continuing to work with the Nuclear Utility Group on Station Blackout, which is developing detailed guidance for the use of utilities in assessing their plants' capabilities.

Shutdown Decay Heat Removal Requirements

The staff continues its study of the adequacy of systems for safely removing decay heat from a reactor core during shutdown and of the value and impact of alternative measures for improving the reliability of the decay heat removal function. Under study are such matters as system reliability, system engineering feasibility, thermal-hydraulic analyses, power plant characterizations, emergency operating procedures, and the vulnerability of the systems to special emergencies, such as fire, flood, earthquake, or sabotage.

A contractor to the NRC has completed six plant studies which will form the basis for the staff's assessment of current decay heat removal capability and of potential risk reduction, together with an estimate of the cost of possible changes to plant design or operation. A technical summary of these studies and a value-impact analyses of alternatives are scheduled to be submitted to the Committee to Review Generic Requirements in February 1988 and issued for public comment in May 1988.

Seismic Qualification of Equipment In Operating Plants

The design criteria and methods employed for the seismic qualification of mechanical and electrical equipment in nuclear power plants have changed significantly during the history of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and to perform intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure that a plant can be brought to a safe shutdown condition following a seismic event. This issue, USI A-46, entails investigation of alternative procedures for ensuring seismic adequacy of equipment in lieu of requiring qualification to current licensing requirements.

The staff evaluated the various methods available for verifying seismic adequacy of equipment in operating nuclear power plants and decided that the use of seismic experience data and of test experience data would prove the most viable and cost-effective way of doing so. The staff concluded from its investigation of the issue that there are three principal areas of concern: (1) the adequacy of equipment anchorages

Table 1. Formerly 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, Rev. 1	March 1984
A-2	Asymmetric Blowdown Loads	NUREG-0609	November 1980
A-6	Mark I Short-Term Program	NUREG-0408	December 1977
A-7	Mark I Long-Term Program	NUREG-0661 NUREG-0661 Suppl.	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	NUREG-0619	November 1980
A-11	Reactor Vessel Material	NUREG-0744, Rev. 1	October 1982
A-12	Steam Generator and Reactor	NUREG-0577, Rev. 1	September 1982
A-24	Qualification of Class 1E 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	SRP 5.4.7	1978
A-36	Control of Heavy Loads Near Spent Fuel	NUREG-0612	July 1980
A-39	SRV Dynamic Loads	NUREG-0802	September 1982
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-46	Seismic Qualification of Equipment in Operating Plants	NUREG-1030 NUREG-1211	February 1987
A-49	Pressurized Thermal Shock	Regulatory Guide 1.154	February 1987

and supports, (2) the functional capability of electrical relays, and (3) equipment unique to nuclear power plants and outside the limits of the experience data base. The NRC staff issued the final technical resolution of USI A-46 on February 19, 1987, as Generic Letter 87-02. Included as attachments to the generic letter were NUREG-1211, "Regulatory Analysis for Resolution of USI A-46," and NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants (USI A-46)." Sixty-nine operating plants that have not been reviewed by current licensing criteria for equipment seismic qualification are required to perform equipment seismic adequacy reviews. Implementation procedures are under development by the Seismic Qualification Utility Group (SQUG) and the Electric Power Research Institute (EPRI). Plant-specific implementation is scheduled to start in fiscal year 1988.

Safety Implications of Control Systems

The staff has completed systemic evaluations of the control systems typically used during normal startup, shutdown, and on-line power operations of nuclear power plants for each of the four nuclear steam supply vendors—Babcock and Wilcox, Westinghouse Corp., Combustion Engineering, and General Electric Co. The purpose of the studies was

Table 2. Schedule for Resolution of Current Unresolved Safety Issues

Number	Title	Schedule for Issuing Staff Report "For Comment" (as of Sept. 30, 1987)	Schedule for Issuing Final Staff Report (as of Sept. 30, 1987)
A-3, 4, 5	PWR Steam Generator Tube Integrity	Completed April 1985	December 1987
A-17	Systems Interactions	March 1988	April 1989
A-40	Seismic Design Criteria	April 1988	May 1989
A-44	Station Blackout	Completed March 1986	March 1988
A-45	Shutdown Decay Heat Removal Requirements	February 1988	December 1989
A-47	Safety Implications of Control Systems	March 1988	April 1989
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	—	March 1988

to identify control systems whose failure could cause either transients or accidents to become more severe than those assumed possible at the time a plant's license specifications are drawn up, or could adversely affect any assumed or anticipated operator action during the course of an event, cause technical specification limits to be exceeded, or cause transients or accidents to occur at a frequency in excess of those established for abnormal operational transients and design-basis accidents. Final reports detailing the staff's review of each of the designs were issued in July 1986.

These studies have identified several control system failures that could cause transients leading to steam generator or reactor vessel overfill, overcooling, overpressure, or overheating events. The final reports evaluating the potential risk of these failures have been issued. In addition, various alternatives for reducing the initiating failure frequency or eliminating the failure mechanism of control systems found to be major contributors to events of concern have been analyzed.

A proposed staff resolution, which includes recommendations for operating plants and for future plants, was under staff review at the close of the report period. The staff plans to publish for public comment a draft of the technical findings report and the proposed resolution of the issue, USI A-47, in 1988.

Hydrogen Control Measures and Effects of Hydrogen Burns

Unresolved Safety Issue (USI) A-48 arose as a result of the 1979 accidentat Three Mile Island (TMI) Unit 2 in Pennsylvania. Approximately 1,000 pounds of hydrogen deflagrated 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. BWR Mark I and Mark II containments were inerted. The owners of BWR Mark III and PWR ice-condenser type plants elected to use igniters as a hydrogen control system described by the rule. The large, dry containment types of reactors—because of their increased hydrogen dilution volume—were not included in the rulemaking, pending completion of the research programs.

Table 3. Issues Prioritized in FY 1987

Number	*Title*	*Priority*
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	HIGH
125.I.1	Availability of the STA	DROP
125.I.4	Plant-Specific Simulator	DROP
125.I.7.b	Realistic Hands-On Training	DROP
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	DROP
125.II.1.a	Two-Train AFW Reliability	DROP
125.II.1.b	Review Existing AFW Systems for Single Failure	HIGH
125.II.1.c	NUREG-0737 Reliability Improvements	DROP
125.II.1.d	AFW Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	DROP
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	DROP
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	DROP
125.II.6	Re-examine PRA Estimates of Core Damage Risk from Loss of All Feedwater	DROP
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	DROP
125.II.10	Hierarchy of Impromptu Operator Actions	DROP
125.II.12	Adequacy of Training Regarding PORV Operation	DROP
127	Testing and Maintenance of Manual Valves in Safety-Related Systems	LOW
128	Electrical Power Reliability	HIGH
130	Essential Service Water Pump Failures at Multi-plant Sites	HIGH
135	Steam Generator and Steam Line Overfill	MEDIUM

USI A-48 was originally developed to assess all reactor types. With rulemaking, the issue has been focused on BWRs with Mark III containments and PWRs with ice-condenser containments.

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 Light-Water-Reactor Accidents," was published early in 1987. The nuclear industry research program of the BWR Hydrogen Control Owners Group was under review by the staff at the close of the report period.

The estimated completion date for USI A-48 is March 1988. A generic summary report will be issued based on research results of both the NRC and the nuclear industry. The report will also address the conclusions and recommendations of the National Research Council.

GENERIC SAFETY ISSUES

The Commission directed the NRC staff to prepare a priority list of all generic safety issues, including TMI-related

Table 4. Generic Safety Issues Resolved in FY 1987

Number	*Title*
91	Main Crankshaft Failure in Transamerica DeLaval Emergency Diesel Generators
I.A.2.6(1)	Long-Term Upgrading of Training and Qualifications—Revise Regulatory Guide 1.8
I.A.3.3	Requirement for Operator Fitness
I.A.4.2(1)	Research on Training Simulators
I.B.1.1	*Organization and Management Long-Term Improvements*
1.B.1.1(1)	Prepare Draft Criteria
1.B.1.1(2)	Prepare Commission Paper
1.B.1.1(3)	Issue Requirements for Upgrading of Management and Technical Resources
1.B.1.1(4)	Review Responses to Determine Acceptability

issues, based on the potential safety significance and cost of implementation of each issue. In December 1983, the listing was approved by the Commission. The guidance is reflected in the NRC Policy and Planning Guidance, the NRC Strategic Plan, and the RES Five-Year Plan.

Priorities of Generic Safety Issues

The NRC 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 subjected to this method was published in "A Prioritization of Generic Safety Issues" (NUREG-0933), which is updated semi-annually (supplements in June and December). The list of issues includes the TMI Action Plan (NUREG-0660) items and Unresolved Safety Issues (USIs); USIs are 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 1987, the NRC identified six new generic issues, established priorities for 19 issues (Table 3), and resolved eight GSIs (Table 4) other than USIs. In addition, 14 GSIs scheduled for resolution were integrated into the action plans for the resolution of other unresolved GSIs. Table 5 contains the schedules for resolution of all unresolved GSIs.

STANDARDIZED AND ADVANCED REACTORS

Advanced Reactor Concepts

The staff continued to review three advanced reactor concepts that were submitted by the DOE. The purpose of the reviews is to determine the licensability of these unique designs. The conceptual designs consist of two advanced Liquid Metal Reactors and one advanced Modular High-Temperature Gas-Cooled Reactor. The staff plans to issue safety evaluation reports on the three advanced reactors in 1988. In addition, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," was prepared to provide further guidance on the staff's advanced reactor review plans.

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.

Table 5. Generic Safety Issues Scheduled for Resolution

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
23	Reactor Coolant Pump Seal Failures	HIGH	11/89
29	Bolting Degradation or Failures in Nuclear Power Plants	HIGH	10/89
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems	MEDIUM	11/89
66	Steam Generator Requirements	NEARLY RESOLVED	TBD
70	PORV and Block Valve Reliability	MEDIUM	08/88
75	Generic Implications of ATWS Events at the Salem Nuclear Plant2RESOLVED	NEARLY	06/89
77	Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains	HIGH	12/88
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	04/88
82	Beyond Design Bases Accidents in Spent Fuel Pools	MEDIUM	12/88
83	Control Room Habitability	NEARLY RESOLVED	04/90
84	CE PORVs	NEARLY RESOLVED	TBD
86	Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	NEARLY RESOLVED	10/87
87	Failure of HPCI Steam Line Without Isolation	HIGH	08/90
93	Steam Binding of Auxiliary Feedwater Pumps	HIGH	10/87
94	Additional Low-Temperature Over-pressure Protection for Light-Water Reactors	HIGH	12/88
99	RCS/RHR Suction Line Interlocks on PWRs	HIGH	04/88
101	BWR Water Level Redundancy	HIGH	03/90
102	Human Error in Events Involving Wrong Unit or Wrong Train	NEARLY RESOLVED	11/88
103	Design For Probable Maximum Precipitation	NEARLY RESOLVED	10/88
105	Interfacing Systems LOCA at BWRs	HIGH	12/88

Table 5. Generic Safety Issues Scheduled for Resolution (continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	HIGH	09/91
115	Enhancement of Reliability of Westinghouse Solid State Protection System	HIGH	05/89
121	Hydrogen Control for Large, Dry PWR Containments	HIGH	02/88
122.2	Initiating Feed and Bleed	HIGH	06/88
124	Auxiliary Feedwater System Reliability	NEARLY RESOLVED	06/88
125.II.7	Re-evaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	HIGH	08/89
128	Electrical Power Reliability	HIGH	12/89
130	Essential Service Water Pump Failures at Multiplant Sites	HIGH	09/89
134	Rule on Degree and Experience Requirements for Senior Operators	HIGH	09/89
135	Steam Generator and Steam Line Overfill	MEDIUM	11/90
A-29	Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage	MEDIUM	08/89
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	MEDIUM	03/89
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	06/90
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	03/89
B-56	Diesel Reliability	HIGH	12/89
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	10/88
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	01/88
C-8	Main Steam Line Isolation Valve Leakage Control Systems	HIGH	02/89
1.A.4.2(4)	Review Simulators for Conformance	HIGH	10/87
1.D.3	Safety System Status Monitoring	MEDIUM	02/89
1.D.4	Control Room Design Standard	HIGH	09/91

1.D.5(3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	01/88
1.D.5(5)	Disturbance Analysis Systems	HIGH	12/87
1.F.1	Expand QA List	HIGH	TBD
II.B.5(1)	Behavior of Severely Damaged Fuel	HIGH	06/94
II.B.5(2)	Behavior of Core Melt	HIGH	06/94
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	MEDIUM	09/92
II.C.4	Reliability Engineering	HIGH	06/88
II.E.4.3	(Containment) Integrity Check	HIGH	02/88
II.E.6.1	Test Adequacy Study	MEDIUM	05/90
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	MEDIUM	01/88
II.H.2	Obtain Technical Data on Conditions Inside TMI-2 Containment Structure	HIGH	03/91
II.J.4.1	Revise Deficiency Report Requirements	NEARLY RESOLVED	12/87
HF 1.1	Shift Staffing	HIGH	06/88
HF 4.1	Inspection Procedures for Upgraded Emergency Operating Procedures	HIGH	07/88
HF 4.4	Guidelines for Upgrading Other Procedures	HIGH	06/89
HF 5.1	Local Control Stations	HIGH	12/90
HF 5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	01/90
HF 8	Maintenance and Surveillance Program	HIGH	11/89

The purpose of the revised policy is to provide the regulatory framework for reference system design certification of nuclear power plant designs that are essentially complete in both scope and level of detail; that cover plant design, construction, and quality assurance programs; that satisfy regulatory requirements before construction begins; and that can be referenced in individual plant applications. Use of certified reference designs in future license applications should enhance plant safety, increase the efficiency of the NRC review process, and reduce complexity and uncertainty in the regulatory process.

The Commission is also developing proposed regulations (10 CFR Part 52) to implement the revised standardization policy. The proposed Part 52 will provide a regulatory framework for certification of reference designs by means of rulemaking, which will obviate the need to reconsider design issues in individual licensing proceedings on future applications that reference the certified designs.

FUEL CYCLE, MATERIALS, AND SAFEGUARDS RESEARCH AND STANDARDS DEVELOPMENT

In 1987, the NRC continued work on the development and assessment of several regulations related to the transportation of radioactive materials, occupational protection from potential radiation exposures associated with low-level waste disposal operations, and safeguards at fuel facilities to protect against theft of weapons-grade nuclear materials. A report (NUREG/CR-4829) was issued in February 1987 to document the level of protection provided by licensed spent fuel casks against transportation accident forces. The following month, the NRC issued a 30-page brochure (NUREG/BR-0111) to make the study's results more accessible to interested parties both within and outside the NRC. The brochure has been widely distributed to Federal and State authorities withresponsibility to ensure that radioactive material shipments are conducted in a manner that protects public health and safety.

A report (NUREG/CR-4938) issued in July 1987 appraised alternative low-level waste disposal methods (i.e., shallow land burial, below- and above-ground vaults, earth-mounded concrete bunkers, augered holes, and mined cavities) to determine how occupational exposures are influenced by site design, operation, and closure. The results indicate that occupational doses do not vary greatly with disposal method but that slight changes in disposal site designs or operations could significantly affect the resulting occupational doses.

During 1987, a proposed rule that would improve physical security at fuel facilities possessing weapons-grade nuclear material was drafted which would ensure that the safeguards requirements at licensed facilities are not only adequate but comparable with requirements at similar facilities operated by the DOE. The final rule is expected to be completed by the end of fiscal year 1988.

DEVELOPING AND IMPROVING REGULATIONS

Develop or Modify Regulations

In a program initiated in 1985 and continued in 1986 and 1987, 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 two-volume research report published in 1986, and Volume 3 in 1987 (NUREG/CR-4330), provided detailed technical assessments of requirements associated with: (1) post-accident sampling system, (2) impregnated charcoal filters, (3) recombiners in BWR Mark I and Mark II, and (4) turbine missiles. NRC staff will recommend whether to eliminate or modify related requirements of marginal safety importance, based on these studies.

Some nuclear power plant licensees have requested amendment of their operating licenses to permit keeping fuel in the reactor for a longer period than is the current practice. In order to evaluate the environmental consequences of this "extended fuel burnup," the NRC commissioned a study documented in a report entitled "The Environmental Consequences of Higher Fuel Burn-up" (NUREG/CR-5009). In the study, various aspects of fuel production, transportation, power generation, and waste management were considered. The overall finding was that there would be no significant increase in the environmental impact associated with the widespread use of extended fuel burnup. The NRC staff will recommend whether and how to modify the existing regulations on fuel burnup.

The International Atomic Energy Agency (IAEA) has developed proposed revisions to the Nuclear Safety Standards (NUSS) Codes of Practice in light of the Chernobyl accident. The revised Codes of Practice were developed by member country advisory groups and the IAEA technical staff. All member countries were requested to conduct a detailed review of the revised Codes of Practice. The State Department requested that the NRC be the lead agency in developing the U.S. position in the review of these Codes of Practice. Participants in the review included the U.S. Department of Energy, the Nuclear Power Regulations Review Committee, members of the public, and members of the NRC staff. The review focused on the manner in which the IAEA severe accident policy was incorporated in the revised Codes of Practice, and on its consistency with U.S. regulatory policy. The U.S. review was completed and forwarded to the IAEA by the State Department. An IAEA advisory group will review the member countries' comments and suggestions in developing final revisions to the NUSS Codes of Practice.

Regulatory Analysis

RES has among its prime concerns the development and implementation of systematic methods to facilitate NRC decision-making. To accomplish this goal, the April 1987 NRC reorganization (see Chapter 1) terminated the Cost Analysis Group and consolidated the Regulatory Impact Analysis function and resources in RES. With the formation of the Committee to Review Generic Requirements in 1981, the issuance of a revised backfitting rule (§50.109 of 10 CFR Part 50), the endorsement of safety goals for nuclear power plants, and new source term research data, the need for regulatory analysis as a means of fostering a more disciplined regulatory process will continue. During the report period, the Commission initiated and completed several safety-related regulatory analyses, using the methods prescribed in the value-impact handbook (NUREG/CR-3568), in both reactor and non-reactor applications. The methods and procedures of the handbook have been incorporated by reference into the revised regulatory analysis guidelines (NUREG/BR-0058) and Manual Chapter 0514 (Management of Plant-Specific Backfitting of Nuclear Power Plants); they should prove useful to the NRC and industry in evaluating the need for and effectiveness of a variety of regulatory actions, including major rulemaking, standards development, and backfitting safety improvements on nuclear power plants.

Independent Review and Control of Rulemaking. In February 1984, the NRC Executive Director for Operations (EDO) directed that all offices reporting to the EDO and responsible for rulemaking must obtain the EDO's approval

Table 6. Rulemakings Processed for Review in Fiscal Year 1987

		Resulting EDO Action		
	Total	Initiated	Continued	Terminated
Reviews completed	11[1]	6	3	1
Rulemakings exempted from review	30[2]	2	-	12
Reviews under way	3	-	-	-
Reviews deferred until FY 1988	13	-	-	-
	57			

[1]One rulemaking was under review by the EDO on September 30, 1987.
[2]Sixteen rulemakings were published as final rules near the scheduled time for independent review.

to begin and continue a proposed rulemaking action. The directive was aimed at ensuring that rulemaking activity was necessary and would be effective, efficient, timely, and of high quality.

RES was given the task of independently reviewing prospective rulemakings and making recommendations to the EDO as to whether to proceed with them. Later, the RES role was expanded to include conducting annual independent reviews and making similar recommendations to the EDO concerning ongoing rulemakings.

During fiscal year 1987, RES was given lead responsibility in the NRC for rulemaking. A total of 57 rulemakings were processed by RES for potential independent review during the period. Of these, full reviews were completed on 11 rulemakings, 30 were exempted from RES review because of final publication or EDO action, and 16 were in progress at the end of the report period or had been deferred until fiscal year 1988. The detailed status of these reviews, as of September 30, 1987, is provided in Table 6.

It is estimated that in fiscal year 1988 there will be approximately 38 rulemakings that will require RES independent review and EDO approval for initiation or continuation.

Timeliness of Rulemaking. RES has established a tracking and feedback system to help the EDO ensure the timeliness of approved rulemaking actions. Existing quarterly updating of rulemaking entries in the NRC Regulatory Agenda (NUREG-0936) was modified to require a timetable for each ongoing rulemaking sponsored by an office reporting to the EDO, and a summary report on the timeliness of the schedules of each rulemaking.

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.

The NRC staff is active 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 206 NRC staff members serve on working groups organized by technical and professional societies.

Proceedings and Litigation

Chapter 10

This chapter covers two major subjects: (1) the activities—with a report on select proceedings—of the NRC's Atomic Safety and Licensing Board Panel and the Atomic Safety and Licensing Appeal Panel, and (2) noteworthy legal actions and litigation involving the Commission, including cases both pending and closed.

ATOMIC SAFETY AND LICENSING BOARD PANEL

In fiscal year 1987, the Atomic Safety and Licensing Board Panel (ASLBP) completed all but three of the extensive operating license proceedings that have been its primary responsibility since the start of the decade. Licensing Boards authorized operating licenses for three new nuclear power plant units and completed a total of 25 complex proceedings. At the same time, the panel addressed a rapidly expanding number of smaller cases related to previously licensed facilities, and began preparations for its biggest case of all, the proposed high-level nuclear waste repository.

The Atomic Energy Act of 1954 requires that a public hearing be held on every application for a construction permit for a nuclear power plant or related facility. In certain circumstances, hearings are also held in connection with operating licenses, license amendments, antitrust issues, enforcement and civil penalty cases, and other matters as directed by the Commission. (See "The Licensing Process," in Chapter 2.) Boards composed of three administrative judges drawn from the Atomic Safety and Licensing Board Panel (ASLBP) perform the Commission's hearing function and render initial decisions in licensing cases; single administrative judges and administrative law judges also hear and decide other matters. These hearings are the Commission's principal public forum in which individuals and organizations can voice their interest in a particular licensing, enforcement, or other matter of public concern and have those interests adjudicated by an independent tribunal.

As of September 30, 1987, the panel included 19 permanent and 22 part-time administrative judges drawn from various professions. There were 14 lawyers, 12 environmental scientists, 7 engineers, 5 physicists, 1 medical doctor, 1 economist and 1 chemist. (See Appendix 2 for the names of panel members.) The Commission appoints administrative judges to the panel based upon recognized experience, achievement, and independence in the appointee's field. Judges are assigned to three member Licensing Boards in cases in which their professional expertise will help to resolve the issues litigated. Generally, Licensing Boards consist of a lawyer as chairman, a nuclear engineer or reactor physicist, and an environmental scientist.

The hearing on a particular application for a nuclear facility license may be divided into several phases, each focusing on a particular licensing concern, for example: (1) health, safety, or the common defense and security aspects of the application, as required by the Atomic Energy Act; (2) environmental considerations, as required by the National Environmental Policy Act (NEPA); and (3) emergency planning requirements. These matters, as well as especially complex technical issues, are frequently the subject of separate initial decisions by the Licensing Boards.

Administration

As cases have become more intensely and actively litigated, and the issues to be decided have grown increasingly complex, the effective management of the logistics of the hearing process has become especially important. To compensate for anticipated restrictions on support personnel, the panel has aggressively pursued the automation of hearing functions. As a result, administrative support for the boards and the panel has been automated. Systems and equipment include IBM Personal Computers and word processors, the LEXIS and WESTLAW automated legal research systems, and a computerized travel and timekeeping system. An internal computerized Hearing Status Report now has a virtually complete data base and is capable of generating valuable case management information. In addition, virtually all ASLBP computer work was transferred during the year from a mainframe at the National Institutes of Health to the panel's personal computers. The conversion has had two principal benefits: (1) elimination of almost $10,000 per year in storage and use charges; and (2) increased flexibility, speed, and usefulness of reports created through in-house programming and production. The panel is consolidating and revising data bases to obtain more accurate evaluations and analyses of operations and management.

The panel's Computer Assistance Project (CAP) to expedite large cases made major strides during the year. Computerization of the Indian Point hearing transcript in 1983 proved that substantial time and labor could be saved in decision writing by computerizing and indexing the full text of the transcript. In place at the outset of a large case, a computerized system would permit, as needed, electronic filing, computerized transcripts, pre-filed testimony, and proposed findings of fact and conclusions of law for enormously expedited record searches, shortened hearings, and faster and more complete decision making. By using resources for the most part already in place, the cost-benefit ratio for large cases should be substantial.

In pursuit of this objective, a Licensing Board for the first time, in a May 22, 1987 Memorandum and Order in the Diablo Canyon (Cal.) case, required parties to submit computer readable diskettes with their hard copy filings. The board sought to expedite the proceeding by capturing three categories of record materials, namely: (1) pre-filed testimony; (2) proposed findings of fact and conclusions of law; and (3) the transcript of the hearing. The electronic capture was intended to assist the parties and the board by making available a full text, an electronically searchable record to aid and expedite the preparation of findings of fact, conclusions of law, and the Initial Decision. The Order was a first step in automating the hearing process in preparation for the high-level waste proceeding. The panel is gradually exploring the myriad steps necessary to resolve the problem of incompatible computer equipment in electronic filing.

The Caseload

During the fiscal year ending September 30, 1987, Licensing Boards conducted 52 proceedings involving nuclear power plants and other nuclear facilities with a construction value well in excess of $44 billion. Forty-eight percent of the proceedings were completed. Some 115 days of hearing were held, comprising 92 days of trial and 23 days of pre-hearing conferences. Twenty-five proceedings were closed while eighteen new cases were opened. The operation of three nuclear power plant units was authorized.

At the same time, however, the panel has continued its efforts to prepare for proceedings involving the construction of a high-level nuclear waste repository. Dozens of well-funded intervening parties are prepared to participate in a case that may involve more than 16 million documents. The panel's efforts to expand its ability to utilize sophisticated computer systems for document and hearing management will be essential if its role in the proceeding is to be effective.

Hearing Procedure

The heavy ASLBP caseload, combined with increasing public awareness and involvement in the licensing process, has made effective hearing management critical to the timely completion of licensing decisions. Using the procedural tools available under Commission regulations, Licensing Boards have more sharply focused efforts to assure that issues for hearing are soundly based and well-defined. Pre-hearing conferences are utilized extensively for the purposes of reviewing and refining proposed contentions, defining the scope of relevant discovery, and developing realistic hearing schedules. The discovery process itself is closely monitored in order to eliminate unnecessary or duplicate efforts and to assure the early resolution of potentially time-consuming disputes. As a result of this active management, over 90 per cent of the contentions filed in operating license proceedings were resolved prior to hearing. Most importantly, however, these efficiencies have been achieved through hearing management practices that insure the fundamental fairness to all parties mandated by law.

Fiscal year 1987 also saw an expansion in the use of informal proceedings presided over by a single judge in materials licensing cases. The panel established a policy in these cases of assigning an administrative judge as advisor to the presiding officer to supplement either legal or technical expertise as needed. The panel judges who have participated in these informal hearings have, by sharing their experiences, developed efficient methods for conducting what was originally a less than clearly defined type of adjudication. They have also contributed to the Commission's effort to draft informal hearing rules, encouraging a combination of simplicity and thoroughness.

Cases of Note

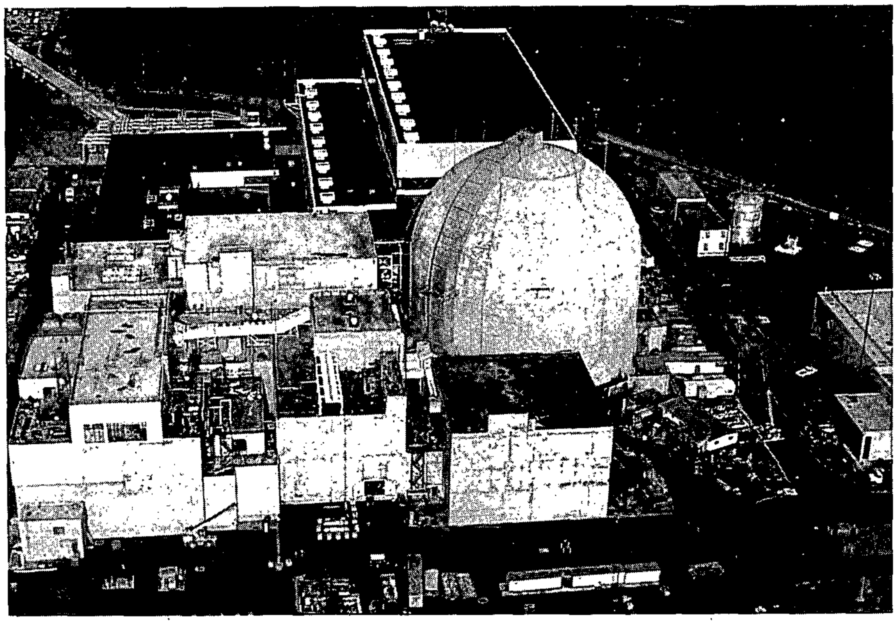
Operating Licenses. In partial initial decisions issued May 13 and May 19, 1987, the Licensing Board in *Braidwood* (Ill.) resolved emergency planning and quality assurance issues in favor of the applicant and authorized a full power license.

The May 13 decision on two emergency planning contentions dealt primarily with the wording and distribution of the applicant's emergency information booklet. The board found no deficiencies in the booklet or the plans for its dissemination serious enough to preclude license issuance. It did, however, find that a better explanation of the relationship between a radioactive plume, weather conditions, and the selection of optimum evacuation routes should be included. The board ordered as a condition of license authorization that a discussion of those topics be incorporated in the next annual revision of the booklet.

The board's May 19 decision dealt with extensive claims of intimidation and harassment of quality control inspectors. This phase of the Braidwood licensing proceeding required nearly 100 hearing days over seven months; encompassed the oral testimony of some 60 witnesses; and resulted in a record of approximately 18,000 pages, including over 500 exhibits.

The majority of the board, in a two-one decision, found considerable evidence of production pressure. However, the board was convinced by the testimony of the inspectors who appeared as witnesses that that pressure did not have any effect on job performance.

The Seabrook Unit 1 nuclear power plant, shown here, was cleared to operate at 5 percent of rated power in a March 7, 1987 partial initial decision by an Atomic Safety and Licensing Board. At year's end, however, other hearings, reviews, and litigation had precluded any action at the plant beyond fuel-loading.



With respect to the ultimate question of whether there was a sufficiently large breakdown in quality assurance procedures to preclude a finding of "reasonable assurance" of safety to the public, the board concluded that there had not been. Their finding was buttressed by the results of two large and independent re-inspection programs, which statistically confirmed the adequacy of the quality control inspectors' performance.

In a Concluding Partial Initial Decision, the *Vogtle* (Ga.) Board found that licenses authorizing operation of the Vogtle Electric Generating Plant should be issued. The board ruled that applicants provided assurance that certain models of solenoid valves that are used to perform safety-related functions are environmentally qualified.

In a supplement to its fourth partial initial decision, the *Limerick* (Pa.) Licensing Board found that arrangements in effect at the State Correctional Institution at Graterford, Pa., for the notification and mobilization of off-duty correctional officers in a radiological emergency were adequate to meet the requirements of NRC regulations. The ruling resolved the final pending issue in Limerick, an issue that had been remanded by the Appeal Board.

In a March 1987 Partial Initial Decision, the Licensing Board in *Seabrook* (N.H.) authorized issuance of a license to operate Unit 1 up to 5 percent of rated power. It resolved three on-site emergency planning and safety contentions related to: (1) applicants' emergency classification and action level scheme; (2) electrical equipment environmental qualification time duration; and (3) applicants' safety parameter display system (SPDS). The low-power license authorization was contingent upon applicants' prior development of maintenance procedures to insure an adequate level of oil continuously present in the riser assemblies associated with the containment water level transmitters.

In *Shoreham* (N.Y.), the Licensing Board denied applicant's motion for summary disposition of the "legal authority" issues. LILCO based its motion on a "realism" argument that State, county and local officials could realistically be assumed to respond in the event of an actual emergency. LILCO argued that this response would render immaterial the inability of utility employees to carry out certain emergency functions under New York State law. In its Memorandum and Order, the Licensing Board reviewed the applicable law on summary disposition and interpreted Commission rulings involving the "legal authority" issues and their effect on the motion for summary disposition. The board found that LILCO had not met the requirements of the summary disposition rule. This decision cleared the way for a full hearing on the "realism" argument.

Show Cause. In the *Sheffield* (Ill.) show cause proceeding, the Licensing Board denied a motion by the licensee for summary disposition. This was the first ruling addressing whether a repository licensee could unilaterally terminate its obligations as to buried low-level waste. The decision was later vacated by the Appeal Board when Illinois became an agreement State.

Civil Penalty. In a June 22, 1987 Memorandum and Order in a civil penalty proceeding related to Three Mile Island, Unit 2 (Pa.), the Administrative Law Judge denied a motion by the Department of Labor (DOL) to quash a deposition subpoena to a retired employee of the department. This unusual case involved "whistle-blower" retaliation allegations over which the NRC and DOL have concurrent jurisdiction. The board held that the regulations invoked by Labor to preclude testimony by its former investigator were "housekeeping" in nature, intended only to permit the Department to control the appearance of its employees pursuant to the demands of litigants. They were not, and could not be, intended to authorize DOL to withhold relevant, discoverable information; in any event, they were inapplicable to a now retired employee.

On March 3, 1987, the presiding Administrative Law Judge issued an Initial Decision imposing a civil penalty on Hurley Medical Center, a community hospital in Flint, Mich. In doing so, the judge aggregated several Severity Level IV violations into a single Severity Level III violation because the separate violations evidenced a general failure to exert adequate management and control over the licensee's radiation safety program.

Materials Licenses. After considering numerous filings and conducting a three-day oral hearing, the Presiding Officer in *Sequoyah* (Okla.) authorized the issuance of a license amendment permitting Sequoyah Fuels Corporation (SFC), a subsidiary of Kerr-McGee Corporation, to operate its facility to convert depleted uranium hexafluoride to depleted uranium tetrafluoride at its Gore, Okla., plant. The authorization was subject to four conditions: (1) to ensure that the automatic telephone emergency notification system will function properly, SFC was required to verify that all residences within a two-mile radius of the facility have telephones and make provisions acceptable to staff to notify any that do not; (2) SFC was required to verify that all telephone numbers listed in its emergency response plan are accurate at each major exercise of the plan; (3) SFC is to maintain the level of staffing outlined in its testimony presented at the hearing and to promptly report any changes in the duties of those individuals to staff; and (4) SFC's president and its general manager are each to spend at least one full workday each month at the facility while it is in operation.

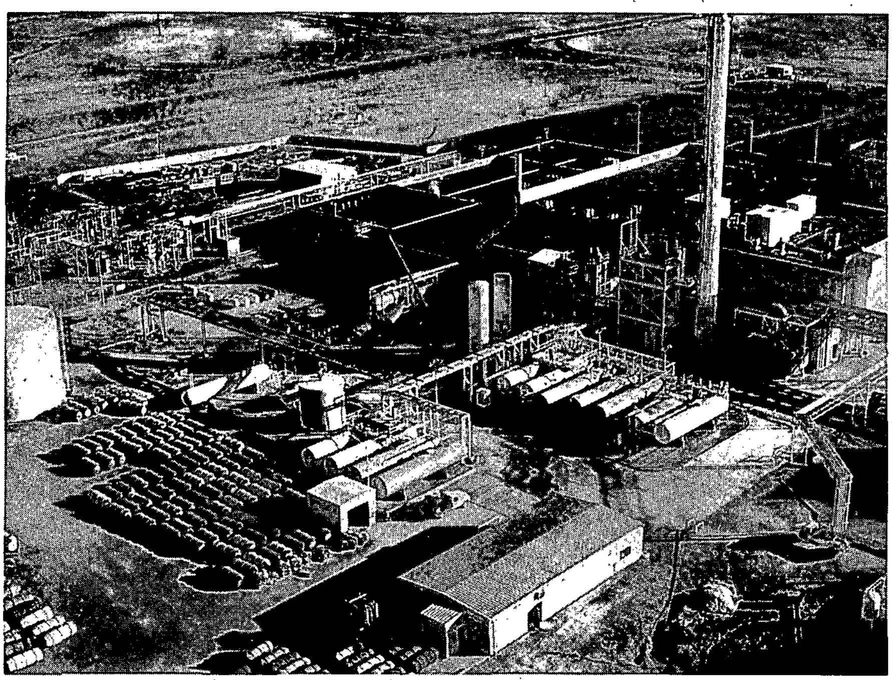
Informal Proceedings. In the *Davis-Besse* (Ohio) proceeding on waste disposal, the single presiding officer, a lawyer, affirmed the decision of the Commission to grant licensee's application to dispose of water treatment sludge and secondary side demineralizer resins by land burial at the site of the Davis-Besse nuclear power plant.

The judge held that reasonable assurance that the public health and safety and environment will be protected when waste is buried can be found if (1) secure confinement of waste at its burial location is assured, even if the waste is a significant source of radioactive or chemically toxic constituents, or (2) the waste itself is not a significant source of hazardous materials, even if the conditions of confinement are not so secure as to guarantee that nothing would escape from the burial site in the future.

The presiding officer in the *Parks Townships* (Pa.) case authorized the staff to issue a license amendment to operate a super-compactor immediately, but ruled that the staff could not issue a license amendment to operate an incinerator at the same site until further testing demonstrated that the incinerator will meet the licensee's criteria for safe operation. The decision concluded the first informal materials license proceeding heard, decided, and written by a panel judge who was not a lawyer.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Atomic Safety and Licensing Appeal Boards, each consisting of three members, perform review functions for the Commission in a wide range of formal adjudicatory proceedings. These include proceedings for the licensing of nuclear power plants and other nuclear facilities. The deci-



The Sequoyah Fuels Corporation plant at Gore, Okla., was the scene, in January 1986, of an accident in which a worker was killed and several injured when a cylinder containing uranium hexafluoride ruptured. After investigation and hearings, the presiding officer of a safety and licensing board authorized issuance of a license amendment placing four new conditions on plant operations. A number of cylinders of the type that burst are visible in the lower left portion of the photo, taken at the facility.

sion of the Appeal Board in these proceedings becomes the final agency order unless the Commission, in its discretion, decides to review it. In the absence of such Commission action, the Appeal Board decision is subject only to judicial review in a Federal court of appeals. The more significant decisions are published in the permanent collection of NRC licensing and other decisions *(Nuclear Regulatory Commission Issuances)*. (See Appendix 2 for the composition of the Atomic Safety and Licensing Appeal Panel (ASLAP), from which appeal board members for a particular proceeding are selected by the panel Chairman.)

The year saw a further decline in the number of licensing proceedings before the Appeal Boards as additional pending cases came to an end in the absence of any new applications for nuclear facility licenses. The decline was balanced to some extent by operating license amendment proceedings involving the expansion of spent fuel pools and other types of proceedings. More and more utilities sought to increase the capacity of their existing spent fuel pools to prevent a shortage of spent fuel storage capacity pending decisions by the Government on other alternatives.

Four operating license proceedings—the Seabrook (N.H.), Vogtle (Ga.), Limerick (Pa.), and Shearon Harris (N.C.) cases—generated the most activity for the Appeal Boards. The Seabrook proceeding alone resulted in eight published decisions in addition to numerous unpublished orders and rulings. Other proceedings giving rise to significant Appeal Board action included those concerned with the Shoreham (N.Y.), Diablo Canyon (Cal.), Turkey Point (Fla.), Three Mile Island (Pa.), Braidwood (Ill.), Comanche Peak (Tex.), and Vermont Yankee (Vt.) nuclear facilities. There was also significant Appeal Board action in the proceeding involving off-site radioactive contamination near the Kerr-McGee Chemical Corporation's facility at West Chicago, Ill.

Seabrook

The Seabrook facility is fully constructed. Before it can be licensed to operate, however, the NRC must find not only that the plant can be operated safely but also that there is reasonable assurance that adequate measures to protect the public can and will be taken in the event of a radiological emergency. To enable the latter determination to be made, Commission regulations require that an applicant submit radiological emergency response plans of State and local governmental entities whose jurisdictions extend to areas within the 10-mile, plume exposure pathway emergency planning zone (EPZ) that surrounds a facility. The Seabrook EPZ lies mostly in New Hampshire but a portion of it extends into Massachusetts. The applicants submitted plans of the State of New Hampshire and its local governments; however, the Commonwealth of Massachusetts and its local governments refused to cooperate in the emergency response planning on the ground that timely evacuation of the EPZ in an emergency was not practicable.

Following a hearing on the safety aspects of the facility, the Licensing Board authorized the issuance of an operating license limited to fuel loading and pre-criticality testing. The Massachusetts Attorney General appealed, disputing that board's authority to allow the issuance of such a license in the absence of any Massachusetts emergency plans. The Appeal Board ruled that, under the pertinent regulations, such a license was not barred. The Commission, however, undertook review of that ruling and declined to uphold the license authorization. According to the Commission majority, the question was not strictly a legal one, but rather one which involved matters of regulatory policy which ultimately it, alone, should decide. It concluded that "sound policy favors requiring the filing of a State, local, or utility plan before any operating license is issued, including a license confined to fuel loading or low-power testing." Subsequently, the applicant filed a "utility" plan for the Massachusetts portion of the EPZ; the Commission thereafter lifted its stay of the low-power license.

Another Licensing Board decision, addressing plant safety issues, authorized low-power operation (up to five percent of rated power) of the facility. Appeals by the Massachusetts Attorney General and others raised the question whether such a license could be issued in advance of the resolution of all emergency planning issues, including review of the plan by the Federal Emergency Management Agency (FEMA) and the Commission. This question had been raised earlier by several of those same parties who sought a stay of that decision but had failed to persuade the Appeal Board that, among other things, they were likely to succeed on the merits. In a later decision on the merits, the Appeal Board decided that, under the Commission rules, the continued existence of emergency planning issues did not, in and of itself, act to bar the issuance of a low-power operating license.

In another appeal, the Appeal Board was called upon to decide whether the schedule adopted by the Licensing Board for the hearings to be held on the New Hampshire emergency response plan was so abbreviated as to deprive the intervenors of due process. In this instance, the Appeal Board agreed with the intervenors' claim and ordered modifications to the schedule.

The question in still another appeal was whether United States Senator Gordon J. Humphrey of New Hampshire could participate in the proceeding under the "interested State" provision of the Commission's Rules of Practice. Under that provision, a State or local government is entitled to participate in a licensing proceeding in a special capacity. The Licensing Board had rejected admission of the Senator in light of the New Hampshire Attorney General's representation of the State in the proceeding. The Appeal Board affirmed the Licensing Board's decision. But recognizing that the Senator might make a worthwhile contribution to the proceeding, it took the unusual step of authorizing him to participate before the Licensing Board as an amicus curiae, presenting his views orally or in writing on any legal or factual issue presented by the pleadings of the parties or the evidentiary record.

Vogtle

The appeal in *Vogtle* dealt with environmental, technical and emergency planning issues. Several of the issues were in the form of contentions rejected at the threshold by the Licensing Board. These included, for example, allegations that the design of the plant failed to take into consideration recent seismic data associated with a claimed geologic fault or the 1886 Charleston earthquake. Several other issues were concerned with allegations of quality assurance deficiencies, defects in equipment, drug and alcohol use by workers at the construction site, and possible contamination of the public water supplies in the area in the event of an accident. With respect to each of these issues, the Appeal Board found that the Licensing Board had acted correctly either in rejecting the allegations without the necessity of a hearing or in ruling that the allegations were not substantiated by the evidence.

Limerick

The emergency response plan for the Limerick facility continued to be the subject of dispute. One major issue concerned whether there would be an adequate number of school bus drivers willing and available to assist in the evacuation of two school districts within the EPZ. Another related to the evacuation of a State prison located within the EPZ. In both cases, the Appeal Board upheld the Licensing Board's findings in favor of the applicant.

Shearon Harris

Following the Licensing Board's authorization of an operating license for the Shearon Harris facility, the intervenors raised a number of issues on appeal. One involved the Licensing Board's findings essentially rejecting the intervenors' allegations of widespread drug use at the plant's construction site. Another involved that board's findings rejecting the intervenors' claim that the applicants' system of public notification in the event of a radiological emergency at the plant did not comply with the Commission's emergency planning requirements. On both of these issues, the Appeal Board agreed with the Licensing Board's findings and conclusions.

In another decision, one of the many issues raised by the intervenors concerned the quality of concrete placements in the Shearon Harris containment building. The Licensing Board had rejected their claim that, among other shortcomings, the concrete pours in the construction of the building had been inadequately tested for strength. Upon review, the Appeal Board found that no violation of the pertinent code or standard occurred. The Appeal Board reviewed other aspects of the Licensing Board's decision and agreed with the Licensing Board's authorization of an operating license for the plant.

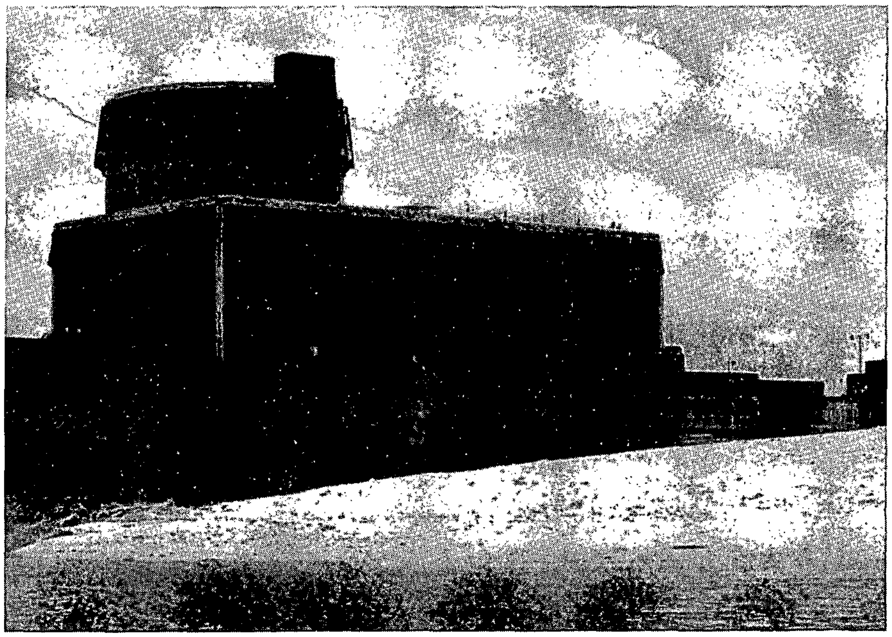
Other Noteworthy Proceedings

Under the Atomic Energy Act, construction of a facility must be completed before the date specified in the construction permit unless the Commission extends the date for good cause shown. In *Comanche Peak,* the applicant had allowed the construction permit for Unit 1 of the plant to expire before plant completion and then sought an extension of the permit. The requested extension was challenged by two intervenors who contended that there was no good cause for the extension. The contention was admitted for hearing and the applicant appealed. Upon review, the Appeal Board affirmed.

Unlike the usual emergency response plan, the plan for the Shoreham facility does not rely on State or local government personnel for its implementation. Following a hearing at which the adequacy of the plan was challenged by the State and county in which the plant is situated, the Licensing Board resolved most of the contested issues in favor of the applicant. The board, however, ruled that the applicant lacked the legal authority to implement material features of the plan—i.e., those normally performed by governmental authorities—with the consequence that an emergency plan in conformity with NRC regulations cannot be carried out. On appeal, the Appeal Board upheld the Licensing Board's conclusions on most issues. On two of the issues, it decided that the proceeding should be sent back to the Licensing Board for further hearings and determination. One of these issues involved the planning basis for those who might seek only monitoring and decontamination in the event of a radiological emergency. The utility had planned for those seeking sheltering but not for those who might need only the other services.

In another decision in the Shoreham operating license proceeding, the matter before the Appeal Board was a petition by FEMA seeking to obtain immediate Appeal Board review of an interlocutory Licensing Board ruling accepting two contentions for litigation proposed by an intervenor. FEMA, a non-party to the proceeding, complained, among other things, that it would be irreparably harmed unless the contentions were excluded from the proceeding. The Appeal Board denied the petition on the ground that the test for interlocutory appellate review had not been met because that agency failed to demonstrate that admission of the contentions either (1) threatened it with immediate and serious irreparable impact, which, as a practical matter, could not be alleviated by a later appeal; or (2) affected the basic structure of the proceeding.

The *Kress Creek* appeal involved radiological materials discharged into Kress Creek from the licensee's Rare Earths



Litigation over emergency planning for the Shoreham nuclear power plant on Long Island, N.Y., continued throughout 1987 and, at year's end, the major issues remained unresolved. Shown in the photo are the radwaste building, turbine building, reactor building, and security building.

Facility in West Chicago, Ill. The question for Appeal Board determination was whether NRC still had jurisdiction over the materials in light of a recently executed NRC agreement with the State of Illinois transferring regulatory jurisdiction of certain radiological materials to the State. Upon review of the agreement and the materials' history, the Appeal Board decided that the materials in question were still subject to NRC regulations. Subsequently, the Commission announced that it would review the decision. Commission action was pending at the close of the report period.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1987 are discussed below. The Commission's actions on export licensing cases are discussed in Chapter 8.

Shoreham Station—Two Major Decisions

In fiscal year 1987, the Commission issued two major decisions concerning the Shoreham (N.Y.) nuclear power plant, both concerning the facility's emergency plan. Litigation concerning this plan was still in progress at the close of the report period.

In *Long Island Lighting Company* (Shoreham Unit 1), CLI-87-04, 25 NRC (June 11, 1987), the Commission denied the request of the utility (Long Island Lighting Co., or LILCO) for an immediate 25 percent license. The utility had asserted that a 25 percent license was needed for adequate electrical power supplies on Long Island during the summer months. In denying the request, the Commission stated that unless the parties were able to propose some means to settle the outstanding emergency planning issues, it could not lawfully grant the utility's request for immediate authorization to increase power from 5 percent to 25 percent of rated capacity.

The Commission concluded that because it would be necessary to resolve new factual issues raised by the request under normal adjudicatory procedures, pursuant to 10 C.F.R. §50.57(c) and 10 C.F.R. Part 2, Subpart G, and because LILCO appeared to desire a more expedited procedure than would be required under those regulations, an immediate authorization was not possible.

In CLI-87-05, 25 NRC (June 11, 1987), the Commission evaluated, under the criteria of 10 C.F.R. §2.734, intervenors' motion to reopen the Shoreham emergency planning record on three issues. First, the Commission granted the motion to reopen as to the withdrawal of WALK radio as the primary emergency broadcast system (EBS) radio station for the emergency plan, but remanded the reopened issue to the Licensing Board with instructions to delay the admission of contentions until receipt of LILCO's modified emergency plan. On the other two issues—(1) the lack of an agreement between the utility and the American Red Cross (ARC) for its participation in emergency response, and (2) the absence of agreements between the ARC and certain shelter owners for the use of shelters in a Shoreham emergency—the Commission denied intervenors' request to reopen. The Commission found that the ARC's charter and policy require it to assist in emergency response whether or not there is an agreement. Moreover, the Commission concluded that movants had not demonstrated that there would have been a materially different result, or that such a result would have been likely, had the absence of the agreements been considered initially. Accordingly, the Commission denied the request to reopen on the two issues.

Braidwood—Full-Power Licenses

In *Commonwealth Edison Company* (Braidwood Units 1 and 2 (Ill.)), CLI-87-07, 25 NRC (June 30, 1987), the Commission conducted, under 10 C.F.R. 2.764(f), an immediate effectiveness review to determine if two Licensing Board decisions should be stayed. The decisions had resolved all contested issues in the proceeding in favor of applicant and authorized the issuance of full-power operating licenses.

The first decision dealt with two emergency planning issues—public information programs, and evacuation of institutions such as nursing homes. The board ruled in favor of the applicant on the public information program, but declared the intervenor in default on the second issue. The second and concluding decision concentrated on a single contention which alleged that certain specified instances of harassment of quality assurance inspectors who inspected electrical system welding had taken place in the last few years at Braidwood. The board was unanimous in concluding that there was reasonable assurance that that part of the electrical system which was installed during the period at issue under the contention could be operated without adverse impact on the public health and safety.

The Commission noted that although the adequacy of the electrical welding system was not one of the contested issues in the proceeding, the Commission's public health and safety responsibilities require it to consider a safety issue discussed by the board, even though the issue was not properly before the board.

The Commission concluded that no safety reasons existed for staying the effectiveness of the board's decisions, and that the decision authorizing issuance of full power operating licenses should become effective, pending completion of the agency's adjudicatory appellate process.

Seabrook Station—Good-Faith Utility Plan Required

In 1987, the Commission issued two significant decisions concerning the Seabrook (N. H.) nuclear power plant. Both decisions concerned the facility's emergency plan for the part of the plume exposure pathway emergency planning zone that lay within Massachusetts. At the close of the report period, litigation concerning the utility plan for Massachusetts was still going on.

In *Public Service Company of New Hampshire* (Seabrook Units 1 and 2 (N.H.)) CLI-87-02, 25 NRC (April 9, 1987), the Commission undertook *sua sponte* review of the authorization of a low power license before the utility applicant had submitted a radiological emergency plan for the facility's entire plume exposure pathway emergency planning zone. The Commission determined not to authorize low-power operation until the applicant has submitted an emergency plan for the Massachusetts sector of the plume exposure emergency planning zone, even though a demonstration of off-site emergency preparedness is not required for low-power operation. The Commission found that in the special circumstances of this case, sound regulatory policy required the filing of such a complete radiological off-site emergency plan prior to issuance of any operating license, including a low-power license, for Seabrook.

The Commission distinguished its decision allowing low power operation in the Shoreham proceeding where a utility off-site emergency plan had been filed, but where there was uncertainty regarding the merits of the emergency plan. The Commission noted that submittal of a complete off-site plan makes possible a summary review to determine if adequate emergency planning is at least in the realm of the possible.

In CLI-87-03, 25 NRC 875 (June 11, 1987), based on its finding that utility applicant's substitute emergency plan was not truly a utility plan, the Commission denied applicants' request to vacate CLI-87-02 as moot and to lift the stay on low power operations at Seabrook. Citing *Long Island Lighting Co.* (Shoreham Unit 1), CLI-86-13, 24 NRC 22 (1986), the Commission emphasized that NRC case law has clearly defined a utility emergency plan as one that provides measures to be taken by the utility to compensate for the absence of governmental participation in emergency planning. The Commission further pointed out that where, as in these circumstances, it has required submittal of an emergency plan in the absence of State and local governmental cooperation in providing some of the materials that normally are essential to support a full power license under Commission regulations, an adequate filing would be one intended for actual implementation as a utility emergency plan and intended to be subjected to staff and FEMA review and litigation on that basis. The Commission also noted that while "realism"—a presumption that State and local governments will respond in the event of a real emergency—may play a role in the ultimate decision on the acceptability of planning that lacks State participation, "realism" cannot fill the void caused by the lack of a utility plan that reflects the utility's compensatory measures and efforts to facilitate the State's participation in the event of an emergency. The utility subsequently filed a plan which satisfied the Commission requirements set forth in CLI-87-03, and the Commission lifted its stay on low-power operations. Other remanded technical issues, however, delayed issuance of a low-power license.

Shearon Harris—Full Power Authorized

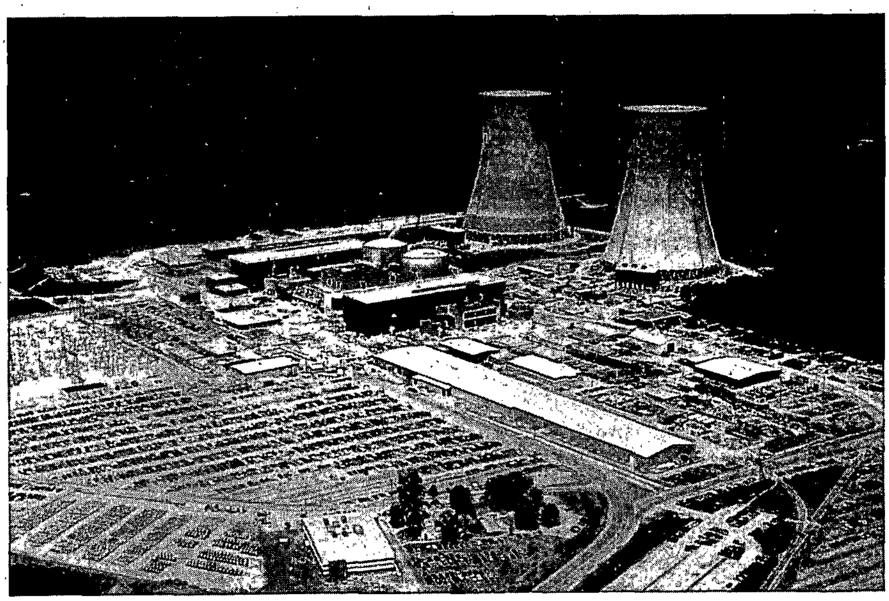
The Commission issued two decisions on the Shearon Harris (N.C.) plant in fiscal year 1987. The first decision, issued at the end of 1986, addressed the issue of whether a hearing must be held with respect to the licensee's request for an exemption from the requirement that a full-scale emergency planning exercise be held one year prior to issuance of a full power operating license. In its second decision, in early January 1987, the Commission authorized issuance of a full power license for the facility.

In *Carolina Power & Light Company and North Carolina Eastern Municipal Power Agency* (Shearon Harris nuclear power plant), CLI-86-24, 24 NRC 769 (1986), the Commission denied petitioners' request for a hearing on applicant's request for an exemption from the NRC's emergency preparedness exercise requirement, 10 C.F.R. Part 50, App. E, Sec. IV.F.1, that a full-scale emergency planning exercise be held one year prior to issuance of a full power operating license. The Commission stated that in this instance the standards to be met for a grant of an exemption from the NRC's licensing requirements were found in 10 C.F.R. §50.12(a)(1) and (a)(2) which provides that (1) the exemption must be authorized by law, not present an undue risk to the public health and safety, and be consistent with the common defense and security; and (2) "special circumstances" must be present. The Commission stressed that a person seeking a hearing on the exemption request must demonstrate that there exists a material issue of fact regarding the appropriate application of either of those standards. The Commission concluded that since petitioners had failed to raise any material issue of fact with respect to whether applicant had met the exemption standards of §50.12, it would not grant a hearing on the exemption request.

In CLI-87-01, 25 NRC 1 (1987), the Commission authorized, as immediately effective, issuance of a full-power license by the NRC staff for the Shearon Harris nuclear facility. The Commission based its decision on: (1) its review of contested safety-related contentions resolved in the remaining Licensing Board partial initial decision not administratively finalized through Commission appellate review; and (2) issues not contested before the Licensing Board, but raised in intervenors' effectiveness comments, at various public meeting presentations, and in a pending 2.206 petition—all of which were resolved in favor of the facility's operation. The Commission found that the issues intervenors sought to raise had been resolved either in Licensing Board, Appeal Board, or Commission rulings on contested matters, or through the staff's analysis of Section 2.206 petitions, and thus did not provide a basis for delaying the Commission's authorization to the Director of Nuclear Reactor Regulation to issue a full-power operating license. The Commission also noted that to provide grounds for a delay of the effectiveness of a Licensing Board decision authorizing issuance of a full-power license, an intervenor's concerns regarding a contested issue, such as management competence, must challenge the board's substantive conclusions regarding the issue. The Commission decided that comments that were no more than speculation about the integrity of a member of the agency's staff responsible for the oversight of the utility's management competence as insufficient grounds to delay its licensing decision.

Perry—Full Power Authorized

In *Cleveland Electric Illuminating Company* (Perry Units 1 and 2 (Ohio)), CLI-86-22, 24 NRC 685 (1986), the Commission authorized the issuance of a full-power license for the Perry Unit 1 nuclear facility based on a determination that the formal adjudicatory proceeding had resolved contested matters related to license issuance, and that various uncontested issues considered outside of the formal proceeding had also been resolved in favor of the plant's operation.



Issuance of a full-power operating license was authorized by the Commission for the Cleveland Electric Illuminating Company's Perry nuclear power plant, Unit 1, following extensive hearings occasioned by a January 1986 earthquake near the Perry site.

After the Perry adjudicatory record had closed, the Governor of Ohio formed a team to review the evacuation plans in light of the accident at Chernobyl and after an earthquake had occurred approximately 10 miles south of the Perry plant. The Commission stated that it appreciated the Governor's desire to improve the status of emergency preparedness and that the NRC itself was continuing to study the implications of Chernobyl. Under NRC regulations, "the issue relevant to licensing of Perry is not whether continued improvements are a useful goal, but whether there is reasonable assurance that adequate protective steps can and will be taken in the event of a radiological emergency.". The Commission made such a "reasonable assurance" determination for Perry based upon its review of findings and determinations by the Federal Emergency Management Agency (FEMA).

The Commission emphasized that where there is reasonable assurance that adequate protective measures can be taken in the event of a radiological emergency, it is neither necessary nor appropriate to postpone the issuance of an operating license on the basis of preliminary State concerns which are without a detailed technical and factual basis and which are being considered outside of a concluded formal adjudicatory proceeding. As to concerns with regard to the plant's seismic design raised after the formal proceeding had closed, the Commission stated that it was satisfied that Perry's seismic design "has adequate safety margins to accommodate the recorded Ohio earthquake of 1986."

Import of South African Uranium—Anti-Apartheid Act of 1986

In 1987, the Commission received petitions asking it to deny pending applications which sought authorization to import South African-origin uranium and also to revoke existing licenses which permitted such imports. At issue was the scope of the Comprehensive Anti-Apartheid Act of 1986's bar on the import of uranium ore and uranium oxide. The primary questions were whether the importation bar extended to: (1) uranium ore and uranium oxide regardless of its intended end use; and (2) uranium ore and uranium oxide which is transformed into uranium hexafluoride or other "substantially transformed" compounds before it is imported into the United States.

After a hearing on written comments received, the Commission, in CLI-87-09, 26 NRC (September 21, 1987), concluded that the proper interpretation of Section 309(a) of the Comprehensive Anti-Apartheid Act of 1986 is one that gives effect to the plain language of the statute—that Congress intended to bar only uranium ore and uranium oxide and the bar did not extend to other forms of uranium. The Commission applied a three-part test commonly employed by the courts and the United States Customs Service in determining whether South African-origin uranium ore or uranium oxide that is transformed into uranium hexafluoride or into enriched uranium hexafluoride in other countries could be considered "substantially transformed" products. The Commission found that uranium hexafluoride and enriched uranium hexafluoride are substantially transformed uranium products and thus are not barred from importation. Based on this interpretation of the Anti-Apartheid Act, the Commission directed the staff to act on four pending import license applications.

In CLI-87-10, 26 NRC (September 21, 1987), the Commission directed the NRC staff to review the existing licenses which permitted imports of South African-origin uranium and to issue immediately effective orders to revoke, suspend, or modify those licenses so that they were consistent with the Commission's interpretation of the Anti-Apartheid Act set forth in CLI-87-9.

JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1987 is summarized below.

Pending Cases

Florida Power & Light, et al. v. NRC, No. 86-1512 (D.C. Cir.).

Wisconsin Electric Power Company, et al. v. NRC, No. 86-1567 (D.C. Cir.).

Arkansas Power & Light, et al. v. NRC, No. 86-1571 (D.C. Cir.).

In these consolidated lawsuits, over 30 utilities have challenged the NRC's Part 171 license fee rule. Promulgated to implement that portion of the Consolidated Omnibus Budget Reconciliation Act of 1985 which requires the agency to collect through annual charges approximately 33 percent of its budget, the rule charges each power reactor licensee a little less than $1,000,000 for fiscal year 1987.

The utilities argue that the NRC has misread the statute: that the 33 percent was intended as a ceiling only; that the generic costs which the NRC included in the cost basis of the rule are not, in the words of the statute, "reasonably related" to the agency's costs in providing "regulatory services" to specific licensees; and that all licensees, not just power reactor licensees, should be charged fees under the statute. The utilities also argue that if the court rules that the agency has read the statute correctly, then the court should alternatively rule that the statute is an unconstitutional delegation by Congress of its power to tax. Finally, the utilities argue that the NRC failed to provide an adequate opportunity for comment on the proposed rule and failed to articulate the factual basis for the final rule in detail sufficient to permit meaningful appellate review.

This litigation is one of the first judicial tests of the recent flood of "user fee" statutes aimed at reducing the

Federal deficit. The court had not yet rendered its opinion at the close of the report period.

Martin v. NRC, (3d Cir. Nos. 85-3444 and 87-3190).
Limerick Ecology Action, Inc. v. NRC (3d Cir. Nos. 85-3431 and 86-3314).
Anthony v. NRC (3d Cir. No. 85-3606).
Limerick Ecology Action, Inc. (3d Cir. No. 87-3508).
Martin v. NRC (3d Cir. No. 87-3565).
These seven consolidated cases challenge various orders issued by the NRC in the completed Limerick operating license proceeding. A briefing schedule has been established for all challenges to the Limerick (Pa.) license, and the case should be fully briefed by the end of the year.

Dellums v. United States (D.C. Cir. No. 87-1531).
A number of Congressmen and several other institutions and individuals have sued the Commission to overturn a pair of September 21, 1987 orders which allow the importation of uranium hexafluoride made from South African uranium ore and uranium oxide. The petitioners sought an emergency stay of the Commission's orders claiming that a scheduled October 15, 1987 shipment of uranium hexafluoride by Advanced Nuclear Fuels will cause them irreparable harm.

Public Citizen v. NRC (D.C. Cir. No. 87-1050).
On November 20, 1986, Public Citizen and six other public interest organizations sued the NRC in U.S. District Court alleging that the NRC has failed to implement section 306 of the Nuclear Waste Policy Act of 1982 ("NWPA"). The central issue is whether the NRC's March 20, 1985 policy statement provides the "regulatory guidance" and "requirements" which section 306 of the NWPA mandates the NRC to promulgate for the training and qualification of nuclear plant personnel. Public Citizen contends that the policy statement, which endorses INPO's training accreditation program, and the NRC's refraining from rulemaking for two years from the date of the policy statement, does not constitute compliance with section 306. U.S. District Judge Charles Richey transferred the case to the D.C. Circuit on January 14, 1987. Briefs have been filed and oral argument in the case has been scheduled for November 19, 1987.

Quivira Mining Company, et al. v. NRC (10th Cir. No. 85-2853).
Environmental Defense Fund v. NRC (10th Cir. No. 86-1235).

The above actions challenge the Commission's amendments to its uranium mill tailings regulations (50 FR 41852 (October 16, 1985)). The amendments conform NRC requirements to standards set by the Environmental Protection Agency ("EPA"). The industry petitioners assert that the amended regulations are an abuse of agency discretion and so must be vacated. The environmental petitioners assert that the NRC failed to fully conform its regulations to EPA's standards, particularly with respect to the standards for ground water protection. The court denied a motion to consolidate these cases and established separate briefing schedules. NRC's brief in *Quivira* was filed in July 1987 and in *Environmental Defense Fund* in August 1987.

Significant Judicial Decisions

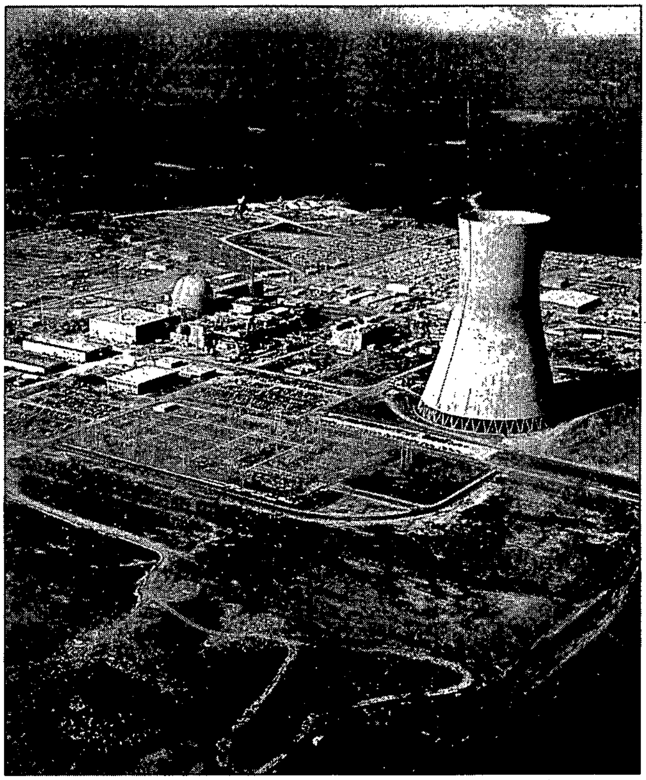
Critical Mass Energy Project v. NRC, No. 86-5647 (D.C. Cir. September 29, 1987), remanded in part, 644 F.Supp. 344 (D.C.D.C. 1986).
In this case the plaintiff sought access to documents generated by the Institute of Nuclear Power Operations (INPO), provided to the agency under a confidentiality agreement, and withheld from the plaintiff under, *inter alia,* Exemption 4 (proprietary information) of the Freedom of Information Act.

The District Court agreed with the NRC's argument that the reports were covered by FOIA Exemption 4 in that they were "commercial" and "confidential" documents. After losing the case in the District Court on all counts, plaintiff appealed to the D.C. Circuit. In an opinion rendered September 29, 1987, the D.C. Circuit upheld the agency's view of the law, which greatly extended the definition of "commercial" and "confidential" as used in the context of Exemption 4, but remanded the case to the District Court for better development of a factual record.

The D.C. Circuit concluded that the INPO documents were "commercial" and then set out a two-part legal test to determine whether they could also be considered "confidential" in a case such as this where competitive injury is not at issue. To be considered confidential, the information at issue "would customarily not be released to the public by the person from whom it was obtained." (The INPO documents met this test.) Secondly, the agency must then make a showing that release of the information would "impair [its] ability to obtain necessary information in the future." The court found it as unable to make a determination on the confidentiality of the records at issue because it found the factual record on these matters to be inconclusive.

Then, breaking new FOIA ground, the court held that, as an alternative to impairment of the NRC's ability to obtain the documents, impairment of other NRC interests could also justify withholding under Exemption 4. In particular, the court held that impairment of agency effectiveness and efficiency, if demonstrated by the record, could justify such a withholding.

Eddleman v. NRC, 825 F.2d 46 (4th Cir. 1987).
On January 20, 1987, Wells Eddleman and others filed a petition to review the Commission's January 12, 1987 licensing of the Shearon Harris Nuclear Power Plant. On August 10, 1987, in a unanimous decision, the United States Court of Appeals for the Fourth Circuit affirmed the agency's grant of a full power license for the facility. By its



In early January 1987, the NRC authorized the issuance of a full-power license for the Shearon Harris nuclear power plant (shown here during construction), near New Hill, N.C. In August 1987, the United States Court of Appeals for the Fourth Circuit upheld the Commission decision and rejected an intervenor petition regarding emergency planning for the facility.

decision the court reaffirmed the validity of the agency's immediate effectiveness procedures and the NRC's determination to grant an exemption from the requirements that a full-scale emergency planning exercise be held one year prior to licensing.

First, the court rejected petitioners argument that the procedural protections appropriate to a full adjudicatory hearing attach to the Commission's immediate effectiveness review. Citing Oystershell Alliance v. NRC, 800 F.2d 1201 (D.C. Cir. 1986) (per curiam), the court declared that the petitioners had no rights to notice and a hearing prior to the licensing decision.

The court also found that petitioners did not have a right, under Section 189a of the Atomic Energy Act, to a hearing on a Section 2.206 petition considered and rejected by the Commission as part of its immediate effectiveness review prior to licensing.

Finally, the court upheld the agency's action in granting, without a hearing, the applicant's request for an exemption from the scheduling aspects of the now superseded requirement that a full-scale emergency planning exercise be held one year prior to full power licensing.

Commonwealth Edison Co. v. NRC, 8 F.2d (7th Cir. 1987).

On November 1, 1985, the Commonwealth Edison Company sued for a declaratory judgment that the NRC's application of its current license fee ceilings to license review work done before the effective date of the current ceilings was illegal under both the Independent Offices Appropriations Act (IOAA) and the due process clause of Amendment V to the United States Constitution.

In its petition, Edison argued that the agency made an impermissibly retroactive application of the 1984 version of Part 170 fees for inspection work done at Edison's Byron (Ill.) and Braidwood (Ill.) plants. The 1984 version raised the ceilings in effect since 1978 on charges for NRC review of applications for operating licenses.

On May 15, 1987, the Court of Appeals for the Seventh Circuit, in part dismissed as untimely and in part denied Edison's petition for review of a bill the NRC had sent Edison under the Part 170 license fee rule. (819 F.2d 750, 7th Cir. 1987.) Subsequently, Edison requested rehearing, which the panel granted. On September 1, 1987, the panel issued its revised opinion on rehearing. In the rehearing opinion the court reversed its earlier jurisdictional holding. Relying on the well established presumption in favor of judicial review of agency action, the court held that it had jurisdiction to review the merits of an NRC rule challenge made within 60 days of the application of the rule to the aggrieved party.

The court rejected, on the merits, however, Edison's claim of illegal retroactivity. The court stated that, although the license fees were computed based on work done by the NRC under the 1978 fee schedule, those fees did not "vest" until the license process application was complete. Thus, the court held that the 1984 fee schedule properly controlled the amount of the license application fee which Edison owed for its Byron and Braidwood plants. Additionally, the court adhered to the NRC's view that it could charge penalties and interest on the unpaid fee due it from Commonwealth Edison.

Union of Concerned Scientists v. NRC, 824 F.2d 108 (D.C. Cir. 1987).

On November 18, 1985, the Union of Concerned Scientists (UCS), and others filed suit against the Commission, seeking to have the court declare the Commission's "backfitting rule," 10 C.F.R. §50.109, null and void, and direct the Commission to issue a rule that would conform to the requirements of the Atomic Energy Act.

On August 4, 1987, the Court of Appeals for the District of Columbia Circuit struck down the Commission's 1985 backfit rule. The court held that the rule was invalid because it permitted the NRC to take costs into account in establishing the "adequate protection" standard. The court read certain language in the rule's Statement of Considerations as suggesting that the Commission intended to use the rule's cost-benefit approach not only on proposals for safety improvements beyond the basic "adequate protection" standard, but also for establishing the "adequate protection" standard itself. The court ruled that, while the NRC need not set the "adequate protection" standard at zero risk levels, that standard must be established on health and safety grounds alone, without reference to cost.

The court did not take issue, however, with the principle—central to the Commission's backfit rule—that where a plant already meets the "adequate protection" standard, the NRC can take the cost of safety improvements into account in deciding whether the plant should be made even safer. The court thus rejected the Union of Concerned Scientists' principal substantive objection to the rule: the claim that the Atomic Energy Act bars the NRC from ever taking cost into account in making decisions on backfitting.

Ohio Citizens for Responsible Energy v. NRC, 803 F.2d 258 (6th Cir. 1986).
Ohio v. NRC, 814 F.2d 258 (6th Cir. 1987).
These actions attacked the Commission's licensing of the Perry plant. The first case dismissed for lack of jurisdiction a pre-licensing lawsuit brought by the Ohio Citizens for Responsible Energy ("OCRE"). The second case upheld the NRC's licensing of Perry on the merits in the face of separate challenges by OCRE and the State of Ohio.

OCRE's claims originated in a motion to reopen the record which it filed in the Perry operating license proceeding in February 1986. The motion raised the issue of the implications for the Perry plant of the earthquake which had occurred 10 miles from the site in January 1986. The NRC denied the motion to reopen, and OCRE appealed. Ohio protested the NRC's refusal to admit it as a party to the proceeding to allow it to contest emergency planning concerns about the Perry plant.

On March 17, 1987, the Sixth Circuit upheld the NRC on the merits (Ohio v. NRC). The court found no abuse of discretion on the part of the NRC in its refusal to reopen the record on OCRE's seismic contentions. With regard to Ohio's emergency planning claims, the court stated that the NRC had neither ignored Ohio's contentions nor acted unreasonably in denying the State leave to intervene as a party to the proceeding. The court accordingly found no basis on which to overturn the NRC's grant of a full-power operating license to the Perry plant.

Management and Administrative Services

Chapter 11

Consolidation of NRC Headquarters

A major milestone was reached during fiscal year 1987 pursuant to the NRC's long sought objective of consolidating all of the NRC's Washington headquarters operations at a single location. Occupancy of the new 18-storey One White Flint North building began in mid-December, 1987 and it is expected that, by early spring of 1988, a total of 1,400 agency employees—or 60 percent of all headquarters staff—will be housed together in a single modern facility. See Chapter 1 for a detailed description of the NRC consolidation.

Changes Within the Commission

The single change on the Commission itself during the report period was the appointment, in August 1987, of Commissioner Kenneth C. Rogers, filling a vacancy created when former Commissioner James K. Asselstine completed his five-year term. Other appointments at the senior-staff level are reported in Chapter 1.

PERSONNEL MANAGEMENT

NRC Staff Ceilings

In fiscal year 1987, the NRC expended a total of 3,376 staff-years in carrying out its mission. This expenditure was 0.2 percent above the budgeted ceiling of 3,369 staff years. Major categories of employees who count against staff year expenditures include: permanent full-time staff, part-time and temporary workers, and consultants.

During fiscal years 1988 and 1989, the NRC ceiling will be 3,250 and 3,180 staff-years, respectively. These numbers reflect a continuing reduction in ceiling and will require limitations on hiring during fiscal year 1988. The Office of Personnel developed a staffing strategy for the NRC overall, as well as for each Office and Region, in order to achieve these reductions.

Recruitment

In fiscal year 1987, the NRC hired 237 and lost 311 permanent full-time employees, for an attrition rate of 9.6 percent per year. The agency's recruitment program included visits to numerous college campuses (including campus "job fairs") and participation in approximately 15 other kinds of job fairs during the year. A total of eight entry-level scientists and engineers were hired and four cooperative education students were converted to permanent employment after graduation.

Incentive Awards

NRC managers recognized high quality work performed by staff members during 1987 with 391 special achievement awards, 448 high quality performance increases, 84 certificates of appreciation and 75 SES bonuses.

Labor Relations

NRC Management and the National Treasury Employees Union agreed to implement an Interim Collective Bargaining Agreement in fiscal year 1987 covering all Articles except "Performance Appraisal," "Reduction-in-Force" and "Salary." Those three articles have been referred to the Federal Labor Relations Authority for a determination as to negotiability. The determination is still pending on "Performance Appraisal" and "Reduction-in-Force," and the agency is appealing the determination that the article on "Salary" is negotiable.

Training and Development

The NRC provides over 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 directed at improving performance and maintaining up-to-date technical proficiency.

In 1987, the NRC continued its emphasis on upward mobility programs and the use of Individual Development Plans to help all employees clarify their career goals and improve their job skills and performance. A Certified Professional Secretary Program, 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. These one-year, part-time programs are designed to assist program participants to acquire or enhance their supervisory and managerial competences. They provided NRC employees with opportunities to complete individual and group activities and developmental work assignments. Other NRC employees participated in job-rotation assignments to enable them to broaden their work experiences and gain a wider perception of the NRC's mission.

The NRC offers extensive supervisor and management development programs for current staff members. A pre-supervisory orientation program is offered to assist employees in the pursuit of career goals leading to supervisory positions. 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 and practices. Over 80 employees were trained in these courses during fiscal year 1987.

Executive Leadership Development

During fiscal year 1987, the NRC initiated an Executive Leadership Development/Management Succession Planning Program. The essential elements of the program—rotational assignments and mechanisms to continually review and evaluate NRC's management depth and back-up—were discussed at the Senior Manager's Conference in the fall. Comments and suggestions obtained during the conference are being incorporated into the final plan. When fully implemented, the program will ensure continuity of executive expertise for the future.

Alcohol and Drug Abuse Program

Since 1979, the NRC Alcohol and Drug Abuse (ADA) Program staff has conducted annual awareness training and periodic supervisors' training for all headquarters and regional personnel. The theme of this year's program was "Back to Basics," which included two films, "What Everyone Should Know About Alcohol" and "The Medical Aspects of Mind-Altering Drugs," as well as an overview of the diseases of alcoholism and drug addiction. In addition to the basic training, Supervisors and managers received instructions in procedures for dealing with problem employees. ADA Program staff individually counseled approximately 50 employees, family members, and supervisors on a variety of alcohol and drug issues, as well as other problems affecting employees' ability to function on the job. The ADA staff also provided input to formulation of the Drug Screening Policy Statement, with respect to treatment and rehabilitation procedures. And a lending library has been established, making books, films and audio tapes available to employees and their families.

NRC INFORMATION RESOURCES

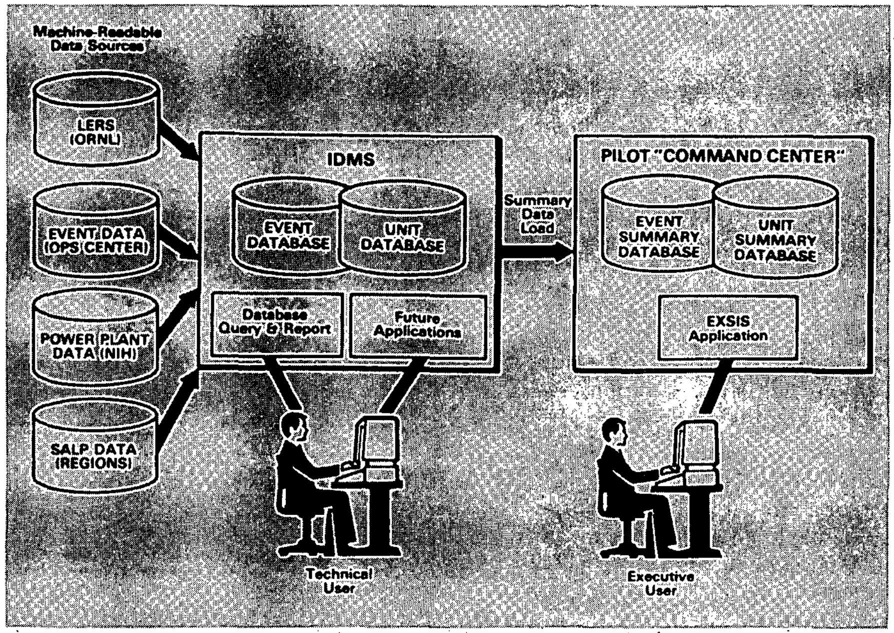
The NRC has a long history as a pioneer among agencies of the U.S. Government in applying state-of-the-art information technology to the fulfillment of its mission. The NRC elements responsible for this application were, during fiscal year 1987, integrated into the new Office of Administration and Resources Management, or ARM (see Chapter 1), the better to wed information resources to the delivery of administrative services agency-wide. The explicit goals of the agency's information systems are to provide NRC management with the information needed to manage agency programs; to ensure that all data gathered by, or contained in, automated systems is consistent, timely and accurate; to provide the capability to process and report information efficiently for the most effective results; and to provide the strategic approaches, equipment and software, as well as the organizational structures, to facilitate achievement of these goals.

Safety Information Network (SINET)

The NRC's information resource planning is based on the principle that comprehensive, reliable, and accessible information is crucial to carrying out the agency's mission. Under the NRC's approach, responsibility for both the availability and integrity of data rests with the NRC organizational units responsible for the collection and validation of the data. Data-users are to have ready access to the data they need with a minimum of technical knowledge or required training. Data are to be managed in a network of subject-oriented data bases in an integrated hardware/software environment, linked by the most current telecommunications technology. This network was named the Safety Information Network (SINET) in fiscal year 1987, in order to stress primary application in the area of safety-related data bases.

The purpose behind the SINET initiative is to collect health and safety information related to NRC licensees and their operations into a centralized data base, and to provide the tools that the NRC staff, both headquarters and regional, will need to obtain instant access to the data, as well as to analyze and display the information in the most relevant, usable mode. The ultimate objective is to assure that the NRC performs its basic mission—protecting public health and safety by assuring adequate safety in civilian nuclear operations—in the most informed, coordinated, efficient, and effective way.

Over a period of two years, the information resources staff set out to identify the data requirements of each office in



This is an overview of how SINET is expected to serve both the technical and executive-level user. The initial machine-readable sources of data for SINET (at left) include the Licensee Event Report (LER) data base from Oak Ridge National Laboratory (ORNL), Systematic Assessment of Licensee Performance (SALP) data stored in BASE files, Event Notification data from the NRC Operations Center computer, and power plant data currently stored at the National Institutes of Health. All of these data files are converted, restructured, and loaded into SINET's "UNIT" and "EVENT" data bases.

the NRC, to analyze existing computer systems and their data content, to define the data needed to adequately support the conduct of NRC business, and to propose an approach to converting existing systems, in their widely dispersed computers, into a single computer system, employing state-of-the-art software tools. Following the modelling phase, a pilot data base was designed and installed, comprising data related to both "units" and "events." This experience demonstrated the usefulness of shared data bases, which is the conceptual essence of the safety information network. Agency management endorsed the concept of centralized data bases by ratifying the award of three-year Systems Development contract in December 1986. Following completion of the pilot program, the information resources management (IRM) staff developed a Request for Proposal for the development of the centralized data bases. A fourth-generation software product was selected and installed for NRC. As a first application, the pilot data base encompassing units and events was installed. Identification of other health and safety information which belongs in the safety information network continues.

When complete, the SINET data base will contain information about the "data entities" shown in Table 1. Each of these entities is a person, place, thing, concept, or event about which the NRC wishes to store data.

During the report period, the first phase of SINET was implemented, using Cullinet's data base management software, IDMS/R. As a result, the items designated * in Table 1 are available for access by NRC staff by the spring of 1988.

Two major information areas, SITE/UNIT and EVENT, contain the following kinds of data:

The SITE/UNIT information includes:

- Descriptive information about sites where NRC licensed facilities are located.
- Basic descriptive and design data about commercial nuclear power plant units.
- Plant performance data (SALP and Performance Indicators).
- Daily status data collected by the NRC Operations Center.
- Monthly operating and outage data.
- Names of key NRC employees associated with each unit, such as the Project Manager and Resident Inspectors.

The EVENT information includes:

- Data associated with reportable events called in to the NRC Operations Center.
- Data submitted in Licensee Event Reports.

Plans for phase II of SINET development include the items designated ** in Table I. As indicated there, the EVENT portion of the data base will be expanded to include:

- Data reported in Preliminary Notifications.
- Data on events found in Daily Reports from the Regions and Headquarters.

Table 1. SINET Data Entities Grouped by Logical Data Base

ALLEGATION DATA BASE

Allegation
Investigation (Case)
Vendor

HARDWARE DATA BASE

**Component
Deficiency
Structure
Test

ENFORCEMENT DATA BASE

Deviation
Enforcement Action
Violation

EVENTS DATA BASE

*Event
Event Report
*LER
*Event Notification
**Preliminary Notification
**Daily Report
**Construction Deficiency Report
**Part 21 Report
**Subevent
Event Followup
Exposure
Release
Threat

VENDOR DATA BASE

Design
Topical Report

ISSUE-DATA BASE

**Backfit Issue
**Bulletin
**Generic Letter
**Generic/US Issue

Issue (General)

SITE/UNIT DATA BASE

**Facility (Non-Reactor)
*Site
*Unit
*Performance Indicator
*SALP Report
*Monthly Operating Report
*Outage
*Daily Status
**System

LICENSING ACTION DATA BASE

License Commitment
License Action
Open Item (Licensing)

INSPECTION DATA BASE

**Inspection
**Inspection Module
**Inspection Program
**Outstanding Item (Inspection)

LICENSE DATA BASE

License
License Applicant
License Application
Licensee
Licensee Plan
Tech Spec/License Condition

OPERATOR DATA BASE

Examination (Operator)
Reactor Operator

RESEARCH DATA BASE
Research Program

RADIOACTIVE MATERIALS DATA BASE

Fuel Assembly
Package
Radioactive Materials
(Accountability)
Shipment

* Phase I ** Phase II

- Data associated with system/component failures or actuations which contributed to an event.
- Data from "Part 21" reports.
- Data from construction deficiency reports.

Later in phase II, work will begin on the implementation of the inspection, issue, and hardware data, in that order. In addition to the design and loading of data into SINET, application software will be written to provide the staff with customized outputs and easy-to-use menu access to the SINET data. Training courses will be developed to train NRC users to query the SINET data base using their personal computers as terminals, prepare output reports of an ad hoc nature, and download data to microcomputers if desired.

To facilitate use of SINET by NRC upper management, ARM has acquired the "Command Center" executive software. This software, currently in use by many Fortune 500 companies, is designed to provide data in format suitable for high level managers. Using Command Center in conjunction with SINET, ARM has developed an Executive Safety Information System (EXSIS), which provides NRC executives with a summarized version of the information in SINET, in a form which they can readily access and use.

Safety Issues Management System (SIMS).

The Safety Issues Management System (SIMS) addresses the basic need of the NRC to effectively manage power reactor safety issues, i.e., identified safety-related problems or concerns, from their inception through to the implementation of corrective measures by licensees and verification thereof by the NRC. These concerns necessarily engage a number of different offices and components of the agency and call for a system with agency-wide scope and access.

In September 1984, the Director of the Regional Operations and Generic Requirements staff requested that the then Office of Resource Management (now part of ARM) develop an agency-wide, cradle-to-grave system for closely tracking the treatment of generic concerns associated with nuclear power reactors, with a view to expanding the new system later to include other NRC-licensed activities, such as the transportation and non-reactor uses of nuclear materials, nuclear waste disposal, and nuclear fuel fabrication.

The system became operational in May 1986, but it was clear from the outset that its effectiveness would depend directly upon the ability of all users to understand and report SIMS data. This would mean not only a thorough understanding of the data base but of reporting capabilities in the system. To this end, a training program was developed, in conjunction with the NRC Training Center, to provide the user-community with the needed tools to take full advantage of the information resource.

It was also clear from the start that certain existing intra-office systems devoted to issues management would have to be subsumed under or replaced by the agency-wide facility. Thus the Office of Nuclear Reactor Regulation (NRR) requested that the Office of Information Resources Management (IRM) undertake an upgrading of its generic issues tracking system and then link it to data available in SIMS. The then Office of Inspection and Enforcement requested that SIMS be expanded to provide the capacity to identify and to facilitate retrieval of documents needed to track implementation and inspection verification data, with respect to generic issues, on a plant-specific basis.

Over a period of 15 months, ARM analysts worked with representatives from other NRC elements and with the Executive Director for Operations (EDO) and the Deputy Executive Director for Regional Operations and Generic Requirements staff to draw up and define all system requirements. A detailed requirements document was approved by all parties in December 1985. By April of 1986, the Safety Issues Management System (SIMS) had been designed, developed, and made operational.

The initial loading of data into SIMS continued through the balance of 1986 and for most of 1987. Formal training in the use of the SIMS computer system was provided to over 100 staff in January and February of 1987. Training sessions were held for all Project Managers in September 1987; regional staff were provided SIMS training in their offices, in August and September of 1987. In all, about 325 NRC staff were introduced to SIMS in formal training sessions. As of the end of the report period, more than 750 generic concerns were logged into SIMS, along with almost 31,000 discrete plant-level items (see figures). NRC staff and management are now routinely updating and using SIMS information.

The overall purpose of SIMS is to provide NRC staff with an effective management information system and procedure which will assure the timely resolution of outstanding generic concerns, as well as the implementation of necessary changes in nuclear power reactor operations. The system gives the appropriate NRC personnel a single source for validated, reliable information on the status of each generic concern, at any time during the often complex process of analyzing, understanding, and resolving the problem. Relevant data are recorded in SIMS from the time an issue is identified and prioritized; through its technical resolution and to the stage, if such eventuates, when requirements are imposed on licensees; through the review, analysis, approval of licensees' proposed corrective actions; through actual implementation of requirements by the licensees; and finally to the verification by the NRC, as necessary, that the action agreed upon has been taken.

The scope of the system encompasses all generic concerns affecting nuclear power reactors, taking generic simply to mean that the concern of problem affects two or more plants. Once power reactor data are routinely available in SIMS, other NRC licensed activities such as transportation, materials, waste disposal, and fuel cycle will be added to the system.

The resolution of a generic issue may include the imposition of requirements on a particular set of licensees. The imposition may be accomplished through issuance of a rule, an order, a generic letter, a bulletin, or an immediate action letter. These "imposition vehicles" are recorded in SIMS as a sub-item under the originating generic issue record.

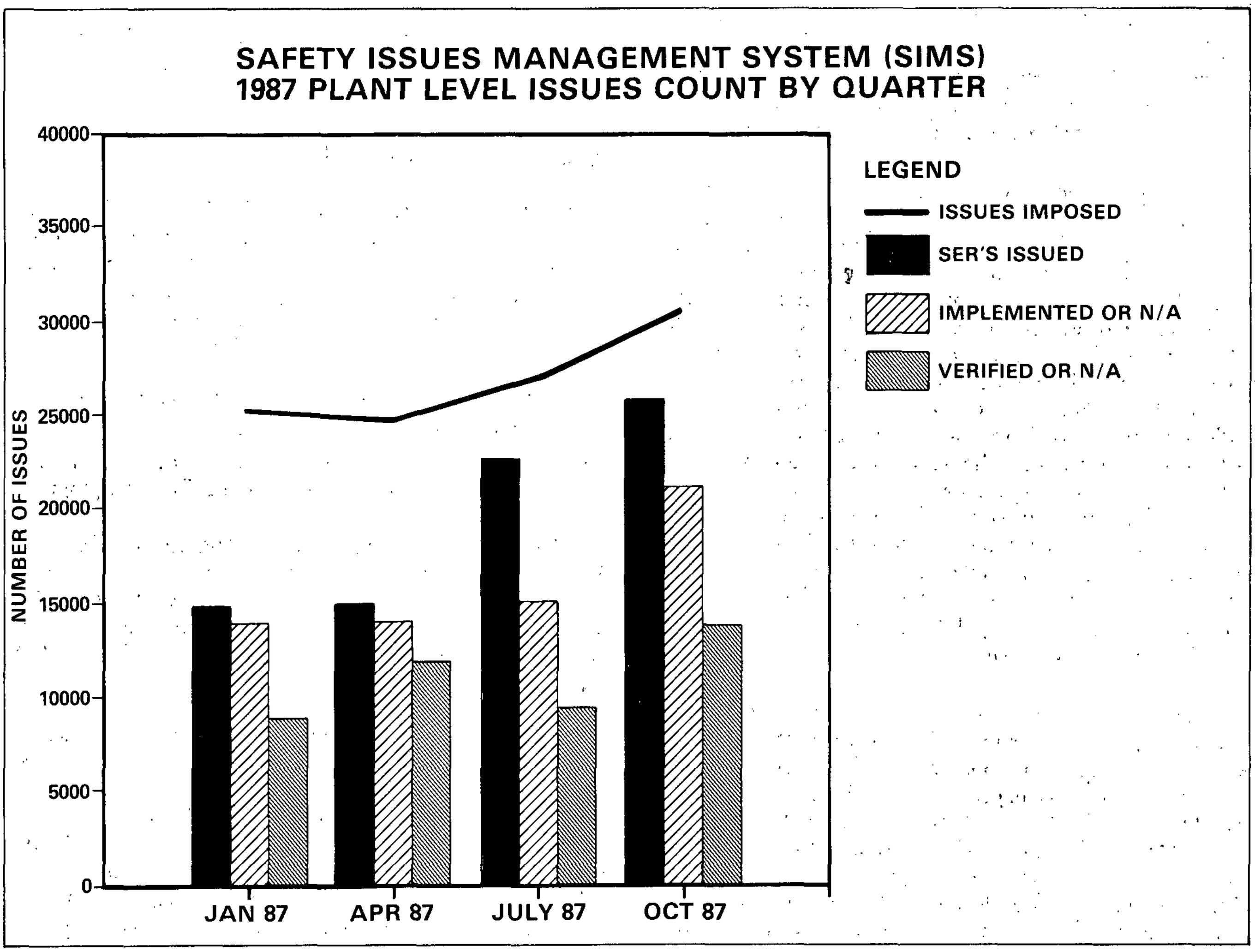
SIMS will also contain rules, orders, generic letters, bulletins, and immediate action letters which are employed independent of any generic issue. In such situations, there is no "generic issue" root record in SIMS; instead, the regulatory vehicle itself is the unique record tracked in SIMS.

All three kinds of generic concerns—(1) generic safety issues, (2) generic issues, or (3) independent regulatory vehicles—may also reference associated guidance or information documents. Such documents typically take the form of Regulatory Guides, NUREGs, Policy Statements, Technical Positions, Information Notices, Standard Review Plan, Inspection Program, Confirmatory Action Letters, and Enforcement Letters. SIMS accommodates reference to all these forms of guidance. In the case of changes to the Standard Review Plan or changes to the Inspection Program (routine or temporary), SIMS carries specific data about the proposed changes and expected date of completion.

In order to provide a comprehensive view of a given nuclear power reactor, SIMS also contains records for all plant specific licensing actions (i.e., licensee request for an amendment that is independent of any resolved generic concern). Both generic and plant-specific license actions can now be presented in one format for NRC management review and analysis.

SIMS was developed using fourth generation Data Base Management System software and operates out of the National Institutes of Health Federal Data Center, in Bethesda, Md. The system deals with data and processes at two levels: issue-level and plant-level. One issue generates up to 130 plant-level entries, needed to track that issue at each affected plant. Issue-level data and processes describe the issue, the resolution, the requirements, and the tracking of milestones, as well as the status associated with the issue. Issue-level data include:

- Basic Information—control number, title, description, priority, status, identifying organization, and sponsoring office.



- Cost Data—dollar estimates of the cost to the NRC to develop, impose, and assure compliance; also, dollar estimates to the public/industry to implement and comply.
- Benefits Data—net changes in public exposure, occupational exposure and core melt frequency.
- Technical Resolution Data—a description of the proposed solution and related generic requirements that are developed.
- Requirements Review and Approval—dates and status for tracking regulatory forms as they are reviewed by CRGR, ACRS, EDO, and the Commission, issued for public comment and published.

Plant-level data and processes are used to identify and track milestones, along with the status of the issue as it is imposed, implemented, and verified at each affected nuclear power plant. Plant-level data include:

- NRC Imposition—date of imposition, date of Safety Evaluation Report, and status codes.
- Licensee Implementation—date and status of license implementation completion.
- Verification—date, status, and inspection report number.
- Facility Data—pertinent data regarding the reactor, i.e., docket number, facility name, NRR Project Manager, Resident Inspector, OL issue/expire date, etc. 4

SIMS is designed to be "user friendly." It is an on-line, menu-driven system, readily accessible from any NRC computer terminal or personal computer with data communications capability. Once the user is logged onto SIMS, all system functions are performed through menu selection screens. Reports are printed at any of NRC's remote high-speed printers. Small reports of a few pages can be directed to a local personal computer for printing if desired. Reports can also be down-loaded to personal computers and saved as files for further local processing. Getting information from SIMS is easy. A user may choose to execute standard reports and receive a hard-copy printout of the data or choose among the standard queries which produce results immediately on the screen. More experienced SIMS users can create and format their own reports or queries via the data manipulation tool called REPORTER. SIMS also has an English language query capability, called ENGLISH, which accepts commands in normal English.

Other major information systems in the area of document control and retrieval include the Document Control System, the Public Document Room Bibliographic and Retrieval System, the Waste Management Transitional Licensing Support System (TLSS), the Congressional Correspondence Retrieval System, and the ASLBP (Atomic Safety and Licensing Board Panel) Pilot Project. A signal achievement under TLSS was the recent development of a system allowing the user to search millions of words of text and retrieve the original image of a document, including graphics, a capability made possible because of encoding by optical character-reader.

Other noteworthy initiatives include a demonstration project exploring uses of optical disk technology for gathering data inside an operating nuclear power plant and several telecommunications projects involving both intra-agency and agency-power plant communication.

OFFICE OF INSPECTOR AND AUDITOR

The mission of the NRC's Office of Inspector and Auditor (OIA) is to assure effectiveness, efficiency, and integrity in all NRC operations. In fiscal year 1987, OIA issued 18 audit reports, containing 78 recommendations, and 21 follow-up audit reports intended to improve the operations of various NRC programs and activities. OIA also issued 36 investigative reports in response to allegations concerning the integrity of NRC operations and employees. Of the investigative matters addressed by OIA during the report period, two were referred to the Department of Justice for consideration and possible prosecution. Some of the OIA reports issued during fiscal year 1987 are summarized below:

Quality Assurance at TVA

As part of an overall review of NRC's activities with respect to operations of the Tennessee Valley Authority (TVA), the Commission requested that OIA review issues concerning quality assurance (QA) at TVA nuclear power plants. The objectives of the review were to determine whether:

(1) The NRC Region II Office (Atlanta) showed favoritism in its treatment of TVA.

(2) There was a regulatory breakdown that contributed to the problems and failure of the TVA QA program.

OIA found no evidence to suggest that Region II showed favoritism to TVA in regulating TVA's QA program. Furthermore, the review did not disclose a regulatory breakdown. Region II did recognize QA problems at TVA and was attempting to effect improvements in TVA's QA operations.

The review found that NRC Headquarters and Region II were not effective in obtaining significant improvement in the operation of the TVA QA program. The reasons for this are complex and involve issues related to NRC's review and approval of licensee QA programs in general. OIA believes the very nature and history of the TVA organization also contributed to NRC's inability to bring about needed changes. OIA's report was issued in December 1986.

Closing Out Expired Contracts

OIA's review of NRC's procedures for closing out expired contracts revealed that the detailed close-out procedures outlined in the Federal Acquisition Regulation were not being initiated in a timely manner. As a result, OIA concluded that NRC needed to place more emphasis on closing out contracts. OIA's January 1987 report contained five recommendations for improving the contract close-out process and freeing unexpended funds to be used to support other NRC needs.

Control of Government Transportation Requests

In a February 1987 audit report, OIA concluded that improved controls were needed for the issuance of Government Transportation Requests (GTRs) at the NRC Warehouse and in two Regional Offices. OIA made five recommendations designed to tighten the controls over GTRs and evaluate the number of GTRs needed in NRC Headquarters and each Region.

NRC's Procedures for Handling Freedom of Information Act Requests

In a March 1987 audit report, OIA concluded that, overall, NRC is meeting the intent of the Freedom of Information Act (FOIA) by ensuring that releasable information is made available for public disclosure. OIA did note that changes were needed in some administrative practices associated with responding to FOIA requests. OIA made seven recommendations which addressed the specific problems identified during the review.

Regulatory Effectiveness Reviews For Operating Reactors

This review was undertaken to determine how effectively and efficiently NRC is carrying out the Regulatory Effectiveness Review (RER) program, which evaluates the adequacy of physical security at licensed nuclear power plants. The audit report, issued in April 1987, disclosed that the RER program is beneficial to both NRC and licensees; however, changes are needed to ensure RER findings are being analyzed to meet the objectives of the program and to improve the way reviews are conducted. The final report contained ten recommendations to address the changes needed in the progam. These included the need for:

(1) Analyses to determine the cause of weaknesses identified through RERs and to assure that backfit issues are properly processed in accordance with NRC policies.

(2) Improvements in the training of RER team members, the communication of findings to licensees, and RER team members' participation in evaluating the effectiveness of actions taken by licensees.

Use of Technical Assistance

In a May 1987 audit report, OIA concluded that the Office of Inspection and Enforcement had strengthened controls over work performed by the Department of Energy's (DOE) national laboratories. The report also noted that additional improvements were still needed in the planning, management, and administration of technical assistance projects in areas such as documentation, project manager training, and accounting for NRC equipment held by DOE labs. OIA made 11 recommendations to improve the management of technical assistance projects in the above areas.

NRC's Contract Operations

In a July 1987 audit report, OIA concluded there was a need to improve NRC's procurement process to better facilitate the acquisition of goods and services at NRC and to more fully comply with applicable laws and regulations. OIA made four recommendations with the objectives of ensuring compliance with the Competition in Contracting Act of 1984, clarifying NRC's delegation of contracting authority provisions, and improving other aspects of the procurement process.

The Thimble Tube Incident At TVA's Sequoyah Facility

This audit was initiated in response to concerns raised by Congressman John Dingell regarding a possible regulatory breakdown by NRC in its handling of an incident at TVA's Sequoyah (Tenn.) facility involving the cleaning of the thimble tubes. OIA's July 1987 report identified problems in three areas:

(1) Programmatic issues that resulted from NRC's inspection and enforcement activities.

(2) Technical issues identified in TVA's Nuclear Safety Review Staff's (NSRS) report on the incident.

(3) Issues resulting from NRC's investigative activities.

OIA concluded that NRC's Region II did not perform a timely and adquate review of the thimble tube incident prior to becoming aware of the NSRS investigation of the incident from an article in *The Wall Street Journal*. OIA's report documented that the timeliness and quality of the enforcement actions taken against TVA as a result of this incident were adversely affected by deficiencies in the NRC enforcement staff's response to the incident. Finally, OIA concluded that the Office of Investigations did not perform a comprehensive investigation as to whether TVA management deliberately failed to adequately report the incident in a Licensee Event Report. OIA made 11 recommendations to correct the problems identified.

NRC LICENSE FEES

In fiscal year 1987, the Commission collected $131.9 million in fees. The Consolidated Omnibus Reconciliation Act (Public Law 99-272) required the Commission to assess and collect fees not to exceed $132.3 million, or 33 percent of its estimated fiscal year 1987 budgeted costs of $400.95 million. The Commission used two different approaches in collecting these fees. 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 routine and non-routine safety inspections. These fees are established under 10 CFR 170 of the Commission's regulations. In addition, Public Law 99-272 authorizes the NRC to assess annual fees to utilities licensed to operate nuclear power plants. These annual fees are established under 10 CFR 171 of Commission regulations. The annual fee assessed in fiscal year 1987 for each plant with an operating license was $838,000.

All license, inspection, and annual fees collected are sent to the Department of Treasury for deposit as miscellaneous receipts. Table 1 shows the totals in the two categories cited.

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 171), which became effective on October 20, 1986. Numerous utilities filed suit challenging the rule in the U.S. Court of Appeals for the District of Columbia. Oral arguments were heard by the court on September 21, 1987.

In other user-fee litigation, the U.S. Court of Appeals for the Seventh Circuit, on September 1, 1987, rejected the challenge of the Commonwealth Edison Co. to amendments to the 10 CFR 170 schedule which had become effective on June 20, 1984. Under the revised fee schedule, the Commission had raised the fee ceilings for operating license reviews. The utility challenged the amendments on the grounds that the fees were being raised retroactively. The court rejected the utility's argument and held that both the fees and penalty with interest charges levied on Commonwealth Edison were proper.

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 Law 95-507, amending the Small Business Investment Act of 1957. During fiscal year 1987:

- It was estimated that $53 million in total prime contracts would be awarded in fiscal year 1987 and that the total amount of all prime contracts with individual dollar values over $10,000 would be $50 million. The actual total prime contracts and actual dollar awards over $10,000 were $49,408,217 and $46,302,471, respectively.
- It was estimated that small business prime awards with dollar values over $10,000 would be $22,000,000, or 44 percent of the total estimate. The actual achievement for small business prime awards with dollar values over $10,000 was $22,264,963, or 48.09 percent of the dollars reflected in the item above.
- The NRC estimated that awards to 8(a) firms would be $9,000,000, or 16.98 percent in fiscal year 1987. Awards to 8(a) firms were actually $7,793,906, or 15.77 percent of the total dollar amount of all prime contracts regardless of dollar value.
- The goal for prime contract awards having a value of $10,000 or more to small disadvantaged business firms other than 8(a) was $1,000,000, or 2 percent. The actual achievement was $257,601, or 0.55 percent of the dollars reported in the first item above, using awards over $10,000 as the base.

Table 2. License Fee Collections—FY 1987

Fees	*Facilities Program*	*Materials Program*	*Total*
10 CFR 171	$86.2 million		$86.2 million
10 CFR 170	$42.3 million	$3.4 million	$45.7 million
TOTAL FEES	$128.5 million	$3.4 million	$131.9 million

- The estimate for prime contract awards to small business concerns owned and controlled by women was $1,300,000, or 2.45 percent. Awards to such firms were $1,359,764, or 2.75 percent of the total dollar amount of all prime contracts regardless of dollar value.
- The goal for subcontract awards to small business was $1,650,000, or 66 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $1,072,690, or 63.15 percent of total subcontracts awarded. The NRC's total subcontract dollar awards goal in fiscal year 1987 was $2,500,000.
- The goal for subcontract awards to small disadvantaged businesses was $60,000, or 2.4 percent. Subcontracting awards to small businesses was $104,006, or 6.12 percent of total subcontract dollars awarded.

During the year, 102 interviews were conducted with firms wanting to do business with the NRC, and 53 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 annual Minority Enterprise Development Week in October 1987, and the MEGA Market-place observance for women business owners in April 1987.

Civil Rights Program

The NRC Consolidated Equal Employment Opportunity (EEO) Program Plan was updated in order to continue to promote affirmative action in NRC employment practices.

The Commission was briefed in July 1986 and February 1987 concerning the status of NRC's EEO/Affirmative Action Plan goals, programs, and accomplishments.

The NRC conducted an active recruitment program during the report period, placing strong emphasis on colleges

The NRC Federal Women's Program sponsored the agency's annual observance of National Women's History Month in March 1987. More than 400 NRC people attended one event in which Dr. Dixy Lee Ray, former Chairman of the U.S. Atomic Energy Commission, was the keynote speaker.

with high quality engineering programs and a good minority and/or female representation. NRC representatives visited college campuses, several of them being predominately minority schools. These recruitment visits included participation in "job fairs" for minorities or women.

The agency continues its "Upward Mobility" efforts to provide developmental opportunities to lower-level employees. The program focuses on the selection of those employees who show the potential to function effectively in professional or para-professional positions with greater growth potential. During fiscal year 1985, five Upward Mobility positions were filled.

An analysis of the EEO accomplishment report, submitted annually by Office Directors and Regional Administrators to the Director, Office of Small and Disadvantaged Business Utilization and Civil Rights (OSDBU/CR), was provided to the Chairman and the Executive Director for Operations to apprise them of the performance of managers in achieving their assigned goals. The Director, OSDBU/CR, continues to function as a non-voting, ex-officio member of the SES Performance Review Board.

Federal Women's Program

During fiscal year 1987, efforts continued under the Federal Women's Program (FWP) to enhance career opportunities for women through self-development and to ensure equal employment opportunity (EEO). In the Regions and in Headquarters, many NRC women availed themselves of the counseling assistance of the FWP Manager (FWPM) in the preparation of applications for employment (SF-171), individual development plans, and EEO-related complaint resolution.

The FWPM and attorneys from the Office of the General Counsel provided a refresher briefing on prevention of sexual harassment to headquarters and regional employees. Over two-thirds of the agency's employees attended the briefings.

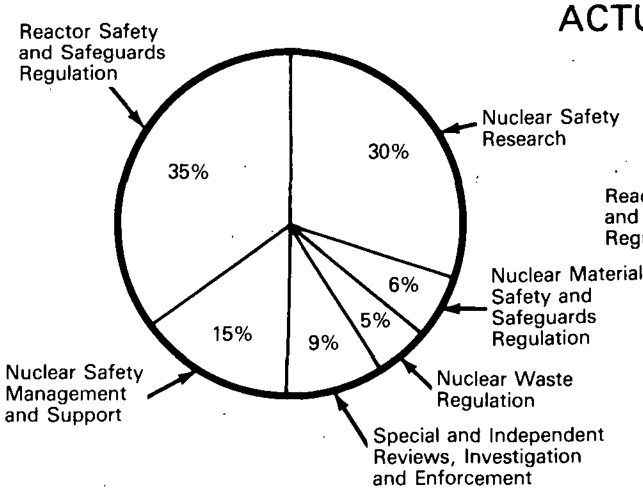
National Women's History Month was observed in March, and two Maryland State Delegates participated in the event; Dr. Dixy Lee Ray, former Chairman of the Atomic Energy Commission, was the keynote speaker. National Secretaries Week was celebrated with Managers and Secretaries attending a luncheon, and, on Women's Equality Day, Ms. Julia M. Walsh, Director of a multi-million dollar investment firm, was NRC's guest speaker.

Several members of the FWP Advisory Committee (FWPAC) attended the annual working conference of the FWPM and regional coordinators. The largest contingent of NRC women ever attended the annual Federally Employed Women's National Training Conference.

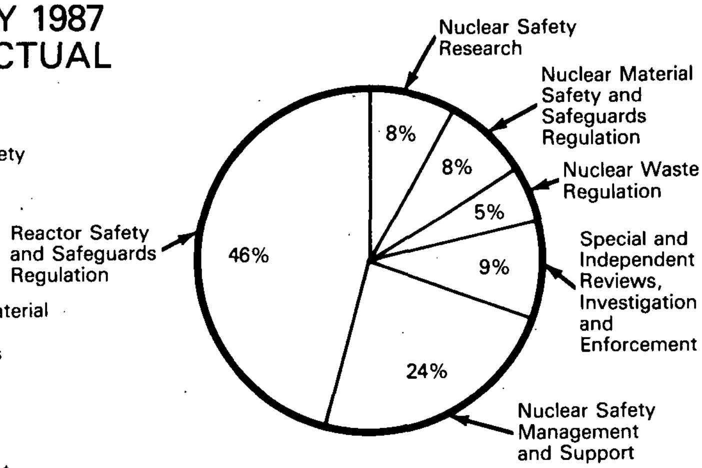
On-site consultation was provided to several Office Directors to assist them in pursuing their EEO initiatives for women. Exit interviews with women leaving the agency continued, in order to determine if their decision to leave was a result of sex discrimination. The FWPM continued monitoring and assessing selections for key positions and the impact on women of major changes in the agency—such as reorganizations, reductions-in-force, hiring freezes, and the consolidation of the NRC.

An FWP goal for fiscal year 1987 was to increase representation of women in the senior and SES positions. Some progress was made, as 14 women moved up from GG-13 positions, six women became GG-15, and one woman entered SES. Some NRC women also participated in the Women's Executive Leadership Program, the Mid-Level Executive Potential Program, and the rotational assignments program.

NRC RESOURCES
FY 1987
ACTUAL

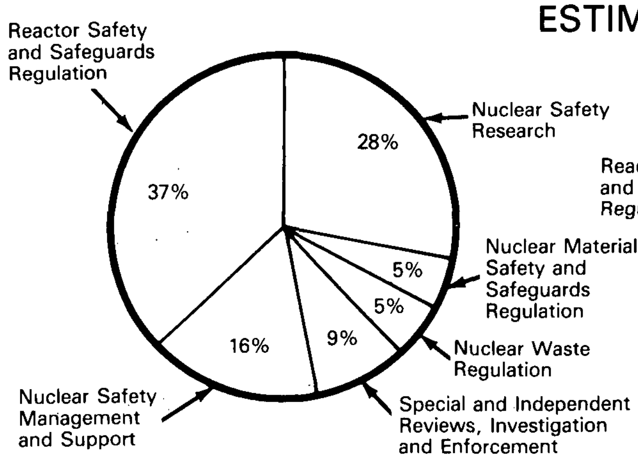


FUNDS 407 MILLION

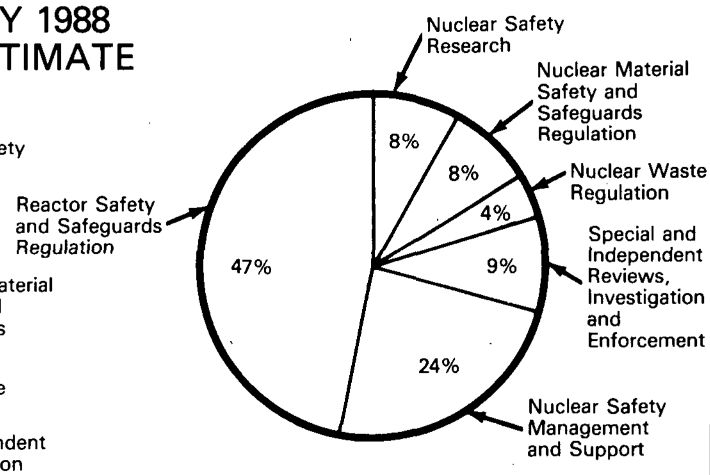


PERSONNEL 3376
(Full Time Equivalent)

NRC RESOURCES
FY 1988
ESTIMATE



FUNDS 401 MILLION



PERSONNEL 3250
(Full Time Equivalent)

Appendix 1

NRC Organization

(As of December 31, 1987)

COMMISSIONERS

Lando W. Zech, Jr., Chairman
Thomas M. Roberts
Frederick M. Bernthal
Kenneth M. Carr
Kenneth C. Rogers

The Commission Staff

General Counsel, William C. Parler
Office of Governmental and Public Affairs, Harold R. Denton, Director
Office of Inspector and Auditor, Sharon R. Connelly, Director
Secretary of the Commission, Samuel J. Chilk
Office of Investigations, Ben B. Hayes, Director

Other Offices

Advisory Committee on Reactor Safeguards, William Kerr, Chairman
Atomic Safety & Licensing Board Panel, B. Paul Cotter, Jr., Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, Victor Stello, Jr.
Deputy Executive Director for Operations (Acting), James M. Taylor
Deputy Executive Director for Regional Operations
and Generic Requirements, James M. Taylor
Assistant for Operations, Thomas A. Rehm

Program Offices

Office of Nuclear Reactor Regulation, Thomas E. Murley, Director
Office of Nuclear Material Safety and Safeguards, Hugh L. Thompson, Director
Office of Nuclear Regulatory Research, Eric S. Beckjord, Director
Office of Enforcement, James Lieberman, Director
Office of Special Projects, Stewart D. Ebneter, Director

Staff Offices

Office of Administration and Resources Management,
William G. McDonald, Director
Office for Analysis and Evaluation of Operational Data,
Edward Jordan, Director
Office of Personnel, Paul E. Bird, Director
Office of Small and Disadvantaged Business Utilization/Civil Rights,
William B. Kerr, Director
Office of Consolidation, John M. Montgomery, Director

Regional Offices

Region I—Philadelphia, Pa., William T. Russell, Regional Administrator
Region II—Atlanta, Ga., J. Nelson Grace, Regional Administrator
Region III—Chicago, Ill., A. Bert Davis, Regional Administrator
Region IV—Dallas, Tex., Robert D. Martin, Regional Administrator
Region V—San Francisco, Cal., John B. Martin, Regional Administrator

The NRC is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and the Nuclear Nonproliferation Act of 1978; and in accordance with the National Environmental Policy Act of 1969, as amended, and other applicable statutes. These responsibilities include protecting public health and safety, protecting the environment, protecting and safeguarding materials and plants in the interest of national security, and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience; and regulatory research. The Commission itself is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

The Executive Director for Operations directs and coordinates the Commission's operational and administrative activities among the program and support staff offices described below and also coordinates the development of policy options for Commission consideration. The EDO reports directly to the Chairman.

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. NRR 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 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. NMSS reviews and assesses safeguards 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 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 Office of Enforcement develops policies and programs for the enforcement of NRC requirements, manages major enforcement actions, and assesses the effectiveness and uniformity of regional enforcement actions.

The 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 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.

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 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 Inspector and Auditor investigates to ascertain the integrity of all NRC operations; investigates allegations of NRC employee misconduct, equal employment and civil rights complaints, and claims for personal property loss or damage; conducts the NRC's internal audit activities; and hears individual employee concerns regarding Commission activities, under the agency's "open door" policy. The Office develops policies governing the Commission's financial and management audit program and is the agency contact with the General Accounting Office on this function. The Office refers criminal matters to the Department of Justice and maintains liaison with law enforcement agencies.

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, GPA coordinates and licenses export-import activity. The Office also administers the agency's program of public information.

SUPPORT STAFF

The Office of Administration and Resources Management directs the agency's programs for preparation of the budget; the accounting and financial systems management, such as payroll and travel expenses; central administrative services, such as rules and records management, facilities and operations support and publications services; and management of centralized information resources, including computer and telecommunications services, document control systems, records management, and library facilities.

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 responsbible for the NRC incident response program and the technical training center, as well as the tracking of licensee performance indicators.

The Office of Small and Disadvantaged Business Utilization/Civil Rights develops and implements the NRC's program in accordance with the Small Business Act, as amended, insuring that appropriate consideration is given to labor surplus area firms and women-owned businesses. The Office develops and recommends NRC policy providing for equal employment opportunity and develops, monitors, and evaluates the affirmative action program to assure compliance with the policy. The Office also serves as contact with local and national public and private organizations with related interests.

The Office of Special Projects exists to ensure that licensed facilities with particularly complex regulatory problems are given comprehensive and timely attention and appropriately high-level direction by NRC. The mission of the Office is short-term.

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.

OTHER ORGANIZATIONAL ELEMENTS

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 Board Panel is a panel of lawyers and others with expertise in various technical fields from which three-member Licensing Boards are drawn to conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, amend, suspend or revoke NRC licenses.

The 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.

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 December 31, 1987, the members were:

CHAIRMAN: DR. WILLIAM KERR, Professor of Nuclear Engineering and Director of the Office of Energy Research, University of Michigan, Ann Arbor, Mich.
VICE-CHAIRMAN: DR. FORREST J. REMICK, Acting Vice President for Research and Graduate Studies and Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.
MR. JESSE C. EBERSOLE, Head Nuclear Engineer (retired), Division of Engineering Design, Tennessee Valley Authority, Knoxville, Tenn.
DR. HAROLD W. LEWIS, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.
DR. CARSON MARK, Division Leader (retired), Los Alamos Scientific Laboratory, Los Alamos, N.M.
MR. CARLYLE MICHELSON, Principal Nuclear Engineer (retired), Tennessee Valley Authority and Director (retired), Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.
DR. DADE W. MOELLER, Professor of Engineering in Environmental Health and Director, Office of Continuing Education, School of Public Health, Harvard University, Boston, Mass.
MR. GLENN A. REED, Plant Manager (retired), Pt. Beach Nuclear Power Plant, Wisconsin Electric Power Company, Two Rivers, Wis.
DR. PAUL G. SHEWMON, Professor and Chairman of Metallurgical Engineering Department, Ohio State University, Columbus, Ohio
DR. CHESTER P. SIESS, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.
DR. MARTIN J. STEINDLER, Director, Chemical Technology Division, Argonne National Laboratory, Park Forest, Ill.
MR. DAVID A. WARD, Chairman, Research Manager, Reactor Safety Research, E. I. du Pont de Nemours & Company, Savannah River Laboratory, Aiken, S.C.
MR. CHARLES J. WYLIE, Chief Engineer (retired), 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 Physicist, 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, 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 HUGH K. CLARK, Attorney (retired), E.I. duPont deNemours & Company, Kennedyville, Md.
JUDGE RICHARD F. COLE, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE GEORGE A. FERGUSON, Physicist, Howard University, 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, ASLBP Attorney, 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 Laboratories, 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, Sanitary Engineer, University of North Carolina, Chapel Hill, N.C.
JUDGE GUSTAVE A. LINENBERGER, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE LINDA W. LITTLE, Environmental Biologist, L.W. Little Associates, Raleigh, N.C.
JUDGE EMMETH A. LUEBKE, ASLBP Physicist, 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), Summerland, Fla.
JUDGE PETER A. MORRIS, ASLBP, Physicist, US Nuclear Regulatory Commission, Bethesda, Md.
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, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

PROFESSIONAL STAFF

CHARLES J. FITTI, Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELVA W. LEINS, Assistant Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DAVID L. PRESTEMON, Legal Counsel to the Panel, 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

An Atomic Safety and Licensing Appeal Board, established September 18, 1969, was delegated the authority to perform the review function that would otherwise be performed by the Atomic Energy Commission in proceedings on applications for licenses or authorizations in which the Commission had a direct financial interest, and in such other licensing proceedings as the Commission might specify.

In view of the increase in the number of proceedings subject to administrative appellate review, the Atomic Safety and Licensing Appeal Panel was established on October 25, 1972, from whose membership three-member Appeal Boards could be designated for each proceeding in which the Commission had delegated its authority to an Appeal Board. At the same time, the Commission modified its rules to delegate authority to Appeal Boards in all proceedings involving the licensing of production and utilization facilities (for example, power reactors).

Pursuant to subsection 201(g)(1) of the Energy Reorganization Act of 1974, the functions performed by Appeal Boards were specifically transferred to the Nuclear Regulatory Commission. The Commission appoints members to the Appeal Panel, and the Chairman of the panel designates a three-member Appeal Board for each proceeding. Recently, the Commission expanded the Appeal Board's review authority to cover, as well, a variety of other formal adjudicatory proceedings including those resulting from orders to show cause and assessing civil penalties. The Commission retains review authority over decisions and actions of Appeal Boards. The Appeal Panel, on October 1, 1987, was composed of the following persons:

FULL-TIME MEMBERS:

ALAN S. ROSENTHAL, Appeal Panel Chairman, U.S. Nuclear Regulatory Commission, Bethesda, Md.
GARY J. EDLES, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.*
CHRISTINE N. KOHL, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS S. MOORE, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
HOWARD A. WILBER, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
*Resigned as of October 4, 1987.

PART-TIME MEMBER:

DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

PROFESSIONAL STAFF:

JOHN CHO, Counsel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS G. SCARBROUGH, Technical Advisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.

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 renders expert opinions regarding 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, 1987, the members were:

RICHARD E. CUNNINGHAM, Chairman, ACMUI, Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Silver Spring, Md.
DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.
DR. SALLY J. DE NARDO, Director, Nuclear Hematology-Oncology, Department of Nuclear Medicine, University of California Davis Medical Center, Sacramento, Cal.
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. B. LEONARD HOLMAN, Chief, Clinical Nuclear Medicine, Department of Radiology, Brigham and Women's Hospital Boston, Mass.
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.
DR. DAVID H. WOODBURY, Director, Nuclear Medicine Section, Wayne County General Hospital, Westland, Mich.

Advisory Panel for the Decontamination of Three Mile Island Unit 2

The Advisory Committee for the Decontamination of Three Mile Island, Unit 2, was established in October 1980. Its purpose is to obtain input and views from the residents of the Three Mile Island area and afford Pennsylvania government officials an opportunity to participate in the Commission's decision-making process regarding cleanup plans for Three Mile Island Unit 2. The Panel consists of the following members representing agencies of the Commonwealth of Pennsylvania, local government authorities in the vicinity of the Three Mile Island facility, the scientific community and persons having their principal place of residence in the vicinity of the facility.

ARTHUR E. MORRIS, Panel Chairman, 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, Chairman, Dauphin County Board of Commissioners, Harrisburg, Pa.
GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, Univeristy Park, Pa.
JOEL ROTH, resident of Elizabethville, 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, 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) at 1717 H Street, N.W., Washington, D.C., for public inspection. Other PDRs on NRC premises include the rooms at the Willste Building, 7915 Eastern Avenue, Silver Spring, Md., and in the five Regional Offices (the latter 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. (N.B. Updated listings of local PDRs may be obtained by writing to the Local Public Document Room Branch, Division of Rules and Records, 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 Power Station
  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
  Jospeph M. Farley Nuclear Plant

- Ms. Nancy Stover
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

ARIZONA

- Ms. Fern Eckhardt, Documents Librarian
  Business and Science Division
  Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde Nuclear Station

ARKANSAS

- Mrs. Delores Pollard, Serials Librarian
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One

CALIFORNIA

- Ms. Margaret J. Nystrom
  Documents Librarian
  Eureka-Humboldt County Library
  636 F Street
  Eureka, Cal. 95501
  Humboldt Bay Power Plant

- Mr. Arthur Pond
  West Los Angeles Regional Library
  11360 Santa Monica Boulevard
  Los Angeles, Cal. 90025
  UCLA Training Reactor

- Ms. Marilee Cogswell
  Documents Librarian
  Sacramento Public Library
  828 I Street
  Sacramento, Cal. 95814
  Rancho Seco Nuclear Generating Plant

- Ms. Judy Horn, Department Head
  University of California
  General Library
  P.O. Box 19557
  Irvine, Cal. 92713
  San Onofre Nuclear Station

- 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 Power Plant

COLORADO

- Miss Shirley Soenksen
  Greeley Public Library
  City Complex Building
  919 7th Street
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Generating Station

CONNECTICUT

- Ms. Florence Munat, Reference Librarian
  Russell Library
  123 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Plant

- Ms. Carolyn Greene
  Waterford Public Library
  49 Rope Ferry Road
  Waterford, Conn. 06385
  Millstone Nuclear Power Station

FLORIDA

- Ms. Julie DeBusk
  Coastal Region Library
  8619 W. Crystal Street
  Crystal River, Fla. 32629
  Crystal River Nuclear Plant

- Ms. Jimmie Anne DeRoss, Librarian
  Charles S. Miley Learning Resources Ctr.
  Indian River Community College
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Plant

- Ms. Karlinne Wulf, Librarian
  Miami-Dade Public Library
  Homestead Branch
  700 North Homestead Blvd.
  Homestead, Fla. 33030
  Turkey Point Plant

- Ms. Esther B. Gonzalez, Librarian
  Urban and Regional Documents Collection Library
  Florida International University
  University Park
  Miami, Fla. 33199
  Turkey Point 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
  County Library
  412 4th Street
  Waynesboro, Ga. 30830
  Alvin W. Vogtle Nuclear Plant

ILLINOIS

- Mrs. Yvonne Cox, Assistant Librarian
  Byron Public Library District
  109 N. Franklin Street
  Byron, Ill. 61010
  Byron Station

- Ms. Cheryle Rae Nyberg
  Assistant Law Librarian
  University of Illinois Law Library
  504 East Pennsylvania Avenue
  Champaign, Ill. 61820
  Clinton Power Station

- Mrs. Betsy Taubert
  Vespasian Warner Public Library
  120 West Johnson Street
  Clinton, Ill. 61727
  Clinton Power Station

- Mr. Earl R. Shumaker, Head
  Government Publications Department
  Founder's Memorial Library
  Northern Illinois University
  DeKalb, Ill. 60115
  Byron Station

- Mrs. Nancy Gillfillian
  Library Director
  Dixon Public Library
  221 Hennepin Avenue
  Dixon, Ill. 61021
  Quad Cities Station
  Sheffield Low-level Waste Burial Site

- Ms. Deborah Trotter
  Reference Assistant
  Morris Public Library
  604 Liberty Street
  Morris, Ill. 60450
  Dresden Nuclear Power Station
  Morris Spent Fuel Storage Facility

- Ms. Evelyn Moyle, Documents Librarian
  Jacobs Memorial Library
  Illinois Valley Community College
  Rural Route 1
  Oglesby, Ill. 61348
  LaSalle County Station

- Ms. Marie Phillips, Supervisor
  Business, Science and Technology Dept.
  Rockford Public Library
  215 North Wyman Street
  Rockford, Ill. 61101
  Byron Station

- Ms. Nancy Barbour, Librarian
  Government Documents Collection
  Wilmington Public Library
  201 South Kankakee Street
  Wilmington, Ill. 60481
  Braidwood Station

- Mrs. Laura Hadjimitsos
  Reference Librarian
  Waukegan Public Library
  128 N. County Street
  Waukegan, Ill. 60085
  Zion Nuclear Power Station

- 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 Energy Center

KANSAS

- Ms. Nannette Martin, Documents Librarian
  Government Documents Division
  William Allan 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 Station

- Mr. Kenneth E. Owen, Head
  Louisiana Collection
  Earl K. Long Library
  University of New Orleans
  Lakefront Drive
  New Orleans, La. 70148
  Waterford Generating Station

MAINE

- Ms. Sue Cereste, Assistant Librarian
  Wiscasset Public Library
  High Street
  P.O. Box 367
  Wiscasset, Me. 04578
  Maine Yankee Atomic Power Plant

MARYLAND

- Ms. Mildred Ward, Library Assistant
  Calvert County Public Library
  Fourth Street
  P.O. Box 450
  Prince Frederick, Md. 20678
  Calvert Cliffs Nuclear Power Plant

MASSACHUSETTS

- Mrs. Marilyn O'Brien
  Library/Learning Resource Center
  Greenfield Community College
  One College Drive
  Greenfield, Mass. 01301
  Yankee Rowe Nuclear Power Station

- Ms. Grace E. Karbott, Reference Librarian
  Plymouth Public Library
  11 North Street
  Plymouth, Mass. 02360
  Pilgrim Nuclear Power Station

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

- Mrs. Marie D. Chulski, Head
  Reference and Information
  Reference/Government Documents Coordinator
  Monroe County Library System
  Monroe, Mich. 48161
  Enrico Fermi Atomic Power Plant

- Ms. Bea Rodgers, Library Assistant
  Maude Preston Palenske Memorial
  Library
  500 Market Street
  St. Joseph, Mich. 49085
  Donald C. Cook Nuclear Power
  Plant

MINNESOTA

- Mr. William L. Johnston, Librarian
  Technology and Science Department
  Minneapolis Public Library
  300 Nicollet Mall
  Minneapolis, Minn. 55401
  Monticello Nuclear Generating Plant

MISSISSIPPI

- Mrs. Gayle Keefe
  Library Technical Assistant
  George M. McLendon Library
  Hinds Junior College
  Main Street
  Raymond, Miss. 39154
  Grand Gulf Nuclear Station

MISSOURI

- Mrs. Evelyn Hillard
  Public Services Librarian
  Callaway County Public Library
  710 Court Street
  Fulton, Mo. 65251
  Callaway Plant

- Mr. Bill Olbrich
  Government Publications Librarian
  John M. Olin Library
  Washington University
  Skinker and Lindell Boulevards
  St. Louis, Mo. 63130
  Callaway Plant

NEBRASKA

- Mrs. Trudy Peaslee
  Auburn Public Library
  1118 15th Street
  P.O. Box 324
  Auburn, Neb. 68305
  Cooper Nuclear Station

- Mr. Patrick R. Esser, Librarian
  Business, Science and Technology Dept.
  W. Dale Clark Library
  215 S. 15th Street
  Omaha, Neb. 68102
  Fort Calhoun Station

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

NEW HAMPSHIRE

- Ms. Pamela Gjettum, Director
  Exeter Public Library
  Front Street
  Founders Park
  Exeter, N.H. 03833
  Seabrook Nuclear Station

NEW JERSEY

- Mrs. Amy Allen, Librarian
  Pennsville Public Library
  190 S. Broadway
  Pennsville, N.J. 08070
  Hope Creek Nuclear Station

- Ms. Elizabeth C. Fogg, Director
  Salem Free Public Library
  112 West Broadway
  Salem, N.J. 08079
  Salem Nuclear Generating Station

- Ms. Lois J. Brown, Reference Librarian
  Reference Department
  Ocean County Library
  101 Washington Street
  Toms River, N.J. 08753
  Oyster Creek Nuclear Power Plant

NEW YORK

- Ms. Blanche Judd, Coordinator
  Reference and Documents Department
  Penfield Library
  State University of New York
  Oswego, N.Y. 13126
  James A. Fitzpatrick Nuclear Power
  Plant
  Nine Mile Point Nuclear Station

- Ms. Cynthia Dana
  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 Power Station

- Mr. Oliver F. Swift
  Municipal Reference Librarian
  White Plains Public Library
  100 Martine Avenue
  White Plains, N.Y. 10601
  Indian Point Station

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 Station

- Mrs. Joe Ann Stephens
  Reference Librarian
  Richard B. Harrison Library
  1313 New Bern Avenue
  Raleigh, N.C. 27610
  Shearon Harris Nuclear Power 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 Power 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 Power Station

OKLAHOMA

- Ms. Valerie Rogers, Library Assistant
  Sallisaw City Library
  101 E. Cherokee St.
  Sallisaw, Okla. 74955
  Kerr-McGee Sequoyah

OREGON

- Mr. Robert Lockerby, Head
  Readers' Services Department
  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 Power Station
- 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 Station
  Peach Bottom Atomic Power Station
- Ms. Sharon Reilly
  Apollo Memorial Library
  219 N. Pennsylvania Avenue
  Apollo, Pa. 15613
  Babcock & Wilcox Parks Township and B&W Apollo
- Mr. Jacques Peterman
  Reference Librarian
  Government Publications Department
  Free Library of Philadelphia
  19th and Vine Streets
  Philadelphia, Pa. 18103
  Limerick Generating Station
- Mrs. Julia Albright
  Interlibrary Loan Librarian
  Pottstown Public Library
  500 High Street
  Pottstown, Pa. 19464
  Limerick Generating Station
- 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
  Room C 207
  Pennsylvania State University
  University Park, Pa. 16802
  Beaver Valley Power Station
  Susquehanna Steam Electric Station
- Mr. Ray Van de Castle
  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
  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
  Columbia, S.C. 29201
  Catawba Nuclear Station
- Ms. Virginia Warr, Librarian
  Nuclear Information Depository
  Hartsville Memorial Library
  220 N. Fifth Street
  Hartsville, S.C. 29550
  H.B. Robinson 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 Station
- 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 Station

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
  P.O. Box 19497
  Arlington, Tex. 76019
  Comanche Peak Steam Electric Station
- Mr. Bill Brock, Library Assistant
  Austin-Travis County Collection
  Austin History Center
  Austin Public Library
  810 Guadalupe Street
  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
  San Antonio, Tex. 78205
  South Texas Project

- Ms. Gloria Southall, Secretary
  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 Power Station

VIRGINIA

- Mr. Gregory A. Johnson
  Senior Public Services Assistant
  Manuscripts Dept.
  Alderman Library
  University of Virginia
  Charlottesville, Va. 22901
  North Anna Power Station

- Mr. Alan Zoellner
  Documents Librarian
  Swem Library
  College of William and Mary
  Williamsburg, Va. 23185
  Surry Power Station
  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. Janet Fullerton
  Reference Librarian
  Richland Public Library
  Swift and Northgate Streets
  Richland, Wash. 99352
  WPPSS Nuclear Projects 1, 2, & 4
  Basalt Waste Isolation Project,
  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 Power Plant

- Ms. Elizabeth J. Helfrich
  Reference Librarian
  LaCrosse Public Library
  800 Main Street
  LaCrosse, Wis. 54601
  LaCrosse Nuclear Power Plant

- Ms. Joan Schmid, Head
  Adult Services
  Joseph Mann Library
  1516 16th Street
  Two Rivers, Wis. 54241
  Point Beach Nuclear Plant

Appendix 4

Regulations and Amendments—Fiscal Year 1987

The regulations of the Nuclear Regulatory Commission are contained in Title 10, Chapter 1, of the Code of Federal Regulations. Effective and proposed regulations concerning licensed activities, and certain policy statements related to them, which were published in the Federal Register during fiscal year 1987, are described briefly below.

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Amendments to Access Authorization Fee Schedule; Publication for Licensee Personnel—Parts 11 and 25

On October 2, 1986 (51 FR 35206), the NRC published an amendment to its regulations, effective immediately, that revised the schedule for publishing access authorization investigation fees charged to licensee personnel who require access to National Security Information and/or Restricted Data and Special Nuclear Material.

Nomenclature Changes to Implement Consolidation of OGC and OELD—Parts 0, 1, 9, 10, 14, 51, and 110

On October 8, 1986 (51 FR 35997), the NRC published an amendment to its regulations, effective immediately, to reflect the changes resulting from the Commission decision to consolidate the Office of the Executive Legal Director into the Office of the General Counsel.

Regional Nuclear Materials Licensing for the United States Air Force—Parts 30, 40, and 70

On October 8, 1986 (51 FR 35999), the NRC published an amendment to its regulations concerning the domestic licensing of source, byproduct, and special nuclear material. The amendment, effective October 1, 1986, extends to Region IV the same authority for the United States Air Force license that it now possesses for nearly all other Federal licensees.

Medical Use of Byproduct Material—Parts 30, 31, 32, 35, and 40

On October 16, 1986 (51 FR 36932), the NRC published an amendment to its regulations modifying the process for licensing and regulating the medical use of byproduct material. The amendment, effective April 1, 1987, provides a single source of requirements related specifically to the medical use of byproduct material. The amendment also provides flexibility for licensees by allowing them to update their day-to-day radiation safety procedures without applying for and receiving a license amendment.

Domestic Licensing of Production and Utilization Facilities; Communications Procedures Amendments—Parts 50 and 51

On November 6, 1986 (51 FR 40303), the NRC published an amendment to its regulations concerning the procedures for submitting correspondence, reports, applications, or other written communications pertaining to the domestic licensing of production and utilization facilities. The amendments, effective January 5, 1987, indicate the correct mailing address for delivery of communications and specify the number of copies required to facilitate action.

Access Authorization for Licensee Personnel—Parts 25 and 95

On December 31, 1986 (51 FR 47204), the NRC published an amendment to its regulations, effective immediately, that requires each current holder of and applicant for an NRC access authorization to complete a new standardized form. The amendment also requires that a security briefing be presented to each person prior to completing the new form.

Imports of Uranium from South Africa—Part 110

On December 31, 1986 (51 FR 47207), the NRC published an amendment to its regulations, effective immediately, concerning the import of uranium from South Africa under the general license. The amendment deletes the general import license with respect to the import of any uranium of South African origin. This action implements provisions of the Comprehensive Anti-Apartheid Act of 1986.

Revision of Specific Exemptions—Part 9

On January 9, 1987 (52 FR 759), the NRC published an amendment to its regulations pertaining to specific exemptions to the NRC's Systems of Records. The amendment, effective immediately, is necessary to reflect recent changes to the regulations following the revision and republication of NRC's Systems of Records notices.

Bankruptcy Filing; Notification Requirements—Parts 30, 40, 50, 61, 70, and 72

On January 12, 1987 (52 FR 1292) the NRC published an amendment to its regulations requiring that licensees notify the NRC if they become involved in a bankruptcy proceeding. The amendment, effective February 11, 1987, is necessary because a licensee's severe financial condition could affect its ability to handle licensed radioactive material. The NRC must be notified if such a situation occurs so that appropriate measures may be taken to protect public health and safety.

Functions of Atomic Safety and Licensing Appeal Board—Part 2

On January 30, 1987 (52 FR 2993), the NRC amended its regulations, effective immediately, to provide for Atomic Safety and Licensing Appeal Board review of all decisions rendered by the Atomic Safety and Licensing Board in formal agency adjudications.

Improved Personnel Dosimetry Processing—Part 20

On February 13, 1987 (52 FR 4601), the NRC published an amendment to its regulations requiring that all licensees using personnel dosimetry devices to comply with NRC regulations have the devices processed by processors that have been accredited by the National Voluntary Laboratory Accreditation Program of the National Bureau of Standards. These amendments, effective February 12, 1988, will improve uniformity and accuracy in personnel dosimetry.

Requirements for Criminal History Checks—Part 73

On March 2, 1987 (52 FR 6310), the NRC published an amendment to its regulations implementing a program for the control and use of criminal history data received from the Federal Bureau of Investigation as part of the criminal history checks of individuals granted unescorted access to nuclear power facilities or access to Safeguards Information by nuclear power reactor licensees. The amendment, effective April 1, 1987, helps assure that individuals with criminal histories reflecting on their reliability and trustworthiness are not permitted unescorted access to a nuclear power facility or access to Safeguards Information.

Licenses and Radiation Safety Requirements for Well Logging—Parts 19, 20, 21, 30, 39, 40, 51, 70, 71, and 150

On March 17, 1987, (52 FR 8225), the NRC published an amendment to its regulations that specifies radiation safety requirements and license requirements for the use of licensed radioactive materials in well logging. The amendment, effective July 14, 1987, consolidates radiation safety requirements for well logging in a new Part 39, establishes clearly stated and specific radiation safety requirements, and promotes the adoption of uniform radiation safety requirements among NRC and Agreement States.

Operators' Licenses and Conforming Amendments—Parts 50 and 55

On March 25, 1987 (52 FR 9453), the NRC published an amendment to its regulations that updates its operator licensing requirements. The amendment, effective May 26, 1987, clarifies the requirements for issuing licenses to operators and senior operators, revises the requirements and scope of written examinations and operating tests for operators and senior operators to include a requirement for a simulation facility, codifies procedures for administering requalification examinations, and describes the form and content for operator license applications.

Implementation of the Convention on the Physical Protection of Nuclear Material—Parts 40, 70, 73, and 110

On March 26, 1987 (52 FR 9649), the NRC published an amendment to its regulations to bring them into accord with the provisions of the Convention on the Physical Protection of Nuclear Material. The amendment, effective March 25, 1987, will result in strengthened protection of shipments of convention-defined material during international transport.

Material Control and Accounting Requirements for Facilities Licensed to Possess and Use Formula Quantities of Strategic Special Nuclear Material—Parts 70 and 74

On March 30, 1987 (52 FR 10033), the NRC published an amendment to its regulations concerning material control and accounting requirements for facilities licensed to possess and use formula quantities of strategic special nuclear material. The amendment, effective April 29, 1987, will strengthen material control and accounting capabilities at affected facilities by requiring more timely detection of anomalies potentially indicative of strategic special nuclear material losses and by providing for more rapid and conclusive resolution of discrepancies.

Restriction Against Ownership of Certain Security Interests by Commissioners, Certain Staff Members, and Other Related Personnel; Vested Pension Interests—Part 0

On April 7, 1987 (52 FR 11026), the NRC published an amendment to its regulations governing the ownership by NRC employees of stocks, bonds, and other security interests in companies engaged in activities relating to the nuclear fuel cycle. The amendments, effective immediately, ensure that only major companies engaged in nuclear fuel cycle activities would be placed on the prohibited stock list. The amendments also address the treatment of vested pension interests held by NRC employees.

Production and Utilization Facilities; Timing Requirements for Full Participation Emergency Preparedness Exercises for Power Reactors Prior to Receipt of an Operating License—Part 50

On May 6, 1987 (52 FR 16823), the NRC published an amendment to its regulations to change the timing requirements for a full participation emergency preparedness exercise for power reactors prior to the issuance of a full-power operating license. The amendment, effective immediately, requires that a full participation exercise, including State and local governments, be held within two years before the issuance of a full-power operating license.

Reporting of Special Nuclear Material Physical Inventory Summary Results—Parts 70 and 74

On May 22, 1987 (52 FR 19303), the NRC published an amendment to its regulations on special nuclear material control and accounting. The amendment, effective June 22, 1987, requires the reporting of the summary results of physical inventories of special nuclear material. This information is used to monitor and assess the material control and accounting performance of NRC licensees.

Changes to Safeguards Reporting Requirements—Parts 70, 72, 73, and 74

On June 9, 1987 (52 FR 21651), the NRC published an amendment to its regulations for the reporting of safeguards events. The amendment, effective October 8, 1987, clarifies reporting requirements for NRC licensees and improves the NRC safeguards event data base by requiring more uniform safe-guards event reports.

Nondiscrimination on Basis of Age in Federally Assisted Commission Programs—Part 4

On July 7, 1987 (52 FR 25355), the NRC published an amendment to its regulations, effective immediately, that implements provisions of the Age Discrimination Act of 1975. The amendment incorporates the basic standards for determining what is age discrimination, describes responsibilities of NRC recipients, and details enforcement procedures necessary to ensure compliance with the Act.

Manufacturers' Registration of Radiation Safety Information for Certain Devices and Sealed Sources—Parts 30 and 32

On July 24, 1987 (52 FR 27782), the NRC published an amendment to its regulations to formally endorse the current administrative practice under which manufacturers of radiation sources and devices containing radiation sources file safety information about their products with the NRC. The amendments, effective August 24, 1987, describe the information necessary for the NRC to evaluate a source or device and state the registrant's responsibility to ensure that distributed products meet the radiation safety specifications filed with the NRC.

Changes in Property Insurance Requirements for NRC Licensed Nuclear Power Plants—Part 50

On August 5, 1987 (52 FR 28963), the NRC published an amendment to its regulations requiring licensees to maintain substantial amounts of onsite property insurance to provide financial security for stabilizing and decontaminating their licensed reactors in the event of an accident. The amendments, effective October 5, 1987, increase the amount of insurance required, impose a modified decontamination priority on any proceeds from the insurance, and require that proceeds subject to the decontamination priority be paid to an independent trustee.

Charges for the Production of Records—Part 9

On August 10, 1987 (52 FR 29504), the NRC published an amendment to its regulations that revised the charges for copying records publicly available at the NRC Public Document Room. The amendment, effective July 9, 1987, reflects the change in copying charges resulting from the agency's award of a new contract to provide this service.

Standards and Procedures for Case-by-Case Exemptions for De Minimis Interests from Prohibition Against Employee's Participation in Particular Matter Affecting Employee's Financial Interests—Part 0

On August 18, 1987 (52 FR 30902), the NRC published an amendment to its regulations, effective immediately, concerning its standards of conduct regulations. The amendment clarifies and elaborates on provisions governing the granting of statutorily authorized case-by-case exemptions for insubstantial interests from the prohibition against an employee's personal and substantial participation in a particular matter the outcome of which would have a direct and predictable effect on the financial interest of the employee.

Statement of Organization and General Information—Parts 0, 1, 2, 7, 9, 10, 11, 15, 19, 20, 21, 25, 30, 35, 40, 50, 51, 60, 61, 70, 71, 72, 73, 74, 75, 95, 150, and 170

On August 21, 1987 (52 FR 31601), the NRC published an amendment to its regulations that revised its statement of organization and general information. The revision, effective August 19, 1987, reflects the completion of a major reorganization of agency offices and assigned functions.

Telephone Reporting of Significant Events Involving Byproduct, Source, or Special Nuclear Material—Part 20

On September 9, 1987 (52 FR 33916), the NRC published an amendment to its regulations regarding the telephone reporting of significant events involving byproduct, source, or special nuclear material. The amendment, effective immediately, requires that all telephone calls reporting significant events be directed to the NRC Operations Center.

Policy and Procedure for Enforcement Actions: Policy Statement—Part 2

On September 28, 1987 (52 FR 36215), the NRC published an amendment to its regulations, effective immediately, that revised its enforcement policy. The amendment further explains enforcement actions involving individuals, describes the criteria to be used for reopening closed enforcement actions, provides for the exercise of discretion in issuing a Notice of Violation or a proposed civil penalty under certain limited circumstances and makes minor deletions and language changes.

REGULATIONS AND AMENDMENTS PROPOSED

Reporting of Special Nuclear Material Physical Inventory Summary Results—Parts 70 and 74

On October 23, 1986 (51 FR 37578), the NRC published a notice of proposed rulemaking that would amend its regulations on special nuclear material control and accounting. The proposed changes would require the reporting of summary results of physical inventories of special nuclear material.

Leakage Rate Testing of Containments of Light-Water Cooled Nuclear Power Plants—Part 50

On October 29, 1986 (51 FR 39538), the NRC published a notice of proposed rulemaking that would update the criteria and clarify questions of interpretation in regard to the leakage rate testing of containments of light-water-cooled nuclear power plants. The proposed rule would aid the licensing and enforcement staff by eliminating conflicts, ambiguities, and lack of uniformity in the regulation of the in-service inspection program.

Requirements for Criminal History Checks—Part 73

On November 7, 1986 (51 FR 40438), the NRC published a notice of proposed rulemaking that would add provisions necessary to implement a program for the control and use of criminal history data received from the FBI as part of criminal history checks of individuals granted unescorted access to nuclear power facilities or access to Safeguards Information by nuclear power reactor licensees.

Production and Utilization Facilities; Timing Requirements for Full Participation Emergency Preparedness Exercises for Power Reactors Prior to Receipt of an Operating License—Part 50

On December 2, 1986 (51 FR 43369), the NRC published a notice of proposed rulemaking that would relax the timing requirements for a full participation emergency preparedness exercise for power reactors prior to the issuance of a full-power operating license. The proposed amendment would require a full participation exercise, including State and local governments, to be held within two years before the issuance of full-power operating license instead of the current requirement that the exercise be held within one year before the license is issued.

Manufacturers' Registration of Radiation Safety Information for Certain Devices and Sealed Sources—Parts 30 and 32

On January 23, 1987 (52 FR 2540), the NRC published a notice of proposed rulemaking that would amend its regulations to formalize the current administrative practice under which manufacturers of radiation sources and devices containing radiation sources file safety information about their products with the NRC. The proposed amendments describe the information the NRC needs for its evaluation of a source or device and states the registrant's responsibility to ensure that distributed products meet the radiation safety specifications filed with the NRC.

Issuance or Amendment; Power Reactor License or Permit Following Initial Decision—Part 2

On February 4, 1987 (52 FR 3442), the NRC published a notice of proposed rulemaking that would specify when a license, permit, or amendment can be issued following an initial decision resolving all issues before the presiding officer in favor of authorizing the issuance or amendment of a license or permit. The proposed changes would simplify and clarify the existing rule and remove language emanating from Three Mile Island related regulatory policies upon which action has been completed.

Emergency Core Cooling Systems; Revisions to Acceptance Criteria—Part 50

On March 3, 1987 (52 FR 6334), the NRC published a notice of proposed rulemaking that would allow the use of alternative methods to demonstrate that the emergency core cooling system would protect the nuclear reactor core during a postulated design basis loss-of-coolant accident. The use of alternative methods is proposed because research performed since the current requirement was issued has significantly improved understanding of cooling system performance.

Licensing of Nuclear Power Plants Where State and/or Local Governments Decline to Cooperate in Offsite Emergency Planning—Part 50

On March 6, 1987 (52 FR 6980), the NRC published a notice of proposed rulemaking concerning its regulations governing offsite emergency planning at nuclear power plant sites. The proposed amendment would allow the issuance of a full-power operating license in limited circumstances even when a utility cannot meet all of NRC's current emergency planning requirements when there is a lack of cooperation by State and/or local governments in developing or implementing offsite emergency plans.

Completeness and Accuracy of Information—Parts 30, 40, 50, 55, 60, 61, 70, 71, 72, 110, and 150

On March 11, 1987 (52 FR 7432), the NRC published a notice of proposed rulemaking that would codify the obligation of licensees and applicants for licenses to provide the Commission with complete and accurate records, and to provide for disclosure of information identified by licensees as significant for licensed activities.

Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees—Parts 30, 40, and 70

On April 20, 1987 (52 FR 12921), the NRC published a notice of proposed rulemaking that would require certain fuel cycle and other radioactive material licensees to revise existing emergency plans. The proposed rule would require that the emergency plans include descriptions of the means and equipment to mitigate the consequences of an accident and to notify offsite response organizations promptly if an accident occurs that might result in a significant release of radioactive material.

Informal Hearing Procedures for Materials Licensing Adjudications—Part 2

On May 29, 1987 (52 FR 20089), 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 material licensing proceedings.

Codes and Standards for Nuclear Power Plants—Part 50

On June 26, 1987 (52 FR 24015), the NRC published a notice of proposed rulemaking that would incorporate by reference certain recent addenda to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The sections that would be incorporated provide rules for the construction of light-water cooled nuclear power plant components and specify requirements for in-service inspection of those components.

Revision of Freedom of Information Act Regulations; Conforming Amendments—Part 9

On August 6, 1987 (52 FR 29196), the NRC published a notice of proposed rulemaking that would amend its Freedom of Information Act regulations to conform to the requirements of the Freedom of Information Reform Act of 1986. The proposed rule would also make the changes necessary to reflect current NRC organizational structure and current agency practices and delegation of authority.

Revision of Backfitting Process for Power Reactors—Part 50

On September 10, 1987 (52 FR 34223), the NRC published a notice of proposed rulemaking that would amend its regulations concerning the backfitting process for power reactors. The proposed amendments are intended to clarify when economic costs may be considered in backfitting nuclear power plants.

ADVANCE NOTICES OF PROPOSED RULEMAKING

Radioactive Waste Below Regulatory Concern; Generic Rulemaking—Parts 2 and 20

On December 2, 1986 (51 FR 43367) the NRC published an advance notice of proposed rulemaking to request public comment on a contemplated amendment that would address the disposal of radioactive wastes that contain sufficiently small quantities or low concentrations of radionuclides that their disposal need not be regulated as radioactive. The NRC is considering generic rulemaking as an efficient and effective means of dealing with the disposal of wastes that do not pose an undue risk to public health and safety or the environment.

Rule on the Submission and Management of Records and Documents Related to the Licensing of a Geologic Repository for the Disposal of High-Level Radioactive Waste; Intent to Form an Advisory Committee for Negotiated Rulemaking—Part 2

On December 16, 1986 (51 FR 45338), the NRC published a notice of intent to announce that it is considering the formation of an advisory committee under the Federal Advisory Committee Act. This committee would develop recommendations for revision of the Commission's discovery rules and selected other rules of practice related to the adjudicatory proceeding for the issuance of a license for a geologic repository for the disposal of the high-level radioactive waste. Specifically, the committee would attempt to negotiate a consensus on proposed revisions related to the submission and management of documents related this licensing proceeding.

Intent to Develop Regulations to Establish Criteria and Procedures for Evaluating Requests for Emergency Access to Low-Level Radioactive Waste Disposal Facilities—Part 62

On January 15, 1987 (52 FR 1634), the NRC published a notice of intent to announce that it is developing regulations to establish criteria and procedures for evaluating requests for emergency access to non-Federal low-level waste disposal facilities.

Definition of "High-Level Radioactive Waste"—Part 60

On February 27, 1987 (52 FR 5992), the NRC published an advance notice of proposed rulemaking to request public comment on its intent to modify the definition of "high level radioactive waste in geologic repositories. The contemplated amendment would modify the definition to follow more closely the statutory definition set out in the Nuclear Waste Policy Act of 1982.

Appendix 5

Regulatory Guides—Fiscal Year 1987

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, revised, 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, 1986, to September 30, 1987.

Division 1—Power Reactor Guides

1.8 Qualification and Training of Personnel for Nuclear Power Plants (Revision 2)

1.63 Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Revision 3)

1.134 Medical Evaluation of Licensed Personnel for Nuclear Power Plants (Revision 2)

1.149 Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations (Revision 1)

1.154 Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors

Division 2—Research and Test Reactor Guides

NONE

Division 3—Fuels and Materials Facilities Guides

3.1 Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material (Revision 2)

3.52 Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Processing and Fuel Fabrication (Revision 1)

3.57 Administrative Practices for Nuclear Criticality Safety at Fuels and Materials Facilities

3.58 Criticality Safety for Handling, Storing, and Transporting LWR Fuel at Fuels and Materials Facilities

3.59 Methods for Estimating Radioactive and Toxic Airborne Source Terms for Uranium Milling Operations

3.60 Design of an Independent Spent Fuel Storage Installation (Dry Storage)

Division 4—Environmental and Siting Guides

4.17 Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories (Revision 1)

Division 5—Materials and Plant Protection Guides

NONE

Division 6—Product Guides

NONE

Division 7—Transportation Guides

NONE

Division 8—Occupational Health Guides

NONE

Division 9—Antitrust and Financial Review Guides

NONE

Division 10—General Guides

10.8 Guide for the Preparation of Applications for Medical Programs (Revision 2)

10.10 Guide for the Preparation of Applications for Radiation Safety Evaluation and Registration of Devices Containing Byproduct Material

10.11 Guide for the Preparation of Applications for Radiation Safety Evaluation and Registration of Sealed Sources Containing Byproduct Material

DRAFT GUIDES

Division 1

EE 006-5 Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants

EE 108-5 Proposed Revision 2 to Regulatory Guide 1.100, Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants

EE 404-4 Environmental Qualification of Connection Assemblies for Nuclear Power Plants

HF 601-4 Proposed Revision 2 to Regulatory Guide 1.114, Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit

MS 021-5 Containment System Leakage Testing

RS 701-4 Best-Estimate Calculations of Emergency Core Cooling System Performance

Division 3

CE 403-4 Proposed Revision 2 to Regulatory Guide 3.44, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Water-Basin Type)

CE 406-4 Proposed Revision 1 to Regulatory Guide 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)

CE 602-4 Proposed Revision 2 to Regulatory Guide 3.1, Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material

WM 503-4 Calculation of Radon Flux Attenuation by Earthen Uranium Mill Tailings Covers

Division 4

WM 408-4 Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste

Division 7

MS 527-4 Proposed Revision 1 to Regulatory Guide 7.8, Load Combinations for the Structural Analysis of Shipping Casks for Irradiated Fuel

Division 8

OP 013-4 Proposed Revision 1 to Regulatory Guide 8.22, Bioassay at Uranium Mills

Division 10

FC 603-4 Guide for the Preparation of Applications for Radiation Safety Evaluation and Registration of Sealed Sources Containing Byproduct Material

Appendix 6

Civil Penalties and Orders—Fiscal Year 1987

Civil Penalty Actions During FY 1987*

Licensee	*Amount*	*Reason*
Reich Geo-Physical, Inc. Billings, Mont. (EA 84-078)	$1,600 proposed in FY 84; imposed in FY 85; paid in FY 87	Violations involving the use of unauthorized material and failure to calibrate survey meters at the required intervals.
Astrotech, Inc. Harrisburg, Pa. (EA 85-086)	$5,000 proposed in FY 85; imposed in FY 86; paid in FY 87	Violations involving the use of technically unqualified personnel to perform licensed activities.
Hurley Medical Center Flint, Mich. (EA 85-089)	$2,500 proposed in FY 85; imposed in FY 86; paid in FY 87	Violations involving the use of licensed material in an unauthorized location, improper disposal of licensed material, and failure to perform surveys.
Toledo Edison Co. (Davis Besse) (EA 85-107)	$900,000 proposed in FY 86; $450,000 imposed and paid in FY 87	Violations which resulted in the failure of the auxiliary feedwater system to perform its intended function.
Texas Utilities Generating Co. (Comanche Peak) (EA 86-009)	$250,000 proposed in FY 86; Licensee paid $200,000 in FY 86; $50,00 was imposed and paid in FY 87	Violations involving significant weaknesses in the implementation of quality programs during construction and weaknesses in cable tray reinspection program.
Florida Power & Light Co. (Turkey Point) (EA 86-020)	$300,000 proposed in FY 86; paid in FY 87	Violations involving failures to satisfy Technical Specification and 10 CFR 50.59 requirements, and to ensure that safety activities were performed in accordance with adequate procedures and drawings.
Florida Power Corporation (Crystal River) (EA 86-037)	$80,000 proposed and paid in FY 87	Violations involving deficiencies in the management of training licensed operators.

*Cases are presented in the order of EA number. Indicated status reflects the situation as of the end of the fiscal year, September 30, 1987. Some pending cases may have been settled by the time of publication.

Licensee	*Amount*	*Reason*
Florida Power & Light Co. (Turkey Point) (EA 86-038)	$50,000 proposed in FY 86; $25,000 imposed and paid in FY 87	Violations involving the unauthorized entry of a plant worker into a locked high radiation area due to numerous procedural violations.
Mercy Hospital Wilkes Barre, Pa. (EA 86-040)	$5,000 proposed in FY 86 and paid in FY 87	Violations involving a misadministration which was not reported to the NRC or other responsible authority and a material false statement.
Nebraska Public Power Dist. (Cooper) (EA 86-044)	$50,000 proposed in FY 86; $25,000 imposed and paid in FY 87	Violations involving numerous security violations, including a degraded vital area and unescorted access by a temporary employee.
South Carolina Electric & Gas (Summer) (EA 86-045)	$50,000 proposed and imposed in FY 86; paid in FY 87	Violations involving a failure to comply with plant technical specifications.
Commonwealth Edison Co. (Zion) (EA 86-049)	$25,000 proposed in FY 86; withdrawn in FY 87	Violations relating to testing and maintenance activities resulting in the isolation of the service water to the bearing oil cooler, rendering the pump inoperable.
Combustion Engineering Windsor, Conn. (EA 86-051)	$15,000 proposed in FY 86; imposed and paid in FY 87	Violations involving unauthorized transfer of licensed material use of licensed material at an unauthorized location.
Duke Power Company (McGuire) (EA 86-052)	$50,000 proposed and imposed in FY 86; paid in FY 87	Violations involving failure to take appropriate measures when a Limiting Condition of Operation was exceeded.
TVA (Browns Ferry) (EA 86-056)	$150,000 proposed in FY 86; paid in FY 87	Violations involving inadequate design of cable tray supports, failure to take corrective actions, and failure to ensure that actions were taken in accordance with appropriate drawings and procedures.
Philadelphia Electric Company (Peach Bottom) (EA 86-059)	$200,000 proposed in FY 86; imposed and paid in FY 87	Violations involving withdrawal of the wrong control rod from the core, inadequate verification of adherence to rod withdrawal program, improper bypassing of the Rod Sequence Control System for a control rod, and the inadequate verification of the rod position before bypassing the Rod Sequence Control System.
Arizona Public Service (Palo Verde) (EA 86-065)	$100,000 proposed in FY 86; imposed and paid in FY 87	Violations involving numerous security violations, including access control, degraded vital area barriers, and failure to report.
Progressive Engineering Consultants Grand Rapids, Mich. (EA 86-079)	$500 proposed in FY 86; imposed and paid in FY 87	Violations involving failure to supervise use of licensed materials, unauthorized transfer of licensed materials, failure to perform surveys, and transportation
Commonwealth Edison Co. (Byron) (EA 86-087)	$25,000 proposed in FY 86; imposed and paid in FY 87	Violating relating to a subcontractor's discharge of an employee for reporting inadequate inspection procedures and the installation of non-radiation-proof seals.

Licensee	*Amount*	*Reason*
Sequoyah Fuels Oklahoma City, Okla. (EA 86-091)	$310,000 proposed and paid in FY 87	Violations involved in an accident during which a cylinder filled with uranium hexafluoride ruptured.
Sacramento Municipal Utility District (Rancho Seco) (EA 86-094)	$375,000 proposed and paid in FY 87	Violations identified as a result of a loss of DC power to the integrated control system.
Southern California Edison Co. (San Onofre) (EA 86-097)	$180,000 proposed in FY 86; paid in FY 87	Violations associated with a loss of power and water hammer event; including rendering inoperable the auxiliary feedwater flow path to the steam generators, failure of licensee's check valve testing program, and failure to take adequate corrective actions.
Cook County Highway Department Chicago, Ill. (EA 86-102)	$500 proposed in FY 86; imposed and paid in FY 87	Violations including the handling of nuclear material by an unauthorized individual, and storage of licensed material in an unauthorized location.
Texas A & M College Station, Tex. (EA 86-105)	$1,250 proposed in FY 86; $833 imposed and paid in FY 87	Violations involving the removal of an experiment from the reactor while it was operating.
Portland General Electric (Trojan) (EA 86-113)	$50,000 proposed and paid in FY 87	Violations involving the failure to main tain operable flow paths for the Residual Heat Removal System for cold leg injection and incomplete inspection by a contract quality control inspector for the installation of a pressurizer safety valve.
Union Electric Company (Callaway) (EA 86-119)	$25,000 proposed in FY 86; paid in FY 87	Violations involving the inoperability of both trains of the intermediate head safety injection system and auxiliary feedwater automatic start on loss of main feedwater pumps.
South Carolina Electric and Gas (Summer) (EA 86-126)	$50,000 proposed in FY 86; paid in FY 87	Violations involving actions rendering the charging pumps inoperable under certain conditions.
Star-Jet Services, Inc. Oklahoma City, Oklahoma (EA 86-134)	$500 proposed in FY 86; imposed and paid in FY 87	Violations involving the failure to perform surveys and failure to properly store licensed materials.
University of Utah Salt Lake City, Utah (EA 86-136)	$3,000 proposed, imposed and paid in FY 87	Violations involving transportation, health physics, safeguards, and operations.
NOW Logging, Perforating, Inc. Enid, Okla. (EA 86-138)	$800 proposed in FY 86; paid in FY 87	Violations based on numerous health physics violations including failure to properly store licensed material, provide survey instruments, provide personnel dosimetry, and conduct leak tests.
Professional Consultants, Inc. Missoula, Mont. (EA 86-140)	$1,000 proposed in FY 86; imposed and paid in FY 87	Violations involving the failure to conduct inventories and failure to perform leak tests.

Licensee	*Amount*	*Reason*
Illinois Power Company (Clinton) (EA 86-143)	$50,000 proposed in FY 86; pending	Violations involving discrimination against an employee for reporting missing safety-related documentation.
General Public Utilities (TMI) (EA 86-146)	$40,000 proposed in FY 86; paid in FY 87	Violations relating to modifications made to the reactor building polar crane without proper engineering review and documentation.
Duke Power Company (Catawba) (EA 86-147)	$50,000 proposed, imposed, and paid in FY 87	Violations associated with a depressurization event during a loss of control room test, involving failure to change plant procedures when components were modified, inadequate control panels resulting in the inadequate labelling and marking of components, and deficient procedures which lacked specific test termination criteria.
Wisconsin Electric Power Co. (Point Beach) (EA 86-148)	$50,000 proposed in FY 87; pending	Violations involving degraded vital area barriers.
Indiana & Michigan Electric Company (D.C. Cook) (EA 86-150)	$25,000 proposed and paid in FY 87	Violations involving the failure to follow procedures in the removal and reinstallation of wires and in the initiation of a noncomformance report to resolve a wiring problem.
Arkansas Power & Light (ANO, Unit 1) (EA 86-151)	$50,000 proposed, imposed and paid in FY 87	Violations involving the modification of the steam supply lines to the turbine driven emergency feedwater pump.
Nurrie Construction Company Muskogee, Okla. (EA 86-152)	$500 proposed, imposed and paid in FY 87	Violations including failure to leak test a sealed source, conduct inventories of licensed material, use shipping papers during transportation of licensed material, and use an authorized radiation protection officer.
Omaha Public Power District (Ft. Calhoun) (EA 86-153)	$15,000 proposed and paid in FY 87	Violation involving an inadequate vital area barrier.
Commonwealth Edison Company (Byron) (EA 86-163)	$25,000 proposed, imposed, and paid in FY 87	Violation involving the installation of a pressurizer code safety valve without its valve disc.
Connecticut Yankee (Haddam Neck) (EA 86-167)	$50,000 proposed and paid in FY 87	Violations involving inadequate control and direct surveillance of work activities and failure to verify the qualifications of HP technicians.
Gamma Diagnostic Laboratory Attleboro Falls, Mass. (EA 86-168)	$5,000 proposed and paid in FY 87	Violations including failure to properly secure or control licensed material, properly post and control access to a high radiation area, and perform adequate surveys.

Licensee	Amount	Reason
Lehigh Testing Laboratory New Castle, Del. (EA 86-170)	$1,500 proposed and paid in FY 87	Violations including failure to completely verify the qualifications of an individual before certifying him as a radiographer and and failure to perform certain audits.
Eastern Virginia Medical Authority (EA 86-172)	$2,500 proposed and paid in FY 87	Violations involving health physics violations. Norfolk, Va.
Omaha Public Power District (Ft. Calhoun) (EA 86-176)	$50,000 proposed and paid in FY 87	Violations associated with a modification of the system admit valve to the turbine-driven auxiliary feedwater pump steam supply system without conducting and documenting an adequate safety evaluation.
University of Wisconsin Madison, Wis. (EA 86-179)	$1,250 proposed, imposed and paid in FY 87	Violations involving failure to conduct appropriate radiological surveys during the course of an experiment and the loss of the material.
Massachusetts General Hospital Boston, Mass. (EA 86-180)	$2,500 proposed and paid in FY 87	Violations including the failure to properly secure or provide continuous surveillance of licensed material, provide appropriate training, and conduct required surveys.
Henry Heywood Hospital Gardner, Mass. (EA 86-181)	$2,500 proposed and paid in FY 87	Violations involving the unauthorized incineration of a generator containing licensed material.
Riverton Hospital Riverton, Wyo. (EA 86-185)	$2,500 proposed, imposed, and paid in FY 87	Violations involving the failure to restrict use of licensed materials to qualified and authorized users and failure to properly follow assay procedures.
University of Missouri Columbia, Mo. (EA 86-191)	$4,000 proposed, imposed and paid in FY 87	Violations involving an extremity overexposure and inadequate radiation hazard evaluation during a pellet handling operation.
New York Power Authority (Indian Point) (EA 86-197)	$50,000 proposed and paid in FY 87	Violations involving the heating up of the reactor above the cold shutdown condition with both recirculation pumps and both containment spray pumps inoperable.
Yellowstone County Surveyor's Office, Billings, Mont. (EA 86-198)	$500 proposed, imposed and paid in FY 87	Violations involving the failure to provide dosimetry to personnel using licensed material, store licensed material properly, maintain personnel exposure records, maintain inventory records, adhere to transportation requirements, and perform leak tests.
Grede Foundries, Inc. Milwaukee, Wisc. (EA 86-201)	$7,500 proposed and paid in FY 87	Violations involving inaccurate statements in two letters to the NRC.
Dairyland Power Cooperative (LaCrosse) (EA 87-002)	$25,000 proposed and imposed in FY 87; pending	Violations involving the protection of safeguards information.

Licensee	Amount	Reason
Soils and Materials Engineers, Inc., Livonia, Mich. (EA 87-003)	$500 proposed and paid in FY 87	Violations involving failure to make an adequate evaluation of apparent radiation doses to an individual in excess of regulatory requirements, adequately train individuals who use licensed gauges, make timely leak tests of sealed sources, properly transport moisture density gauges, and store moisture density gauges only at authorized locations.
Philadelphia Electric Company (Peach Bottom) (EA 87-005)	$50,000 proposed and paid in FY 87	Violations involving termination of a health physics technician for engaging in protected activity.
Daly Memorial Hospital Hamilton, Mont. (EA 87-006)	$1,250 proposed and paid in FY 87	Violations including failure to supervise the use of licensed material, have operable survey instruments, and maintain proper records.
Duke Power Company (McGuire) (EA 87-008)	$50,000 proposed and paid in FY 87	Violations involving the failure to establish and maintain the operability of the nuclear service water system and performance of an inadequate 10 CFR 50.59 evaluation.
Illinois Power Company (Clinton) (EA 87-011)	$75,000 proposed and paid in FY 87	Violations involving the quality assurance program for maintenance and modifications and the protection of the circulating water screenhouse against flooding.
Indiana & Michigan Electric Company (D.C. Cook) (EA 87-013)	$50,000 proposed and paid in FY 87	Violations involving the operations staff rendering both independent emergency core cooling system subsystems inoperable while the unit was operating.
Duke Power Company (Oconee) (EA 87-014)	$25,000 proposed in FY 87; pending	Violations involving inadequate design control to assure that the emergency feedwater pumps would remain operable under design basis transients.
Commonwealth Edison Company (Byron) (EA 87-016)	$50,000 proposed and paid in FY 87	Violations involving failure to obtain prior NRC approval of a change in acceptance criteria for a reactor coolant flow coast-down test from that described in the FSAR.
Norland Instruments Atkinson, Wis. (EA 87-018)	$500 proposed and paid in FY 87	Violations involving failure to control licensed material in an unrestricted area, comply with Department of Transportation requirements, and follow proper procedures.
Program Resources, Inc. Frederick, Md. (EA 87-019)	$2,500 proposed and paid in FY 87	Violations involving a radiation overexposure.
Huron Mercy Hospital Port Huron, Mich. (EA 87-021)	$2,500 proposed, imposed, and paid in FY 87	Violations involving the failure to perform bioassays, perform instrument calibration tests, perform leak tests, and maintain proper records.

Licensee	*Amount*	*Reason*
Babcock & Wilcox Co. Lynchburg, Va. (EA 87-024)	$6,250 proposed and paid in FY 87	Violations including failure to perform adequate surveys, perform required audits, calibrate instruments, and adequately train radiation workers.
Commonwealth Edison Company (Dresden) (EA 87-026)	$25,000 proposed and paid in FY 87	Violations relating to the control of plant modifications.
Georgia Power Company (Hatch) (EA 87-027)	$50,000 proposed and paid in FY 87	Violations involving the loss of water from the spent fuel pools and the failure to follow procedures, including the closing of one valve supplying air to the fuel transfer canal seals and the calibration of the transfer canal leak detection switch.
Radiation Sterilizers, Inc. Park, Cal. (EA 87-028)	$10,000 proposed in FY 87; $7,500 imposed; pending	Violations involving failure to maintain Menlo warning devices in operable condition, failure to check irradiation cells for personnel prior to exposing the source, and failure to utilize personnel access control tags.
Cintichem, Inc. Tuxedo, N.Y. (EA 87-030)	$12,500 proposed and paid in FY 87	Violations involving accumulated radiation exposure.
LTV Steel Co., Inc. Cleveland, Ohio (EA 87-031)	$2,000 proposed, imposed, and paid in FY 87	Violations involving the failure to maintain licensed material under constant surveillance and control and multiple failures to inventory licensed material every six months as required.
Rappahannock General Hospital Kilmarnock, Va. (EA 87-034)	$750 proposed and paid in FY 87	Violations involving repeat failure of the licensee's Radiation Safety Committe to meet quarterly as required.
Halliburton Company Duncan, Okla, (EA 87-035)	$1,000 proposed in FY 87; pending	Violations involving unauthorized use of byproduct material and failure to calibrate survey instruments, properly instruct individuals involved in operations using licensed materials, maintain materials accountability records, maintain records of survey results, and post documents and notices.
Newport News Shipbuilding and Dry Dock Company Newport News, Va. (EA 87-038)	$2,000 proposed and mitigated in full in FY 87	Violations involving an overexposure during retrieval of a radiography source.
Florida Power & Light Company (Turkey Point) (EA 87-040)	$75,000 proposed and paid in FY 87	Violations involving a sleeping security guard and an unescorted visitor in a vital area.
Cleveland Clinic Foundation Cleveland, Ohio (EA 87-042)	$2,500 proposed and paid in FY 87	Violations including the failure to notify the NRC regarding a therapeutic misadministration.

Licensee	*Amount*	*Reason*
Niagara Mohawk Power Corp. (Nine Mile Point) (EA 87-045)	$50,000 proposed and paid in FY 87	Violations involving numerous examples of failure to follow station procedures, evaluate test results, and perform an adequate radiation survey and provide adequate radiation surveillance.
Florida Power Corporation (Crystal River) (EA 87-047)	$50,000 proposed and paid in FY 87	Violations involving a sleeping security guard and failure to check security badge authorizations against the access authorization list.
Power Authority of New York (Fitzpatrick) (EA 87-048)	$75,000 proposed and paid in FY 87	Violations involving radiation exposure protection requirements.
Commonwealth Edison Company (Dresden) (EA 87-049)	$25,000 proposed and paid in FY 87	Violations involving the inadequate monitoring of reactor water temperature while shutdown.
Detroit Edison Company (Fermi) (EA 87-050)	$100,000 proposed and paid in FY 87	Violations involving significant deficiencies in the implementation of the surveillance testing program.
Commonwealth Edison Company (Dresden) (EA 87-053)	$50,000 proposed and paid in FY 87	Violations involving the failure to ensure the integrity of the protected area barrier.
Comdustrial Roofing, Inc. Hatfield, Pa. (EA 87-054)	$500 proposed and paid in FY 87	Violations involving unauthorized storage of a moisture density and the failure to secure the gauge against unauthorized removal.
Centro Oftalmologico Metropolitano San Juan, Puerto Rico (EA 87-058)	$750 proposed, imposed, and paid in FY 87	Violations involving the use of licensed material by unauthorized users.
Portland General Electric (Trojan) (EA 87-060)	$50,000 proposed and paid in FY 87	Violations involving failure to establish and implement radiation protection procedures, train workers, perform radiation surveys, and maintain records of radiation survey results.
Arkansas Power & Light (ANO, Unit 1) (EA 87-062)	$25,000 proposed in FY 87; pending	Violations involving newly developed information regarding the inoperable pressurizer code safety valve.
Southern California Edison Co. (San Onofre) (EA 87-063)	$100,000 proposed and paid in FY 87	Violations involving an overexposure and the failure to control byproduct material and report the overexposure in a timely manner.
PTL Inspectorate, Inc. Pittsburgh, Pa. (EA 87-065)	$5,000 proposed and paid in FY 87	Violations involving failure to maintain direct surveillance of a high radiation area at a temporary field site.
E. I. DuPont de Nemours & Co. Boston, Mass. (EA 87-069)	$12,500 proposed and paid in FY 87	Violations involving an individual not following the licensee's procedures and receiving an overexposure.

Licensee	Amount	Reason
Wheeling Hospital, Inc. Wheeling, W. Va. (EA 87-074)	$2,500 proposed in FY 87; pending	Violations involving inadequate management oversight and control of the radiological safety program.
Hitwell Surveys, Inc. Parkersburg, W. Va. (EA 87-075)	$750 proposed and paid in FY 87	Violations including unauthorized individuals using byproduct material, excessive radiation levels on the exterior of the transport vehicles, inoperable survey equipment, and sealed sources not tested for leakage.
Eastern Testing and Inspection, Incorporated Pennsauken, N. J. (EA 87-079)	$6,500 proposed in FY 87; pending	Violations including the failure to maintain an audible warning signal to a permanent radiography cell in an operable status, properly use their dosimeters, and inadequate audit activities.
Aztec Laboratories City, Mo. (EA 87-084)	$500 proposed and imposed in FY 87; pending	Violations including failure to make Kansas necessary surveys, report an event involving licensed material, and secure licensed material in an unrestricted area from unauthorized removal.
Philadelphia Electric Company (Peach Bottom) (EA 87-088)	$50,000 proposed and paid in FY 87	Violations involving failure to meet fire protection requirements.
Arkansas Power & Light (ANO, Unit 1) (EA 87-090)	$75,000 proposed in FY 87; pending	Violations involving breaches in vital area barriers and sleeping security guards.
General Public Utilities (Oyster Creek) (EA 87-092)	$205,000 proposed and paid in FY 87	Violations involving the suppression chamber-torus vacuum breakers, reactor building-torus vacuum breakers, and procedures for making temporary variations to the facility.
Norwalk Hospital Norwalk, Conn. (EA 87-093)	$2,500 proposed and imposed in FY 87; pending	Violations involving storage of food in an area where radioactive material was used, disposing of licensed material improperly, and not wearing protective clothing.
Goodwell, Inc. Upton, Wyo. (EA 87-094)	$1,000 proposed and paid in FY 87	Violations involving implementation of the radiation safety program.
Kermit Butcher Elkins, W. Va.	$500 proposed in FY 87; pending	Violations involving lack of control and loss of licensed material.
Florida Power and Light Co. (Turkey Point) (EA 87-097)	$100,000 proposed and paid in FY 87	Violations involving the failure to take prompt corrective action for a reactor coolant leak and meet the required prerequisites prior to performing core alteration activities.
Florida Power and Light Co. (Turkey Point) (EA 87-098)	$75,000 proposed in FY 87; pending	Violations involving the failure to maintain positive access control over personnel and equipment and perform an adequate vehicle search.
Georgia Power Company (Vogtle) (EA 87-100)	$200,000 proposed in FY 87; pending	Violations involving failure to implement adequate compensatory measures, follow security procedures, and maintain positive access control.

Licensee	*Amount*	*Reason*
Duke Power Company (Oconee) (EA 87-101)	$25,000 proposed and paid in FY 87	Violations involving failure to properly implement a start-up procedure which resulted in high pressure injection trains and reactor building cooling units being inoperable.
Niagara Mohawk Power Corp. (Nine Mile Point) (EA 87-106)	$2,500 proposed in FY 87; pending	Violations involving shipment of material with external surface radiation levels in excess of the regulatory limit.
Yankee Atomic Electric Company (Yankee Rowe) (EA 87-107)	$25,000 proposed and paid in FY 87	Violations involving the failure to take appropriate compensatory measures, notify the appropriate personnel when the security system was degraded, and complete security records.
Sequoyah Fuels Corporation City, Okla. (EA 87-108)	$8,000 proposed in FY 87; pending	Violation involving a material false Oklahoma statement in a letter to the NRC.
Carolina Power & Light (H. B. Robinson) (EA 87-112)	$50,000 proposed in FY 87; pending	Violation involving the failure to control valve lineup activities.
St. Luke's Radiologist, Inc. Cleveland, Ohio (EA 87-113)	$1,250 proposed and paid in FY 87	Violations involving a failure of a technician to thoroughly review the patient's dose computation sheet, and as a result to administer an excessive dose.
Georgia Power & Light (Vogtle) (EA 87-115)	$50,000 proposed in FY 87; pending	Violations involving the improper evaluation of component and system operability and the failure to take prompt corrective action.
Consolidated NDE, Inc. Woodbridge, N. J. (EA 87-121)	$5,000 proposed in FY 87; pending	Violations involving failure to maintain direct surveillance of a high radiation area and to properly post an access point to the area with a required warning sign.
ATEC Associates of Virginia, Inc. Alexandria, Va. (EA 87-126)	$400 proposed and paid in FY 87	Violations involving failure to control licensed material.
Tidewater Memorial Hospital Tappahannock, Va. (EA 87-127)	$2,500 proposed in FY 87; pending	Violations involving failures of the Medical Isotopes Committee (MIC) and the Radiation Safety Officer (RSO) to perform reviews, to calibrate survey meters, and test the dose calibrator for accuracy and linearity on a quarterly basis.
Detroit Edison Company (Fermi) (EA 87-133)	$75,000 proposed in FY 87; pending	Violations involving an uncontrolled heatup of the reactor in violation of Technical Specifications.
Northern States Power (Prairie Island) (EA 87-138)	$25,000 proposed in FY 87; pending	Violations involving failure to verify that the power supply breaker for a safety injection pump was in the full racked position.
E. L. Conwell and Company Bridgeport, Pa. (EA 87-141)	$1,000 proposed and paid in FY 87	Violations involving failure to secure or maintain constant surveillance of licensed material in unrestricted areas.

Orders Issued During FY 87*

Licensee	*Date*	*Reason*
Met-Chem Testing Laboratories Laboratories of Utah, Inc. Salt Lake City, Utah (EA 86-124)	December 30, 1986	Order Modifying License and Order to Show Cause (Effective Immediately) Reason: Deliberate forging of records to cover up a radiation overexposure.
Kedarnath B. Joshi, M.D. Highland Waterford Medical Services Pontiac, Mich. (EA 86-139)	December 23, 1986	Order to Show Cause Reason: Use of licensed material at an unauthorized site and deliberate deception of a supplier of licensed material.
Advanced Medical Systems, Inc. Geneva, Ohio (EA 86-155)	October 10, 1986	Order Suspending License and Order to Show Cause (Effective Immediately) Reason: Service and maintenance performed on teletherapy equipment without proper NRC authorization, training, radiation detection and monitoring equipment, or service manuals.
Eastside Radiology Imaging and Therapy Center Willoughby Hills, Ohio (EA 86-156)	October 10, 1986	Order Modifying License (Effective Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualified individuals.
VA Medical Center Radiation Therapy Center Allen Park, Mich. (EA 86-157)	October 10, 1986	Order Modifying License (Effective Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualifed individuals.
Ball Memorial Hospital Department of Radiology Muncie, Ind. (EA 86-158)	October 10, 1986	Order Modifying License (Effective Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualified individuals.
VA Hospital Orange, N. J. (EA 86-159)	October 10, 1986	Order Modifying License (Effective East Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualified individuals.
VA Bronx Bronx, N. Y. (EA 86-160)	October 10, 1986	Order Modifying License (Effective Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualified individuals.

*Cases are presented in the order of EA number. Indicated status reflects the situation as of the end of the fiscal year, September 30, 1987. Some pending cases may have been settled by the time of publication.

Licensee	Date	Reason
Munson Medical Center Traverse City, Mich. (EA 86-161)	October 10, 1986	Order Modifying License (Effective Immediately) Reason: Service and maintenance performed on teletherapy devices by unauthorized and unqualified individuals.
Northern States Power Co. (Prairie Island) (EA 86-164)	October 20, 1986	Order to Show Cause Reason: To show cause why radios used to provide background music in the control rooms should not be removed.
Halliburton, Co. Duncan, Okla. (EA 87-35)	September 29, 1987	Order Modifying License Reason: Unauthorized use of by-product material and failure to: calibrate survey instruments, instruct individual involved in operations using licensed material, maintain material accounting records, maintain records of survey results, and post documents and records.
A-1 Inspection, Inc. Evanston, Wyo. (EA 87-041)	April 10, 1987	Order Temporarily Suspending License (Effective Immediately) and Order to Show Cause Reason: Hiring an individual to conduct radiography without assuring that the individual was qualified and without adding the individual to the license.
Milford Memorial Hospital Milford, Del. (EA 87-044)	June 15, 1987	Order Modifying License (Effective Immediately) Reason: Falsification of daily constancy checks of the dose calibration by the licensee's technologists and falsification of records of Radiation Safety Committee meetings by the Radiation Safety Officer.
Philadelphia Electric Company (Peach Bottom) (EA 87-046)	March 31, 1987	Order Suspending Power Operation and Order to Show Cause (Effective Immediately) Reason: Periodic sleeping by members of the operations control room staff.
U.S. Testing Co., Inc. Unitech Services San Leandro, Cal. (EA 87-052)	June 17, 1987	Order Modifying License (Effective Immediately) Reason: Individuals performing radiographic operations without being properly trained/certified, unreported personnel overexposures, failure to accurately or completely record required information on utilitization logs, failure to perform radiation surveys, and performing an unauthorized source transfer.
Well Logging, Inc. Notice Nowata, Okla. (EA 87-099)	August 24, 1987	Confirmatory Order Modifying License and of Violation Reason: Failure to survey storage locations, job sites, and transportation vehicles, unauthorized use of licensed material, and failure to maintain various records.

Licensee	Date	Reason
Advanced Medical Systems, Inc. Geneva, Ohio (EA 87-139)	July 23, 1987	Order Modifying License (Effective Immediately) Reason: Decontamination of licensee's facility.
Veterans Administration Edward Hines, Jr. Medical Center Hines, Ill. (EA 87-150)	August 24, 1987	Order to Show Cause Why License Should Not Be Modified (Effective Immediately) Reason: Failure to report a diagnostic misadministration, actions to conceal that misadministration, and efforts to impede the NRC investigation.
Precision Materials Corp. Edison, N. J. (EA 87-156)	September 14, 1987	Order Modifying License (Effective Immediately) Reason: Uncertainty regarding the operation of the licensee's irradiator.
Log-Tec Cleveland, Okla. (EA 87-172)	September 14, 1987	Order Suspending License (Effective Immediately) Reason: Proprietor deceived an NRC inspector about the use of licensed material.
Finlay Testing Laboratories, Inc. Aiea, Hawaii (EA 87-186)	September 29, 1987	Order Suspending License (Effective Immediately) Reason: Transportation of licensed material in a passenger aircraft and failure to maintain proper records.

Appendix 7

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1987)

The following is a listing of the 124 nuclear power reactor electrical generating units which were in operation or under construction in the United States as of December 31, 1987, representing a total capacity of approximately 114,000 MWe, of which about 17,000 MWe was not yet operational. Reactor types are indicated as follows: BWR—boiling water reactor, PWR—pressurized water reactor, HTGR—high temperature gas-cooled reactor. 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, 1987. At that time, there were 110 commmercial nuclear reactors in the United States with operating licenses (including one, the Seabrook (N.H.) nuclear power plant, licensed to load fuel only), and 14 units for which construction permits were in effect (although construction of some of these has been postponed indefinitely).

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ALABAMA						
Decatur	Browns Ferry Nuclear Power Plant Unit 1	1,065	BWR	OL 1973	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant Unit 2	1,065	BWR	OL 1974	Tennessee Valley Authority	1975
Decatur	Browns Ferry Nuclear Power Plant Unit 3	1,065	BWR	OL 1976	Tennessee Valley Authority	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 1	804	BWR	OL 1977	Alabama Power Co.	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 2	814	PWR	OL 1981	Alabama Power Co.	1981
Scottsboro	Bellefonte Nuclear Plant	1,235	PWR	CP 1974	Tennessee Valley Authority	1993
Scottsboro	Bellefonte Nuclear Plant Unit 2	1,235	PWR	CP 1974	Tennessee Valley Authority	1995
ARIZONA						
Wintersburg	Palo Verde Nuclear Generating Station Unit 1	1,304	PWR	OL 1984	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 2	1,304	PWR	OL 1985	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 3	1,304	PWR	OL 1987	Arizona Public Service Co.	1988
ARKANSAS						
Russelville	Arkansas Nuclear One Unit 1	836	PWR	OL 1974	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One Unit 2	858	PWR	OL 1978	Arkansas Power & Light Co.	1980

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
CALIFORNIA						
San Clemente	San Onofre Nuclear Generating Station Unit 1	436	PWR	OL 1967	So. Calif. Ed. & San Diego Gas & Electric Co.	1968
San Clemente	San Onofre Nuclear Generating Station Unit 2	1,100	PWR	OL 1982	So. Calif. Ed. & San Diego Gas & Electric Co.	1983
San Clemente	San Onofre Nuclear Generating Station Unit 3	1,100	PWR	OL 1983	So. Calif. Ed. & San Diego Gas & Electric Co.	1984
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 1	1,084	PWR	OL 1984	Pacific Gas & Electric Co.	1985
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 2	1,106	PWR	OL 1985	Pacific Gas & Electric Co.	1986
Clay Station	Rancho Seco Nuclear Generating Station Unit 1	873	PWR	OL 1974	Sacramento Municipal Utility District	1975
COLORADO						
Platteville	Fort St. Vrain Nuclear Generating Station	330	HTGR	OL 1973	Public Service Co. of Colorado	1979
CONNECTICUT						
Haddam Neck	Haddam Neck Generating Station	555	PWR	OL 1967	Conn. Yankee Atomic Power Co.	1968
Waterford	Millstone Nuclear Power Station Unit 1	654	BWR	OL 1970	Northeast Nuclear Energy Co.	1971
Waterford	Millstone Nuclear Power Station Unit 2	864	PWR	OL 1975	Northeast Nuclear Energy Co.	1975
Waterford	Millstone Nuclear Power Station Unit 3	1,156	PWR	OL 1985	Northeast Nuclear Energy Co.	1986
FLORIDA						
Florida City	Turkey Point Station Unit 3	646	PWR	OL 1972	Florida Power & Light Co.	1972
Florida City	Turkey Point Station Unit 4	646	PWR	OL 1973	Florida Power & Light Co.	1973
Red Level	Crystal River Plant Unit 3	806	PWR	OL 1977	Florida Power Corp.	1977
Ft. Pierce	St. Lucie Plant Unit 1	817	PWR	OL 1976	Florida Power & Light Co.	1976
Ft. Pierce	St. Lucie Plant Unit 2	842	PWR	OL 1983	Florida Power & Light Co.	1983
GEORGIA						
Baxley	Edwin I. Hatch Plant Unit 1	757	BWR	OL 1974	Georgia Power Co.	1975
Baxley	Edwin I. Hatch Plant Unit 2	771	BWR	OL 1978	Georgia Power Co.	1979

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
GEORGIA—(continued)						
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 1	1,100	PWR	OL 1987	Georgia Power Co.	1987
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 2	1,100	PWR	CP 1974	Georgia Power Co.	1989
ILLINOIS						
Morris	Dresden Nuclear Power Station Unit 2	772	BWR	OL 1969	Commonwealth Edison Co.	1970
Morris	Dresden Nuclear Power Station Unit 3	773	BWR	OL 1971	Commonwealth Edison Co.	1971
Zion	Zion Nuclear Plant Unit 1	1,040	PWR	OL 1973	Commonwealth Edison Co.	1973
Zion	Zion Nuclear Plant Unit 2	1,040	PWR	OL 1973	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Station Unit 1	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Cordova	Quad-Cities Station Unit 2	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Seneca	LaSalle County Nuclear Station Unit 1	1,078	BWR	OL 1982	Commonwealth Edison Co.	1984
Seneca	LaSalle County Nuclear Station Unit 2	1,078	BWR	OL 1983	Commonwealth Edison Co.	1984
Bryon	Byron Station Unit 1	1,120	PWR	OL 1984	Commonwealth Edison Co.	1985
Byron	Byron Station Unit 2	1,120	PWR	OL 1986	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 1	1,120	PWR	OL 1986	Commonwealth Edison Co.	1988
Braidwood	Braidwood Unit 2	1,120	PWR	OL 1987	Commonwealth Edison Co.	1988
Clinton	Clinton Nuclear Power Plant Unit 1	950	BWR	OL 1986	Illinois Power Co.	1987
IOWA						
Pala	Duane Arnold Energy Center Unit 1	515	BWR	OL 1974	Iowa Elec. Power & Light Co.	1975
KANSAS						
Burlington	Wolf Creek	1,150	PWR	OL 1985	Kansas Gas & Elec. Co.	1985
LOUISIANA						
Taft	Waterford Steam Electric Station	1,151	PWR	OL 1984	Louisiana Power & Light Co.	1985
St. Francisville	River Bend Station Unit 1	934	BWR	OL 1985	Gulf States Utilities Co.	1986
MAINE						
Wiscasset	Maine Yankee Atomic Power	810	PWR	OL 1972	Maine Yankee Atomic Power Co.	1972

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
MARYLAND						
Lusby	Calvert Cliffs Nuclear Power Plant Unit 1	825	PWR	OL 1974	Baltimore Gas & Elec. Co.	1975
Lusby	Calvert Cliffs Nuclear Power Plant Unit 2	825	PWR	OL 1976	Baltimore Gas & Elec. Co.	1977
MASSACHUSETTS						
Rowe	Yankee Nuclear Power Station	175	PWR	OL 1960	Yankee Atomic Elec. Co.	1961
Plymouth	Pilgrim Station Unit 1	670	BWR	OL 1972	Boston Edison Co.	1972
MICHIGAN						
Big Rock Point	Big Rock Point Nuclear Plant	64	BWR	OL 1962	Consumers Power Co.	1963
South Haven	Palisades Nuclear Power Station	635	PWR	OL 1971	Consumers Power Co.	1971
Laguna Beach	Enrico Fermi Atomic Power Plant Unit 2	1,093	BWR	OL 1985	Detroit Edison Co.	1988
Bridgman	Donald C. Cook Plant Unit 1	1,044	PWR	OL 1974	Indiana & Michigan Elec. Co.	1975
Bridgman	Donald C. Cook Plant Unit 2	1,082	PWR	OL 1977	Indiana & Michigan Elec. Co.	1978
MINNESOTA						
Monticello	Monticello Nuclear Generating Plant	525	BWR	OL 1970	Northern States Power Co.	1971
Red Wing	Prairie Island Nuclear Generating Plant Unit 1	503	PWR	OL 1973	Northern States Power Co.	1973
Red Wing	Prairie Island Nuclear Generating Plant Unit 2	500	PWR	OL 1974	Northern States Power Co.	1974
MISSISSIPPI						
Port Gibson	Grand Gulf Nuclear Station Unit 1	1,250	BWR	OL 1982	Mississippi Power & Light Co.	1985
Port Gibson	Grand Gulf Nuclear Station Unit 2	1,250	BWR	CP 1974	Mississippi Power & Light Co.	Indef.
MISSOURI						
Fulton	Callaway Plant Unit 1	1,188	PWR	OL 1984	Union Electric Co.	1985
NEBRASKA						
Fort Calhoun	Fort Calhoun Station Unit 1	478	PWR	OL 1973	Omaha Public Power District	1973
Brownville	Cooper Nuclear Station	764	BWR	OL 1974	Nebraska Public Power District	1974

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
NEW HAMPSHIRE						
Seabrook	Seabrook Nuclear Station Unit 1	1,198	PWR	OL 1986	Public Service of N.H.	Indef.
Seabrook	Seabrook Nuclear Station Unit 2	1,198	PWR	CP 1976	Public Service of N.H.	Indef.
NEW JERSEY						
Toms River	Oyster Creek Nuclear Power Plant Unit 1	620	BWR	OL 1969	GPU Nuclear Corp.	1969
Salem	Salem Nuclear Generating Station Unit 1	1,079	PWR	OL 1976	Public Service Elec. & Gas Co.	1977
Salem	Salem Nuclear Generating Station Unit 2	1,106	PWR	OL 1980	Public Service Elec. & Gas Co.	1981
Salem	Hope Creek Generating Station Unit 1	1,067	BWR	OL 1986	Public Service Elec. & Gas Co.	1986
NEW YORK						
Indian Point	Indian Point Station Unit 2	864	PWR	OL 1973	Consolidated Edison Co.	1974
Indian Point	Indian Point Station Unit 3	891	PWR	OL 1975	Power Authority of the State of New York	1976
Scriba	Nine Mile Point Nuclear Unit 1	610	BWR	OL 1969	Niagara Mohawk Power Co.	1969
Scriba	Nine Mile Point Nuclear Unit 2	1,080	BWR	OL 1986	Niagara Mohawk Power Co.	1988
Ontario	R. E. Ginna Nuclear Power Plant Unit 1	470	PWR	OL 1969	Rochester Gas & Elec. Co.	1970
Brookhaven	Shoreham Nuclear Power Station	820	BWR	OL 1984	Long Island Lighting Co.	Indef.
Scriba	James A. FitzPatrick Nuclear Power Plant	810	BWR	OL 1974	Power Authority of the State of New York	1975
NORTH CAROLINA						
Southport	Brunswick Steam Electric Plant Unit 2	790	BWR	OL 1974	Carolina Power & Light Co.	1975
Southport	Brunswick Steam Electric Plant Unit 1	790	BWR	OL 1976	Carolina Power & Light Co.	1977
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 1	1,180	PWR	OL 1981	Duke Power Co.	1981
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 2	1,180	PWR	OL 1983	Duke Power Co.	1984
Bonsal	Shearon Harris Plant Unit 1	915	PWR	OL 1986	Carolina Power & Light Co.	1987

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
OHIO						
Oak Harbor	Davis-Besse Nuclear Power Station Unit 1	874	PWR	OL 1977	Toledo Edison-Cleveland Electric Illum. Co.	1977
Perry	Perry Nuclear Power Plant Unit 1	1,205	BWR	OL 1986	Toledo Edison-Cleveland Elec. Illum. Co.	1987
Perry	Perry Nuclear Power Plant Unit 2	1,205	BWR	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	Indef.
OREGON						
Prescott	Trojan Nuclear Plant Unit 1	1,080	PWR	OL 1975	Portland General Elec. Co.	1976
PENNSYLVANIA						
Peach Bottom	Peach Bottom Atomic Power Station Unit 2	1,051	BWR	OL 1973	Philadelphia Elec. Co.	1974
Peach Bottom Station Unit 3	Peach Bottom Atomic Power	1,035	BWR	OL 1974	Philadelphia Elec. Co.	1974
Pottstown	Limerick Generating Station Unit 1	1,065	BWR	OL 1984	Philadelphia Elec. Co.	1986
Pottstown	Limerick Generating Station Unit 2	1,065	BWR	CP 1974	Philadelphia Elec. Co.	1990
Shippingport	Beaver Valley Power Station Unit 1	810	PWR	OL 1976	Duquesne Light Co. Ohio Edison Co.	1976
Shippingport	Beaver Valley Power Station Unit 2	852	PWR	OL 1987	Duquesne Light Co. Ohio Edison Co.	1987
Goldsboro	Three Mile Island Nuclear Station, Unit 1	776	PWR	OL 1974	GPU Nuclear Corp.	1974
Berwick	Susquehanna Steam Electric Station Unit 1	1,052	BWR	OL 1982	Pennsylvania Power & Light Co.	1983
Berwick	Susquehanna Steam Electric Station Unit 2	1,052	BWR	OL 1984	Pennsylvania Power & Light Co.	1985
SOUTH CAROLINA						
Hartsville	H. B. Robinson S.E. Plant Unit 2	665	PWR	OL 1970	Carolina Power & Light Co.	1971
Seneca	Oconee Nuclear Station Unit 1	860	PWR	OL 1973	Duke Power Co.	1973
Seneca	Oconee Nuclear Station Unit 2	860	PWR	OL 1973	Duke Power Co.	1974
Seneca	Oconee Nuclear Station Unit 3	860	PWR	OL 1974	Duke Power Co.	1974
Broad River	Virgil C. Summer Nuclear Station Unit 1	900	PWR	OL 1982	So. Carolina Elec. & Gas Co.	1984
Lake Wylie	Catawba Nuclear Station Unit 1	1,145	PWR	OL 1984	Duke Power Co.	1985
Lake Wylie	Catawba Nuclear Station Unit 2	1,145	PWR	OL 1986	Duke Power Co.	1986

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
TENNESSEE						
Daisy	Sequoyah Nuclear Power Plant Unit 1	1,128	PWR	OL 1980	Tennessee Valley Authority	1981
Daisy Plant Unit 2	Sequoyah Nuclear Power	1,148	PWR	OL 1981	Tennessee Valley Authority	1982
Spring City	Watts Bar Nuclear Plant Unit 1	1,165	PWR	CP 1973	Tennessee Valley Authority	1988
Spring City	Watts Bar Nuclear Plant Unit 2	1,165	PWR	CP 1973	Tennessee Valley Authority	1989
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station Unit 1	1,150	PWR	CP 1974	Texas Utilities	1988
Glen Rose	Comanche Peak Steam Electric Station Unit 2	1,150	PWR	CP 1974	Texas Utilities	1989
Bay City	South Texas Nuclear Project Unit 1	1,250	PWR	OL 1987	Houston Lighting & Power Co.	1987
Bay City	South Texas Nuclear Project Unit 2	1,250	PWR	CP 1975	Houston Lighting & Power Co.	1989
VERMONT						
Vernon	Vermont Yankee Generating Station	504	BWR	OL 1972	Vermont Yankee Nuclear Power Corp.	1972
VIRGINIA						
Gravel Neck	Surry Power Station Unit 1	775	PWR	OL 1972	Va. Electric & Power Co.	1972
Gravel Neck	Surry Power Station Unit 2	775	PWR	OL 1973	Va. Electric & Power Co.	1973
Mineral	North Anna Power Station Unit 1	865	PWR	OL 1976	Va. Electric & Power Co.	1978
Mineral	North Anna Power Station Unit 2	890	PWR	OL 1980	Va. Electric & Power Co.	1980

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
WASHINGTON						
Richland	WPPSS No. 1 (Hanford) Supply System	1,266	PWR	CP 1975	Wash. Public Power	Indef.
Richland	WPPSS No. 2 (Hanford) Supply System	1,103	BWR	OL 1983	Wash. Public Power	1984
Satsop	WPPSS No. 3 Supply System	1,242	PWR	CP 1978	Wash. Public Power	Indef.
WISCONSIN						
Two Creeks	Point Beach Nuclear Plant Unit 1	495	PWR	OL 1970	Wisconsin Electric Power Co.	1970
Two Creeks	Point Beach Nuclear Plant Unit 2	495	PWR	OL 1971	Wisconsin Electric Power Co.	1972
Kewaunee	Kewaunee Nuclear Power Plant	515	PWR	OL 1973	Wisconsin Public Svc. Corp.	1974

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