U.S. Nuclear
Regulatory
Commission

1984 Annual Report

NRC

June 12, 1985

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

Dear Mr. President:

This Annual Report for 1984 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 1984, with additional treatment of events after that period where circumstances warranted.

Respectfully

Nunzio J. Palladino

Nunzio J. Palladino
Chairman

NRC

U.S. NUCLEAR REGULATORY COMMISSION

1984 Annual Report

PREVIOUS REPORTS IN THIS SERIES

1975 NRC Annual Report, published April 1976
1976 NRC Annual Report, published April 1977
NUREG-0400, *1977 NRC Annual Report,* published April 1978
NUREG-0516, *1978 NRC Annual Report,* published February 1979
NUREG-0690, *1979 NRC Annual Report,* published March 1980
NUREG-0774, *1980 NRC Annual Report,* published March 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
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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 relate to the benefits, costs, and risks of nuclear power." (See Chapter 1 for overall policy and planning guidance. Specific goals concerning nuclear power reactors are also discussed in Chapters 2 and 3; operating experience and assessment in Chapter 4; fuel cycle in Chapter 5; safeguards in Chapter 6; waste management in Chapter 7; inspection, enforcement and emergency preparedness in Chapter 8, nuclear nonproliferation in Chapter 10; and nuclear regulatory research in Chapter 11.)

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

"(1) insuring the safe design of nuclear power plants and other licensed facilities ...." (For reactors, see Chapters 2, 3 and 11; for materials facilities, devices and transportation packages, see Chapters 5 and 11; for waste facilities, see Chapters 7 and 11.)

"(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, 10, and 11.)

"(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 6, 8 and 11.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities ...." (See Chapters 1, 7 and 11.)

"(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 11.)

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 2.)

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 10.)

ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 170i 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 9.)

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 13.)

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.)

1984 Highlights/1985 Planning

CHAPTER 1

This is the 10th annual report of the U.S. Nuclear Regulatory Commission (NRC). The NRC was created by enactment of the Energy Reorganization Act of 1974 as an independent agency of the Federal Government. The five Commissioners are nominated by the President and confirmed by the U.S. Senate, and 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 and of the environment, the safeguarding of nuclear materials and facilities from theft and sabotage, and safe transport and disposal of nuclear materials and wastes. The NRC accomplishes its purposes through the licensing of nuclear reactor operations and other possession and use of nuclear materials, the issuance of rules and regulations governing licensed activities, and inspection and enforcement actions.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1984 (October 1983 through September 1984) 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. Other statutory reporting requirements related to the report are set forth on the preceding page.

This chapter deals with significant agency personnel changes and with certain noteworthy events which took place during the report period (and which are covered in greater detail within the body of the report). The report period is fiscal year 1984, i.e., October 1, 1983 to September 30, 1984; some coverage is given in this chapter to events occurring in the last quarter of the calendar year 1984. Also set forth in this chapter, in condensed form, is the policy and planning guidance for fiscal year 1985; the guidance document is drawn up yearly by the Commission and a copy is distributed to every member of the NRC staff.

Changes Within Commission and Senior Staff

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

Commissioner Victor Gilinsky's term ended on June 30, 1984, and on July 5, 1984, Lando W. Zech was appointed to the Commission, bringing it back to its full strength of five members.

In September 1984, Sharon R. Connelly was appointed Director, Office of Inspector and Auditor, succeeding James J. Cummings.

In October 1984, Robert D. Martin was appointed Regional Administrator of Region IV, Dallas, Texas, succeeding John T. Collins.

Noteworthy Events of 1984

The following are some of the more significant events or actions taken by the Nuclear Regulatory Commission during the report period:

Power Reactor Regulation. Seven operating licenses were granted during fiscal year 1984, five of them being full power authorizations. This brought the total number of licensed power reactors in the United States to 86. The overall volume of licensing activity during this time comprised more than 2,400 separate actions. The period also saw the introduction of integrated implementation scheduling for licensing actions and initiation of a pilot integrated safety assessment program (see "Improving the Licensing Process" in Chapter 2). Technical resolutions for 13 generic safety issues were completed in fiscal year 1984, and priorities for the resolution of 20 others were defined. Of the 27 "Unresolved Safety Issues" which have been identified, final technical resolution has been achieved for 15. Many of the remaining 12 are nearing final resolution. (See Chapter 2.) The reactor vessel head of Three Mile Island Unit 2 (Pa.) was successfully removed during the period, ahead of schedule (see Chapter 3). The quality assurance study mandated by the Congress was carried out and submitted to Congress in April 1984 (see Chapter 8).

Inspection and Enforcement. About 2,300 inspections of operating power reactors were conducted during the report period; special attention was given to maintenance and surveillance activity. Inspections of power reactor units under construction totaled about 1,400. On the enforcement side, a total of 135 cases were handled during the fiscal year. The NRC issued 73 civil penalties for licensee violations, with assessments that totaled over $2.3 million. (See Chapter 8.)

On July 5, 1984, Lando W. Zech, Jr., was appointed to the Commission for a five-year term, filling the vacancy left by the departure of Commissioner Victor Gilinsky on June 30, 1984. Commissioner Zech, a retired U.S. Navy Vice Admiral and former Deputy Chief of Naval Operations for Personnel, had earlier served as commanding officer of both surface ships and submarines, including the nuclear submarine U.S.S. Nautilus and the fleet ballistic missile nuclear submarine U.S.S. John Adams.

More than 250 safeguards inspections were conducted at power reactor sites during the report period, plus 119 at fuel cycle facilities and 41 at non-power reactor locations. (See Chapter 6.)

NRC Regionalization Complete. In a policy statement issued in February 1984, the Commission declared that its program to enlarge the role of the NRC Regional Offices in certain licensing and support functions formerly handled in NRC Headquarters is essentially complete. (See Chapter 13 of this report and the *1983 NRC Annual Report*, pp. 150-152, for background and details.) A pilot program involving decentralizing certain technical reviews of operating reactor license amendments is still under way and will be reviewed by the Commission following a two-year trial. In its policy statement, the Commission announced that project managers in the Office of Nuclear Reactor Regulation would not be transferred to the Regions, except that the project manager for Fort St. Vrain (Colo.) will remain in Region IV (Dallas) and limited licensing authority would be exercised out of that Regional Office. The Commission also announced that non-power reactor licensing would not be decentralized, nor would license fee management.

It was the judgment of the Commission that the major goals of decentralization had been realized: better coordination between licensing staff and regional inspectors, better service to applicants and licensees, better liaison with State and local government and with the public, strengthened capability for responding to incidents, and others.

Responding to Transport Accidents. In March 1984, the Commission issued a policy statement on the NRC's role in responding to accidents and incidents involving the transportation of nuclear materials. Under the law, the NRC is authorized to license and regulate the receipt, possession, use and transfer of nuclear material (whether source, byproduct or special material). The Department of Transportation (DOT) is legally required to regulate for safety in the transport of hazardous materials, including radioactive materials. A Memorandum of Understanding adopted in June 1979 delineates the roles of the two agencies in regulating the transport of nuclear materials, but it does not define the specific responsibilities of each in dealing with transportation accidents or incidents, beyond identifying the NRC as "lead agency" for investigating the cause of any actual or reported leak of radioactive material. The policy statement, issued for comment, prescribes that when an accident or incident involving radioactive material is reported to DOT's National Response Center, the NRC will, on notification by DOT, respond by taking the following actions: call the State agency responsible for controlling a transportation accident site to protect public health and safety and ensure that it is aware of the situation; offer technical assistance in the form of information, advice and evaluation to the State officials; assure that the Department of Energy and other affected Federal agencies are aware of the situation; monitor the situation until normality is restore provide information in response to any queries regarding NRC-approved packages; if the shipper is an NRC licensee, ensure that the shipper gives complete and accurate information to authorities; take on the "lead agency" role in investigating any instances of actual or suspected leakage of radioactive materials regulated by the NRC; make recommendations, as requested, to emergency response personnel regarding radioactive hazards.

Financial Qualification Revision. In 1982, the Commission eliminated its existing requirement that a review establishing the financial qualifications of an applicant for both a construction permit and operating license be carried out as part of the licensing process. That action was challenged in the D.C. Circuit and, in February 1984, the court remanded the rule back to the Commission, finding it inconsistent for various reasons. In April 1984, the Commission proposed revised financial qualification requirements that would, in effect, reinstate the previous requirement that applicants for a construction permit be financially qualified to conduct the activities authorized by the permit. There being no pending applications for a permit, the Commission felt that this approach would afford the NRC an opportunity to give further study to the subject while still responding to the court's remand. At the operating license stage, however, the Commission continued to believe that the regulated status of electric utilities constituted a reliable basis for determining financial qualification and that a case-by-case review was unnecessary. Thus the revised rule would not require financial qualification review for operating license applicants who are regulated public utilities whose rates are set by State commissions to permit recovery of all reasonable costs of serving the public, or who are authorized to set their own rates. All others would be subject to case-by-case appraisal. (See "Judicial Review," Chapter 12.)

The Commission is also seeking public comment on an alternative approach which would completely eliminate financial qualification reviews for all license or permit applicants, on the ground that past experience indicates that such reviews probably do not provide any significant additional assurance of safety for the public.

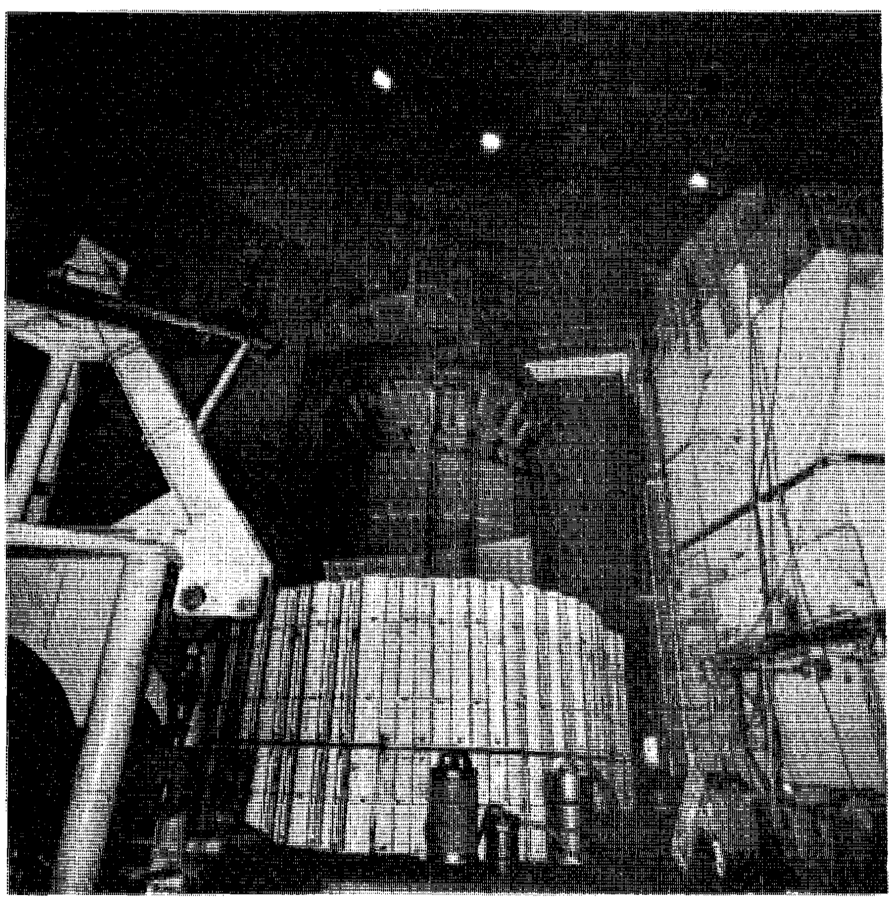
Changing the Licensing Process. The NRC's Regulatory Reform Task Force was formed in November 1981 to examine the NRC's licensing process for the design, siting, construction and operation of nuclear power plants and other nuclear facilities. Legislative proposals submitted to the Congress in 1983 were largely the product of the task force's efforts. In April 1984, the task force issued a draft report to the Commission containing further suggestions for reform of the licensing process. The report was sent for review to two groups established for that purpose—the internal Senior Advisory Group and, from outside the agency, the Ad Hoc Committee for the Review of Nuclear Reactor Licensing Reform Proposals. In April 1984, the Commission decided to seek public comment on the package of suggestions before deciding how to proceed with them. The Commission was especially interested in having reactions to these suggestions:

(1) Establishment of a screening board to determine if a hearing should be held, to rule on petitions to intervene in a hearing, and to rule on the admissability of contentions in a hearing.

(2) Improvements in the conduct of the hearing process itself by control of discovery (information gathering); changes in the way evidence is presented, with emphasis on written submissions; a provision which would permit the establishment of panels of technical experts to determine if there is a technical basis for concluding that a proposed contention raises a genuine issue of disputed fact; and a limitation on Licensing Board authority to raise, on its own initiative, issues to be considered in a hearing.

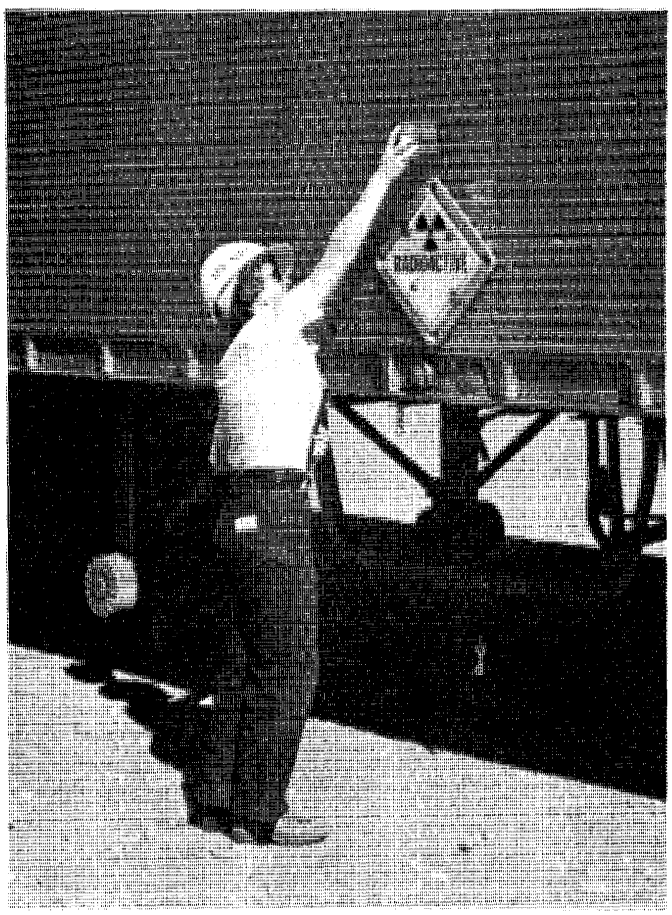
(3) Improvements in the decision-making process by elimination of the Appeal Boards as independent appeal

Highlighting activities associated with the Three Mile Island Unit 2 cleanup during 1984 was the successful removal (ahead of schedule) and storage of the reactor vessel head in July. The vessel head and service structure in shielded storage are shown here. Shielding is provided by surrounding sand columns.



tribunals, leaving that function to the Commission itself; prohibition of consideration of generic factual issues in more than one hearing involving a similar reactor or facility; limiting intervenors to the filing of findings of fact and conclusions of law—or filing exceptions to initial decisions—only on issues placed in controversy by the intervening party; and elimination of the current requirement that an Appeal Board or the Commission review initial decisions authorizing reactors to operate at more than 5 percent of full power before the initial decision can be made effective.

Emergency Planning Requirements Revised. In August 1984, certain changes in emergency planning requirements went into effect. Under the new provisions, each State and local government with jurisdiction within 10 miles of a nuclear plant is expected to test its emergency plan every two years, instead of once a year, as before. The new rule provides for remedial exercises if the emergency plan is not satisfactorily carried out during the biennial exercise.



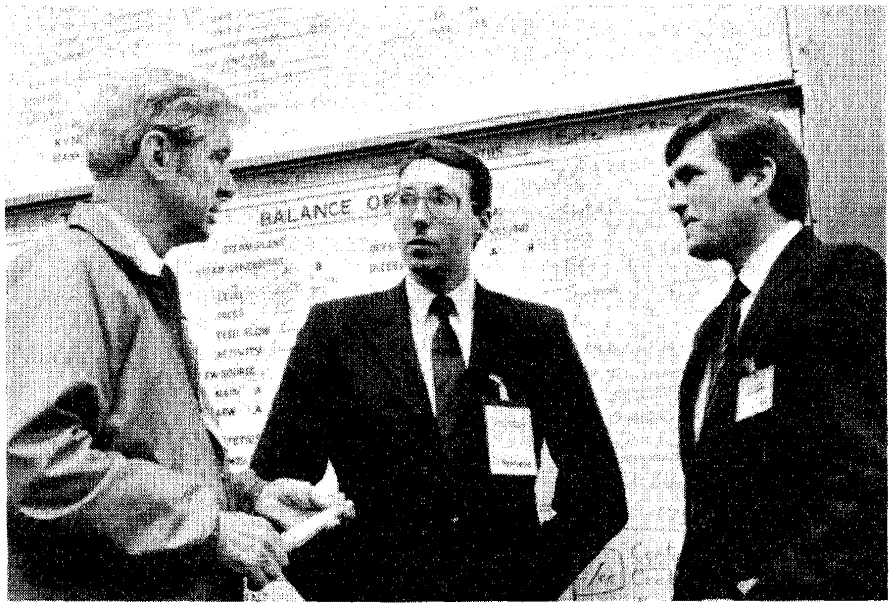
The roles of two Federal Government agencies in regulating the transportation of radioactive materials were clarified in March 1984 when the Commission issued a policy statement further defining the responsibilities of the NRC and the Department of Transportation in dealing with accidents or incidents. This photo shows an NRC inspector monitoring the markings on a radioactive material carrier.

Proposed Changes in Backfitting Rule. Backfitting is a process involving either plant-specific change or changes applicable to one or more classes of licensed nuclear facilities. In December 1984, the Commission proposed to amend its requirements regarding the backfitting of commercial power reactors and certain other licensed nuclear facilities. The revised rule would define backfitting as the imposition of new regulatory requirements, or the modification of previous regulatory requirements applicable to a facility, by means other than rulemaking after (1) the date of issuance of a construction permit (if such comes after the effective date of a final rule), or (2) six months before docketing of an application for an operating license (if a construction permit were issued before the effective date of a final rule), or (3) the date of issuance of the operating license for a facility.

The existing rule requires backfitting if the agency "finds that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security." The proposed rule would require backfitting only when the agency "determines, based on a systematic and documented analysis of the relevant and material factors...that there is a substantial increase in the overall protection of the public health and safety or the common defense and security to be derived from the backfit and the direct and indirect cost of implementation for that facility are justified in view of this increased protection." The factors considered to be "relevant and material" in the evaluation include, among others, (1) the potential reduction in risk to the public from the accidental off-site release of radioactive materials, (2) the potential impact on radioactive exposures of facility employees, (3) the installation and continuing costs associated with the backfit, including the cost of downtime or of construction delay, (4) the potential safety impact in terms of increased complexity in the plant or in operations, including the effect on other proposed or existing requirements, (5) the estimated impact on NRC resources associated with a backfit, and the availability of such resources, (6) the potential impact of the differences in facility types, designs, and age in assessing the relevancy or practicality of a proposed backfit, and (7) whether the proposed backfit is interim or final and, if interim, the justification for imposing an interim backfit requirement. The new rule would also require a backfitting analysis when new requirements are imposed on licensees through issuance of an amendment to a license.

The Commission could, however, decide that an immediate imposition of a backfit requirement, without the prescribed systematic analysis, was warranted in a given instance to protect public health and safety or the common defense and security.

All requirements, including backfit requirements, proposed by the NRC staff related to one or more classes of reactors must be reviewed by the Committee to Review Generic Requirements (CRGR). The committee seeks to eliminate unnecessary demands on licensees by ensuring that the need for a new requirement can be demonstrated



August 1984 saw the NRC implement changes in nuclear plant emergency planning requirements. Earlier, NRC Commissioners and senior representatives of the Federal Emergency Management Administration (FEMA) visited full field exercises across the country. In the photo shown, NRC Commissioner James Asselstine and FEMA Deputy Administrator Bernard McGuire are briefed by NRC Region II Administrator James P. O'Reilly during a test of the emergency response plan for the St. Lucie Nuclear Plant in Florida.

by those proposing it. (See the *1982 NRC Annual Report*, pp. 1-3, for full description of CRGR's structure and review process.) Through its review, the CRGR seeks assurance that a proposed requirement is (1) necessary for public health and safety, (2) likely to result in a net safety improvement, and (3) 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 (EDO) that the proposed requirement be approved, disapproved, modified, or conditioned in some way. It also makes recommendations as to the method and scheduling of implementation. The EDO considers CRGR recommendations, as well of those of cognizant NRC offices, in deciding whether a requirement shall be imposed. From its inception in November 1981 through December 1984, the CRGR has held 70 meetings and considered a total of 114 separate issues.

Policy and Planning Guidance for 1985

In order to set forth the principles underlying its regulatory policies, enunciate its major policies and major objectives, and thereby provide a common basis for the developing of programs, setting of priorities, and allocating of resources throughout the agency, the Commission yearly publishes a Policy and Planning Guidance document. The document seeks to provide the guidance whereby NRC staff offices can develop program plans and objectives that are consistent with the Commission's purposes and with one another. The document is distributed to every member of the NRC staff.

The following is a condensed treatment of basic themes in the NRC policy and planning guidance for 1985.

Regulatory Philosophy. The basic mission of the NRC is to regulate those who make use of or produce nuclear materials for commercial purposes so that they give adequate protection to the public health and safety, the common defense and security, and the environment. The Commission recognizes that, in carrying out this mission, its actions can affect the nation's energy supplies and the interdependent energy supply system of which nuclear energy is a significant part. Consistent with that recognition, the Commission will continue to pursue predictability and stability in the licensing process and will impose new requirements on existing licensees only in accordance with NRC backfitting policy (see discussion earlier in this chapter). The nuclear industry could also pursue the benefits of greater stability and predictability by commitment to the development of standardized nuclear plant design.

The Commission believes that it has an obligation to license a nuclear project which has satisfied the rigorous reviews and requirements of the licensing process, but it also believes it must provide an accessible avenue for the expression of public concerns and for an adequate response to them.

Over the next two years (and depending on the availability of resources), the NRC's larger objectives include implementing a safety goal; achieving technical resolution of current unresolved safety issues; seeking licensing reform legislation, and implementing reforms which have been identified; completing the reassessment of radioactive source terms and, if appropriate, implementing a revised and more realistic source term and revising existing regulations as warranted; developing and implementing a policy on severe accidents; setting forth a procedure for review and approval of standard plant designs and preapproval of plant sites; developing procedures for reactivating a project, should an applicant utility desire to do so, on which construction and licensing activity had been suspended; determining an appropriate way to incorporate industry safety initiatives into nuclear regulation; pursuing improvements in construction quality assurance policy; ensuring that adequate and timely actions are taken to carry out NRC's responsibilities under the Nuclear Waste Policy Act of 1982; and implementing a policy for early review of advanced reactor concepts and designs.

For the balance of the 1980s, the NRC will revise regulations so as to emphasize performance rather than prescriptive requirement, adopt permanent procedures for the review and approval of standardized plant designs, consider organizational changes which reflect long-term objectives, and—to the extent warranted—modify relevant regulations in accordance with new source terms; the NRC will also place greater reliance on industry self-regulation.

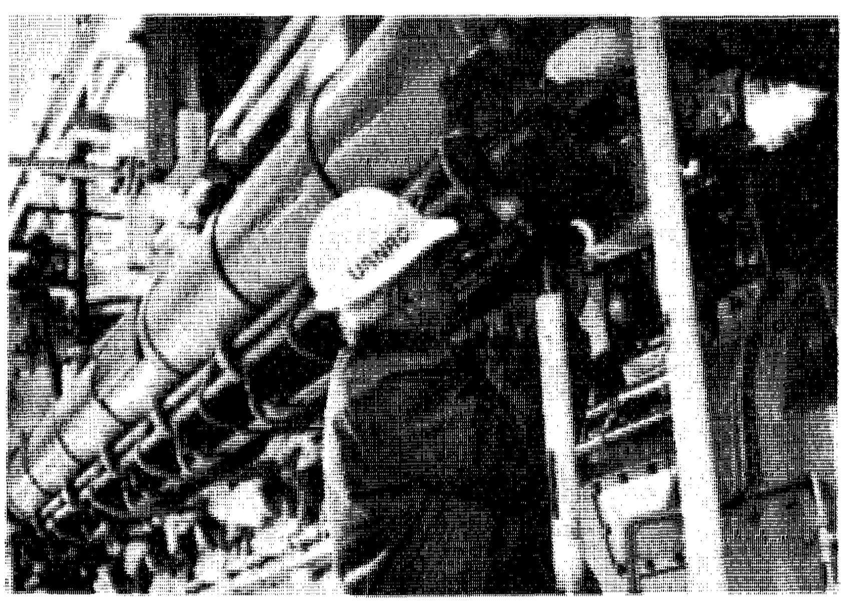
The six themes put forward in the 1985 policy and planning guidance elaborate on some of the objectives cited above. A brief discussion of each of these themes follows.

Assuring Safe Operation of Facilities. The NRC's fundamental task is, as it has been from the inception of the agency in January 1975, to make sure that existing nuclear power plants and those coming on line operate safely. Assuring that these facilities are adequately designed, built and tested prior to operation, and that operating facilities maintain adequate levels of protection, remains the highest priority of the agency. While the industry bears the prime responsibility for safety in all phases of design, construction and operation of nuclear plants, the NRC should give high priority to the development of commercial reactor operating expertise within the agency through training, hiring and close communication with industrial experts. The formulation of a severe accident policy and the early resolution of outstanding technical issues are of major importance in this area.

Planning Guidance: On-site inspection of operating reactors should continue to focus on plant operations, including maintenance activities. The regular analysis of operational data and systematic assessment of licensee performance will be used to help focus NRC activities, to provide for a more efficient allocation of resources, and to assess the licensee's management of its plant. Priority attention is to be given those licensees with low performance ratings. The NRC will closely monitor the first two years of new plant operation, especially if the licensed utility has little or no prior experience in nuclear plant operation. Through 1985, the agency should actively pursue the recruitment of experienced personnel and also provide appropriate training opportunity to the staff in reactor plant operations. A report to the Commission on the results of this effort will be due by the end of the year.

The collecting, analyzing and disseminating of operational data remains of great importance, as does the emphasis on quality assurance at every phase of building and running a nuclear facility. The NRC should emphasize to licensees that theirs is the responsibility to assure that their vendor-supplied equipment and services are adequately inspected and their quality assured. By its own inspection effort, the NRC will satisfy itself that both licensees and vendor organizations are meeting their responsibilities. By the end of 1985, the staff will, to the extent practicable, issue draft technical resolutions for currently identified unresolved safety issues for public comment. The staff should expeditiously implement the Commission's severe accident policy.

Improving Regulation of the Nuclear Industry. A number of subject areas related to improved regulation of the industry are covered in the policy and planning guidance for 1985. Some discussion of each follows.



NRC Policy and Planning Guidance for fiscal year 1985, developed and refined during 1984, stressed that on-site inspections of operating reactors will focus on maintenance activities as well as other plant operations. NRC inspector Ted Rebelowski is shown during an inspection at the Millstone Nuclear Power Station, Waterford, Conn.

• **New Requirements.** The NRC must continue to be aware of the large volume of requirements imposed on the nuclear industry and to be careful that each new proposed requirement represents a positive contribution to safety in itself and in the context of the entire body of regulations. Licensees should be allowed the flexibility to select the most cost-effective ways of satisfying NRC safety objectives, particularly for plant-specific requirements. In general, safety issues which affect numerous licensees should be addressed in the context of rulemaking rather than case-by-case. And unresolved safety issues should be promptly and energetically addressed, according to priority of safety significance.

Planning Guidance: The Committee to Review Generic Requirements (CRGR) should continue to review proposed requirements and make recommendations on them to the Executive Director for Operations, who has the overall authority and responsibility for managing backfit requirements. (See above.) Existing requirements should be reviewed to see if some could be eliminated without compromise to safety and to see that those which are necessary and effective are being implemented in a timely manner. Implementation schedules should be worked out with each licensee and take into account the licensee's other priorities and ability to perform the necessary engineering, evaluation and design. Where practicable, cost-benefit analysis should be employed.

• **Preparing to License Future Facilities.** The Commission intends to reconsider the legislative package forwarded to the Congress in February 1983 before resubmission of the proposals to the 99th Congress in 1985.

Planning Guidance: The Regulatory Reform Task Force will give support to the reappraisal of the legislative package by the Commission and continue its examination of the hearing stages of the licensing process to find ways of making them more efficient and effective. The NRC staff should develop procedures for the licensing of projects presently postponed (which may necessitate further legislative proposals).

• **Standardization.** The NRC recognizes the several advantages to the development and use of standardized nuclear power plant designs. The standardized approach will redound to the benefit of public health and safety in a number of ways: by concentrating the resources of designers, engineers and vendors on particular approaches to design problems; by stimulating standardized programs of construction practice and quality assurance; by improving the training of personnel and fostering more effective maintenance and operation; and by providing for more efficient and effective licensing and inspection.

Planning Guidance: The NRC should, by 1986, develop the capability to review and license new standardized nuclear plant designs and to review and pre-approve potential plant sites. For the rest of the 1980s, the NRC shall maintain that capability. In general, the NRC should study how best to bring about standardized nuclear plants and propose ways to the Commission by which to encourage industry moves in that direction.

• **Investigations.** On request of the Commission, the Executive Director for Operations or a Regional Administrator, or on its own initiative, the Office of Investigations shall investigate significant allegations of wrongdoing by other than NRC employees or contractors. Any evidence of possible criminal activity uncovered in the course of an investigation shall be referred to the Department of Justice.

Planning Guidance: The Office of Investigations should, by mid-1985, develop criteria for initiating and terminating an investigation, in coordination with the Executive Director for Operations. Information uncovered in the course of an investigation which has safety implications should be referred to the appropriate NRC office immediately.

• **Enforcement.** The NRC seeks a firm and fair enforcement policy, applied efficiently and uniformly through the Regional Offices, which will assure compliance with NRC requirements; appropriate corrective action by licensees, when that is indicated; and a greater likelihood of future compliance. The licensee must not benefit by violating regulations. Credit should be given for the prompt reporting of deficiencies and correction thereof.

Planning Guidance: The report of the Commission's Ad Hoc Advisory Committee for Review of the Enforcement Policy is to be published in 1985. The committee will be reviewing comments and recommendations on the effectiveness of the policy by the staff, licensees and the public.

• **Timely Licensing of Facilities.** The NRC intends that its regulatory process be an efficient and cost-effective one, and that unwarranted delay will be eliminated with no compromise to an assurance of adequate safety. The Commission reaffirms its 1981 statement urging Licensing Boards to take positive steps toward the more efficient conduct of licensing hearings.

Planning Guidance: Staff reviews and public hearings should not be a factor that could unnecessarily delay the startup of a completed nuclear facility.

• **Safety Goals.** The two-year trial period for the proposed safety goals and related guidance began in 1983 and ends in 1985 (see *1982 NRC Annual Report*, pp. 4 and 7). These preliminary goals were not to be used as a basis for regulatory decisions during the trial period.

Planning Guidance: The staff will provide the Commission with its recommendations on proposed safety goals by early 1985, together with a detailed discussion of their regulatory implications. The NRC will continue to work toward the objective of defining to the industry and the public the acceptable limits for nuclear plant risks. The staff should propose revisions to NRC regulations in light of the safety goal, where indicated, to reflect the more general performance objectives for nuclear power plants.

• **Radioactive Source Terms and Siting Policy.** Before proceeding with new siting regulations for nuclear plants, the Commission decided first to seek a better definition of the safety goals (see preceding item) and a more accurate characterization of the source terms, i.e., the inventories of radioactive materials that could be released in nuclear reactor accidents.

Planning Guidance: A systematic analysis of the release and transport of radioactive material should yield a more accurate understanding of the source terms associated with postulated reactor accidents. A draft reassessment of these should be available by early 1985, and, if warranted, implementation of revised source terms and re-evaluation of regulations should begin by the end of the year. Any need for revision of siting rules or of other existing or proposed regulations, e.g., emergency preparedness, should also be evaluated once the source terms have been validated by an effective peer review procedure.

- **Transportation.** The transportation of nuclear and radioactive materials is an important area of NRC regulatory responsibility.

Planning Guidance: The staff should assure that NRC regulatory activity in the transportation of radioactive materials is coordinated with that of other Federal agencies, pursuant to an integrated Federal program that protects public health and safety without unwarranted impact on the regulated industry.

- **Advanced Reactors.** The Commission is looking to approve essentially complete standard plant designs and, while not taking part in the development of new designs, will maintain the capability to review and appraise such designs as may be proposed. The Commission will make known which factors it considers important for advanced reactor concepts, in order to minimize uncertainties in the process.

Planning Guidance: The Commission will establish an advanced reactor policy by March 1985. The staff will develop draft guidelines on changes to general design criteria and regulations for advanced reactors by the end of 1986.

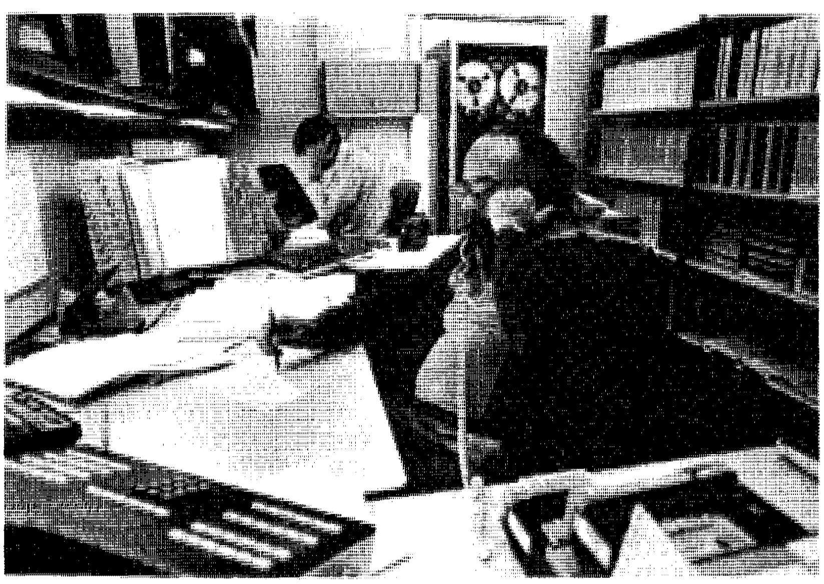
- **Protecting Nuclear Material and Facilities.** In the area of domestic safeguards, regulations should be based on the same defense-in-depth philosophy that governs safety regulations. The emphasis should be on performance requirements rather than on prescriptive measures, thus allowing the licensee to select the most cost-effective ways to meet NRC requirements. Implications of all safeguard requirements for overall safety shall be evaluated.

In the area of international safeguards, the NRC will carefully discharge its licensing responsibility to ensure effective controls are applied to the import and export of nuclear materials, equipment and facilities.

Planning Guidance (Domestic): NRC staff evaluation of domestic and foreign experience in protecting nuclear materials from theft or sabotage will continue to be the main basis for changes in safeguards regulations, though necessary change can be made at any time. Staff will continue its independent assessment seeking assurance that licensee safeguards plans meet NRC objectives and also that NRC regulations do in fact support those objectives. The staff will report annually to the Commission on its assessments and also on comments received regarding the use of low-enriched fuel in research reactors.

Planning Guidance (International): The NRC should continue to facilitate timely processing of export license applications to nations which adhere to effective nonproliferation policies and will continue to meet commitments for implementation of international safeguards at U.S. licensed facilities. The Commission continues to endorse a reduction to the maximum extent possible of the use of highly enriched uranium in both domestic and foreign reactors, as affirmed in its policy statement of August 1982.

- **Nuclear Materials.** Nuclear materials must receive regulatory attention commensurate with the hazard they



As a matter of explicit policy and planning for fiscal year 1985, the Commission and the NRC will seek a more accurate characterization of the radioactive materials that might be released in a nuclear reactor accident. Once this is done, other regulations—particularly emegency preparedness guidelines—may be revised. The NRC recently upgraded the capabilities of its Operations Center (shown here) to respond to such emergencies.

present to the public and to the users of such materials. Materials regulation should emphasize performance requirements over prescriptive requirements.

Planning Guidance: Regulations to consolidate and streamline the safety requirements associated with the medical use of byproduct nuclear materials and well-logging should be promulgated by the end of 1985, together with regulatory guidance, standard review plans and inspection procedures. Efforts to improve radiography safety—especially through establishment of performance standards—and to improve training and inspections programs should be completed by July 1986.

- **Cleaning Up Three Mile Island Unit 2.** The expeditious and safe cleanup of the damaged reactor at Three Mile Island Unit 2 (Pa.) remains high among NRC safety priorities. The staff will continue to maintain oversight of activities at the site and is ready, if necessary, to give direction to ensure a safe decontamination of the facility and timely removal of radioactive materials. The NRC should work closely with the Department of Energy (DOE) to get whatever technical information on severe accidents can be garnered from study of the reactor core.

Planning Guidance: The TMI Program Office staff of the NRC will continue monitoring the cleanup of the plant, and also the activities undertaken pursuant to the agreement under which the DOE is removing and disposing of solid nuclear wastes. The staff should help assure they are safely and expeditiously removed from the site and should assist the DOE in developing plans for off-site disposition of the damaged core.

- **Managing Nuclear Waste.** Nuclear waste management is a critical concern of the agency and the nation. The NRC will give the needed licensing and regulatory oversight to the program of the Executive Branch, as provided in the Nuclear Waste Policy Act of 1982 (see Chapters 5 and 7). The NRC will act on the premise that, in the absence of unresolved safety concerns, the regulatory program will not delay implementation of the Executive Branch's program. To that end, NRC staff will keep in close communication with DOE and others involved and thus identify required actions and lead times early in the planning process. Should it appear that, for lack of resources or other reasons, schedules cannot be maintained, the staff will promptly inform the Commission so that the required notification of DOE and the Congress can be made. The staff shall also monitor implementation of the low-level radioactive waste legislation and apprise the Commission of any problems calling for Commission action, with recommendations for such.

Planning Guidance: The staff shall assess the need for a general Memorandum of Understanding with DOE to spell out the interactive roles of the two agencies in implementing the Nuclear Waste Policy Act of 1982. The results of the assessment should be reported to the Commission by mid-1985. The staff should review existing and proposed regulations and make whatever changes are indicated to bring them into conformance with the Act. A similar canvass should be made in this are when the standards of the Environmental Protection Agency are published. The staff should make timely review of any utility proposals for adding spent fuel capacity for interim storage and assure that there will be no regulatory delay in the matter affecting reactor operations.

- **Research.** The purpose of the NRC research program is to provide a technical basis for rulemaking and regulatory decisions, to support licensing and inspection activities, to assess the feasibility and effectiveness of proposed safety improvements, and to increase understanding of those phenomena relevant to regulatory actions for which analytic methods are needed. There should be continued emphasis on obtaining research results useful to the regulatory process, as distinct from those which, though of intrinsic interest, are extraneous to or of limited relevance to pressing regulatory problems and issues. The severe accident research program must provide timely information for the Commission's decision-making process in this area.

Planning Guidance: Budgetary resources allocated for research should go to support a program balanced between research to reinforce or revise current regulatory bases and conceptual research leading to improvements in reactor safety, waste management and other licensed activities. The staff should be alert to research results indicating a need for changes in regulations, either because they are too stringent or not stringent enough. Wherever possible, the NRC should join with or coordinate its research with programs of industry groups, other government agencies or foreign research projects.

Reactor Regulation

CHAPTER 2

The Office of Nuclear Reactor Regulation (NRR) is responsible for reviewing applications for construction permits and operating licenses for nuclear reactors and for issuing such permits and licenses after consideration by the Advisory Committee on Reactor Safeguards, Atomic Safety and Licensing Boards and Appeal Boards, and the Commission. It is also responsible for regulation of operating reactors. These functions require resolution of generic and specific issues with regard to safety, the environment, and antitrust matters.

This chapter summarizes NRR activities during fiscal year 1984, under the following headings: Status of Licensing, Improving the Licensing Process, Human Factors, Unresolved Safety Issues, Safety Reviews, Protecting the Environment, and Antitrust Activities. Also included in this chapter is a section on the activities of the Advisory Committee on Reactor Safeguards.

Status of Licensing

Applications For Operating Licenses For Power Reactors

Seven power reactor facilities were licensed during fiscal year 1984. One of these received a fuel load license, and one received a low-power license only; three facilities received low-power licenses followed by full-power licenses, and one facility received a full-power license. In addition, one facility had its low-power license reinstated and subsequently received a full-power license; this licensing action was challenged in court (see below). Four safety evaluation reports, six draft environmental impact statements and three final environmental impact statements also were issued during the fiscal year. All plants under construction have operating license applications under review; the reviews are targeted for completion on a schedule consistent with plant completion. There are currently 43 units undergoing an operating license review. Some of these plants have been indefinitely delayed and eventually may be cancelled. Eight units were cancelled during fiscal 1984.

Several applications have experienced special problems which are covered later in this chapter, under "Safety Reviews."

Applications For Construction Permits Or Manufacturing Licenses

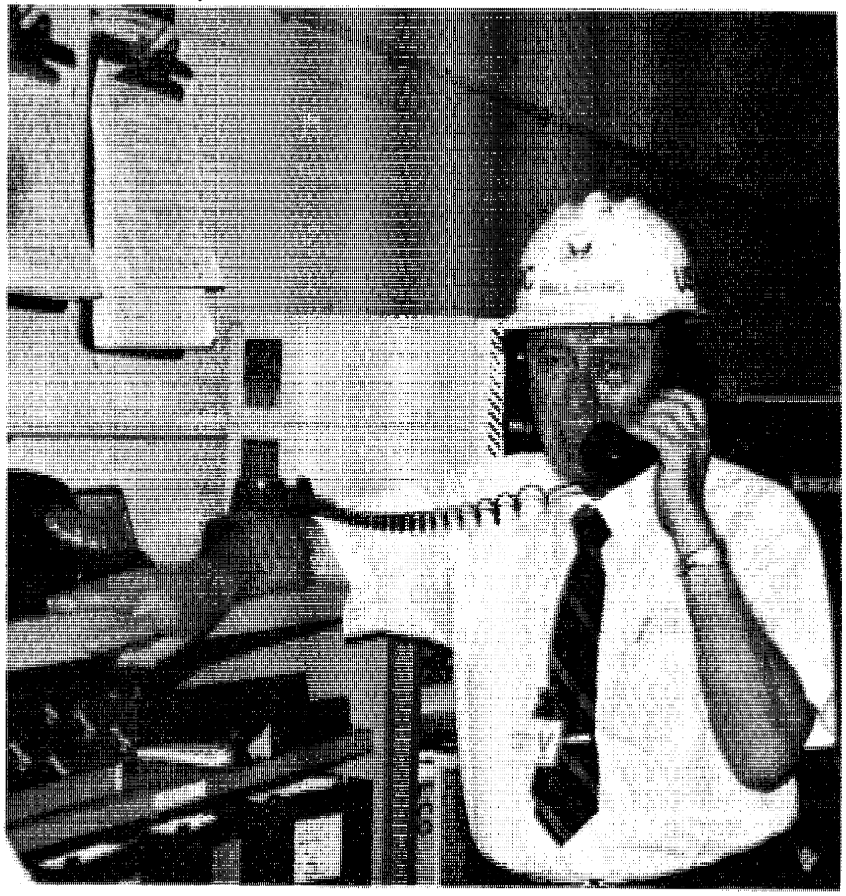
No construction permits were issued during fiscal year 1984, nor was there any activity related to a manufacturing license. Utilities announced the cancellation of the following eight units for which construction permits had been issued: Hartsville Units A1 and A2 (Tenn.); Clinton Unit 2 (Ill.); Riverbend Unit 2 (La.); Yellow Creek Units 1 and 2 (Ala.); Shearon Harris Unit 2 (N.C.); and Zimmer Unit 1 (Ohio). There are no construction permit applications under review.

Licensing Actions for Operating Power Reactors

By the end of fiscal year 1984, 86 power reactors were licensed to operate. Several types of post-licensing actions can affect operating reactors, including license amendment requests, public hearings, exemption requests to regulations, new regulations which are specifically backfitted on operating reactors, orders for modification of a license, new generic activities, and review of information supplied by a licensee for the resolution of technical issues. With the publication of the "Clarification of TMI Action Plan Requirements" (NUREG-0737) in fiscal year 1981, the inventory of these kinds of actions increased dramatically, up to approximately 5,400 by the beginning of fiscal year 1982. To reduce this inventory, the NRC established strong management controls over the issuance of new requirements and assigned additional resources to the review of pending actions. However, unexpected events—such as the trip-breaker problem at the Salem (N.J.) facility—have created a large number of additional licensing actions which, when added to those normally needed for each licensee, has extended the time required to reduce the inventory of pending licensing actions to desired steady-state levels. The inventory had been reduced to approximately 3,800 items by the end of fiscal year 1984.

Licensing Actions for Non-power Reactors

There were 66 non-power reactors in use for research, training, and test purposes, licensed for operation by the NRC, at the start of the fiscal year. In addition, 19 applica-



NRC Chairman Nunzio J. Palladino visited the Grand Gulf nuclear power plant at Port Gibson, Miss., in July 1984, prior to a Commission vote on a full power license. The Commission granted the full power license on July 31, 1984.

tions for operating license renewals were awaiting either the start or completion of staff review. During fiscal year 1984, renewals were issued for 11 of these operating licenses, two applications were withdrawn and one new application was received. The seven renewal application remaining were under active review and scheduled for completion during the first half of fiscal year 1985. Elimination of the backlog permits starting reviews of all future applications for renewals as they are received. Three new operating license renewal applications and and one new construction permit application are expected during fiscal year 1985.

Three contested renewal proceedings have been ongoing for the past several years. During fiscal year 1984, one of these was resolved and concluded; hearings on the renewal for the second were terminated because the renewal application was withdrawn, and any further hearing activity is on hold pending submittal and approval of a decommissioning plan; and the third was in the prehearing stage at the end of the fiscal year. Two new requests for decommissioning/license termination were submitted during the fiscal year, one license was terminated and 12 facilities were either in the process of decommissioning or in a possession-only (non-operating) status.

Special Cases

Restart of TMI-1. The Commission has determined that the plant design and procedures are adequate for purposes of restarting Three Mile Island Unit 1 (Pa.). The Commission had previously determined (fiscal year 1983) that emergency planning for the facility is adequate. The remaining issues in contention, management competence and integrity, were before the Commission at the close of the report period.

During 1984, the Appeal Board remanded three of these management issues to the Licensing Board for further hearings: (1) the adequacy of licensee's training program, (2) the May 9, 1979 mailgram from licensee official Herman Dieckamp to Congressman Udall concerning the "pressure spike" during the TMI-2 accident, and (3) reactor coolant system leak rate testing practices at TMI-1 prior to the March 1979 accident at TMI-2. In addition, the Appeal Board had previously remanded to the Licensing Board certain allegations from a former control room operator charging improprieties in determining testing practices related to reactor coolant system leak rate at TMI-2 prior to the accident. Concurrent with the ongoing Licensing Board remanded proceedings, the Commis-

Table 1. Licenses Issued in FY 1984 for Operation of Nuclear Power Plants

APPLICANT	*FACILITY*	*FUEL LOAD*	*LOW POWER*	*FULL POWER*	*LOCATION*
Duke Power Co. et. al.	Catawba-1	07/18/84	-	-	York Co., S. Carolina
Pennsylvania Power & Light Co.	Susquehanna-2	-	03/23/84	06/27/84	Berwick, Pa.
Union Electric Co.	Callaway-1	-	06/11/84	-	Steedman, Mo.
Commonwealth Edison Co.	LaSalle-2	-	12/16/83	03/23/84	LaSalle Co., Ill.
Washington Public Power Supply System	WNP-2	-	12/20/83	04/13/84	Richland, Wash.
Mississippi Power & Light Company	Grand Gulf-1	-	-	08/31/84	Clairborne Co., Miss.
Pacific Gas & Electric Co.	Diablo Canyon-1	-	04/13/84*	09/06/84**	San Luis Obispo, Cal.

*Reinstated
**Challenged in U.S. Court of Appeals

sion undertook review of these issues, in September 1984, to determine whether further hearings are warranted and, if so, what their scope should be. The modifications and repairs affecting plant readiness are essentially complete. In July 1984, a Licensing Board conducted evidentiary hearings on the operability of the repaired steam generators. The board's decision was pending at the close of the report period. During 1984, NRC staff reviewed nine investigations conducted by the NRC Office of Investigations into matters identified as relevant and material to an evaluation of the TMI-1 licensee's management integrity. Based upon this review, the NRC staff concluded that the TMI-1 licensee can and will operate TMI-1 without undue risk to the health and safety of the public.

Indian Point Hearings. Hearings were held in 1983 by an Atomic Safety and Licensing Board (ASLB) to determine whether Indian Point Units 2 and 3 (N.Y.) should be shut down or other action taken. The record was closed on April 29, 1983.

The Commission's questions before the board addressed the character and magnitude of the risk to public health and safety posed by severe reactor accidents at Indian Point Units 2 and 3. To respond to these questions, the licensees published, in March 1982, a full-scope probabilistic risk assessment. The NRC staff commissioned a review and reanalysis of the character and likelihood of severe reactor accidents based upon the licensees' submittal. The staff also performed an independent analysis of the radiological releases and off-site consequences associated with a spectrum of postulated severe reactor accidents.

On October 24, 1983, the ASLB issued its findings, conclusions and recommendations. The Commission subsequently, on November 8, 1983, solicited comments on the board's findings and recommendations from the parties to the Indian Point Special Inquiry. These comments were received by the Commission on February 6, 1984. Several public Commission meetings were held in early summer and fall of 1984 regarding the ASLB findings and comments by the parties. The Commission review was continuing at the close of the report period.

Limerick Hearings. Hearings were held in 1984 by an Atomic Safety and Licensing Board (ASLB) to assess the environmental impact of severe reactor accidents at the Limerick Generating Station, Units 1 and 2 (Pa.). To support its application for an operating license for Limerick, the Philadelphia Electric Company had submitted a plant-specific probabilistic risk assessment which included external accident causes—such as earthquakes, fires and floods—as well as internal plant failures. This assessment is the first done for a boiling water reactor at a densely populated site, and the first for a Mark II containment. For these hearings, the NRC staff commissioned a thorough review of severe accidents, based upon the applicant's probabilistic risk assessment. The staff performed an independent analysis of the probabilities of a spectrum of representative severe reactor accidents, and of the radiological releases and the off-site consequences thereof. These releases and consequences were evaluated by methods derived from the Reactor Safety Study. On August 29, 1984, the ASLB issued its findings permitting Limerick to proceed towards obtaining an Operating License.

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 Guide 1.70), 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 important 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.

Clinch River Breeder Reactor. On October 26, 1983, the U.S. Senate tabled an amendment which would have authorized continued funding of the Clinch River Breeder Reactor (CRBR) project. Accordingly, the applicants (the Department of Energy, the Tennessee Valley Authority, and Project Management Corporation) terminated the project and formally notified NRC of this decision on December 27, 1983. NRC activities associated with CRBR licensing were subsequently terminated. A review of the applicants' plan for site redress was conducted by NRR and an approval letter was issued on June 5, 1984. Termination of the CRBR legal proceedings is awaiting final action from the ASLB.

Improving the Licensing Process

Standardization

On December 21, 1983, the NRC issued a Final Design Approval (FDA) for Combustion Engineering's System 80 nuclear steam supply system reference design, as described in CESSAR-F. This FDA, the second issued by the NRC, allows CESSAR-F to be referenced in operating license applications for plants that referenced the System 80 Preliminary Design Approval (PDA) at the construction permit stage of the licensing process. During fiscal year 1984, the staff continued its review of the application by the General Electric Company for approval of the severe accident portion of its BWR/6 Nuclear Island Design, as described in GESSAR II. Also during fiscal year 1984, Westinghouse Electric Corporation continued technical discussions with the NRC, and on October 24, 1983, tendered an application for a PDA for its Advanced Pressurized Water Reactor design, as described in RESAR-SP/90. The NRC found the application acceptable and it was docketed on May 19, 1984.

Decentralization

Responsibility for the review of about 149 licensing actions was transferred to the five Regional Offices of the NRC during fiscal year 1984. This brings the total number of licensing action reviews transferred to the regions since fiscal year 1982 to approximately 506. These reviews include inservice testing, emergency exercise exemption requests, organizational changes, snubber surveillance, degraded grid voltage testing, and plant-specific issues. The region conducts technical reviews, makes site visits when appropriate, and prepares Safety Evaluation Reports for the Office of Nuclear Reactor Regulation. A two-year pilot program to evaluate the effectiveness of these regional reviews was initiated in July 1984 with the issuance of NUREG-1075, "Decentralization of Operating Reactor Licensing Reviews."

In fiscal year 1983, responsibility for review of changes made to a facility's security plan under section 50.54(p) of the regulations was transferred to Regions I and II. In fiscal year 1984, this responsibility was transferred to the remaining regions. Also in fiscal year 1983, the licensing authority for all Fort St. Vrain Nuclear Generating Station licensing actions, except those involving generic issues or exemptions to regulations, was delegated to Region IV. This is the first instance of regionalized reactor licensing authority; it is being carefully evaluated to determine whether program objectives are being met. Decisions on further decentralization of reactor regulation are currently in abeyance pending results of evaluations of the program to date.

Continued hearings on possible risks to public health and safety posed by the Consolidated Edison's Indian Point Units 2 and 3 (N.Y.) resulted in a number of licensee proposals which were under consideration by the Commission at year's end. NRC inspectors made repeated visits to the plants to inspect emergency plans and drills. Here, Thomas E. Murley, NRC Regional Administrator for Region I, heads a Regional response team at an Indian Point emergency plan exercise.

Coordination of Regulatory Requirements

The NRC staff has taken steps to integrate implementation schedules for new requirements with the schedules for existing requirements. The program is designed to take into account the overall effect of these schedules on plant operation and utility resources. Participation in the program is voluntary on the part of the licensee. The lead plant for this program is the Duane Arnold facility in Iowa, which was issued a license amendment in 1983 to implement a plan providing integrated scheduling of plant modifications. In 1984, the Pilgrim plant (Mass.) joined the program and was issued an appropriate license amendment. The NRC staff is encouraging other licensees to negotiate similar arrangements; the staff is currently reviewing plans for integrating schedules submitted by several other utilities.

Backfitting

On June 22, 1983, the Commission approved a set of directions to the NRC staff for controlling plant-specific backfitting measures required of licensees of operating nuclear power reactors. The Commission directed the staff to develop an appeal process to provide an opportunity for operating reactor licensees to discuss any areas of disagreement with a staff-proposed requirement. The Commission also directed the staff to conduct a study of the feasibility of and alternatives for applying backfit controls to plants for which a construction permit, but not an operating license, has been issued.

The staff developed a set of procedures for managing plant-specific backfitting requirements for operating reactors. Subsequently, a policy statement describing actions the Commission had taken to control backfitting and an advance notice of proposed rulemaking were published for public comment in the *Federal Register* on September 28, 1983.

In October 1983, the staff was directed to use these procedures for managing plant-specific backfitting requirements for operating reactors on an interim basis. In early 1984, the procedures were sent to both applicants and licensees by NRR. These procedures stipulate that each proposed requirement for improvement of safety involving a new staff position or a change in an existing staff position must be approved by the appropriate NRC management before it is sent to the licensee. The licensee then has the option of using the appeal process, if he disagrees with the requirement. The appeal process consists of three stages: the first is recourse to the appropriate Assistant Director of the Division of Licensing, and the second is to the Director of the Division of Licensing. If the outcome of this appeal process is unsatisfactory to the licensee, he can then appeal to the Director of Nuclear Reactor Regulation (NRR). The Director, NRR, will then request the staff to prepare a cost-benefit analysis of the requirement which will be forwarded to the licensee. The Director, NRR, will decide upon the suitability of the backfit after consideration of the cost-benefit analysis and other pertinent information. In early 1984, the NRC staff developed a system to track the status of each backfitting item for operating reactors and licenses under review. The progress of the resolution of these items is reflected in a monthly status report to the Director, NRR.

The interim procedures were published in the *Federal Register* on April 20, 1984, for public comment. These comments have been reviewed and have led to some minor changes in the procedures. The staff is currently implementing these modified interim procedures. The Commission will be reviewing the modified interim procedures and providing direction to the staff on the adoption of a final set of procedures.

Priorities of Generic Safety Issues

The NRC continued to use the methodology described in the *1982 NRC Annual Report* (p. 29) for determining the priority of generic safety issues. In December 1983, a comprehensive list of the issues subjected to this method was published in NUREG-0933, "A Prioritization of Generic Safety Issues." This list includes items from the TMI Action Plan (NUREG-0660) and Unresolved Safety Issues (USIs), which are discussed in more detail later in this chapter. Priorities for other issues were published in Supplement 1 to NUREG-0933 in July 1984. The results of the NRC's continuing effort in identifying significant safety issues to be resolved will be included in future Supplements to NUREG-0933.

Thirty-five new generic safety issues were identified in fiscal year 1984, including 16 Human Factors Program Plan items. Priorities for 20 issues were established in fiscal year 1984 and these issues are listed in Table 2. Other than USIs, 13 issues were resolved in fiscal year 1984 and these are listed in Table 3. Currently, the total number of generic safety issues that remain unresolved stands at 77. The schedules for the resolution of these issues are shown in Table 4.

Advanced Reactors

Growing interest among the Commissioners, the Department of Energy and the industry in the design and deployment of advanced reactors with greater inherent safety than the current generation of light-water reactors (LWRs) was in evidence in 1984. Accordingly, early in the year, the Commissioners initiated consideration of a policy statement on advanced reactors which would provide guidance on future activities in this area. In addition, an Advanced Reactors Group (ARG) was established within NRR to act as the focal point for future interaction with DOE and industry on the design and licensing requirements for these advanced designs (i.e., liquid metal cooled reactors, gas cooled reactors and water cooled reactors which differ significantly from current generation LWRs). The ARG is also to provide guidance to NRC research regarding required support in the area of advanced reactors. The initial phase of interaction with DOE and industry was begun in 1984 and was primarily directed toward familiarizing the NRC with the various advanced reactor concepts under evaluation. Plans for future interaction were also discussed.

Human Factors

A major concern of the NRC is with those activities in which human performance is a key element in the safe operation and maintenance of nuclear power plant equipment or facilities. These include staffing and qualifications of personnel, training, licensing of operators, procedures, man-machine interfaces, and the management and organization of plants.

Revision 1 to the NRC Human Factors Program Plan was published in September, 1984 (NUREG-0985, Revision 1), to develop the technical bases for regulatory action. During fiscal year 1984, the plan activities were reviewed by the NRC Human Factors Review Group (HFRG) to assure coordination of human factors activities throughout the NRC. As part of the coordination function, NRC Human Factors activities were discussed in detail with the Institute for Nuclear Power Operations (INPO), the Electric Power Research Institute (EPRI) and the Department of Energy (DOE). The objective of these discussions was to ensure that related industry and government human factors developments were not duplicated by the NRC, and that each group would be cognizant of the others' activities. Each of these groups—INPO, EPRI and DOE—was invited to participate in the regular Program Reviews held by the HFRG. In 1984, the nuclear power industry established the Nuclear Utility Management and Human Resources Committee (NUMARC). NRC human factors activities were discussed with NUMARC representatives, and the industry group began participation in HFRG program reviews and other related actions.

Professional Activities

To continue improving the quality of its technical knowledge, the NRC staff participates in professional, industrial, and international working groups and meetings. The staff participated in meetings this year dealing with nuclear operations (the American Nuclear Society), psychology (the American Psychological Association), human factors (the Human Factors Society) and artificial intelligence (the American Association for Artificial Intelligence). The staff also participated in efforts to identify industrial standards, involving the American Nuclear Society's Reactor Operations and Support Systems Subcommittee (ANS-3), the American Society of Mechanical Engineers' Operations and Maintenance Committee, and the Human Factors Society Standards Committee. Internationally, the NRC staff has exchanged information on national human factors programs in meetings of the Committee for Safety of Nuclear Installations of the Organization for Economic Cooperation and Development.

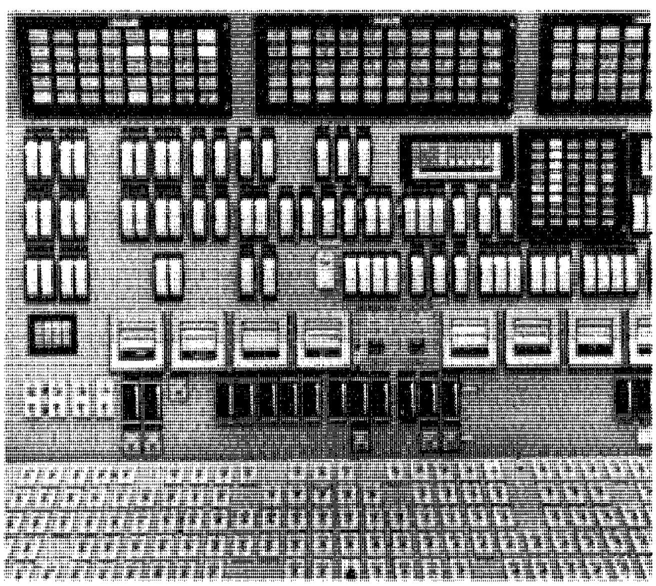
Operational Safety

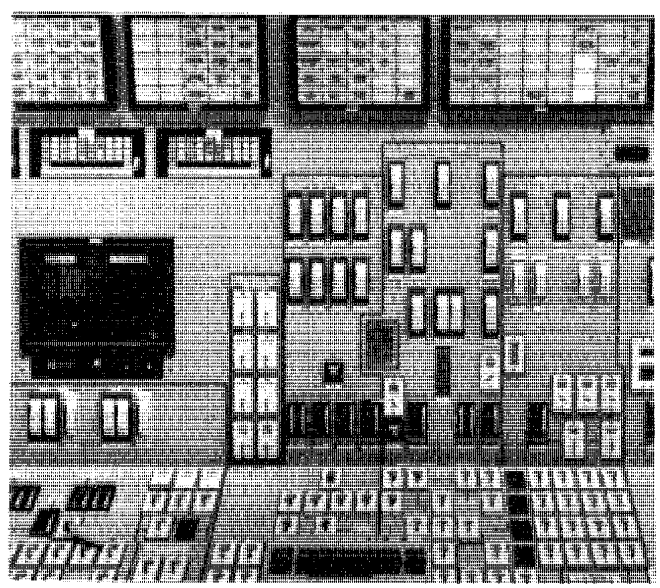
The NRC staff regularly investigates the contributions, both positive and negative, that plant personnel make to reactor plant events. These efforts focus on such human factors issues as operator qualifications, training, staffing, and also operational aids, such as procedures. Last year, a single event—the automatic reactor trip failure at the Salem (N.J.) plant—generated major human factors concerns (see *1983 NRC Annual Report*, p. 42). In contrast, this year there were several events of lesser significance in the human factors area. For example, the staff investigated the human factors component in the issue of the reliability of diesel generators at Shoreham (N.Y.) and Grand Gulf (Miss.), the main generator hydrogen explosion at Rancho Seco (Cal.), the steam generator boil-dry at Davis-Besse (Ohio), the loss of AC power during start-up testing at Susquehanna (Pa.), and the reactor trip with complications at Trojan (Ore.). All of these events involved human factors performance to some extent, and both the NRC and the industry are coming to recognize that human factors can play a sometimes very significant role in reactor events. Both will continue to support the application of human factors principles in promoting and maintaining operational safety.

Staffing and Qualifications

The NRC staff, in 1984, prepared a Final Commission Policy Statement (SECY-84-355) regarding Engineering Expertise on Shift. This policy statement allows licensees to combine the functions of the senior operator and the shift technical advisor, thus permitting the integration of engineering expertise into the normal operating crew.

In addition, NRC staff developed a proposed rule on training and qualifications (SECY-84-76). The qualifications element of the proposed rule would require licensees and applicants to ensure that all operating personnel have qualifications appropriate to the performance requirements of the job to which they are assigned. Regulatory Guide 1.8 will endorse the minimum qualification standards established by ANSI/ANS 3.1-1981 as a means of implementing the new rule. However, Regulatory Guide 1.8 does allow for a determination of qualifications using a systems approach to training, which is the reason for the training element of the proposed rule.





A review group of the Nuclear Utility Management and Human Resources Committee (NUMARC) made recommendations that resulted in changes in control room arrangements and panel markings at the Shearon Harris Unit 2 plant at Bonsal, N.C. These photos show the facilities before the changes (at left) and afterward (at right).

Table 2. Issues Prioritized in FY 1984 (20)

Number	*Title*	*Priority*
34	RCS Leak	DROP
35	Degradation of Internal Appurtenances in LWRs	LOW
36	Loss of Service Water	NEARLY-RESOLVED
43	Contamination of Instrument Air Lines	DROP
44	Failure of Saltwater Cooling System	COVERED IN 43
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	NEARLY-RESOLVED
49	Interlocks and LCOs for Redundant Class 1E Tie Breakers	MEDIUM
53	Consequences of a Postulated Flow Blockage Incident in a BWR	DROP
60	Lamellar Tearing of Reactor Systems Structural Supports	COVERED IN USI A-12
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	MEDIUM
66	Steam Generator Requirements	NEARLY-RESOLVED
68	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	HIGH
69	Make-up Nozzle Cracking in B&W Plants	NEARLY-RESOLVED
70	PORV and Block Valve Reliability	MEDIUM
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NEARLY-RESOLVED
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	LOW
82	Beyond Design Basis Accidents in Spent Fuel Pools	MEDIUM
90	Technical Specifications for Anticipatory Trips	LOW
92	Fuel Crumbling During LOCA	LOW
B-65	Iodine Spiking	LOW

Note: HIGH, MEDIUM and NEARLY-RESOLVED priority issues are allocated resources for resolution. DROP and LOW priority issues are not allocated resources for resolution.

Table 3. Generic Safety Issues Resolved in FY 1984

Number	*Title*
12	BWR Jet Pump Integrity
20	Effects of Electromagnetic Pulse on Nuclear Plant Systems
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System
45	Inoperability of Instrumentation due to Extreme Cold Weather
50	Reactor Vessel Level Instrumentation in BWRs
69	Make-up Nozzle Cracking in B&W Plants
B-10	Behavior of BWR Mark III Containments
B-26	Structural Integrity of Containment Penetrations
B-60	Loose Parts Monitoring System
I.A.1.4	Operating Personnel and Staffing: Long-term Upgrading
II.A.1	Siting Policy Reformulation
II.E.5.2	Transient Response of B&W Designed Reactors
III.D.2.5	Public Radiation Protection Improvement: Office Dose Calculation Manual

During the report period, the NRC issued Generic Letter 84-16 on the adequacy of on-shift experience for Near Term Operating License Applicants which endorsed an industry proposal on the subject, with several clarifications.

The NRC is working with the industry's NUMARC group (see above) to develop non-prescriptive approaches in this and other areas involving "human factors."

Training

The NRC staff continued work on various proposed versions of the training and qualifications rulemaking, which was set in motion by Section 306 of the Nuclear Waste Policy Act of 1982 (P.L. 97-425). Because of certain industry concerns regarding the rulemaking, publication of a draft rule has been delayed. The Commission continued to ask for modifications and staff responses to individually proposed versions of a rule or policy statement.

The staff made several visits to utility training programs undergoing review by an industry (INPO) Accreditation Team. The visits afforded the staff a better understanding of the industry's accreditation process and will be useful in formulating a strategy for evaluating the adequacy of the accreditation program to assure improved performance of utility personnel through training.

Operator Licensing

The operator licensing function was decentralized prior to the start of fiscal year 1984. All licensing examinations are now scheduled and administered through the NRC regional offices. During fiscal year 1984, 500 new licenses and 438 license renewals were issued for reactor operators. For senior operators, 667 new licenses and 891 license renewals were issued. There were also 127 instructor certifications granted. In addition, the Regional Offices conducted requalification examinations at 23 facilities.

Significant experience was gained in the use of the new Operator Licensing Examiner Standards (NUREG-1021). Feedback from the regional offices on use of the standards has led to revisions both to improve their clarity and to extend the guidance to non-power reactor operator license candidates. To assure consistent application of the examiner standards for administering examinations at power reactors, audits of contract examiners and program reviews of each regional office were conducted. Based on these pilot audits, the staff developed assessment standards intended to document the criteria and methods for evaluating the operator licensing function in the regions. These standards are expected to be used by the staff for audits conducted in fiscal year 1985.

Table 4. Generic Issues Scheduled for Resolution

A. NRR Issues

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
14	PWR Pipe Cracks	NEARLY RESOLVED	12/84+
22	Inadvertent Boron Dilution Event	NEARLY RESOLVED	10/84
23	Reactor Coolant Pump Seal Failures	HIGH	07/86
29	Bolting Degradation or Failure in Nuclear Power Plants	HIGH	09/85
36	Loss of Service Water (Calvert Cliffs Unit 1)	NEARLY RESOLVED	05/86
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	NEARLY RESOLVED	02/86
49	Interlocks and LCOs for Class 1E Tie Breakers	MEDIUM	09/88
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems	MEDIUM	10/86
61	SRV Discharge Line Break Inside the Wetwell Airspace of BWR Mark I and Mark II Containments	MEDIUM	03/87
65	Component Cooling Water System Failure	HIGH	07/86
66	Steam Generator Requirements	NEARLY RESOLVED	09/85
68	Loss of AFWS Due to AFW Steam HELB	HIGH	10/87
70	PORV and Block Valve Reliability	MEDIUM	12/84+
77	Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains	HIGH	08/86
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convention Cooldown	MEDIUM	02/86
82	Beyond Design Bases Accidents in spent Fuel Pools	MEDIUM	09/86
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	MEDIUM	06/86
A-30	Adequacy of Safety Related DC Power Supplies	HIGH	07/85+
A-41	Long Term Seismic Program	MEDIUM	10/84
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	MEDIUM	09/86

+Schedules may extend beyond date shown.

Table 4. Generic Issues Scheduled for Resolution
(continued)

A. NRR Issues

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
B-6	Loads, Load Combinations, Stress Limits	HIGH	10/84+
B-54	Ice Condenser Containments	MEDIUM	10/84
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	05/85
B-56	Diesel Reliability	HIGH	01/85
B-58	Passive Mechanical Failures	MEDIUM	10/84+
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	09/87
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	05/86
C-8	Main Steam Line Isolation Leakage Control	HIGH	12/86
C-11	Assessment of Failure and Reliability of Pumps and Valves	MEDIUM	10/84+
I.A.2.2.	Training and Qualifications of Operations Personnel	HIGH	04/85+
I.A.2.6	Long-Term Upgrading of Training and Qualifications	–	–
(1)	Revised Regulatory Guide 1.8	HIGH	04/85+
(4)	Operator Workshops	MEDIUM	09/85
I.A.2.7	Accreditation of Training Institutions	MEDIUM	01/85
I.A.3.4.	Licensing of Additional Operations Personnel	MEDIUM	09/85
I.A.4.2. (4)	Review Simulators for Conformance	HIGH	04/85+
I.B.1.1.	Organization and Management of Long Term Improvements*	–	–
(1)	Prepare Draft Criteria	See HFPP 6.1 and 6.2	
(2)	Prepare Commission Paper	See HFPP 6.1 and 6.2	
(3)	Issue Requirements for the Upgrading Management and Technical Resources	See HFPP 6.1 and 6.2	
(4)	Review Responses to Determine Acceptability	See HFPP 6.1 and 6.2	
HFPP—6.1	Establish NRC Position on Management and Organization at Operating Nuclear Power Plants	MEDIUM	11/85+

*Human Factors Program Plan Issues No. 6.1 and 6.2 have replaced TMI Issues I.B.1.1 (1 through 4).

Table 4. Generic Issues Scheduled for Resolution
(continued)

A. NRR Issues

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
HFPP—6.2	Development of NRC Assessment Materials for Evaluating Management and Organization at Nuclear Power Plants	MEDIUM	02/86
I.C.9	Long-Term Program for Upgrading of Procedures	MEDIUM	07/85
I.D.3	Safety System Status Monitoring	MEDIUM	12/84+
I.G.2	Scope of Test Program	MEDIUM	10/84
II.B.6	Risk Reduction for Operator Reactors at Sites with High Population Densities	HIGH	11/84
II.C.2	Continuation of Interim Reliability Evaluation Program	HIGH	07/85
II.E.4.3	(Containment) Integrity Check	HIGH	12/84+
II.E.6.1	Test Adequacy Study	MEDIUM	11/86
III.D.2.3 (1)	Develop Procedures to Discriminate Between Sites/Plants	NEARLY RESOLVED	10/84
(2)	Discriminate Between Sites and Plants that Require Condition of Liquid Pathway Interdiction Techniques	NEARLY RESOLVED	10/84
(3)	Establish Feasible Method of	NEARLY RESOLVED	10/84
(4)	Prepare a Summary Assessment	NEARLY RESOLVED	10/84
III.D.3.1	Radiation Protection Plans	HIGH	09/85
IV.E.5	Assess Currently Operating Plants	HIGH	07/85

B. Non-NRR Issues

3	Set Point Drift in Instrumentation	NEARLY RESOLVED	06/85
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NEARLY RESOLVED	02/86
B-17	Criteria for Safety Related Operator Actions	MEDIUM	10/84+
I.A.3.3.	Requirement for Operation Fitness	HIGH	01/85
I.A.4.2 (1)	Research on Training Simulators	HIGH	03/85
I.B.1.1. (6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	MEDIUM	02/86
I.B.1.1. (7)	Issue Regulatory Guides 1.33 and 1.8	MEDIUM	02/86

Table 4. Generic Issues Scheduled for Resolution
(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
I.D.4	Control Room Design Standard	MEDIUM	11/84
I.D.5 (3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	12/87+
I.D.5 (5)	Disturbance Analysis Systems	MEDIUM	11/84+
I.F.1	Expand QA List	HIGH	09/85+
II.B.5 (1)	Behavior of Severely Damaged Fuel	HIGH	06/85
II.B.5 (2)	Behavior of Core Melt	HIGH	06/85
II.B.5 (3)	Effect of Hydrogen Burning and Explosions of Containment Structure	MEDIUM	09/85+
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	HIGH	12/84
II.C.1	Interim Reliability Evaluation Program	HIGH	01/85
II.C.4	Reliability Engineering	HIGH	12/87+
II.E.2.2.	Research on Small Break LOCA and Anomalous Transients	MEDIUM	09/86+
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	MEDIUM	9/84+
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	06/86+
II.J.4.1	Revise Deficiency Report Requirements	NEARLY RESOLVED	06/85+
III.A.1.3 (2)	Maintain Supplies of Thyroid-Blocking Agent for Public	NEARLY RESOLVED	10/84+
III.A.3.4	Nuclear Data Link	MEDIUM	10/84+

Efforts to develop a valid, job-related examination and examination procedure continued during the report period. The NRC completed a catalog of the kinds of knowledge and ability required of PWR reactor operators and senior operators and an examiner's handbook for developing examinations. The examiner standards will be revised to reflect this information pending completion of a similar BWR catalog and pilot testing of the examiner's handbook. In addition, the computerized examination question bank has been modified to enhance its usefulness to all examiners. Use of the catalog, handbook, and question bank are expected to improve the reliability and efficiency of the NRC examination process.

Emergency Operating Procedures

The NRC long-term program for upgrading emergency operating procedures (EOPs) is in the implementation stage. Shortly after the TMI accident, the staff embarked on an effort to improve the technical accuracy and completeness of the EOPs and to incorporate human factors principles in the presentation of the technical material. Last year, owners groups for all four vendors of nuclear power plants had satisfactorily reanalyzed accidents and transients and developed generic technical guidelines for their plants. Coordinated with this effort was the issuance of the NRC's long-term plan requiring all plants to revise EOPs based on approved technical guidelines and NRC guidance on incorporating effective human factors practices into procedure design.

The industry-wide implementation of revised EOPs has made significant progress during the fiscal year. The required Procedures Generation Packages for over half of the plants have been submitted to the NRC. These packages describe the applicant/licensee's programs for adapting the generic technical guidelines to the individual plant and the plans to use human factors considerations in the production of procedures. The NRC program allows implementation of upgraded procedures prior to completion of NRC review, so all plants are expected to be using revised procedures within two years.

The review of the generic technical guidelines last year identified certain technical issues to be resolved. During the year, the NRC staff initiated programs working with the owners groups to encourage further improvements in accident recovery strategies. To date, three of the four vendor owners groups have submitted revised technical guidelines, and these are under review. The BWR Owners Group revision was approved with a few more items to be addressed in the future. These technical guidelines are also the basis for the selection of parameters for the Safety Parameter Display System (SPDS) to be incorporated into nuclear power plant control rooms.

In addition to the improvements in EOPs, the NRC staff is working to improve other operating procedures and maintenance procedures.

Man-Machine Interface

During the report period, the NRC continued to evaluate the human factors aspects of man-machine interfaces to minimize design-induced errors in nuclear power plants. The basic requirements for detailed control room design reviews (DCRDR) and the safety parameter display system (SPDS) were continued in Generic Letter 82-33 (Supplement 1 to NUREG-0737—Requirements for Emergency Response Capability). The NRC has received an additional 27 plans during fiscal year 1984 for detailed control room design reviews, representing 37 operating units. By the end of fiscal year 1984, 51 detailed control room design reviews were started by various utilities, representing 106 units. NRC staff conducted an additional 16 in-progress audits in fiscal year 1984. The staff reviewed and commented on 45 plans, representing 75 units; received 18 summary reports, covering 32 units; issued 14 Safety Evaluation Reports, representing 27 units; and conducted five Pre-implementation Audits, covering seven units. In addition, the staff completed control-room preliminary design analyses for seven applicants for operating licenses in fiscal year 1984. During fiscal year 1984, the NRC received 53 SPDS Safety Analysis Reports, representing 77 units; and the staff issued 11 Safety Evaluation Reports, covering 16 units. These efforts will continue through fiscal year 1987.

Review of the DCRDR revealed a significant technical issue related to satisfying the task analysis requirement of Supplement 1 to NUREG-0737, operator information and control needs. The staff met with representatives of three of the NSS vendor Owners Groups to discuss the task analysis requirement and identified additional analyses and documentation which were needed for review. During fiscal year 1984, the staff also completed human factors evaluations of three NRC Regional Operation Centers to provide recommendations for improving the human engineering of the man-machine interfaces as well as the Technical Support Center (TSC) and Emergency Operations Facility (EOF) at San Onofre Nuclear Generating Stations 2 & 3 (Cal.) and the Kewaunee Nuclear Power Plant (Wis.).

The Office of Nuclear Regulatory Research was asked to develop a proposed rule for a new General Design Criterion (GDC) on Human Factors Requirements for Operability and Maintainability. The proposed GDC would be intended mainly for new plants, but it would also apply to major retrofits to existing plants. Drafts of the proposed GDC were reviewed by the NRR staff in fiscal year 1984 and development of the criterion will continue in fiscal year 1985. Other documents published or planned include: the "Computerized Annunciator System Recommendations" (NUREG/CR-3987); a study of "Safety System Status Monitoring" (NUREG/CR-3621), which identified a number of human factors shortcomings that present critical opportunities for operator error; Regulatory Guide 1.47, which will be revised to include guidelines on how to improve safety systems status monitoring; and a report on "Human Factors Deficiencies in the Design of Local Control Stations and Operators Interfaces in Nuclear Power Plants" (NUREG/CR-3696). Based on this latter report, the need for regulatory action regarding local control stations will be determined.

An NRR Working Group was formed to review present control room habitability requirements and to investigate the need for new requirements. A regulatory recommendation will be developed by the group.

Management and Organization

The NRC continued to explore the matter of management and organization at nuclear power plants. A set of guidelines and an accompanying workbook were developed to enhance consistency in NRC review of the various management and organization schemes proposed by applicants for operating licenses. An analysis of the possible relationship between organizational characteristics and safe performance of nuclear power plants was performed. The NRC received and assessed several proposals from the Nuclear Utility Management and Human Resources Committee (NUMARC) for industry self-improvement in the management effectiveness area. The NRC initiated efforts to identify methods of assessing management effectiveness at operating nuclear power plants. Preliminary work on the development of a set of safety performance indicators was completed. The NRC has pursued open communication with other U.S. and foreign organizations concerned with the effect of the management of nuclear power plants on their safe operation.

In addition to scheduled operating licensing reviews, more than 80 technical reviews were conducted during the report period for plant-specific licensing actions on organization, management, staffing and training. Approximately 60 of the actions were technical specification changes with the remainder being operator requalifications program reviews.

Unresolved Safety Issues

Section 210 of 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). A total of 27 USIs have been identified, and a final technical resolution has been achieved for 15 of these (see Table 5). Technical resolution of the remaining 12 USIs involves (1) development of technical findings and the incorporation of such findings into new licensing requirements in the NRC Regulations, Standard Review Plan, Regulatory Guides, or other official guidance; (2) provision for a plan for implementation of the technical resolution to plants in operation or under construction if required, (3) preparation of a regulatory analysis of any new requirements and a review by the Committee to Review Generic Requirements (CRGR); (4) provision of a public comment period after CRGR review, followed by discussion and disposition of the comments received in a final report; and (5) provision for a second review of the resolution by the CRGR after public comments have been addressed.

SUMMARY OF STATUS

The USIs that are being actively worked on are listed in Table 6, together with the present schedule for technical resolution. A summary of the status of USIs is published quarterly in NUREG-0606.

PROGRESS REPORTS

The following are progress reports on each of the Unresolved Safety Issues under active consideration or resolved during fiscal year 1984. For background on these issues, see the *1983 NRC Annual Report*, pp. 17-22.

Water Hammer

Water hammer events introduce high pressure pulses in fluid systems and can be caused by collapse of steam pockets in voided lines, water slugs resulting from steam condensation, pump startup in voided lines and inadvertent valve closures. The frequency of water hammer occurrence in nuclear plants has been relatively low and damage has generally been limited to piping support systems. Underlying causes have been about equally divided between design deficiencies and operator induced water hammer. Identified design deficiencies have been corrected and current plants generally employ proven design concepts (e.g., J-tubes in PWR steam generators and "keep-full" systems in BWRs). The staff has concluded its technical evaluation of this issue, USI A-1, and these findings are published in NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants." As described more completely in NUREG-0933, Revision 1, "Regulatory Analysis for USI A-1, Water Hammer," this safety issue has been resolved through revision to the sections of the Standard Review Plan (SRP) dealing with those systems which experienced water hammer; the revision is intended to assure that proven design concepts will be maintained and emphasizes the need for operator training and awareness.

PWR Steam Generator Tube Integrity

Steam generator tube degradation in pressurized water reactors continues to be a matter of concern. (See the section on steam generators later in this chapter). The proposed resolution for the problem (USIs A-3, A-4, and A-5) was discussed with the Advisory Committee on Reactor Safeguards and the NRC Committee to Review Generic Requirements in October 1983. At the close of the report period, the proposed resolution was under consideration by the Commission before being issued for public comment.

Systems Interactions

Adverse Systems Interactions are events that may jeopardize the independent functioning of nuclear power plant safety systems. NRR staff efforts on systems interactions during fiscal year 1984 were directed toward establishing the potential safety significance of these types of events and exploring possible ways to anticipate such interactions.

An investigation has been made of the operating experience at U.S. nuclear power plants for adverse system interaction events and to establish possible patterns among the events. A draft report on "Survey and Evaluation of System Interaction Events and Sources" from Oak Ridge National Laboratory was completed in July 1984 and is being reviewed by the staff. Further work is planned to establish the potential safety significance of these types of events.

The investigation of systems interaction search methods using Indian Point Unit 3 (N.Y.) has been completed by Brookhaven National Laboratory and Lawrence Livermore National Laboratory. The staff will use the results of these studies to help develop guidelines for acceptable search methods.

Seismic Design Criteria

Rapid advancements in state-of-the-art technology in seismic design over the past decade have made it neces-

Table 5. Formerly Unresolved Safety Issues for Which A Final Technical Resolution has been Achieved

Title		*Report Number*	*Date*	*Implementation Status*[2]
A-1	Water Hammer	NUREG-0927 Rev. 1 NUREG-0993 Rev. 1	March 1984	No new requirements for operating plants. Revised SRP sections address requirements for any new application (see NUREG-0993, Rev. 1).
A-2	Asymmetric Blowdown Loads	NUREG-0609	November 1980	Resolution on final twelve operating plants will complete review of all operating plants.
A-6	Mark I Short Term Program	NUREG-0408	December 1977	Complete.
A-7	Mark I Long Term Program	NUREG-0661	July 1980	Licensees have designed and are installing modifications to meet the Commission's Order date for each operating plant with Mark I containment. Modification have been completed on more than one-half of the 22 plants affected.
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808	August 1981	Implemented as a part of the OL review of each Mark II containment.
A-9	Anticipated Transients	NUREG-0460	September 1980	The final rule (49FR5752) was published in the Federal Register on June 26, 1984. Guidance for implementation on all plants is included in the final rule.
A-10	BWR Feedwater Nozzle	NUREG-0619	November 1980	Complete.
A-11	Reactor Vessel Material	NUREG-0744 Rev. 1	October 1982	Implementation on a case-by-case basis as needed.
A-12	Steam Generator and Reactor Coolant Pump Supports	NUREG-0577 Rev. 1	September 1982	No implementation on operating plants required.
A-24	Qualification of Class IE Safety Related Equipment	NUREG-0588 Rev. 1	July 1981	Implementation in accordance with the new rule 10 CFR 50.49 is continuing.
A-26	Reactor Vessel Pressure Transient Protection	NUREG-0224	September 1978	Complete.
A-31	Residual Heat Removal	SRP[1] 5.4.7	1978	Implemented as part of the review for each operating license application. No backfit to operating reactors is planned.
A-36	Control of Heavy Loads Near Spent Fuel	NUREG-0612	July 1980	Detailed implementation for each licensee is continuing.
A-39	SRV Dynamic Loads	NUREG-0802	September 1982	Implemented as part of the OL review of each Mark II and Mark III containment.
A-42	Pipe Cracks in Boiling Water Reactors	NUREG-0313	July 1980	Actions required for each licensee on a case-by-case basis in accordance with operating experience.

[1]Standard Review Plan (NUREG-0800).
[2]See NUREG-0606 for further detail on implementation.

Table 6. Schedule for Resolution of Current Unresolved Safety Issues

Task No.	*Unresolved Safety Issue*	*Schedule for Issuing Staff Report "For Comment" (as of September 30, 1984)*	*Schedule for Issuing Final Staff Report (as of Sept. 30, 1984)*
A-3,4,5	PWR Steam Generator Tube Integrity	December 1984	June 1985
A-17	Systems Interactions	September 1985	March 1986
A-40	Seismic Design Criteria	January 1985	July 1985
A-43	Containment Emergency Sump	Complete May 1983	May 1985
A-44	Station Blackout	December 1984	January 1986
A-45	Shutdown Decay Heat Removal Requirements	July 1985	February 1986
A-46	Seismic Qualification of Equipment in Operating Plants	December 1984	August 1985
A-47	Safety Implications of Control Systems	September 1985	June 1986
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	--------	June 1986
A-49	Pressurized Thermal Shock Rulemaking Complete Technical Resolution	Completed February 1984 August 1985	December 1984 March 1986

sary to update the NRC acceptance criteria for seismic design of structures, systems, and components of nuclear plants. Lawrence Livermore Laboratory conducted a study comparing NRC Seismic Design Criteria with the current state-of-the-art knowledge. Results of their study were published in NUREG/CR-1161, entitled "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria" issued in May 1980. NRC review of these recommendations resulted in proposed changes in Sections 2.5.2, 3.7.1, 3.7.2, and 3.7.3 of the Standard Review Plan. Incorporation of the proposed changes should eliminate potential sources of non-conservatism and excessive conservatism and result in seismic design that reflects an up-to-date understanding of this technology.

The proposed changes will apply to new applications for construction permits. The seismic design of safety-related above-ground steel tanks has, however, been identified as a potential backfit requirement for operating plants, OL applications and CP holders. The proposed resolution of this issue, USI A-40, includes a recommendation that certain licensees be required to report seismic design information on these tanks to the NRC to enable the staff to review the seismic design of each tank.

The proposed changes in the Standard Review Plan and proposed backfit requirements for safety-related tanks are being developed.

Containment Emergency Sump

Following a loss-of-coolant accident (LOCA), long-term heat removal must be maintained by operation of residual heat removal pumps and containment spray pumps. The water source for these systems in pressurized water reactors is the containment emergency sump, and in boiling water reactors it is the intake pipe in the suppression pool or wetwell. Three concerns in the post-LOCA period are: possible air ingestion, debris blockage of the sump or intake screens and ingestion of small size debris into the pumps. Air ingestion could lead to loss of pumping capacity; debris blockage could lead to loss of net positive suction head (NPSH) margin; and ingestion of particulates could affect pump seals. The debris concern stems from the LOCA jet capability to destroy insulation; this debris then is transported to the sump screen, resulting in high pressure drops with a corresponding loss of NPSH.

These concerns have been investigated extensively through full-scale sump experiments, plant surveys and analyses. The findings show that air ingestion potential is low and not as significant as previously postulated. Debris blockage effects are dependent on plant design and on the insulation materials used. A revision to Regulatory Guide 1.82 has been proposed that would provide a method for performing a plant-specific evaluation which properly addresses the design differences in each plant. Evaluations of the types of residual heat removal and containment spray pumps in use shows that these pumps can tolerate ingestion of insulation debris and other types of particulates of the size that can pass through the debris screens used in nuclear plants.

Technical findings by the NRR staff (NUREG-0897) and value/impact analysis (NUREG-0869) for the proposed requirements were published for public comment in April 1983, along with proposed revision to Regulatory Guide 1.82 and Standard Review Plan Section 6.2. These documents have been revised to reflect information received during the for comment period. The proposed final resolution was under review by the Committee to Review Generic Requirements at the close of the report period.

Station Blackout

Concurrent loss of off-site and on-site emergency alternating current power sources is referred to as station blackout (USI A-44). Many safety systems required for decay heat removal from the reactor core are dependent on the availability of these power sources. Oak Ridge National Laboratory issued a draft report on the subject in July 1984, entitled "Collection and Evaluation of Complete and Partial Losses of Off-Site Power at Nuclear Power Plants." Two other NUREG/CR technical reports were published in 1983: one on on-site emergency diesel generator reliability, and the other on analyses of station blackout accident sequences. Based on the results of these investigations, recommendations to resolve USI A-44—which include proposed rulemaking and a new regulatory guide on station blackout—have been prepared by the NRC staff. In addition, the staff is preparing an "Evaluation of Station Blackout Accidents at Nuclear Power Plants" (NUREG-1032), that summarizes the technical findings related to USI A-44. In May 1984, the Committee to Review Generic Requirements recommended that the proposed rule, the proposed regulatory guide and draft NUREG-1032 be issued for public comment. The Notice of Proposed Rulemaking, including a supporting regulatory analysis, is being prepared for review by the Commission prior to issuance for public comment.

Shutdown Decay Heat Removal Requirements

A program has been established to evaluate the safety adequacy of systems for removing decay heat from a reactor core during shutdown and to assess the value and the impact of alternative measures for improving the reliability of those systems. In order to accomplish these objectives, numerous tasks and subtasks have been identified, including system reliability assessments, system engineering feasibility studies, thermal-hydraulic analyses, power plant characterizations, reviews of emergency operating procedures, and evaluation of the vulnerability of the systems to special emergencies such as fire, flood, earthquake and sabotage. Work on these aspects continues. A meeting of specialists from 13 countries for the purpose of exchanging information on decay heat removal systems was held in Wurenlingen, Switzerland, on April 25-29, 1983, with the cooperation of the Nuclear Energy Agency and the Swiss Government. During the latter part of September 1984, follow-up visits were arranged to those countries (Netherlands, Belgium, Germany, Switzerland and Italy) with design information particularly relevant to decay heat removal systems.

The two following contractor interim milestone reports have been issued in final form:

(1) "Grouping of LWRs for Evaluation of Decay Heat Removal Capability," NUREG/CR-3713, dated June 1984.

(2) "Cost-Benefit Analysis of Adding a Feed and Bleed Capability to Combustion Engineering PWRs," NUREG/CR-3421, dated August 1983.

Six other contractor draft interim milestone reports have been prepared and are under review by the NRC staff.

Work during fiscal year 1985 will concentrate on: (1) assessing the engineering feasibility, costs and other impacts of alternative decay heat removal (DHR) design which would provide a safety benefit in operating power plants, and (2) performing value-impact evaluations of the most promising alternative DHR system measures for a spectrum of plant types.

Seismic Qualification of Equipment in Operating Plants

This safety issue, USI A-46, is the concern that the margins of safety provided by equipment in operating nuclear power plants under seismically induced loads may vary considerably. Design criteria and methods for seismic qualification of equipment have undergone significant change during the history of the commercial nuclear power program. The seismic capability of equipment in operating plants, therefore, should be reassessed to assure the plant can be brought to a safe shutdown condition when subjected to a design basis seismic event. The objective of USI A-46 is to develop seismic qualification methods and/or acceptance criteria that can be used to assess the capability of mechanical and electrical equipment in operating nuclear power plants to perform their intended safety function during and/or after a seismic event. This issue entails investigation of alternative procedures for assuring seismic adequacy of equipment. This and other tasks studied are described in NUREG-1030 titled, "Seismic Qualification of Equipment in Operating Nuclear Power Plants."

A utility group, the Seismic Qualification Utilities Group (SQUG), has collected and evaluated seismic experience data from industrial facilities and conventional power plants that had undergone strong motion earthquakes. In 1983, the SQUG proposed to NRC management the formation of the Senior Seismic Review Advisory Panel (SSRAP) to provide expert opinion and advice on the use of experience data. The idea was endorsed by the NRC and the SSRAP was formed in June 1983. In February 1984, the SSRAP released its report which describes their findings and recommendations on eight classes of equipment. Basic conclusions of their study were stated by SSRAP as follows:

(1) Equipment installed in nuclear power plants is generally similar to, and at least as rugged as, that installed in conventional power plants.

(2) This equipment, when properly anchored, can, with some reservations, be said to have an inherent seismic ruggedness and a demonstrated capability to withstand significant seismic motion without structural damage.

(3) For this equipment, the ability to function after the strong shaking has ended has also been demonstrated, but the absence of relay chatter during strong shaking has not been demonstrated.

The NRC staff worked closely with the SQUG and the SSRAP during the collection and evaluation of the seismic experience data and concurs with the SSRAP conclusions.

Although equipment is inherently rugged and not susceptible to seismic damage, failures due to seismic loads can occur if equipment is not adequately supported or anchored. This need to review anchorage and supports was also identified by the NRC Systematic Evaluation Program. Therefore, the proposed resolution will include a requirement to verify that equipment is adequately anchored and supported.

During 1984, technical work was completed on task A-46 and a proposed resolution was developed by the staff.

Safety Implications of Control Systems

Systematic evaluations are being performed by NRC staff of control systems that are typically used during normal startup, shutdown and on-line power operations of nuclear power plants for each of the four nuclear steam supply system vendors (i.e., Babcock and Wilcox, Westinghouse Corp., Combustion Engineering, and General Electric Co.). The purpose of this study (USI A-47) is to determine whether failures in these systems could significantly affect the safety of the plant. The evaluations of the design of a boiling water reactor plant and of a pressurized water reactor (PWR) plant have been completed by the Idaho National Engineering Laboratory, and the draft reports are currently under review by the NRC staff. In addition, the Oak Ridge National Laboratory is in the final stages of completing its evaluation of a different vendor's pressurized water reactor. The evaluation of the fourth plant design (also a PWR) is in the early stages of review.

These studies have identified several control system failures that could potentially result in undesired transients leading to steam generator or reactor vessel overfill, overcooling and increased pressure events. To assess the contribution of these potential failures to public risk, analyses are being performed to determine the safety significance of the control system failures that have been identified. These analyses are expected to be complete in early 1985. After completion of the technical work, recommendations may be indicated to to assure that control system failures do not pose an unacceptable risk. A proposed staff resolution, including any recommendations for operating plants or new applications, is expected in September 1985.

Hydrogen Control Measures and Effects Of Hydrogen Burns on Safety Equipment

Postulated reactor accidents that result in a degraded core, such as the one at Three Mile Island Unit 2 in 1979, can result in generation and release to the containment of large quantities of hydrogen, which can burn or explode under certain conditions (USI A-48). Consequently, the NRC determined that additional hydrogen control measures should be considered for all nuclear power plants. A final rule for Mark I and II containments for boiling water reactors was published on December 2, 1981, requiring that these containments be inerted by the replacement of air inside the containment with nitrogen.

With respect to Mark III containments for boiling water reactors and ice condenser containments for pressurized water reactors, a proposed final rule was submitted to the Commission for review during the report period. The proposed final rule requires improved hydrogen control systems that can handle large amounts of hydrogen during and following an accident. Because of the greater inherent capability of the large dry containment designs to accommodate large quantities of hydrogen released during an accident, the staff also proposed that rulemaking related to large dry containments could be deferred pending completion of NRC and industry sponsored research programs.

Extensive research has been undertaken by the nuclear industry and by NRC on hydrogen combustion. The large scale hydrogen combustion tests conducted at the Nevada Test Site were completed in early 1984. The results of these tests are being evaluated by the NRC staff and its consultants. In addition, the Mark III Owners Group is sponsoring a 1/4-scale hydrogen test program. The tests are currently scheduled to be conducted in early 1985. The results of these 1/4-scale tests will be evaluated and considered in the resolution of certain open items regarding Mark III containments. Based on the schedule of the hydrogen test program indicated above, the completion date of USI A-48 is estimated to be in late 1986.

Pressurized Thermal Shock

Pressurized Thermal Shock (PTS) events involve unintended rapid cooling of the steel reactor pressure vessel to a low temperature concurrent with or followed by repressurization of the water inside the vessel. If the toughness of the vessel wall has been decreased excessively by the neutron irradiation that occurs during normal power generation, severe PTS events could cause rupture of the vessel and potential melting of the nuclear core that is contained within the vessel.

After extensive analyses performed by both the NRC staff and by several nuclear industry groups, the NRC staff concluded that: (1) the risk due to PTS events is related directly to the "reference temperature," which is a measure of ductility loss and is determined from the reactor vessel material properties, the high energy neutron irradiation at the reactor vessel wall, and the duration of reactor operation; and (2) the risk due to PTS events is acceptably low if the "reference temperature" has not exceeded a certain specified screening limit, which has been defined by the staff

To ensure that nuclear plants do not operate with unacceptable PTS risk, the NRC issued a proposed rule for public comment on February 7, 1984, which would amend its regulations to: (1) establish a screening criterion related to the fracture resistance of pressurized water reactor (PWR) vessels during pressurized thermal shock (PTS) events; (2) require analyses and schedule for implementation of neutron flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plant operation beyond the screening criterion will be considered.

The NRC also has prototype analyses for three nuclear plants nearing completion. These analyses will form the bases for preparation of NRC guidelines to licensees who may be required to perform plant-specific PTS risk analyses to justify any proposed operation beyond the screening limit.

The PTS rule is scheduled for issuance in final form before the end of 1984, and the guidance and acceptance criteria will be published in late 1985 or early 1986, well before any plant would near the screening limit.

Safety Reviews

Other significant safety aspects of nuclear power plant operation are discussed below, including both general programs that involve a number of reactor systems in numerous plants and specific concerns that involve a particular system, safety feature, or plant.

TMI Action Plan

The accident at Three Mile Island Unit 2 in 1979 led to a thorough review of NRC regulatory and licensing requirements for nuclear power plants. A TMI Action Plan was issued as NUREG-0660, and the requirements approved for implementation at plants in operation or under construction were later clarified in NUREG-0737. Approximately 95 percent of these requirements for operating reactors have now been acted on, and 85 percent of required actions have been reviewed by NRC staff. TMI Action Plan requirements for plants under construction are being implemented as part of the licensing process, while those for operating reactors are confirmed by NRC orders. The items not covered by NUREG-0737 have been addressed in NUREG-0933, which sets priorities for generic issues.

Supplement 1 to NUREG-0737, the requirements for emergency response capabilities, was sent to all licensees on December 17, 1982. Discussions were held with the utilities at regional meetings to negotiate implementation schedules. By June 12, 1984, the schedules were confirmed by issuance of Confirmatory Orders for all licensed light water reactors.

Systematic Evaluation Program

The Systematic Evaluation Program (SEP) is an ongoing program to assess the adequacy of the design and operation of older nuclear power reactors, to compare them with current safety criteria, and to provide a basis for integrated and balanced decisions on proposed procedural or plant modifications. Integrated plant safety assessments have been completed for nine of the plants. The assessment of San Onofre 1 (Cal.) is scheduled to be completed in fiscal year 1985. The assessment of Dresden Unit 1 (Ill.) was deferred initially because of an extended plant shutdown; Commonwealth Edison Co., the licensee, has announced its intention to decommission Dresden Unit 1 because of the estimated costs to restart the plant.

In June 1984, the Commission approved a new program, the Integrated Safety Assessment Program (ISAP), which will be undertaken in lieu of the previously proposed continuation of SEP (Phase III) and the conduct of the National Reliability Evaluation Program. The objectives of ISAP are: to provide an integrated, cost-effective implementation program; to provide the technical bases to resolve all outstanding licensing actions, including the significant topic reviews from SEP Phase II; to establish overall plant improvement schedules; and to provide the benchmark by which future regulatory actions can be judged, on a plant-specific basis. A pilot program will be undertaken with about four licensees over a period of two years. The pilot program is designed so that preliminary results will be available in about a year for a Commission review of the program effectiveness. This program will not be implemented until a cost-benefit evaluation of the results of SEP Phases I and II is reviewed by Congress, in accordance with Public Law 98-50. That evaluation was submitted to Congress by the Commission on May 18, 1984.

Severe Accident Policy

Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious off-site consequences. Throughout fiscal year 1984, a number of meetings were held by the staff with the full committee of the Advisory Committee on Reactor Safeguards (ACRS) and its Class 9 Subcommittee to exchange views on issues related to the further development of severe accident policy for both future and existing plants. Working sessions were also held with representatives of the Industry Degraded Core Rulemaking Program (IDCOR), which was set up by the nuclear utility industry to develop the technical basis for determining whether changes in regulatory requirements are needed to reflect severe accident considerations. The meetings between IDCOR and the staff focused on the definition of the most important technical issues of relevance to severe accidents and to compare IDCOR's independent models and assessments of severe accident behavior with work sponsored by the NRC.

On September 19, 1984, the staff forwarded to the Commission for their review and approval a recommended "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants." The revised policy statement differs from the earlier proposed policy statement of April 1983 (see *1983 NRC Annual Report*, pp. 2 and 5) in two key respects

(1) The new policy statement recommended by the staff includes a sharpened focus on future reactor designs and sets aside severe accident rulemaking for existing plants unless and until new safety information should expose severe accident vulnerabilities that would warrant it.

(2) It is structured for the concurrent publication of a companion staff report, "NRC Policy On Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulations" (NUREG-1070).

A copy of the draft report (NUREG-1070) has been forwarded to the Commission. Complementing the recommended policy statement on severe accidents, the report provides an expanded discussion of numerous interrelated ongoing severe accident programs. Among them are: the Severe Accident Research Program (NUREG-0900); the Source Term Program; the development of Safety Goals and the Probabilistic Risk Assessment Reference Document (NUREG-1050); the resolution of Unresolved Safety Issues and Generic Safety Issues; and the integration of insights from IDCOR, foreign reactor and regulatory experience as well as the staff review of new reactor designs. A separate chapter of the report provides an overview of, and staff response to, public comments and the views and recommendations received from the ACRS, including those provided in the ACRS letter of July 18, 1984. The report also includes a short appendix on the treatment of uncertainty in the severe accident program and a more detailed appendix on current information bearing on the need for generic design changes or further regulatory changes affecting nuclear power plants. The latter provides a rationale for the differential policy treatment of existing and future plants and an up-to-date information base to support a number of critical premises or assumptions underlying the basic strategies of the Policy Statement.

The proposed Policy Statement, with its emphasis on procedures and criteria for staff review of new standard plant designs, is intended to facilitate and stabilize review procedures. A staff review of severe accidents for the GESSAR II design for forward referenceability is nearly complete. The staff also has been involved with the pretendering review of an application for Westinghouse Electric Corporation's advanced pressurized water reactor design—RESAR-SP/90.

Probabilistic Risk Assessment

Probabilistic risk assessments (PRAs) continue to be used to gain insight into the importance of certain potential safety issues and to identify strengths and weaknesses in nuclear power plants. For example, as a result of the information learned during its performance of the PRA for the Limerick plant ((Pa.), the applicant made several design changes to reduce accident vulnerabilities. In addition, the applicant volunteered to make three additional improvements resulting from a staff review which provided evidence that these additional improvements, that are beyond licensing requirements, would be cost-effective in further reducing the risk to core damage accidents. The applicant plans to use the Limerick PRA throughout plant life, for the purpose of discriminating potential issues of central importance to reactor safety from those of marginal importance to safety.

A review of the Zion (Ill.) Probabilistic Safety Study was performed by Sandia National Laboratory and reported in NUREG/CR-3300, issued in May 1984. Insights regarding potential design and operational improvements that may have safety benefits at the Zion plants (Ill.) are being developed by the NRC staff based on the Sandia review. The reviews of the GESSAR-II and Millstone Unit 3 (Conn.) PRAs are nearing completion and reports documenting the results of the reviews will be published in the fall of 1984 and the winter of 1984/1985 respectively. The review of the Millstone Unit 3 PRA has involved an estimate of core-damage likelihood in the calculation of off-site consequences for potential severe accidents in the Draft Environmental Statement. As in the case of Zion units, insights regarding potential design and operational improvements that may have safety benefits at Millstone Unit 3 are being developed by the NRC staff.

A probabilistic assessment has been submitted for the Seabrook Plant (N.H.) and is currently under review. The review of the Shoreham PRA (N.Y.) is scheduled for completion in early 1985.

To facilitate future probabilistic assessments, and to obtain some measure of consistency among the studies that may be submitted to the NRC, the NRC staff has issued a draft report, NUREG/CR-2815, that provides guidance on scope, probability estimating techniques, and documentation. This guidance is being updated to reflect evolving knowledge on externally initiated events such as earthquakes and fires and treatment of uncertainties. The NRC staff has also developed a companion draft document (NUREG/CR-3485) to standardize the review of PRAs by the staff.

Probabilistic assessments provided risk perspectives for several issues that were processed by the NRC staff. These insights were an important adjunct to the engineering assessments that were made for each of the issues.

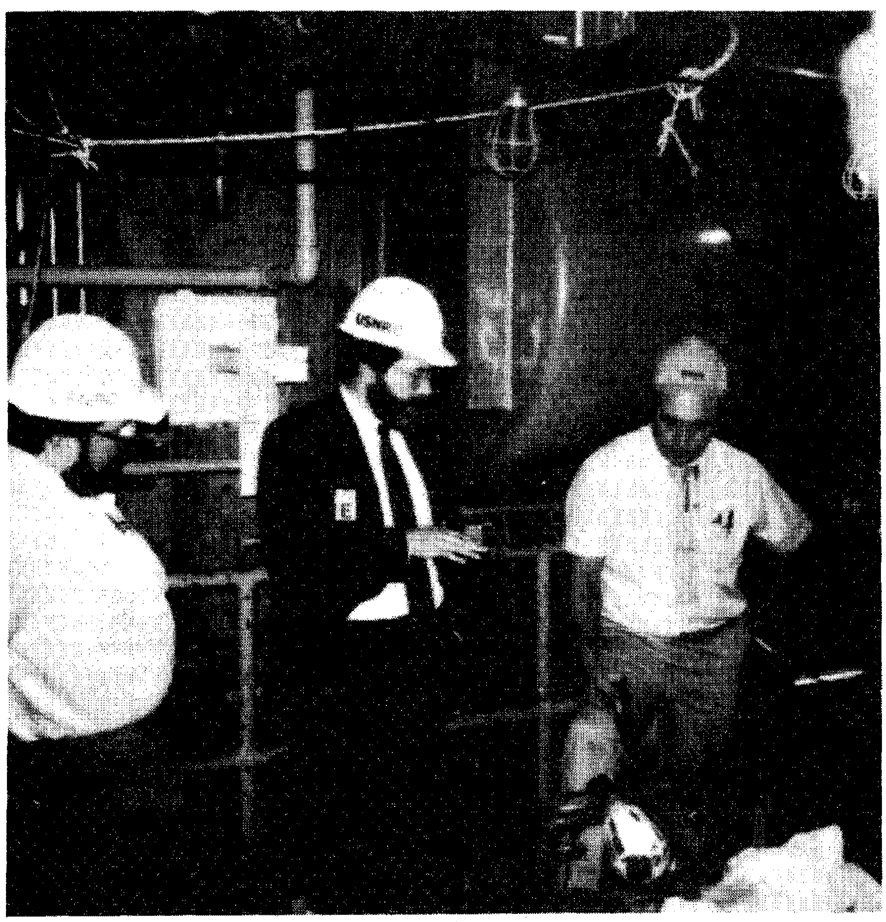
Examples of the more significant studies include:

- Consideration of the addition of power operated relief valves on new plants designed by Combustion Engineering.
- Limiting conditions of operation for diesel generators at the Zion plants.
- Consideration of flooding from internal fluid system failures at Shoreham.
- Consideration of testing frequency for reactor protection systems.

Probabilistic assessments are routinely used in setting priorities for new issues according to their safety significance, and the ranking process is also used to allocate resources for addressing the issues. Similarly, probabilistic assessments are used in weighing alternative solutions to generic safety issues. In addition, the probabilistic assessments provided an important basis, along with deterministic engineering judgment, for the regulatory analyses of new recommendations proposed by the staff.

Reactor Containment

A special study of reactor containment characteristics was undertaken during the year, with particular attention to leak rates during a postulated severe accident within the containment structures. Six containment types were



NRC review continued at year's end of a probabilistic risk assessment for the Shoreham nuclear power station, Brookhaven, N.Y. Here Regional Administrator Thomas E. Murley (Region I) inspects piping at the plant.

studied—large dry, subatmospheric, ice-condenser, and pressure suppression containments of Mark I, Mark II and Mark III types. A draft report has been prepared and discussed with industry representatives. Special attention was paid to the behavior of containment hatches under pressure and temperature associated with severe accident conditions. A preliminary finding of the study is that, in a severe accident sequence, reactor containments are likely to leak before there would be catastrophic failure. Results of the study will be incorporated in the reassessment of accident source terms and the Commission Policy on Severe Accidents.

Special Study of Power-Operated Relief Valve Functions

During the year, the staff evaluated the need for providing a rapid primary system depressurization capability, in particular using power-operated relief valves in current plants designed by Combustion Engineering (CE). The staff analysis was published February 1984, in NUREG-1044, "Evaluation of the Need for a Rapid Depressurization Capability for CE Plants." This evaluation was deemed important because information developed since the accident at Three Mile Island Unit 2 suggests that power-operated relief valves (PORVs) in the CE plant design would enhance the overall capability of pressurized water reactors (PWRs) to accommodate transients and accident events. Also, PWRs designed by other vendors (Westinghouse and Babcock and Wilcox) include at least one PORV in their design. The evaluations—which were both probabilistic and deterministic, reflecting engineering analysis and judgment—did confirm the ability of current PWR designs without PORVs to meet regulatory requirements related to design basis accidents only.

Following this determination, probabilistic analyses were performed for severe accident scenarios, and these studies indicated that core-melt frequencies could be significantly reduced for current designs of CE plants by the addition of PORVs. The staff then prepared a value-impact analysis for the potential addition of PORV capability that suggested there would be a real but not overwhelming advantage in equipping these plants with a rapid depressurization capability. The value of such a retrofit is not so large as to suggest unambiguous cost-effectiveness, nor does it suggest an urgent need for risk reduction.

As part of its program to resolve generic unresolved safety issues affecting nuclear power plants, the NRC staff is conducting a detailed study of shutdown decay heat removal requirements, designated USI A-45. The staff concluded that the decision regarding PORVs for these CE plants should be deferred until the resolution of USA A-45 is completed. Because part of the benefit of the PORVs was predicated on their ability to provide an alternate decay heat removal path ("feed and bleed"), any improvements in decay heat removal capability that might be introduced as a result of the A-45 assessment could reduce the assumed net benefit of PORVs. Moreover, the events for which PORVs could prove to be of benefit are of low probability. The technical aspects of this problem, which are complex, are addressed in the Severe Accident Research Program (NUREG-0900, January 1983).

Generic Resolution of Reactor Trip Breaker Events

In February 1983, Salem Unit 1 (N.J.) underwent two events in which the reactor failed to shut down (trip) in response to automatic trip signals. An NRC Task Force was formed to investigate the causes and conditions leading to these failures. Its findings were published in NUREG-1000 "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," volume 1, with recommendations for remedial actions in volume 2. These recommendations were further organized into itemized requirements in Generic Letter (GL) 83-28 which was sent in June 1983 to all applicants and licensees requesting that they show compliance with these requirements. In February 1984, a preliminary review of the responses indicated that a large percentage of the them were incomplete. A good deal of effort was expended in fiscal year 1984 in getting applicants and licensees to complete their responses so that a thorough review of compliance could proceed.

Both the Babcock & Wilcox and the Westinghouse Owner's Groups developed generic design change packages to meet the hardware requirements of G.L. 83-28. These generic design change packages were reviewed and accepted by the NRC during February and March of 1984. In addition, eight plant-specific concerns for Babcock & Wilcox plants and thirteen such concerns for Westinghouse plants need to be resolved by the applicant or licensee for each plant. The current status is that 38 of 43 plant actions have been reviewed. Of these, 11 have been found acceptable, 21 are still not acceptable (because their response remains incomplete), six are under review, and five licensees have not yet submitted responses. Eighteen Safety Evaluation Reports have been written this year for G.L. 83-28 responses.

In September 1984, a technical assistance contract was awarded to develop review guidelines and acceptance criteria for the remaining G.L. 83-28 items. This program consists of three consecutive projects: Project I, the current phase, is to be completed by February 1985 and the review of these items for operating reactors under Project II will begin in January 1985 and be completed by March 1987. Review of the items for Operating License applicants under Project III would begin in February 1985 and be completed by March 1987. NRC Staff effort on the review of Item 4.3 is expected to be completed by December 1985.

The goal of the staff's review program is to resolve all G.L. 83-28 items for each plant reviewed prior to writing the safety evaluation reports and thus to avoid having to carry forward open items as operating condition limitations on operating reactors or as license conditions for operating license applicants.

Statistical Methods for LOCA Analysis

In November 1983, the staff approved the use of statistical methods for determining the adequacy of Emergency Core Cooling System evaluation models for design basis accidents. Statistical analysis techniques used in conjunction with best estimate thermal hydraulic models are used to estimate a realistic upper-bound peak cladding temperature (PCT) for loss of coolant accidents (LOCAs). The upper-bound PCT serves as a guide for assessing the conservatism of the licensing basis for PCT calculations. The licensing-basis PCT must be determined in accordance with 10 CFR 50, Appendix K. Anticipated benefits from use of the methodology include improved emergency procedures, better fuel utilization, and lower operating costs. These benefits are realized by reducing excessive conservatisms in licensing calculations. That results in more realistic predictions of accident parameters for LOCA scenarios and permits the relaxation of overly restrictive operational limits. The statistical analyses supported by experimental data also provide confirmation that overall public safety is not compromised. In May 1984, the NRC staff approved a specific statistical methodology proposed by the General Electric Co. for boiling water reactors (BWRs). The methodology is applicable for 27 operating reactors and all BWRs currently under construction. Licensees and applicants are expected to begin use of the statistical methodology during fiscal year 1985.

Loose Part Monitoring System

Studies of the design guidance and review criteria for loose part monitoring programs under NRC Generic Issue B-60 resulted in the publication of Regulatory Guide 1.133, Revision 1, "Loose Part Detection Program for the Primary System of Light Water-Cooled Reactors," May 1981. An effective loose parts monitoring system (LPMS) provides early indication of loose part impacts and alerts the plant operators to take appropriate action to minimize the risk of consequential damage to equipment and reactor components. All Construction Permit and Operating License applications under review by the staff after January 1, 1978, have been reviewed for conformance to Regulatory Guide 1.133, which was available in draft form at that time. Studies were continued under B-60 regarding the backfit implementation to operating reactors licensed prior to January 1, 1978, when the criteria and licensing commitments to loose part monitoring programs were not well defined. The issue was broadened to include the role of a LPMS in preventing steam generator tube rupture events when operating experience revealed that detectable loose parts in either the primary or secondary side of steam generators could cause severe damage leading to rupture of steam generator tubes. The studies of the B-60 issue, completed January 10, 1984, concluded that an LPMS backfit for conformance to Regulatory Guide 1.133 will not be required for operating reactors licensed prior to January 1, 1978.

Because of the economic benefit that can be derived from the avoidance of the costs of equipment repair and replacement and of power replacement because of reactor downtime for repairs, the staff believes there is sufficient incentive for licensees to upgrade their loose part monitoring programs voluntarily. A number have already done so. A Generic Letter was prepared on September 25, 1984, for transmittal to all licensees to recommend voluntary review of their loose part monitoring programs. This completes the staff action on Generic Issue B-60.

Steam Generators

Degradation of the heat-exchanger tubes in steam generators manufactured by the vendors of pressurized water reactors has been a concern for several years. Tube degradation results from a combination of problems related to mechanical design, materials selection, fabrication techniques, and secondary system design and operation.

In June 1984, a publication (NUREG-1063) was issued which discusses the operating experience of steam generators covering the two-year period of 1982 through 1983. This is the fourth in a series of steam generator operating experience reports issued by the staff (NUREG-0886 of February 1982, NUREG-0571 of March 1980, and NUREG-0523 of January 1979).

The most significant steam generator event occurring during fiscal year 1984 was the failure of a Fort Calhoun (Neb.) steam generator tube. During a March 1984 outage, the licensee conducted helium leak tests in an effort to locate a small 0.2 gpd leak in steam generator "B" which was detected about three weeks before the outage. These tests were not successful in locating the small leak. During the outage, extensive eddy current testing was conducted as part of the licensee's planned in-service inspection program. A total of nine tubes were preventively plugged, based on eddy current data. On May 16, 1984, the unit was conducting a hydrostatic test in preparation for returning to power operation. The cold-leg temperature was 398°F. The primary system pressure was 1,800 psi and the steam generator pressure was 200 psi. While plant personnel were closely watching steam generator "B" for indications of the small leak experienced before shutdown, an unanticipated increase in water level indicated a tube failure. The maximum leak rate was later estimated at 112 gpm. A high leak rate persisted for approximately ten minutes, while the primary pressure was decreased and the main steam line isolation valve associated with steam generator "B" was closed.

The failed tube was found in the second peripheral row from the outside. The failure was a 1-inch-long axial "fishmouth" opening along the bottom of the hot-leg side of the horizontal run at the top of the "U". It was located between the scallop bars in the vertical batwing support. Sections of the failed tube and adjacent tube were removed for laboratory analysis.

Analysis revealed the failure to be attributable to intergranular stress corrosion cracking (IGSCC) from the outside, through 95 percent of the wall thickness, with the remaining 5 percent evidencing ductile tearing. The tube cross section was ovalized, with elongation of 0.046 to 0.122 inch on the major axis (along with the plane of the fracture) and compression of 0.045 to 0.070 inch on the minor axis. An additional defect through approximately 50 percent of the wall, was found inch from the hot-leg end of the fishmouth failure. A review of the eddy current data revealed that the failed tube had an indication of a defect through 99 percent of the wall and should have been plugged; the analyst who reviewed the eddy current tape had missed the indications. The failed tube was removed from the steam generator for destructive analysis. Upon return to power, no leakage was detected, which strongly suggests that the failed tube was the one that was leaking before the outage.

Pipe Cracks at Boiling Water Reactors

Intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping in boiling water reactors (BWRs) has been observed for many years. However, extensive IGSCC in large diameter piping was reported, for the first time in the United States, at Nine Mile Point Unit 1 (N.Y.) in March 1982. The cracking was found in the heat-affected zones of the reactor coolant recirculation piping welds. To resolve this concern of cracking in large diameter piping, the NRC issued Inspection and Enforcement (I&E) Bulletins 82-03 and 83-02 in October 1982 and March 1983, respectively, requiring augmented piping inspection in operating BWRs. After extensive IGSCC was reported in several operating BWRs, Orders were issued to five operating BWR licensees to accelerate the inspection schedules for their facilities. The inspection results, including the reinspection results of several BWR plants, have shown extensive cracking in welds of large diameter piping of both recirculation and residual heat removal systems at many operating BWR plants. Only four BWR plants (Oyster Creek (N.J.), Big Rock Point (Mich.), Duane Arnold (Iowa) and Browns Ferry Unit 3 (Ala.)) did not show cracking in large diameter piping.

The extent of IGSCC in piping is generally influenced by three causative factors: the environmental conditions existing in the BWR reactor coolant system; stresses in the piping, including residual stresses induced by welding; and the degree of sensitization in the materials. So far, the inspection results did not show any clear correlation with total operating time, because some plants with a relatively brief operating history show extensive cracking.

The joint effort by NRC and industry in training and qualifying ultrasonic testing (UT) personnel has greatly improved the reliability in the detection of the IGSCC. Recently, training courses for UT flaw sizing were provided by the Electric Power Research Institute at its facility in Charlotte, N.C. To be qualified for flaw sizing, UT personnel must pass the Institute's examinations.

The NRC staff short-term approach to assure continued safe operation of affected facilities was detailed in SECY-83-267C. These staff short-term reinspection and repair criteria—as modified by ACRS (Advisory Committee on Reactor Safety) comments—were issued on April 19, 1984, as Generic Letter 84-11 to all licensees of BWR facilities for use in inspection subsequent to I&E Bulletins 82-3, 83-02, or the individual Order issued on August 26, 1983.

Some BWR licensees are opting to replace their IGSCC affected piping. To facilitate replacement, the NRC issued the Generic Letter 84-07, on March 14, 1984, which transmitted licensing procedural guidance to BWR licensees for piping replacement under 10 CFR 50.59. The guidance covers the engineering design, materials, fabrication, and installation of replacement piping. Nine Mile Point Unit 1 (N.Y.) has completed recirculation piping system replacement; Monticello (Minn.), Pilgrim Unit 1 (Mass.), Peach Bottom Unit 2 (Pa.) and Hatch Unit 2 (Ga.) are undergoing piping replacement; Vermont Yankee, Cooper (Neb.) and Brunswick Units 1 and 2 (N.C.) will replace at least a portion of the affected piping systems during their next refueling outage.

Recommendations for the NRC long-term technical position on BWR pipe cracking were developed by the Task Group on Pipe Cracking, under the auspices of the NRC Piping Review Committee and were published in the final report of NUREG-1061 Volume 1. The final report is being distributed for public comment. The Task Group report concludes that the IGSCC in large diameter piping in BWR plants is not a new phenomenon; however, it is a serious problem requiring some changes to current regulatory practice. The report also recommends that the fixes should consist of measures to combat all three, or at least two, causative factors to be fully effective. The recommended degree of augmented inspection should depend on the material and process used at each weld. The extent and frequency of examinations should therefore depend on the degree of material resistance to IGSCC and the effectiveness of any processes used to reduce the susceptibility to cracking.

The NRC staff long-range plan is to bring all plants to a condition that allows them to be inspected at frequencies specified in 10 CFR 50.55a(g) without relying upon the augmented inservice inspections. Pending modification because of public comments, this goal is scheduled to be achieved within two fuel cycles, starting from the end of the current fuel cycle for all plants. To comply with this long-range plan, the Task Group report recommends that all piping should be made of IGSCC resistant materials or uncracked non-resistant materials with the residual tensile stresses in the weld eliminated by either induction heating stress improvement (IHSI) or other means judged to be fully effective and the reactor water chemistry environment modified by hydrogen additions to further reduce the potential for cracking.

To implement this long-range plan, NUREG-0313, Revision 1, will be revised to incorporate the recommendations made by the present Pipe Crack Task Group, and the public and the NRC internal comments. After revising NUREG-0313, Revision 1, a generic letter incorporating this implementation document will be sent to all BWR licensees asking for their proposal for bringing their plant(s) into compliance with 10 CFR 50.55a(g).

Spent Fuel Pool Modifications

Several licensee requests to increase on-site spent fuel pool storage capability were made this year, and more are expected in fiscal year 1985. Requests from Maine Yankee, Trojan (Ore.), R. E. Ginna (N.Y.), St. Lucie 2 (Fla.), and Virgil Summer (S.C.) were approved by NRC this year. Actions on Turkey Point Units 3 and 4 (Fla.) and Yankee Rowe (Mass.) are pending. NRC has approved more than 90 such actions over the past several years. The Maine Yankee approval, in addition to providing for increased on-site storage capability by reracking, allows for fuel assembly pin consolidation of up to 20 standard assemblies. These actions are necessary until the Department of Energy begins accepting licensee spent fuel at a permanent repository or equivalent facility.

Safety evaluations of deterministically derived accident scenarios include those resulting from fuel handling, rack and gate drop, cask drop and/or tip, heavy load drops, and tornado missiles. Additionally, NRC has recently initiated action on Generic Issue 82, "Study of Beyond Design Basis Accidents in Spent Fuel Pools," to investigate the plausibility and conditionally dependent consequences and risks of hypothetical accidents beyond the present design bases. The timeliness of this generic issue was underscored by an event on August 21, 1984 at the Haddam Neck plant (Conn.), where the refueling cavity water seal failed. This could have led to partial spent fuel pool drainage with a resultant uncovering of a short length of some stored fuel.

Instrumentation to Detect Inadequate Core Cooling

In December 1982, all licensees of pressurized water reactors were directed (by Letter or Order) to install instrumentation for detection of Inadequate Core Cooling, as described in TMI Action Plan Item II.F.2 of "Clarification of TMI Action Plan Requirements" (NUREG-0737, November 1980). The required instrumentation consists of upgraded subcooling margin monitors, upgraded core-exit thermocouples, and a reactor coolant inventory tracking system (ITS). Staff review and approval of licensee submittals describing the design and schedule for implementing the final instrumentation system is required. In addition, an implementation review of the final system, including its integration into emergency operating procedures, is required for each plant subsequent to installation of the ITS. The implementation reviews for plant-specific approval of ITS installations are being performed as NRR multi-plant action item F-26. During fiscal year 1984, the staff completed its implementation review of the inadequate core cooling instrumentation systems for 10 operating plants. These included completed ITS installations consisting of Westinghouse differential pressure systems for Summer Unit 1 (S.C.), McGuire Units 1 and 2 (N.C.), North Anna Units 1 and 2 (Va.), Surry Units 1 and 2 (Va.), and Salem Units 1 and 2 (N.J.). Yankee Rowe (Mass.) received an exemption from the requirement for an ITS because of unique design characteristics. Implementation reviews for 16 additional plants (including several with Combustion Engineering Heated Junction Thermocouple systems) are expected to be completed in fiscal year 1985.

Occupational Exposure Data

The NRC staff has been tabulating the annual average occupational doses at light water reactors (LWRs) since 1969. Between 1969 and 1973, the annual average doses for pressurized water reactors (PWRs) exceeded those for boiling water reactors (BWRs). Since 1974, however, the annual average doses at BWRs have exceeded those at PWRs. Although both PWR and BWR annual dose averages have fluctuated over the years, the overall trend between the mid-1970s and 1980 was one of increasing annual dose averages. During the two-year period from 1980-1982 the annual dose average for LWRs dropped by nearly 11 percent. However, in 1983, the annual dose average for LWRs increased by nearly 7 percent. This overall increase was due to a 12 percent increase in BWR doses in 1983 and a 2 percent increase in PWR doses for the same year. Special maintenance jobs contributing to BWR doses in 1983 included torus modifications, recirculation system pipe inspections and repair or replacement, and TMI modifications. A large contributing factor to PWR doses was steam generator repair work (tube sleeving, replacement, and plugging).

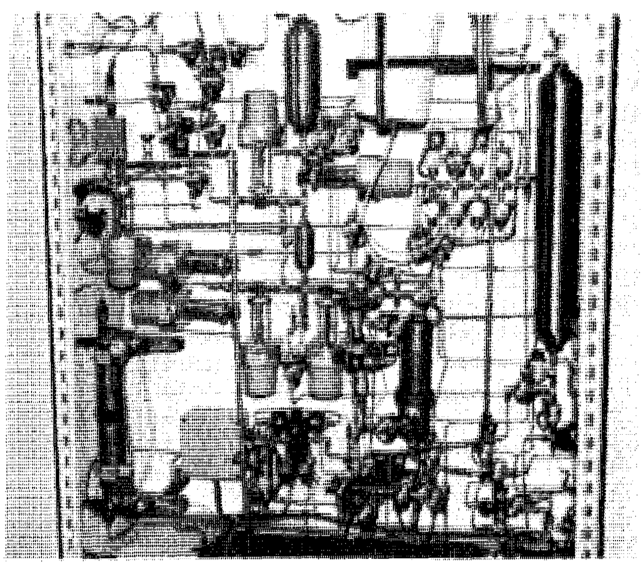
The NRC has several ongoing contracts with Brookhaven National Laboratory in the area of occupational doses reduction at LWRs. One of the objectives of these studies is to identify and evaluate high-dose maintenance tasks in LWRs. This will allow the NRC and industry to focus on major dose-reduction targets in an effort to reduce LWR doses. Other objectives are to determine the cost-effectiveness of certain dose reduction techniques that have been applied at only a few LWRs, determine the extent of use of high-reliability, low-maintenance equipment at LWRs, and compare the occupational doses at U.S. and foreign LWRs.

Source Terms: Releases of Radionuclides in Severe Accidents

Estimated releases of radioisotopes to the environment (i.e., "source terms") were undertaken for the GESSAR II standard plant Severe Accident Safety Evaluation Report, based principally on a study that employed the methodology and sample calculations provided by contractors to the Accident Source Term Program Office. The GESSAR II estimates considered ranges of fission product retentions in the primary systems, suppression pool, and containment. Fission product aerosol generation during core-concrete interactions were also evaluated. These sample calculations are summarized in a draft report by the Battelle Columbus Laboratories on "Radionuclide Release under Specific Accident Conditions" (BMI-2104, Vols. II-VI, July 1984). The methodology for source term reassessment is being independently reviewed by a study group of the American Physical Society. The results of this review will be reported by the study group in early 1985.

The GESSAR II source term evaluations indicate that few accident scenarios result in off-site radiation doses high enough to produce early radiation illness or fatality. These few derive from a combination of very low probability events that include the most rapidly evolving accidents with early containment failure; the worst case assumptions regarding uncertainties surrounding causal parameters in the accident evaluation models; and the least favorable combinations of weather conditions for dose dispersion. From the evaluations of severe accident scenarios initiated by events internal to the plant (e.g., equipment failure or operator error), the staff concluded that the risk of early fatality for the GESSAR II design is small compared to that predicted by the Reactor Safety Study (RSS). This conclusion is based on recent advances in source term evaluation methodology and to certain design improvements in GESSAR II over those in the BWR plant used as a model in the RSS.

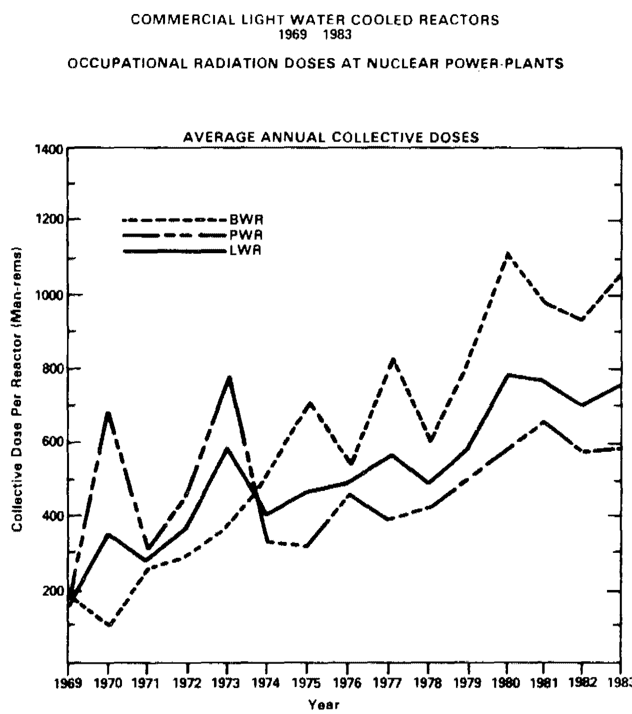
The industry, notably the Industry Degraded Core Rulemaking Program (IDCOR), has sponsored research to improve source term evaluation methodology which augments the research sponsored under NRC's Severe Accident Research Program (NUREG-0900). Throughout 1984, technical exchange meetings were held between IDCOR representatives and the NRC staff to identify and resolve significant differences in the codes and models for source term evaluation. While some disagreements remain, a consensus emerged from these meetings that, for the more probable severe accident sequences studied, the source terms are likely to be less than those predicted by the RSS.



Shown above is the internal cabinet arrangement of post-accident liquid sampling system equipment at one of the Oconee nuclear station units at Seneca, S.C.

Achieving "ALARA" in Occupational Radiation Exposure

NRC efforts directed towards developing effective efforts to reduce radiation exposures to levels "as low as reasonably achievable"(ALARA) in the operation of commercial power reactors include: regulatory action, radiological safety reviews, radiation protection/ALARA inspections, and interaction with industry. All proposed regulatory actions over the full range of regulations, guidelines, and generic and other safety issues require that occupational doses incurred as a result of implementing these actions be considered along with other decision criteria. Thus, license applications and amendments re-



quire a staff radiological safety/ALARA review utilizing NRC standards. Each NRC Region conducts inspections in radiation protection/ALARA to identify possible deficiencies and desirable improvements at each facility. In addition, the NRC staff has participated in cooperative efforts with industry to achieve mutual goals in radiation protection/ALARA.

Coordination with INPO. A "Coordination Plan for Radiological Protection Activities," prepared and approved by the NRC and the Institute of Nuclear Power Operations (INPO) in March 1983, supports alternative regulatory concepts that recognize the contributions of industry self-policing programs, to the extent that such programs are effective and consistent with NRC responsibilities. The plan recognizes the INPO program of radiological protection evaluations and also INPO's assistance activities for member utilities. One main goal of this INPO effort is to minimize occupational radiation exposure in the nuclear industry. During fiscal year 1984, the NRC staff participated in a number of observer visits during INPO site evaluations of radiation protection and ALARA. NRC and INPO staff also participated in information exchanges regarding radiation protection performance data and evaluation criteria. The NRC staff will evaluate the progress and success of this INPO/industry effort, and will adopt final evaluation criteria for a September 1985 summary evaluation.

Radioactive Effluents Summary and Analysis

The program for implementing Radiological Effluent Technical Specifications (RETS) in operating reactors continued during the reporting period. By the end of fiscal year 1984, about 8 percent of the operating nuclear reactors had received technical approval for their specifications. Many of the plants implemented their RETS during fiscal year 1984. The balance of those that have been approved are to be implemented during 1985. In addition, regulatory action on the remaining twenty percent is expected during 1985.

Licensees submit radioactive effluent reports on a semi-annual basis and NRC publishes summary reports entitled "Radioactive Materials Released from Nuclear Power Plants." These reports contain nuclide-by-nuclide summaries of airborne and liquid effluents, as well as quantities of solid wastes shipped off-site. Doses to members of the public are estimated. In addition to the semi-annual effluent reports, licensees submit an annual radiological environmental operating report to the NRC containing the results of their radiological environmental monitoring programs. The semi-annual effluent reports and the annual radiological environmental operating reports are available for public inspection in local Public Document Rooms.

Evaluation of Control Room Habitability

Since 1980, the Advisory Committee on Reactor Safeguards (ACRS) has held several meetings with the NRC staff to discuss the subject of control room habitablity. These meetings have occasioned a number of ACRS letters expressing specific concerns, to which the staff has responded. On August 15, 1983, a plan, jointly developed by the Offices of Nuclear Reactor Regulation (NRR) and Inspection and Enforcement (IE), that would address the latest ACRS concerns and recommendations was approved. This program plan was implemented by the Control Room Habitability Working Group during fiscal year 1984. The Working Group reviewed the ACRS concerns and recommendations and recommended actions to be taken to address these and other concerns raised *sua sponte*. In a report on "Control Room Habitability" to be published in December 1984, the Working Group made a number of recommendations on the subject:

- Maintenance of the control room environment to be considered equal in importance to the maintenance of control room equipment.
- Centralization of review responsibility for control room habitability so that interaction between NRC review branches yields a systems integration approach.
- Assurances that review branches independently verify the adequacy of control room designs.
- Modification of certain Technical Specifications to correct present errors in the specifications and to cover more adequately areas addressed in the Safety Evaluation Report.
- Establishment of interactions between branch reviewers and regional inspectors.
- Identification of methods to enhance external input to regulatory policy on control room habitability, including industry feedback.

An important part of the program plan involves the survey of operating reactors and plants nearing completion to determine how control room designs are transformed to as-built systems and what testing protocol is actually used by control room operators. This work was completed in August 1984 and a contractor report is scheduled for publication during the first quarter of fiscal year 1985.

Testing Methods for Activated Carbon

Laboratory analysis of activated carbon in ventilation filter systems at nuclear plants is required by the Technical Specifications for Engineered Safety Features (ESF). Because of the safety importance of these filter systems, the frequency of such tests is at least once per refueling cycle (about 18 months) or, more frequently, based upon the hours of operation of the system and the occurrence of chemical and physical processes that potentially could degrade the performance of the activated carbon. In addition, plant procedures may require that many non-ESF activated carbon filter units installed in normal ventilation systems be tested. The purpose of the laboratory analysis is to ensure that the carbon adsorbers are capable of operating at an efficiency at least equal to that assumed in the NRC staff's Safety Evaluation Report (SER). If the laboratory analysis shows that the carbon material has a removal efficiency for radioiodine less than the value specified in the Technical Specifications or by plant-specific procedures, then the carbon in the filtration system must be replaced.

In 1982, the Committee on Nuclear Air and Gas Treatment of the American Society of Mechanical Engineers (ASME) invited a number of laboratories from the United States and elsewhere to participate in a round-robin testing program of several nuclear-grade activated carbon samples, both new and used. The disparity of results of the round-robin tests raised two concerns about the reliability of laboratory test results on ESF and non-ESF systems that NRC licensees are now receiving:

- Are laboratories performing analyses of carbon in accordance with American Society for Testing Materials (ASTM) standards?
- What problems exist with the test method or laboratory performance in applying this method?

During fiscal year 1984, the NRC contractor reviewed the previous round-robin test data, performed initial sensitivity tests on various parameters considered important in the ASTM test method, and conducted a workshop with international participation to review preliminary findings. As a result, agreement was reached with the various laboratories to participate in a new inter-laboratory comparison to be conducted early next year.

Fire Protection

The NRC fire protection rule for nuclear power plants became effective on February 17, 1981. It required all licensees of plants licensed prior to January 1, 1979, to submit plans and schedules for meeting the applicable requirements, a design description of any modifications proposed to provide alternative safe-shutdown capability, and any requests for exemption from specific requirements of the rule. For plants licensed after January 1, 1979, the criteria of the Standard Review Plan, which includes the requirements of the fire protection rule, are used in the NRC staff review prior to issuing a license.

The licensees for 69 plants licensed prior to January 1, 1979, were required to respond to the rule. Exemptions were requested for 64 of the plants, and modifications to provide alternative safe-shutdown capability were proposed for 55 plants. By the end of fiscal year 1984, licensing action on the exemption requests for 57 plants and approval of modifications for alternative shutdown capability for 55 plants had been completed. The licensees for nine plants where exemption requests were denied proposed modifications for alternative shutdown capability during fiscal year 1984. Additional requests for exemptions and modifications to provide alternative safe-shutdown capability have been received in response to inspection results and to the issuance of Generic Letter 82-33 giving staff positions regarding conformance with certain provisions of Appendix R.

The regional offices have continued to inspect for compliance with the fire protection rule. During fiscal year 1984, two inspections at sites with plants licensed prior to January 1, 1979, were conducted. Some of the scheduled inspections were postponed when the licensees filed requests for technical or schedular exemptions or revised proposals for modifications.

The regional offices have begun to inspect plants licensed after January 1, 1979, to verify as-built configurations for conformance with the plant's fire hazards analysis. The six plants inspected were in the operating license review process, and the results were taken into account in the issuance of the licenses.

Regional office inspections of operating plants to verify compliance with NRC fire protection requirements have identified significant items of non-compliance. Similar inspections at plants to be licensed also have identified several deficiencies which have resulted in delays in the implementation of the complete fire protection program.

The Nuclear Utility Fire Protection Group has requested interpretations of NRC fire protection requirements on six subjects associated with the implementation of Appendix R to 10 CFR 50. They have raised many questions concerning issues which they say are hampering the licensees implementation of NRC fire protection requirements. The NRC conducted workshops in each Region to respond to these questions. The results of these workshops are now being reviewed with the Commission.

The NRC staff, with the assistance of a contractor, is reviewing present fire protection guidelines using probabilistic risk assessment techniques to evaluate the importance of each of their major elements. In addition, a fire protection research program has been initiated to study control room fires, fire risk analysis and the effects of fire suppression systems on safety systems.

Operational Safety Assessment

Assessment of the significance of unanticipated events at operating reactors involves NRC Regional and Headquarters offices. Prompt reviews and technical support are provided on issues and events of possibly immediate safety concern. In addition, the NRC staff has been called on frequently to review event sequences against licensing analyses, evaluate plant and operator performance during events, identify generic safety implications, review licensee analyses, and evaluate corrective actions prior to plant restart.

Examples of such events occurring in fiscal year 1984 at operating reactors are:

(1) Through wall crack in vent header inside BWR containment torus at Hatch Unit 2 (Ga.) on February 3, 1984.

(2) Stuck open steam side code safety valve at Davis-Besse (Ohio) on March 2, 1984.

(3) Six of 37 control rods failed to scram at Fort St. Vrain (Colo.) on June 23, 1984.

(4) Loss of all AC electric power at Susquehanna Unit 2 (Pa.) on June 26, 1984.

Equipment Qualification

The NRC requires that equipment important to safety be qualified to operate under seismic, dynamic and environmental conditions that may be associated with an earthquake or an accident. To date, most efforts in this area have been addressed to the environmental qualification of electrical equipment. The NRC staff, with the assistance of a contractor, evaluated the environmental qualification of electrical equipment for 71 operating reactors. Technical evaluation reports for these reactors were completed by the contractor in 1983 and were used by NRC staff as a basis for preparing safety evaluation reports.

A recent rule (Section 50.49 of 10 CFR Part 50, effective February 22, 1983, establishes specific requirements for environmental qualification of electric equipment and sets a deadline by which the equipment must be qualified. During fiscal year 1984, NRC staff met the licensees of the 71 operating plants previously reviewed to discuss identified qualification deficiencies and their resolution. Safety evaluation reports (71) will be completed by December 1984. The analyses and documentation will be audited by the staff during follow-up inspections of the licensees' environmental qualification files.

NRC staff continues to have the assistance of the Brookhaven National Laboratory and the Idaho National Engineering Laboratory in performing plant site audits and preparing safety evaluation reports needed to review applications for operating licenses. Ten site audits were conducted during fiscal year 1984 and an estimated 10 more are to be conducted in fiscal year 1985.

Geosciences Activities in Safety Reviews

Most nuclear power plants are in the eastern or central United States, which, unlike California, is an intra-plate region where relatively little is known about the sources and causes of earthquakes. This lack of knowledge has resulted in some controversy, when either hypotheses change or new information appears, in postulating design earthquakes . For example, increased uncertainty about the likelihood of the recurrence of earthquakes the size of the 1886 Charleston, S.C., event (magnitude 7.0 to 7.5) in other parts of the eastern seaboard has led to several extensive probabilistic assessments of seismic hazard—in addition to the long range deterministic programs—for all nuclear power plants east of the Rocky Mountains. The NRC and its contractor, Lawrence Livermore National Laboratory, have undertaken a program to characterize seismic hazard for this region on a probabilistic (as distinct from deterministic) basis. This project is an outgrowth of an earlier study performed as part of the NRC's Systematic Evaluation Program (SEP). As in the SEP, the fundamental characteristic of the methodology is to use expert opinions for all the input data. The NRC recently published a report (NUREG-CR/3756) on interim progress that the NRC and the contractor have made in the short-term probabilistic assessments of seismic design earthquakes in the eastern seaboard. The Electric Power Research Institute (EPRI) is also conducting an assessment of seismic hazard in the eastern U.S. During fiscal year 1984, NRC geologists and seismologists attended five workshops which were part of EPRI's research. A significant result from the deterministic program has been the recognition by U.S. Geological Survey (USGS) scientists of geologic evidence for pre-1886 earthquake-induced liquefaction near Charleston, S.C.

Recent ground motion recordings in the East have also raised questions as to the nature and source of high frequency ground motion in that part of the country. One of the earthquakes from which high ground motions were recorded was the January 1982 (magnitude 5-3/4) New Brunswick earthquake, which is the largest earthquake to have occurred in that part of Canada or New England in historic times. The NRC is providing funds to the USGS to study this event and its aftershocks. NRC geologists conducted a reconnaissance of the epicentral area in October 1983.

Field investigations have recently been made of the Meers fault in southwestern Oklahoma, approximately 11 miles northwest of Lawton. This fault is of interest because it may be the only known example of recent major tectonic surface rupture in the central and eastern United States. The northwesterly-trending fault is relatively young, is at least 26 kilometers in length and has up to 5 meters of apparent topographic offset. The region around the Meers fault has a low seismic activity with no known events definitely associated with the fault. Applying standard formulas indicates a possibility of generating an earthquake of magnitude 6-to-7. The implications of such an event on the Meers fault was assessed for existing nuclear power plant sites. The NRC is currently funding additional field work, including aerial photography, trenching, and radiometric age dating in an attempt to establish the capability of the Meers fault. If it is found to be capable, the staff will then evaluate the relevance of the fact to an understanding of eastern U.S. seismicity.

The staff has a continuing interest in the effects of new information on earthquake tectonics on the Diablo Canyon nuclear power plant (Cal.). This concern is being addressed through a license condition that requires an assessment during the next three-to-four years of all new geologic and seismic information related to the seismic design basis. Along with the investigations to be conducted by the utility to satisfy the requirements of the licensing condition, the staff will perform an independent geologic and seismic re-evaluation of the site in addition to reviewing the work of the licensee during the same time period. Staff geologists conducted a reconnaissance of the site and region around Diablo Canyon in June 1984.

On October 28, 1983, an earthquake of magnitude 6.9 occurred near Chalis, Ida. This area was visited by a geologist and seismologist from NRC. The earthquake occurred on a fault previously assumed to be capable. Damage to structures in the area was minimal due to sparse population in the epicentral area. The staff observed cracking in buildings and highway surfaces, fault offset, sand boils, and rock slides.

The staff also visited the epicentral area of the April 24, 1984, magnitude 5.8 earthquake in the Halls Valley of California. This event was associated with the Calaveras fault, which has generated similar size historic events. The earthquake caused heavy damage to a bridge built across the fault. The highest horizontal peak ground acceleration (1.3g) ever recorded from an earthquake was reported from the Coyote Dam abutment station. There was little damage observed to engineered structures during the earthquake. For example, a recently constructed IBM computer center about 10 kilometers from the rupture sustained no observable damage and experienced no interruption of computer operation.

During the year, staff members visited Egypt, Israel, Austria, Italy and Taiwan to assist in geologic and seismic aspects of nuclear plant regulation.

NRC Dam Safety Program

The NRC regulates safety-related dams at nuclear power plant sites. The NRC Dam Safety Officer (DSO) represents the NRC on the Federal Interagency Committee on Dam Safety (ICODS) to assist the Federal dam safety program. The NRC Dam Safety Advisory Group, which is under the direction of the DSO and includes representatives from the NRC regional offices, has developed a schedule and identified long-range activities and resources which the group believes are needed for the NRC and the licensees to implement the "Federal Guidelines for Dam Safety" for the dams at the nuclear plant sites and for uranium mill tailings dams. The NRC Dam Safety Advisory Group is completing a Commission Paper on Dam Safety at NRC Licensed Facilities.

In July 1984, coordination efforts by the DSO resulted in an agreement to revise a portion of Regulatory Guide 3.11, "Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills" and Staff Position WM-8201. The revisions will accommodate concerns expressed by the Mine Safety and Health Administration (MSHA) on hydrologic criteria and provide clearer guidance to the operators of uranium mill facilities on the proper design-basis storm to be taken into account and also in determining the hazard classification of a tailings dam project.

Structural Engineering

The NRC staff continued, in 1984, to investigate allegations related to nuclear plant structural soundness, to resolve design and construction deficiencies, to provide support to the Regions and other offices and process licensing actions. Salient cases included the following:

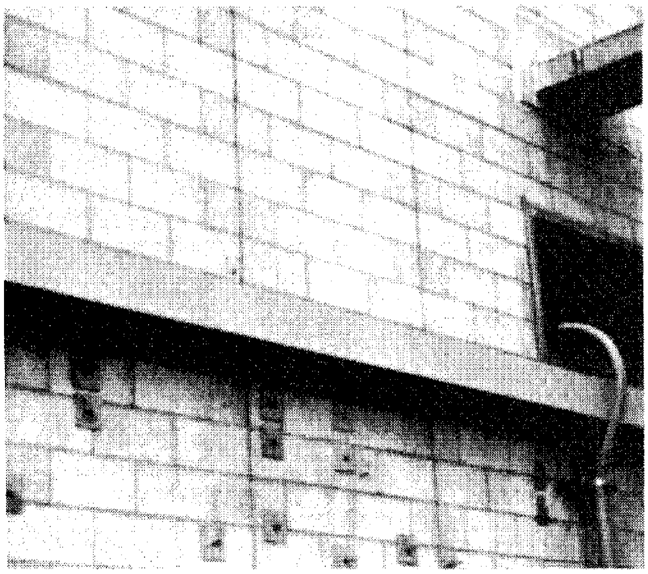
Relicensing of Diablo Canyon Unit 1 (Cal.) required extraordinary investigative effort because of the discovery, in September 1981, of errors in the seismic design of the plant structures and equipment. The NRC staff participated in evaluating numerous allegations on a variety of structural subjects and , with the aid of consultants at the Brookhaven National Laboratory, successfully resolved all those submitted before July 1984.

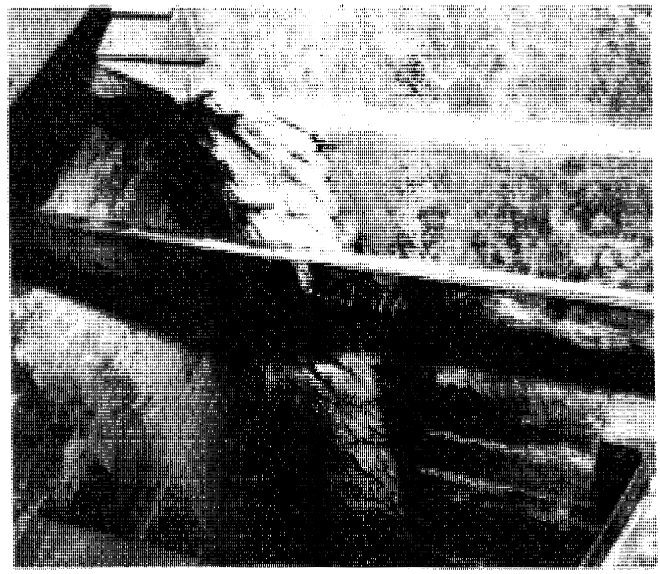
The NRC staff participated in the review of the extent of the alleged quality assurance/quality control program breakdown in the construction of Waterford Unit 3 (La.) and in the evaluation of its effects on the integrity of the safety-related structures. The major concern has been cracking of the reinforced concrete foundation mat and the structural adequacy of safety-related masonry walls. Similarly, evaluation of allegations at Comanche Peak plant (Tex.) is continuing.

A review of the structural integrity of the diesel generator building (DGB) at Midland (Mich.) concluded that in spite of the existence of extensive cracks, the DGB is structurally sound. However, as a precautionary measure, a monitoring program was recommended.

At Limerick (Pa.), the capacity of "category I" structures to withstand blast overpressure resulting from a postulated rail train accident nearby involving explosive cargo was reassessed, as were the effects of a cooling tower overturning or collapsing on other safety-related structures. In addition, the staff evaluated allegations of design inadequacy in floor slabs in the Limerick reactor building.

After structural deficiencies were found in some of the masonry walls of the Trojan power plant (Ore.) in 1980, the staff evaluated the adequacy of masonry walls at other operating reactors. In fiscal year 1983, Safety Evaluation Reports on masonry walls were issued for Oconee 1, 2, 3 (S.C.), Kewaunee (Wis.), Fitzpatrick (N.Y.), Ft. Calhoun (Neb.), Peach Bottom 2 and 3 (Pa.), Calvert Cliffs 1 and 2 (Md.), Maine Yankee (Me.), and Farley 1 (Ala). Structural design audits were conducted at Marble Hill (Ind.), Hope Creek (N.J.), and Nine Mile Point 2 (N.Y.). In addition, Integrated Design Inspections were conducted at Sea-





The NRC, with the help of various independent research organizations, continued in 1984 to review alleged structural deficiencies in a number of plants. Shown above are two types of corrective measures taken by licensees in response to such reviews. At left is an exterior view of the turbine building masonry wall at San Onofre Unit 1 (Cal.) showing the attachment of a structural steel beam to strengthen the wall. At right, an expansion joint is cut in a screenhouse wall abutting the reactor wall at Unit 1 of the Nine Mile Point nuclear power plant at Scriba, N.Y.

brook (N.H.) and River Bend (La.) concerning execution of the design, verification of the quality assurance and quality control programs, review of the documentation controls and systems, and qualification of the personnel responsible for the structural design.

With the assistance of Franklin Research Center, the staff has been reviewing license amendment applications for high density spent fuel racks and has completed the review for McGuire Units 1 and 2 (N.C.), Summer (S.C.), Oyster Creek (N.J.) and St. Lucie 2 (Fla.). Plants still under review are Ginna (N.Y.) and Turkey Point 3 and 4 (Fla.).

Vertical tendons of the Ginna containment were found to have undergone excessive prestress losses. The vertical tendons are connected through flexible conduit couplings to rock anchors which are grouted in the foundation rock. Whether the unexpectedly large prestress loss has any connection with the rock anchor is difficult to determine, since there is no way to inspect or test the rock anchors. The licensee conducted an extensive investigation of the causes and has submitted the results to the NRC staff for review. Because of the uniqueness of the problem, the NRC staff has sought expert opinion. The NRC staff is also investigating a reactor vessel tendon corrosion problem at the Fort St. Vrain Plant (Colo.). As a consequence of these problems and the learning gained from them, the regulatory guide and the technical specifications for tendon surveillance are under major revision.

Foundations

Midland Nuclear Power Plant. Underpinning of the Auxiliary Building and Service Water Pump Structure foundations had been proceeding at the Midland (Mich.) nuclear power plant until July 1984, when the applicant announced a decision to shut down further design and construction activities. The applicant intends to maintain the CP and OL application to provide options regarding future licensing activity. The underpinning foundation work, which was estimated to be about 35 percent complete at the Auxiliary Building, was necessary to replace the inadequately compacted plant fill. Underpinning consists of replacing the supporting fill with reinforced concrete piers and permanent foundation walls that are extended from the base of the existing building foundations down to a deeper, sufficiently competent natural foundation soil.

Field work necessary to leave the partially completed remedial work in a safe condition was completed by August 1984, except for draining of the approximately 880-acre cooling pond. Settlement monitoring of the Auxiliary Building which is supported by both jacks on completed concrete piers and plant fill soils will be continued indefinitely. Draining the cooling pond requires approval of the State of Michigan and is necessary to remove the major source of groundwater recharge to the remaining underpinning foundation excavations. Deactivation of temporary construction dewatering systems will proceed after the cooling pond is drained.

Palo Verde Nuclear Generating Station. Water leakage from pressurized temporary utility lines beneath the foundations of the auxiliary buildings at the Palo Verde plant (Ariz.) has caused erosion of supporting foundation soils near the pipes. The leakage is thought to be a result of breaks in the water lines caused by pipe corrosion and settlement-induced stresses at critical points of structure loading. Some movement of soil particles occurred because of the piping water pressures, as soil and water was deposited in several locations in the deeper building sections where openings (called seismic gaps) are located between buildings. The gaps are conservatively allowed for in design to permit potential movements of structures under severe earthquake conditions.

The applicant corrected the problem by forcing grout into the voids and spaces left by soil erosion. An evaluation of the adequacy of the applicant's remedial grouting program and any potential long-term effects on safe bearing capacity and soil compressibility characteristics was expected to be completed by the staff and its consultant late in 1984.

Waterford Nuclear Plant. In July 1977, at the Waterford (La.) plant, a number of east-west oriented cracks moistened with weeping water were discovered at the top of the concrete mat supporting all safety class structures within the ringwall for the containment structure. In May 1983, additional cracks and accompanying weeping water were discovered in the base mat outside the containment structure. Some of those cracks were found to extend to vertical walls and to extend up those walls. A motion was filed with the appropriate Licensing Board in 1983 regarding the previous assessments by the NRC staff of the adequacy of the foundation mat. This led the NRC to initiate an unprecedented review program encompassing all related licensing, inspection, hearing and allegation issues.

Two groups were formed to review outstanding base mat related issues. One group assembled at the Waterford site to gather information and review documentation related to civil/structural construction aspects in order to assess the validity, safety significance and generic implications of the allegations and to determine whether the problems that occurred during construction had rendered the design assumptions invalid. The other group, comprised of NRC staff and assisted by consultants from Brookhaven National Laboratory, was to review base mat related matters and to perform or identify the necessary analyses and actions needed to assure its adequacy.

The on-site review group has identified four foundation base mat issues that require further efforts on the part of the applicant. The staff anticipates that satisfactory resolution of these issues can be reached.

The design review group has found that the numerous construction difficulties encountered may have caused some differential settlements which, in turn, may have contributed directly or indirectly to the observed cracking of the foundation mat. Difficulties encountered during construction included: (1) local high groundwater conditions incompletely controlled by the dewatering system, which may have caused soil disturbances, mud spurt, standing water in some areas, and difficulties in compaction of the clam shell filter under the facility; (2) the measured heave that was 2-to-4 times the anticipated maximum heave used in the design; (3) the variable degree of compaction reached in the six clam shell filter strips; and (4) certain significant changes in effective soil pressures due to dewatering. The effects of these construction difficulties had not been considered in the applicant's analyses. To assess the significance of the cracks, a series of ultrasonic tests were performed to gauge the depth, width and orientation of the prominent cracks of the base mat. At the request of the staff, the applicant engaged a contractor to perform non-destructive tests at the site. The results indicate that three primary east-west oriented cracks run from the shield wall to the side walls, and vary in depth in an undulating manner, from 2-to-4 feet to as much as 9-to-10 feet in certain locations. The applicant is currently evaluating the test results and will propose appropriate actions. The staff is staying in close contact with the applicant.

Assigning dedicated task groups to resolve problems is an innovation for the NRC which is proving effective in upgrading licensing efficiency and eliminating unnecessary delay.

Transamerica Delaval Diesel Generators

During a load test on August 12, 1983, the main crankshaft on one of the three emergency diesel generators (EDGs) at the Shoreham Nuclear Power Station (N.Y.) broke in two. The EDGs at Shoreham were manufactured by Transamerica Delaval, Inc. (TDI), which has also supplied 54 other EDGs to 14 other nuclear power plant sites in the United States.

During the evaluation of the failure and subsequent repairs of the Shoreham engine, information related to the operating history of TDI engines and the quality assurance (QA) program of the manufacturer came to light which calls into question the reliability of all TDI diesels. As a result, 13 nuclear utilities formed an Owners Group to address the issue through a program consisting of three elements.

The first program element, known as Phase I, consists of a review of a limited number of components with known design and/or manufacturing concerns that warrant attention on an accelerated schedule. Phase I was essentially completed in fiscal year 1984.

Phase II of the program—Design Review/Quality Revalidation (DR/QR)—is a review of a larger set of engine components to assure the adequacy of their design and manufacture, including specifications, quality control and assurance and operational surveillance and maintenance. The first report in Phase II was filed in July 1984 and the remainder are to be submitted in fiscal year 1985.

In addition to these two program elements, the Owners Group has proposed engine teardowns and inspections at all plants. These inspections will provide the data required for the DR/QR program. For plants that will be licensed on an interim basis, the inspections are also used for assessing the condition of an engine after preoperational tests to determine whether interim licensing can proceed. Since the Owners Group is scheduled to be phased out early in 1985, they will issue a DR/QR in some cases based on an inspection performed on a similar engine and will rely on the licensee to validate the DR/QR report when the engine is torn down for inspection.

The NRC staff Safety Evaluation Report (SER) was issued on August 13, 1984, outlining the elements that will be considered with regard to the TDI diesels for licensing or restart of a plant on an interim basis, pending completion of Phases I and II. Licensing or restart can proceed on an interim basis, i.e., until the next refueling outage, under the condition specified in that SER.

The interim basis for licensing will be reviewed after completion of Phase I and Phase II of the program to see what modifications need to be made to the license conditions. A final SER will be issued for each of the plants that are being licensed or restarted on an interim basis. For plants where the Phase I and Phase II program is scheduled to be completed sufficiently ahead of the licensing or restart of the plant, a final TDI Diesel SER will be developed which encompasses the results of Phase I, Phase II and the operational history of a lead engine.

Upon completion of all the DR/QR SERs (Phase II), the staff will review the results to determine if any generic or plant-specific implications may exist for each TDI plant and will modify the final SERs accordingly.

Beginning in early 1984, NRC teams initiated inspections for possible cracks in reactor containment vent headers. Vent headers transport steam that might escape from a pressurized reactor coolant system into the suppression pool. The actions came about as a result of the discovery of such cracks at the Edwin I. Hatch plant, Unit 2, in Georgia. Shown here are members of a team inside the reactor torus of the Millstone nuclear power station in Connecticut. The men are standing on a catwalk above the suppression pool and directly beneath an 18-inch opening in the containment air/nitrogen purge system. They are checking out the possibility of cracks caused by liquid nitrogen impingement and the possible need for debris screens on such containment penetrations.

Vent Header Cracking Incident

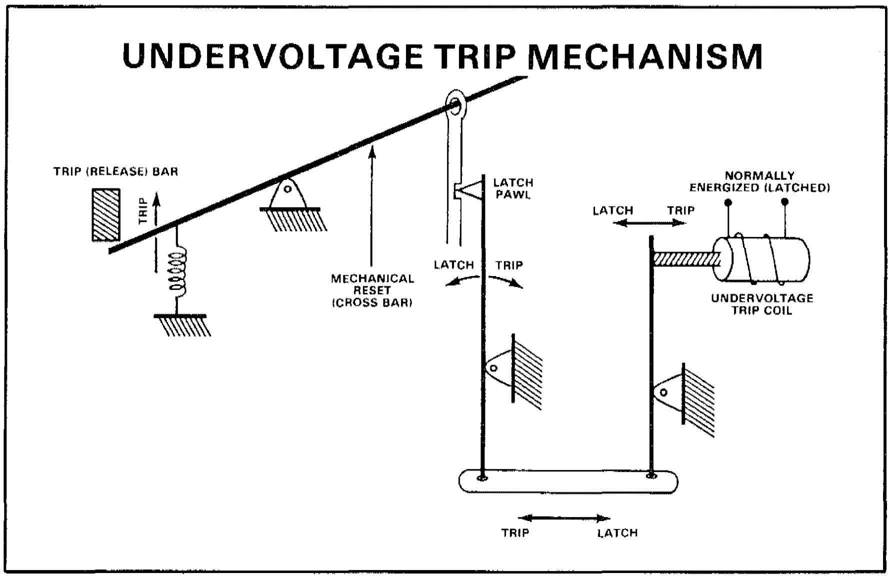
An abnormal occurrence was reported by Georgia Power Company at Hatch 2 (Ga.) on February 3, 1984. (See "Abnormal Occurrences" in Chapter 4.) Hatch 2 is a boiling water reactor (BWR) with a Mark I containment design. The primary containment for this design is a pressure suppression system consisting of a drywell (a steel structure housing the reactor vessel, recirculation system, and connections to the reactor coolant system boundary); and a wetwell (a steel pressure vessel in the shape of a torus below the drywell). Eight circular vent pipes form a connection between the drywell and the torus. The vent pipes exhaust into a 54-inch diameter continuous vent header from which 80 downcomer pipes extend downward into the water in the torus. During a routine visual inspection of the torus interior, the licensee discovered an open circumferential crack in the vent header. This crack was determined to continue almost completely around the vent header and to be through-wall. Because of this event, the NRC required all operating nuclear power plants with Mark I containments that were in a shutdown mode to inspect the torus vent header for any material failures and report the results to the NRC. These inspections revealed no cracks.

The location of the crack at Hatch 2 was directly below a nitrogen line outlet. Nitrogen is released inside the primary containment in order to create an inert atmosphere during plant operation. The nitrogen supply system is designed to evaporate liquid nitrogen and warm the nitrogen gas before it is discharged. Apparently, this system had failed during inerting of the containment, thus discharging cold nitrogen directly onto the vent header. The impingement of the cold nitrogen created a brittle-fracture type of failure in the vent header material. Stresses generated by the cooling of the header contributed to crack initiation and propagation.

Corrective actions being performed by the licensee include repairing the damaged vent header and modifying the inerting system piping configuration and temperature controls, along with the inerting procedures. The NRC and GE are working together to provide the industry with recommended actions for BWR Mark I and II owners to implement in order to prevent the occurrence of a similar event.

Salem Scram Events and Resultant NRC Actions

On February 25, 1983, a signal that water in one of the steam generators was too low generated a reactor trip signal to the Salem Unit 1 (N.J.) reactor, during a routine startup following a refueling outage. Both reactor trip breakers failed to open until operators manually tripped the reactor about 30 seconds later. At that time, the



This diagram shows the functional operating scheme of the mechanism in the Salem nuclear generating station, Unit 1 (N.J.), reactor trip breakers.

reactor trip breakers opened and the control rods dropped into the reactor core to bring the reactor to a stable shutdown condition. It was found that the reactor trip breakers had failed to open because of mechanical binding of the latch mechanism in the undervoltage trip attachment on the circuit breakers.

A number of short term actions were taken through Bulletins and an Information Notice issued by the NRC Office of Inspection and Enforcement (IE). IE Bulletins 83-01 and 83-04 were issued to require testing of all circuit breakers in reactor trip systems with an undervoltage trip attachment. IE Information Notice 83-18 described the failures of the reactor trip breakers discovered in the testing required by the IE Bulletins. In December 1983, the NRC issued IE Bulletin 83-08 to assure proper operation of circuit breakers with undervoltage trip attachments being used in safety-related applications other than as reactor trip breakers.

A detailed review of the Salem event and the licensee's corrective actions was performed prior to authorizing restart of the facility. This review is documented in NUREG-0916. In addition, an NRC Task Force was established to determine the generic implications of the Salem events. This work is documented in NUREG-1000 Volumes 1 and 2. As a result, Generic Letter 83-28 was issued describing intermediate-term actions to be taken by licensees. The actions address issues related to reactor trip system reliability and general management capability; specifically: (1) post-trip review, (2) equipment classification and vendor interface, (3) post-maintenance testing, and (4) reactor trip system reliability.

During fiscal year 1984, the NRC staff began its review of the initial responses to Generic Letter 83-28. The schedules for implementation of these actions are being developed consistent with the goal of integrating new requirements with other existing plant programs, considering the unique status of each plant and the relative safety importance of the improvements. The NRC staff in its review of the responses to Generic Letter 83-28, has approved the installation and operation of the automatic shunt trip for about half of the applicable operating PWRs. The shunt trip attachment will greatly improve the reliability of the circuit breaker, as its function is redundant to the function of the existing undervoltage trip attachment. The staff is encouraging each licensee to install the automatic shunt trip at the first refueling following staff review and approval of their design.

The NRC staff has been working with industry owners groups to improve reliability of circuit breakers using undervoltage trip attachments. The review of reported breaker failures indicates most failures can be attributed to the undervoltage trip attachments and their associated linkages.

Cooling Water Intake Clogged by Jellyfish

In late August and early September, an unusually large number of very large jellyfish clogged the cooling water intake screens at St. Lucie, Units 1 and 2 (Fla.). The clogging problem made it necessary, first, to reduce power on the one operating unit and, subsequently, to go to hot standby when the screens were damaged.

The event coincided with unusual environmental conditions including very calm weather, little to no wind and a stable sea, which persisted for a period of over one week. The potential problem was first noted by the licensee's biological consultant during regularly scheduled monitoring of the intake canal for the presence of sea turtles.

Subsequent aerial reconnaissance by the licensee showed the jellyfish to be concentrated for a distance of about 50 miles along the Florida coast.

The jellyfish which caused the intake clogging problem at St. Lucie are common and generally regarded as a nuisance in coastal waters. The two most abundant kinds—estimated to run in the hundreds of thousands of individuals at St. Lucie—were reportedly *Aurelia* and *Chrysasora*. Specimens up to one foot in diameter were observed.

Similar concentrations of jellyfish may occur again at St. Lucie if the right environmental conditions reoccur. To reduce the potential damage from this kind of event, the licensee has installed modified screens and more trash rakes to increase the cleaning efficiency at the intake screening structures. The licensee also is investigating means by which jellyfish can be floated to the water surface in the intake canal and contained by a floating "skimmer" boom for subsequent removal. The NRC staff is investigating this intake blockage event as a part of the ongoing technical resolution of Generic Issue 51, "Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems."

Protecting the Environment

Environmental Impact Assessment

The staff prepared its largest annual number of OL application environmental impact reviews of nuclear power plant sites during fiscal year 1984, publishing the results as either Draft or Final Environmental Statements.

These reviews entailed analysis of the environmental consequences of nuclear power station operation and involved visiting the construction sites and attending public meetings near the sites to receive input from local citizens and government officials on the scope of the NRC environmental impact review. Natural resource issues identified during the review process and addressed in the environmental statements included: water diversion and modification to creeks (Limerick); synergism of chlorine and thermal effluents (Harris); entrainment and impingement of aquatic biota (Limerick and Millstone); bio-fouling and its control (Hope Creek); fishery impacts from normal operation (all cases) and from accidents (WNP-3, Limerick, and River Bend); threatened and endangered species (shortnose sturgeon at Limerick and Hope Creek, and mussels at Braidwood); farmland disturbances from cooling tower drift (Hope Creek); and impacts to terrestrial vegetation from cooling pond fogging (Braidwood), cooling tower drift (Nine Mile Point), erosion (River Bend), and transmission line routing (several cases). Some issues required further analysis and disclosure via the NRC formal public hearing process, resulting in NRC staff supplying affidavits or testimony to the Licensing Board proceedings. Litigated issues concerned: water diversions (Limerick); control of bio-fouling by Asiatic clams (River Bend); health effects of coal particulates released from fuel-cycle facilities, thermal plume analysis, and chlorine effluent effects (Harris); cooling tower drift contamination of ground water (Hope Creek); electric field effects (Braidwood); and cooling tower drift impact on crops (Palo Verde).

An invasion of coelenterates in the late summer of 1984 clogged the cooling water intake screens at the St. Lucie plant at Ft. Pierce, Fla. Among the corrective actions taken, "skimmer booms" (shown here) were used to remove the jellyfish.

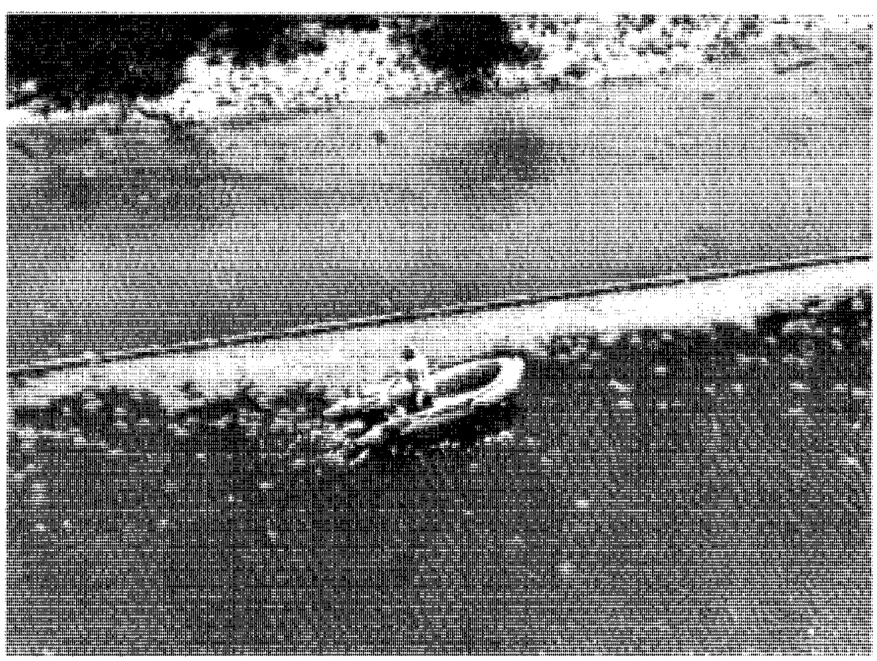


Table 7. Draft and Final Environmental Impact Statements—FY 1984

Site	*Document*	*NUREG No.*	*Publication Date*
Shearon Harris-1&2	FES	0972	10/83
WNP-3	DES	1033	12/83
Limerick-1&2	FES	0974	4/84
Hope Creek-1	DES	1074	6/84
Braidwood-1&2	DES	1026	12/83
Braidwood-1&2	FES	1026	7/84
Millstone-3	DES	1064	7/84
River Bend-1&2	DES	1073	7/84
Nine Mile Point-2	DES	1085	7/84

Marine Borer Damage at Oyster Creek

Oyster Creek Nuclear Generating Station (N.J.) has been operating since 1969. In 1974, the NRC prepared a Final Environmental Statement (FES) related to issuance of a full-term license. The FES determined that station operation contributed to an enhanced population of marine wood-boring organism (principally shipworms) and to borer-related damage to submarine wooden structures (pilings, bulkheads, marinas) in Oyster Creek. The FES recommended the removal of all untreated and infested wood from the area influenced by plant operation and called for studies to quantify the contribution of the nuclear plant to the spread of borer activity to other areas of Barnegat Bay.

The licensee initiated extensive studies of marine wood-borers in June 1975. These studies are continuing. In addition, NRC sponsored an independent study by the Academy of Natural Sciences of Philadelphia, from September 1976 through December 1982. A final report was published in October 1983 (NUREG/CR-3446), entitled "Ecological Studies of Wood-Boring Bivalves and Fouling Organisms in the Vicinity of the Oyster Creek Nuclear Generating Station." The general conclusions of these studies are summarized below.

- Four species of molluscan shipworms were found in the area: two native species, *Bankia gouldi* and *Teredo navalis,* and two introduced tropical-subtropical species, *Teredo bartschi* and *Teredo furcifera*.
- Plant operation created a suitable environment in terms of temperature, salinity, etc., for the survival of the non-native species in the station area, once they were introduced. Their introduction, however, was unrelated to station operation.
- The non-native species differed enough from the native species for all four species to coexist, so the survival of the additional species increased the total amount of wood-boring damage.
- The area affected by thermal effluents included Oyster Creek, Forked River from its mouth upstream through the South Fork (via recirculated effluent), and the shoreline of Barnegat Bay from Forked River to Waretown Creek. A nursery effect was created in which organisms bred in Oyster Creek were broadcast into this area. The non-native species never became established outside the range of the thermal plume.
- During the period 1971-1976, shipworm outbreaks were worse, due to higher station effluent temperature, than in subsequent years. The NRC imposed mitigative measures of reduced effluent temperature (through dilution pumping) and removal of most untreated and infested wood in Oyster Creek. These measures reduced the degree of shipworm damage, especially that due to the native species, but did not eliminate it (or the shipworms) from Oyster Creek or Forked River.
- Breeding populations of the non-native *Teredo furcifera* disappeared completely from the sampling areas after 1978. The winter-spring 1982-83 station outage apparently also eliminated the second non-native species, *Teredo bartschi,* from the sampling area.

- The native shipworm species are present throughout Barnegat Bay. Operation of the Oyster Creek plant influences these species in the thermal effluent area, but apparently not in areas of the bay unaffected directly by station effluents.

The FES projections on the introduction of shipworms into Forked River were verified. Shipworms, especially the non-native species, also have spread to Barnegat Bay, but only to those areas of the western bay under the immediate influence of the plant's thermal effluents. An apparent elimination of non-native species after a prolonged station outage, and the absence of any effect from the plant on shipworms outside of the thermal plume area, suggest that wide-spread shipworm impact has not in fact occurred. Reasons for this appear to be mitigation in the form of reduced effluent temperature and wood removal (during 1976), enhanced by a winter station outage that killed the less tolerant non-native shipworm species. While measures taken by the licensee have decreased the suitability of the area as shipworm habitat, the area still is habitable. The NRC-sponsored study found that as long as there is any unprotected wood in the area influenced by station discharges, a breeding population of borers will be maintained under the plant's present operating conditions. The study concluded that the best course of action is for the licensee to continue to assist local affected property owners in replacing wooden structures with properly treated wood. Such an effort will not only aid affected people, but also will serve to reduce the inhabitable substrata for borers and, therefore, the potential for future problems.

Health Effects of Coal Particulates

NRC environmental reviews for power reactor applications include consideration of the potential impacts from the supporting uranium fuel cycle. The staff's assessment of the fuel cycle impacts are based primarily on the natural resource use and effluent release values given in Table S-3 of 10 CFR Part 51. During the year, the adequacy of the staff's assessment for health effects of coal particulates released at the Table S-3 level of 1154 metric tons per year per 1,000 MWe of nuclear generating capacity was raised and admitted as a hearing contention in the Shearon Harris (N.C.) power reactor licensing proceeding. In admitting the intervenor's contention, the Licensing Board ruled that the staff had not demonstrated in its Environmental Impact Statement (EIS) and subsequent filing for summary disposition that the health effects issue had been generically resolved by the background information developed in the uranium fuel cycle rulemaking proceeding.

The staff supplemented its EIS treatment of the issue with written and oral testimony before the board in the Harris Operating License proceeding. Topics under consideration were the ambient air quality changes attributable to coal particulates released in the vicinity of coal-fired power plants supporting the U.S. Department of Energy uranium enrichment facilities, the populations that would be exposed to increased coal particulate concentrations, and the human health effects from the exposures on local populations at risk. The staff's supplemental analysis concluded that the projected health effects are small and, moreover, the health effects impacts are much smaller than the uncertainties associated with the available health risk assessment models and the predictions derived from the models. Conservatively high estimates indicate that the expected annual increases of acute respiratory disease incidents are less than 0.005 per cent increase over the baseline for the local populations and that annual mortality increases are in the range of zero to 0.09, or less than 0.03 percent increase over the baseline. The Licensing Board has yet to rule on the evidence presented on this issue.

Nuclear power facilities are frequently visited by NRC Commissioners in person. Greeted on his arrival at the Braidwood plant in Illinois is Commissioner James K. Asselstine. NRC environmental reviews addressed, among other concerns, threats to water life (mussels) at the plant site.

In responding to this issue, the staff has identified a need to update Table S-3 non-radiological effluent values for the fossil-fuel requirements of the uranium fuel cycle. The updated values will reflect current control technologies for effluent releases and restrictions on effluents established by the Clean Air Act and Clean Water Act.

Cooling Tower Salt Drift

The Atomic Safety and Licensing Board has re-opened the environmental hearings concerning the granting of an operating license for the Palo Verde Nuclear Generating Station (Ariz.). A group of ranchers contend that cooling tower salt drift from the three circular mechanical draft towers will detrimentally effect agricultural crop production in the vicinity of the plant. The crops under consideration are cotton, alfalfa, barley and wheat. The applicant (Arizona Public Service Company, *et al*) has funded a research effort with the University of Arizona to determine potential salt drift impacts from the plant. NRR staff will be evaluating the results of the research effort and preparing testimony for the hearings currently scheduled for January 1985. The staff will address the potential impacts of salt drift at varying distances and directions from the facility.

Meteorological data and drift deposition models will be used to estimate annual deposition rates in the vicinity of Palo Verde. Biological effects on vegetation from salt drift will be addressed based on drift deposition estimates. The analysis is precedent-setting in that nuclear plant operational data are largely from power plants located in coastal areas with high rainfall and having brackish water as the source of cooling water.

Socioeconomic Impacts of Nuclear Power Plants

NRC environmental impact statements include evaluation of local and regional social and economic impacts of nuclear power plant and associated transmission lines. In the course of the licensing reviews, several historic and archeological sites were identified as potentially eligible for inclusion in the National Register of Historic Places. In fulfillment of its responsibilities as a Federal licensing agency, the NRC submitted for each site a request for determination of eligibility to the keeper of the National Register of Historic Places. Several of these sites have been determined eligible and appropriate protection plans have been developed by the concerned utility. During the year, a reanalysis and updating of population estimates in the vicinity of operating plants was nearly completed.

Antitrust Activities

As required by law since December 1970, the NRC has conducted prelicensing 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 that transfer an ownership interest in a nuclear facility to one or more additional applicants are subject to antitrust review.

An application for an operating license is not subject to formal antitrust review unless the NRC first determines that "significant changes" in the applicant's activities have occurred since the review of the application for a construction permit (see 47 *FR*9983 for procedures used). During fiscal year 1984, five analyses were completed for determination of significant changes. In each instance, the finding was that the changes that had occurred were not significant in an antitrust context.

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 fiscal year 1984, the NRC closed out enforcement actions pertaining to a civil penalty requested against the Cleveland Electric Illuminating Co. in the Davis Besse (Ohio) and Perry (Ohio) nuclear plant licenses. Requests for enforcement of antitrust conditions for the Diablo Canyon (Cal.) and Farley (Ala.) nuclear plants were under consideration as of September 30, 1984.

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, each application for a construction permit or an operating license for power reactors, 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, all 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 chemistry and chemical engineering, electrical engineering, mechanical engineering, structural engineering, reactor operations, reactor physics, and environmental health.

During fiscal year 1984, the Committee completed its annual report to Congress on the NRC Safety Research Program for fiscal year 1985, and its annual report to the Commission on the Safety Research Program and Budget for fiscal year 1987.

Members appeared and presented testimony to the Subcommittee on Energy and the Environment of the House Committee on Interior and Insular Affairs in an oversight hearing on the NRC's budget request for fiscal years 1984 and 1985. Testimony was also given to Subcommittees on Energy Development and Applications, and Energy Research and Production of the House Committee on Science and Technology on the conversion of domestic research and test reactors to low enriched uranium fuel.

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

- The need for PORVs in certain nuclear steam supply systems designed by the Combustion Engineering Company.
- of the "tau effect" in the Diablo Canyon seismic analysis.
- A response to the GAO concerning NRC's PRA-related research programs.
- The BWR piping reinspection program and proposed repair of cracked piping.
- NRC policies on decentralization of licensing activities.
- DOE guidelines on waste repository sites.
- Quantification of seismic design margins.
- *De minimis* values for radiation exposure.
- Plant quality and quality assurance.
- NRC enforcement policy.
- Containment performance guidelines.
- Application of probabilistic risk assessment.
- TMI-2 clean-up activities.
- Loss of coolant accident evaluation model for plants designed by the General Electric Company.
- The Office for Analysis and Evaluation of Operational Data (AEOD) trends and patterns analysis program.
- The use of low-enrichment fuel in research and test reactors.
- Severe accident policy.
- NRC's policy concerning the need for on-shift engineering expertise.

The Committee's activities during the report period reflected the continuing license activity within the Commission and included four reports on requests for nuclear power plant operating licenses, one review of an operating plant evaluated as part of the Systematic Evaluation Program, one review each of a request for a full-term license conversion and for a license renewal and power-level increase. In addition, the Committee provided four separate reports on various aspects of the Diablo Canyon review, including the seismic design bases, the peer review group report, and proposed license conditions.

In addition to its reports on licensed reactors and operating license applications, the Committee provided advice to NRC on 12 proposed rules, criteria, or regulatory guides, including:

- Codes and standards for nuclear power plants.
- Instrument setpoints for safety-related systems.
- Standard for protection against radiation.
- The addition of a human factors criterion to the general design criteria.
- Environmental qualification of electrical equipment.
- Bioassay for tritium.
- Test and calibration of radiation protection instrumentation.
- Residual radioactive contamination limit for decommissioning.
- Inservice inspection of prestressed concrete containment.
- Containment leak test requirements.
- Disposal of high-level waste in geologic repositories.

The Committee also provided advice on proposed resolutions for seven generic or unresolved safety issues, including:

- Steam generator tube integrity.
- LOCA-related hydrodynamic loads in Mark III containments.

- Seismic qualification of equipment in operating plants.
- Shutdown decay heat removal requirements.
- Reactor coolant pump seal leakage.

The Committee commented as well on the NRC Staff's proposed priority ranking for newly identified generic issues and on a categorization of generic and licensing issues proposed by the Electric Power Research Institute (EPRI) for application of unresolved generic items to future standardized plant designs.

The Committee provided a report to the NRC on the training and qualification of personnel in nuclear power plants consistent with the provision of Public Law 97-425, "Nuclear Waste Policy Act of 1982."

In accordance with the procedures embodied in NRC Manual Chapter 4125, the Committee reviewed Differing Professional Opinions concerning radiation monitoring capability in NRC's Region V office and the Generic Westinghouse Safety Parameter Display System.

In performing the reviews and preparing the reports cited above, the ACRS held 12 full Committee meetings and 87 subcommittee and working group meetings. Members of the Committee also participated in several conferences and visits to exchange safety-related information with foreign regulatory and developmental bodies.

On July 2-3, 1984, the ACRS held a meeting with the Reaktor-Sicherheitskommission (RSK) (Reactor Safety Committee of the Federal Republic of Germany) to discuss safety-related issues of mutual interest. During the meeting, held near Munich, Federal Republic of Germany, specific items included design of safety systems for light water reactors, pipe crack experience, severe accidents, operation of nuclear power plants, technical support for nuclear power plant operations, and the use of on-line computers.

Cleanup at Three Mile Island Unit 2

CHAPTER 3

Fiscal year 1984 was marked by significant progress in the cleanup of the accident damaged Unit 2 reactor at the Three Mile Island Nuclear Power Station (TMI) near Harrisburg, Pa. Numerous technical accomplishments were highlighted by the successful removal and storage of the reactor vessel head in July 1984. Prospective funding for future recovery activities was enhanced through additional commitments. Through increased efforts, General Public Utilities Nuclear Corporation (GPU) was able to complete activities previously delayed by funding limitations and allegations regarding polar crane safety. As a result, GPU is currently projecting the initiation of fuel removal activities in July 1985 with completion of the cleanup scheduled for mid-1988.

During fiscal year 1984, the reactor building polar crane was load tested and later used for the removal of the reactor vessel head, the placement of the head on its storage stand, and the placement of the internals indexing fixture (IIF) on the reactor vessel. Sonar and video inspection data were collected to assess core conditions in preparation for future plenum assembly removal and defueling of the reactor. The processing and shipment of radioactive wastes continued in support of cleanup activities as did dose reduction efforts aimed at keeping worker radiation exposures as low as reasonably achievable.

Cleanup Funding

Progress was also made in securing additional funding for future cleanup activities. The Edison Electric Institute, representing the utility industry, pledged funds totalling $25 million per year for six years, beginning in January 1985. A group of Japanese utility companies pledged a contribution of $18 million ($3 million for six years) to the cleanup, making the total level of funding for cleanup activities during 1984 approximately $95 million. The additional commitment of funds will help to eliminate some of the funding constraints to an expeditious cleanup of TMI Unit 2, which is one of the NRC's highest safety priorities. The TMI Program Office continues to monitor cleanup activities from the site and will continue to provide the oversight necessary to ensure the prompt decontamination of the facility and safe removal of radioactive materials from the site.

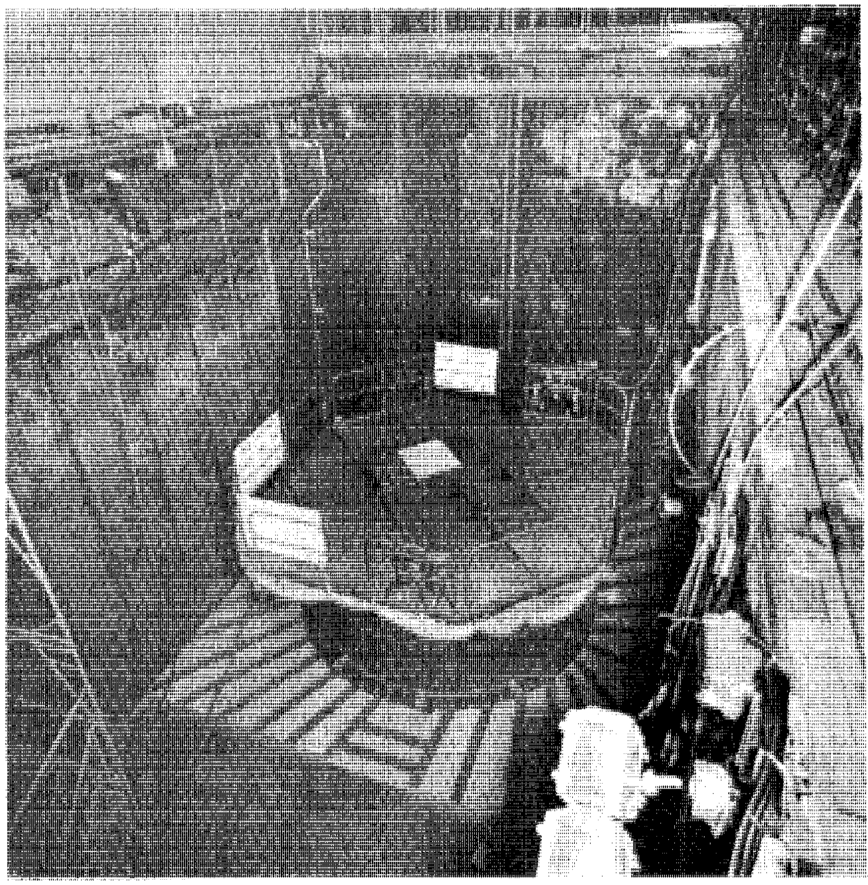
Reactor Building Activities

Workers entered the TMI-2 reactor building 167 times during fiscal year 1984 in the performance of numerous cleanup activities. Entries during the first quarter of the fiscal year were limited to one per week due to funding constraints and focused primarily on collection of reactor coolant samples. In early 1984, more frequent entries were made to prepare for and conduct the load test of the polar crane, to take additional core debris samples, to partially detension the reactor vessel head studs, and to perform video mapping of the reactor vessel internals. Reactor building entries during the third quarter of fiscal year 1984 were conducted to perform activities in preparation for reactor vessel head lift. These activities included depressurization and draindown of the reactor coolant system, refueling canal seal plate installation, control rod drive lead screw parking, auxiliary fuel handling bridge modifications and modification of the IIF. Shielding, radiation monitors and television cameras were installed to support head lift. During the last quarter of the fiscal year, reactor building activities included scabbling of the floor at the 347-foot elevation to reduce dose rates, the removal and storage of the reactor vessel head and the operation of the IIF water processing system to reduce radionuclide concentrations in the reactor coolant. Dose reduction efforts continued in preparation for plenum assembly inspection and pre-removal activities.

Reactor Building Polar Crane

At the end of fiscal year 1983, the Office of Investigations (OI) issued its report on the allegations regarding the safety of the polar crane and other cleanup-related issues. The staff reviewed the OI findings and concluded that the specific deficiencies cited did not result in a significant increase in risk to the public health and safety. The staff also recommended the implementation of a detailed action plan to correct the identified administrative and procedural deficiencies. An Enforcement Action resulting from polar crane refurbishment activities was issued on February 3, 1984.

On November 18, 1983, the staff approved the licensee's safety evaluation for the refurbishment and use of the Reactor Building Polar Crane. The crane was successfully load tested on February 29, 1984, when a test assembly weighing 214 tons was lifted and moved along predetermined test paths.



A major milestone in the cleanup of the Three Mile Island Unit 2 accident was the removal of the reactor pressure vessel head in July 1984. The photo shows a post-removal overhead view of the water-filled Internal Indexing Fixture (IIF), which is covering the open reactor vessel, and a shielded work platform lowered onto the IIF. The IIF was filled with five feet of water for radiation shielding.

Reactor Vessel Head Lift

A major cleanup milestone was achieved in late July 1984 when the reactor pressure vessel head was removed and placed in shielded storage. The polar crane was used to lift the head, place the head on the storage stand, install the cylindrical IIF over the open reactor vessel and lower the shielded work platform onto the IIF. Prior to work platform installation, the IIF was filled with five feet of water to provide radiation shielding over the exposed plenum.

Inspection of Reactor Core

A scale model of the damaged core was constructed in late 1983 based on sonic measurements obtained from inside the reactor vessel. This topographic model provided the most accurate indication of the extent of core damage to date. The volume of the cavity in the damaged area of the core was measured at 330 cubic feet or 26 percent of the original core volume. The bottom of the cavity ranges from 5-to-6 feet below the top of the core and the cavity extends to the core, forming wall in several areas. Forty-two of the original 177 fuel assemblies appear to contain some full-length fuel rods, but 23 of those 42 have less than 50 percent of the rods intact. The sonic mapping also revealed several partial fuel assemblies hanging from the underside of the plenum and indicated some distortion of the core forming wall. In early 1984, a comprehensive video mapping of the core was made and additional core debris samples were taken. The accurate characterization of core conditions provided by these activities has facilitated the planning of subsequent cleanup operations such as plenum removal preparatory activities, including the separation of unsupported partial fuel assemblies, which are scheduled to begin in October of 1984.

Waste Management

During fiscal year 1984, the Submerged Demineralizer System (SDS) and the EPICOR-II system continued to be used to process radioactive water in support of cleanup activities. The SDS was used primarily to process reactor coolant, reactor building sump water, and water generated during the decontamination of the "A" spent fuel pool. The EPICOR-II system typically was used to polish the effluent from the SDS. The SDS and EPICOR-II system processed approximately 532,000 and 272,000 gallons of water, respectively, during the year. Regarding the disposition of solid radioactive wastes generated by SDS and EPICOR-II operations, three SDS liners and 32 EPICOR-II liners were shipped to Hanford, Wash., during the year.

Occupational Exposure

GPU continued efforts to keep worker exposures as low as reasonably achievable during fiscal year 1984. These efforts consisted of extensive pretask planning and mock up training for each task, the use of radiation shielding, and the application of decontamination and dose reduction techniques. The effectiveness of the decontamination and dose reduction methods was demonstrated during the last quarter of the fiscal year. In July, workers entered the reactor building without respiratory protection for the first time since the accident, and subsequent entries were made without respirators, in accordance with ALARA principles. The head lift operation resulted in a cumulative worker exposure of 15 person-rem, compared to the staff's prediction of between 60 and 220 person-rem. Dose rates in the reactor building were restored to pre-lift levels following head lift and subsequent IIF installation and waterfilling. Since the completion of head lift, scabbling (the mechanical removal of a thin layer) of the concrete reactor building floor has resulted in a measured 50 percent reduction in local dose rates.

In January 1984, the TMIPO issued a draft supplement to the Final Programmatic Environmental Impact Statement (NUREG-0683), which revised the staff's earlier estimates of occupational radiation exposure resulting from the cleanup. The total radiation dose to cleanup workers is currently estimated to range between 13,000 and 46,000 person-rem as opposed to earlier estimates of 2,000 to 8,000 person-rem. The higher estimates resulted from a more accurate characterization of radiation fields in the reactor building based on numerous worker entries. Delays in the cleanup complicated decontamination efforts because radiation sources became more deeply entrained in building surfaces; as a result, early dose reduction efforts were less successful than anticipated. Although the staff's revised dose estimates are significantly higher than the previous estimates, the staff still concludes that the environmental impact is insignificant and the cleanup should proceed as expeditiously as possible, to reduce the potential for radiation release to the environment and to assure that TMI-2 does not become a long term waste disposal site.

Advisory Panel on TMI Cleanup

The Advisory Panel for the Decontamination of Three Mile Island Unit 2, comprised of citizens, scientists and local and state government officials, was formed by the NRC in 1980 in order to gain input from area residents regarding major TMI cleanup activities. (See Appendix 2 for a list of members). On November 29, 1983, NRC Chairman Nunzio J. Palladino appointed Arthur E. Morris, Mayor of Lancaster, Pa., as chairman of the Advisory Panel, upon the resignation of the previous chairman, John Minnich. During fiscal year 1984, the panel held eight public meetings in Harrisburg, Pa., and met twice with the NRC Commissioners in Washington, D.C. The principal topics addressed by the panel during the year included cleanup funding, occupational radiation exposure, polar crane repairs and reactor vessel head lift.

Operational Experience

CHAPTER 4

ANALYSIS AND EVALUATION OF OPERATIONAL DATA

NRC's Office for Analysis and Evaluation of Operational Data (AEOD) was established several months after the accident at TMI-2. The office reports directly to the Executive Director for Operations and is dedicated to the collection, assessment, and feedback of operational data to the NRC and industry.

The mission of the office is to analyze and evaluate operational safety data associated with all NRC-licensed activities. These include commercial power reactors, and radioactive material and fuel cycle licensees. The office also coordinates the overall NRC operational data program and serves as the focal point for interaction with outside and foreign organizations performing similar work. Among the AEOD's specific activities are the following:

- Screen U.S. and foreign operational events for significance; systematically and independently analyze these events; seek trends and patterns that indicate potential safety problems; and develop and track AEOD recommendations for action by other NRC offices.
- Develop and coordinate agency guidance on Licensee Event Report requirements and monitor the effectiveness of the Nuclear Plant Reliability Data System.
- Develop and maintain computerized storage and retrieval systems for reactor and non-reactor operational data, including foreign data.
- Prepare and coordinate the quarterly *Report to Congress on Abnormal Occurrences* (NUREG-0090), as required by Section 208 of the Energy Reorganization Act of 1974; the bi-monthly *Power Reactor Events* reports; the monthly *Licensee Event Report Compilation;* and other feedback documents.
- Prepare reports of U.S. events for transmittal to the Nuclear Energy Agency's Incident Reporting System.
- Serve as principal point of contact with the Advisory Committee on Reactor Safeguards (ACRS), the Institute of Nuclear Power Operations (INPO), and the Nuclear Safety Analysis Center (NSAC) on matters involving the collection and evaluation of operational data.

AEOD is part of an integrated NRC program to review operating experience to identify specific events and generic situations where the margin of safety established by design through the licensing process has been degraded, and to identify and implement corrective actions that will restore the original margin of safety. AEOD's focus and involvement in the program are to provide a strong in-house technical capability in analysis of operating experience, independent of regulatory activities associated with licensing, inspection, or enforcement.

NRC Handling of Operational Data

Domestic. On January 1, 1984, 10 CFR 50.73 became effective. This rule modified and codified the Licensee Event Report (LER) system, which previously had been defined in technical specification requirements. AEOD, in addition to developing and coordinating 50.73, prepared detailed guidance documents (NUREG-1022 and NUREG-1022, Supplement 1), and resolved numerous questions regarding the intent and interpretation of the new rule. An analysis of LERs submitted in January 1984 in response to 50.73 concluded that: 50.73 was being implemented correctly; the number of LERS would be reduced by about 50%; the LER data for the first time would permit a more systematic analysis of operational events; and additional analysis was warranted on such issues as reactor scrams, emergency safety features actuations, and total system failures.

AEOD, under contract with the Nuclear Operations Analysis Center (NOAC) at Oak Ridge, Tenn., operates and maintains the Sequence Coding and Search System (SCSS), a computerized storage and retrieval system for LER data. Its objective is to encode all of the relevant technical information provided by the licensee in the LER, with sufficient "tags" so that the individual pieces can be retrieved. The NOAC has completed the basic development of the SCSS. A User's Guide to the system was distributed in September 1984, allowing access by all NRC offices. During the year, about 2800 LERs were added to the SCSS data base, which now contains over 13,000 LERs received since 1981. Efforts initiated during the year to backfit the data base to include LER data from 1980 will be completed during fiscal year 1985.

A Trends and Patterns Program Plan for analyzing LER data was developed and published in March 1984. The program uses statistical techniques to detect trends or patterns from incidents of low individual significance that may signify an unrecognized safety concern. The program encompasses the SCSS, and in the future will include the Nuclear Plant Reliability Data System (NPRDS), a voluntary, industry run system for reporting failures of safety components, which INPO is implementing and the NRC is monitoring.

The program has produced a report on software used to analyze the SCSS data (NUREG/CR-3824), and an exploratory analysis of 1981 SCSS data. In progress are a pilot analysis of reactor trips, a categorical analysis of 1981-1983 SCSS data, and an analysis aimed at taking advantage of the reporting improvements brought about by the inauguration of 50.73. An approach tailored to NPRDS data is also being developed for application early in 1985.

As part of the NPRDS evaluation program, semiannual evaluation reports were forwarded to the Commission in January and July 1984 (SECY-84-44 and SECY-84-44A). These reports noted that, while utility participation in NPRDS continues to improve in terms of the number of participants and the timeliness of reporting, the rate of improvement has slowed and, in a few areas, little or no additional improvement was noted. To achieve a fully operational status, NPRDS needs continued INPO management attention and action.

Foreign. In fiscal year 1984, the NRC continued efforts to increase the number and usefulness of foreign experience reports that are received. The agency also participated in the exchange of operational event information with other countries through the Nuclear Energy Agency and through bilateral agreements. An NRC program at the NOAC systematically screens and assesses selected foreign information for its applicability to the U.S. program, and to abstract it for computerized data filing.

Analysis of Non-Reactor Operational Experience

In addition to the screening and analysis of reactor operational experience, AEOD reviews the non-reactor operational experience associated with the activities and facilities licensed by the Office of Nuclear Material Safety and Safeguards and by Agreement States (see Chapter 9). In addition, AEOD conducts studies from a human factors perspective on both reactor and non-reactor operational events.

Semiannual Report to the Commission

At the close of fiscal year 1984, AEOD submitted its first semiannual report to the Commission (AEOD/S405). The Commission was subsequently briefed on the substance of the report. Based on extensive screening, analysis and feedback of operational experience, AEOD registered the following comments and observations in its report:

General Observations:

(1) Operational data analysis and feedback is vital, yet the lessons are sometimes lost over time.

(2) The quality of operational data reports is improving.

(3) While trends and patterns analysis is more difficult than expected, it warrants continued emphasis.

(4) In addition to the NRC and industry activities, licensee personnel having responsibility for plant operations also play a large role in assuring the effectiveness of operational data collection, assessment and feedback programs.

(5) There is heightened sensitivity to operational events and the lessons of experience today, compared with the period prior to the TMI-2 accident.

Reactor-related Observations:

(1) Operating experience continues to identify a wide range of system interactions.

(2) Operating experience indicates that a total loss of a safety system is not a rare event.

(3) Operating experience shows that operating practices, particularly those associated with maintenance and surveillance, are frequently deficient.

(4) Operating experience continues to indicate that multiple independent failures do occur.

(5) Operating experience has indicated component performance and reliability problems.

(6) Operating experience indicates that Emergency Safeguard Systems (ESF) are frequently challenged.

Non-reactor-related Observations:

(1) Medical misadministrations continue to occur.

(2) Personnel over-exposures continue to occur.

The AEOD semiannual report presents and discusses examples to support the comments and observations set forth above.

ANALYSIS OF REACTOR OPERATIONAL EXPERIENCE

AEOD is responsible for screening LERs and other pertinent event documentation; identifying events of par-

Table 1. AEOD Reports Issued During FY 1984

CASE AND SPECIAL STUDIES

Designation	*Subject*	*Issued*
C401	Low Temperature Overpressure Events at Turkey Point Unit 4	3/84
C402	Operating Experience Related to Moisture Intrusion in Electrical Equipment at Commercial Power Reactors	5/84
C403	Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982	5/84
C404	Steam Binding of Auxiliary Feedwater Pumps	7/84
C405	Breaching of the Encapsulation of Sealed Well Logging Sources	9/84
P401	Operating History Overview for Diesel Generators in Nuclear Service	2/84
P402	AEOD Trends and Patterns Program Plan	3/84
P403	AEOD Trends and Patterns Evaluation Report on Preliminary Assessment of LER Reporting Under 10 CFR 50.73	5/84
S401	Human Error in Events Involving Wrong Unit or Wrong Train	1/84
S402	Pressure Locking of Flexible-Disk Wedge-Type Gate Valves	7/84
S403	Annual Report of USNRC Participation in the Nuclear Energy Agency Incident Reporting System During 1983	6/84

ticular significance; conducting appropriate engineering evaluations and case studies for significant events; and formulating appropriate recommendations for action by other NRC offices. AEOD provides a in-house engineering capability for examining operational events at U.S. and foreign light-water commercial reactors. AEOD technical studies and evaluations are conducted based upon the screening of U.S. and foreign event reports, and on the knowledge and experience of the technical staff. Review efforts are normally initiated after a licensee report is available; thus, AEOD activities are normally independent of and occur later than, the prompt action that may be initiated by the Region or Program Offices to investigate an operating event and determine the need for immediate licensee response or generic action.

During the 1984 period, 11 special studies and case studies (see Table 1), and more than 60 engineering evaluations and technical reviews (see Tables 2 and 3), were completed by AEOD. Subjects examined in the evaluations and reviews included failure of split-wedge and double-disk gate valves, instrument sensing inoperabilities caused by reversed differential pressure instrument sensing lines, mechanical pipe snubber failures, and potential common mode failure or degradation in redundant trains of standby gas treatment systems. AEOD also reviews reactor events from a human factors perspective. Selected case studies performed by AEOD on reactor operational experiences are summarized below.

Human Errors

In January 1984, AEOD published a major study of "Human Error in Events Involving Wrong Unit or Wrong Train" (AEOD/S401), identifying 27 events between 1981 and August 1983 where a safety system was lost because action was taken on the wrong unit or train. Although most events had limited actual safety significance, these events could have been significant under different circumstances. Nineteen events resulted during maintenance or surveillance testing and 16 of these occurred near full power. As a result of this study, licensees were provided direct guidance and information in NRC Information Notices 84-51 and 84-58. AEOD also issued an engineering evaluation involving undetected unavailability of the turbine-driven AFW train, focusing on human factors considerations leading to the failure of the turbine-driven pump. As a result of this report, NRR defined both short and long-term actions to correct the problems identified. Subsequently, in the second half of 1984, NRC Information Notice 84-66 was issued on this subject.

Table 2. Reactor Engineering Evaluations and Technical Reviews

Engineering Evaluations	*Subject*	*Issued*
E325	Vapor Binding of Auxiliary Feedwater Pumps at Robinson 2	11/21/83
E326	Steam Voiding in Oconee 3 on June 13, 1975: A Precursor Event to the TM12 Accident	11/28/83
E327	Gaseous Releases from Waste Gas Disposal System	11/28/83
E401	Temporary Loss of All AC Power Due to Relay Failures in Diesel Generator Load Shedding Circuitry at Fort St. Vrain	1/04/84
E402	Water Hammer in Boiling Water Reactor High Pressure Coolant Injection Systems	1/10/84
E403	Deficiency in Automatic Switch Company (ASCO) Spare Parts Kits for Scram Pilot Solenoid Valves	1/17/84
E404	Failures in the Upper Head Injection System	2/28/84
E405	Common Mode Failure of HPCI Steam Flow Isolation Capability at Browns Ferry	3/22/84
E406	Mechanical Snubber Failure	3/22/84
E407	Initiation and Indication Circuitry for High Pressure Coolant Injection (HPCI) Systems	3/26/84
E323 Revision 1	Load Reduction Transient at the Salem Nuclear Power Plant, Unit 2 on January 14, 1982	3/27/84
E408	Reversed Differential Pressure Instrument Sensing Lines	4/13/84
E409	Operating Experience Involving Air in Instrument Sensing Lines	5/16/84
E410	Operational Experiences Involving Standby Gas Treatment Systems Which Illustrate Potential Common Cause Failure or Degradation Mechanisms	5/21/84
E411	Failure of Anti-Cavitation Device in Residual Heat Removal Service Water Heat Exchanger Outlet Valve	5/22/84
E412	Adverse System Interation with Domestic Water Systems	5/25/84
E413	Natural Circulation in Pressurized Water Reactors	5/25/84
E414	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	5/31/84
E415	Overcooling Transient	6/6/84
E416	Erosion in Nuclear Power Plants	6/11/84
E417	Loosening of Flange Bolts on RHR Heat Exchanger Leading to Primary to Secondary Side Leakage	7/02/84
E418	Feedwater Transients During Startup at Westinghouse Plants	7/24/84
E419	Failures of Fischer-Porter Transmitters Used in Safety-Related Systems	8/16/84
E420	Operational Experiences Involving Shorted Lamp Sockets of Indication Lights	8/23/84
E421	Loss of Pressurizer Heaters During Precore Hot Functional Testing	8/27/84
E422	High Pressure Coolant Injection System Performance at E. I. Hatch Units 1 and 2	8/27/84
E423	Failure of Large Hydraulic Snubbers to Lock up	9/20/84
E424	Failure of Anchor Bolt on Diesel Generator Day Tank at Davis Besse 1	10/1/84
T332	Reactor Vessel Drainage at Grand Gulf Unit 1	10/7/83
T333	Degradation of Saltwater Cooling System Caused by a Loss of Instrument Air at San Onofre 1	10/31/83
T334	Reactor Vessel Drainage at Grand Gulf Unit 1	11/15/83

Engineering Evaluations	*Subject*	*Issued*
T335	Simultaneous Safety Injection Actuation Signal and Recirculation Actuation Signal at San Onofre 3	11/15/83
T336	Design Deficiency Resulting in Isolation of Both Loops of the Emergency Condenser System at Nine Mile Point 1	11/17/83
T337	Water Hammer in the Main Feedwater System Resulting in a Feedwater Line Crack at Maine Yankee	11/21/83
T338	Water Leak Through Containment Spray Block Valves at San Onofre 1	11/28/83
T339	Redundant Emergency Core Cooling System (ECCS) Pump Room Air Coolers Out of Service for 22 Hours at Calvert Cliffs Unit 1	11/29/83
T340	Evaluation of a Control Rod Mismanipulation Event at Hatch Unit 2	12/2/83
T341	Corrosion of Carbon Steel Pipe in Service Water Headers	12/19/83
T401	Failures of Containment Air Monitors at Farley 1 and 2	3/21/84
T402	Chemical Contamination of Primary and Secondary Systems in Light Water Reactors	3/22/84
T403	Set Point Drift of Barton Model 288 Switches	3/23/84
T404	Cable Fire and Loss of Control Power to Engineered Safeguards (ES) Valves	4/13/84
T405	Cold Weather Events 1983–1984	4/25/84
T406	Improper Spare Parts Procurement Event at Grand Gulf Unit 1	4/25/84
T407	Failure of 4 kV Circuit Breaker to Trip	4/30/84
T408	Diesel Generator Inoperability Due to Overheating of Ventilation Cowling	5/7/84
T409	Multiple Failures of Bell and Howell Dual Potentiometer Modules Which Occurred at the Fort Calhoun Nuclear Station	5/10/84
T410	Injection Valve for the High Pressure Coolant Injection System Failure to Open during a Surveillance Test	5/11/84
T411	Contamination of the Nitrogen System at Sacramento Municipal Utility District (SMUD)	6/18/84
T412	Failure of an Access Door Between the Drywell and the Wetwell	6/18/84
T413	Failure of Fire Damper in Safeguards Ventilation System	6/28/84
T414	Station Operating Restrictions for Lost or Out of Service Power Transformers Through Which electrical Power is Supplied to the Emergency Buses	7/12/84
T415	Destruction of Charging Pump	7/17/84
T416	Loss of ESF Auxiliary Feedwater Pump Capability at Trojan on January 22, 1983	8/1/84
T417	Excessive Cooldown Rate Event at LaSalle 1	8/2/84
T418	Events Involving Fires or Other Related Abnormalities in Motor Control Centers with Aluminum Bus Bars	8/6/84
T419	Contamination of Snubber Bleed Screw and Lockup Poppet Valve	8/20/84
T420	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	8/23/84
T421	Design Deficiency in Standby Gas Treatment System	8/23/84
T422	Inoperability of Safety Injection Pump at Salem 1 on October 17, 1983	8/29/84

Table 3. Non-Reactor Engineering Evaluations

Designation	*Subject*	*Issued*
N306	Potentially Leaking Americium-241 Sources manufactured by Amersham Corporation	12/1/83
N307	Nonreactor Event Report Database for the Period January June 1983	12/28/83
N204D	Report on Medical Misadministrations for January 1983 through June 1983	5/8/84
N401	Nonreactor Event Report Database for the Period July–December 1983	6/11/84
N402	Events Involving Undetected Unavailability of the Turbine-Driven AFW Train	6/15/84
N403	Report on Medical Misadministrations for July 1983 December 1983	7/3/84

Moisture Intrusion in Electrical Equipment at Commercial Reactors.

Intensive NRC staff reviews have covered the importance of safety-related electrical equipment qualification. These reviews have resulted in the issuance of NUREG-0588, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment of Operating Reactors," and NRC Bulletin 79-01B regarding environmental qualifications of safety-related electrical equipment in harsh environments.

Numerous occurrences of safety-related equipment failures resulting from moisture intrusion, mostly in a mild environment, were reported to the NRC after the issuance of NRC Bulletin 79-01B in January 1980. The equipment included, but was not limited to, electrical wiring termination boxes, junction boxes, and pressure switch instrumentation. The majority of the reported failures involved boiling water reactors, and the failed equipment was located outside of primary containment in the reactor building basement. The major causes of these events were (1) loss of the environmental protection boundary, generally as a result of maintenance activities; and (2) inadequate protection from moisture sources. The modes of component failure were shorting (or grounding) and corrosion.

The AEOD analysis of these events provided recommendations which could reduce the frequency of safety-related electrical equipment failures resulting from moisture, emphasizing the importance of a well-planned maintenance/surveillance program to protect the equipment.

Plant Systems Interaction Event At Hatch Unit 2

AEOD studied a plant transient at Hatch Unit 2 on August 25, 1982, in which a series of systems interactions during post-scram recovery operations resulted in sustained and uncontrolled loss of hot pressurized reactor coolant outside primary containment. The study concluded that (1) the Hatch event can be viewed as a precursor for a similar but more limiting postulated accident sequence that has recently been comprehensively reviewed on a generic basis by the NRC staff, and (2) if the staff positions and guidance which resulted from the earlier generic review are implemented on a plant-specific basis, adequate preventive and mitigation measures will have been provided for both the Hatch event and the more limiting postulated accident scenario.

In addition, the underlying causes for a number of the specific equipment failures and problems which occurred during the Hatch event were found to be significant in that they had been addressed in official USNRC correspondence to the Hatch licensee and other boiling water reactor licensees years before this event. This study thus concluded that each of the equipment problems that occurred during the Hatch event could have been prevented, and the significant plant response consequences avoided, had adequate corrective actions been implemented in response to the previous communications. To correct this situation, AEOD suggested followup corrective measures which addressed several of the specific areas that appeared to be in need of attention.

Steam Binding of Auxiliary Feedwater Pumps

This study identified a concern regarding leakage of motor-operated or check valves between the main feedwater system and the auxiliary feedwater (AFW) system at PWRs. These valves can leak, allowing hot feedwater to enter the AFW, flash to steam in the low temperature, low pressure environment and thereby cause possible steam binding of the AFW pumps when they are called upon to start.

There were 22 events involving backleakage from the main feedwater system to the AFW system since 1981 at six operating PWRs in the U.S. and one foreign plant. These events involved the misoperation or failure of about 60 check valves and five motor-operated valves installed to prevent reverse flow. The study identified this issue as a potential common mode failure of all AFW pumps. Steam binding of the AFW pumps is presently an undetectable failure. Pump unavailability due to this failure mode is a significant contributor to risk of core melt in PWRs.

The study recommended that NRC take action to require the regular monitoring of the AFW system to detect leakage and ensure that the AFW line conditions are well below saturation conditions. This information was quickly fed back to all licensees by means of NRC Information Notice 84-06 and has been identified by NRR as a high priority generic issue.

ANALYSIS OF NON-REACTOR OPERATIONAL EXPERIENCE

Two Semiannual Medical Misadministration Reports were issued, covering events from calendar year 1983. These studies continue to indicate that over 98 percent of medical misadministrations involve either the administration of the wrong radiopharmaceutical to a patient or the administration of a radiopharmaceutical to the wrong patient.

In addition, a preliminary case study on Breaching of Encapsulation of Sealed Well Logging Sources was issued in March 1984. This study analyzed five ruptures of sealed well logging sources that occurred between August 1982 and September 1983. This study served as a key input to the NRC interoffice group drafting revisions to 10 CFR Part 39 on well logging.

ABNORMAL OCCURRENCES

AEOD prepares the quarterly *Report to Congress on Abnormal Occurrences* (NUREG-0090), which feeds back significant event information to licensees, government agencies, and the public. These reports are available from the NRC/GPO Sales Programs, Washington, D.C. 20555. (For a description of NRC's requirements under law to report abnormal occurrences, see the *1980 Annual Report*, p. 82). The reports issued during fiscal year 1984 were Vol. 6, No. 2 (April-June 1983); Vol. 6, No. 3 (July-September 1983); Vol. 6, No. 4 (October-December 1983); and Vol. 7, No. 1 (January-March 1984). These reports covered eight occurrences at nuclear power plants, five occurrences at Agreement State licensees, and ten occurrences at other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.). The nuclear power plant occurrences and some of the occurrences at the other NRC licensees and Agreement State licensees are summarized below.

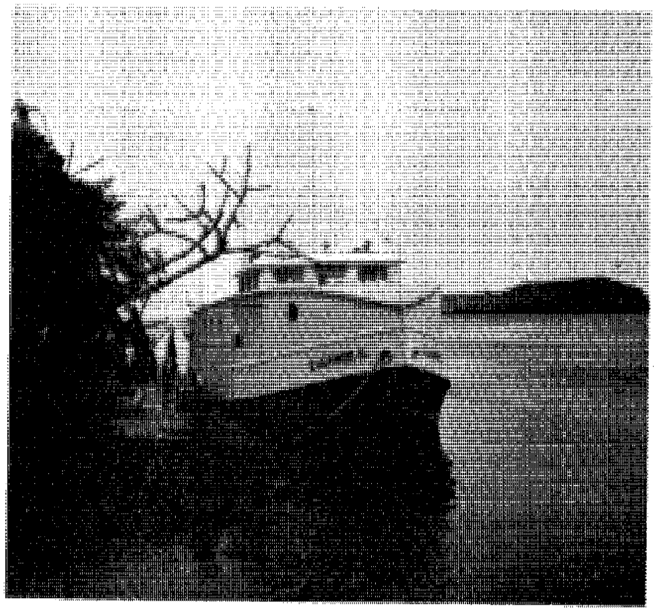
Unavailability of the Auxiliary Feedwater System. On April 19, 1983, the licensee for Turkey Point Units 3 and 4 reported that the auxiliary feedwater (AFW) system on Unit 3 had been inoperable for five days while the plant was operating at 100 percent power. The event was considered significant since AFW flow is expected to initiate automatically upon loss of normal feedwater flow. If normal feedwater flow is interrupted without initiation of AFW flow, proper operator actions become crucial to ensure that the core is not damaged.

The occurrence resulted from incorrect tagging of valves during operational testing of new alternate steam supply lines, which required extensive tagging and manipulation of certain isolation valves. Corrective actions by the licensee included verifying all accessible Unit 3 safety-related flow paths, instrumentation, and main electrical alignments; instructing operators on the need for independent verification, the importance of AFW operations activities, and the importance of performing assigned jobs with accuracy and completeness; and performing monthly walkdowns of all accessible safety system flowpaths.

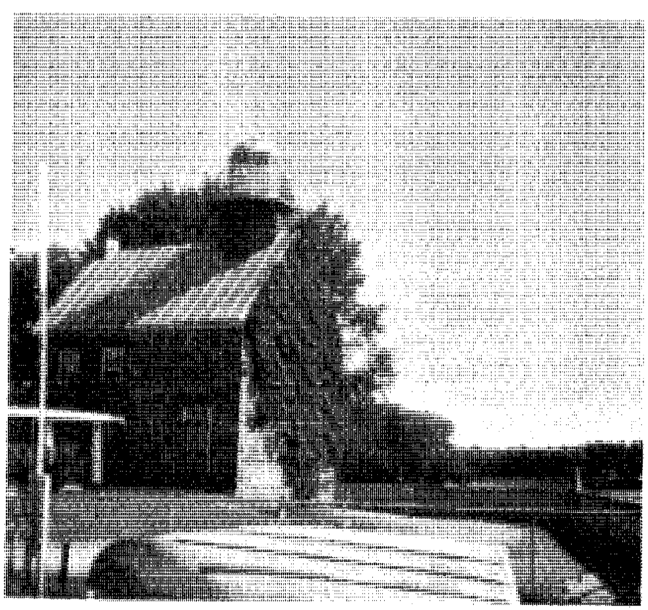
On August 15, 1983, the NRC sent to the licensee a notice of violation and a proposed imposition of civil penalty of $100,000, which the licensee paid.

Large Diameter Pipe Cracking in Boiling Water Reactors. On March 23, 1982, an event was reported which involved leakage from welds on two nozzles connecting recirculation system piping to the reactor vessel of Nine Mile Point Unit 1. The leakage was discovered during performance of a routine hydrostatic pressure test prior to return to operation from a scheduled maintenance outage.

Subsequent inspections and evaluations showed extensive intergranular stress corrosion cracking (IGSCC) in heat affected zones near weld areas of the large (28-inch) diameter reactor coolant recirculation system. The licensee decided to replace the recirculation piping in all five recirculation loops, all ten safe ends, and branch piping as warranted. The replacement material is of a type less susceptible to IGSCC. The findings at Nine Mile



Since 1982, the NRC and industry research agencies have studied problems associated with large diameter pipe cracking in boiling water reactors. These efforts resulted from a 1982 event which occurred at the Nine Mile Point nuclear power plant on the shore of Lake Ontario, near Oswego, N.Y., in which welds on recirculation system piping nozzles began to leak. All such piping has since been replaced at Nine Mile Point, and research and training to address the problem continued in 1984, using original pipe segments from the plant. The fishery tug shown above is in the Oswego Harbor near the plant. It is the last of its kind on the lake; both commercial and recreational fishing, however, at lake locations such as the one near the Selkirk Harbor Light shown below, remain important aspects of life in the upstate New York area.



Point Unit 1 were the first examples of major cracking in large diameter piping in the U.S. (cracking in large diameter piping had been reported on some foreign reactors).

The NRC issued Bulletin 82-03, Revision 1, in October 1982 for action by nine BWR plants scheduled for refueling outages in late 1982 and early 1983. Inspections pursuant to this bulletin showed cracking in five of the first seven plants examined, prompting issuance of Bulletin 83-02 in March 1983. This bulletin required augmented inspection of welds in the recirculation system piping, using ultrasonic testing (UT) inspection procedures of demonstrated effectiveness, for all plants beyond those identified in Bulletin 82-03, Revision 1, at their next refueling or extended outage, but no later than January 1984. In conjunction with these bulletins, joint efforts by the NRC and industry were begun to train and qualify inspection personnel, using improved UT procedures on well-characterized pipe cracks in pipe segments removed from Nine Mile Point Unit 1, to assure higher reliability in the inspection process.

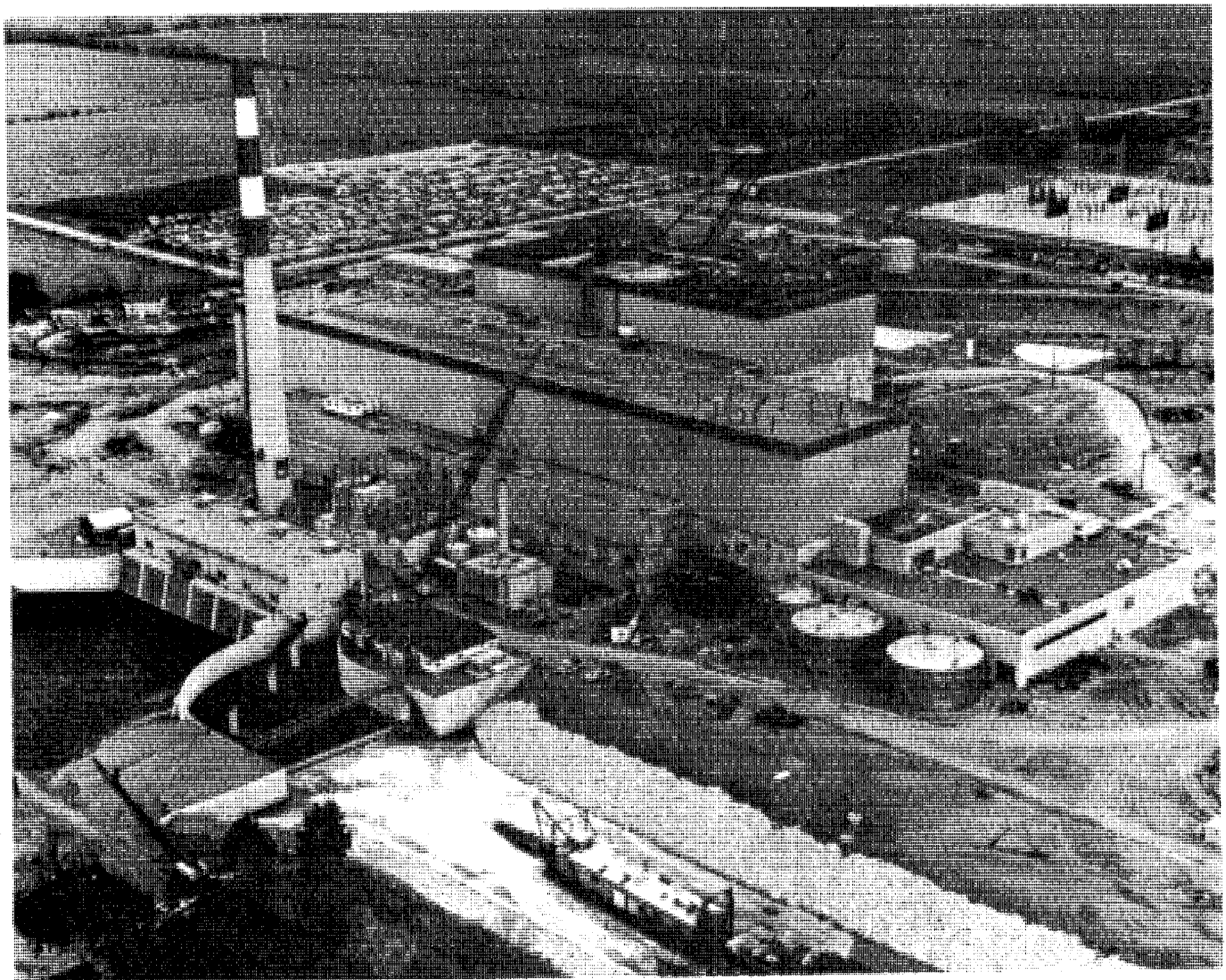
During the remainder of the report period, inspections showed extensive pipe cracking at several BWR plants. Repair or replacement of piping was performed as necessary.

Uncontrolled Leakage of Reactor Coolant Outside Primary Containment. In May 1984, the AEOD staff completed a case study of a plant systems interaction event which occurred at Hatch Unit 2 on August 25, 1982 (see "Technical Studies," earlier in this chapter). As described in the report, a complex series of systems interactions occurred during post-scram recovery operations. Primary coolant discharged through a partially stuck-open scram discharge volume drain line isolation valve into the equipment drain system, subsequently discharging to the open areas of the reactor building through an open drain hub. Even though the isolation valve was only partially open, this represented a direct flow path for the primary coolant, and indicates the potential for an even more significant degradation of the primary coolant boundary. The resultant harsh environment in the reactor building shut down the operating reactor core isolation cooling system (a system important to safety). Although adequate core cooling capability was available to protect fuel integrity during the event, had the isolation valve failed completely, and had the leakage been larger or significantly prolonged, the possibility existed that other vital equipment located in the reactor building could have been threatened.

The main steam line isolation valve manufacturer, Rockwell International, had previously investigated the cause of similar valve failures at Hatch and other facilities, and had recommended three potential solutions to the disk-to-stem disassembly problem for the Rockwell valves. These recommended actions had either not been finalized or not been adequately evaluated and implemented for Hatch at the time of the event. The licensee replaced the entire disk and stem assembly in both the inboard and the outboard isolation valves on the affected steam line, and performed several other corrective actions including equipment overhead and procedure changes.

The main steam line isolation valve disk-to-stem disassembly problem had been the subject of NRC Information Notice 81-28 issued on September 3, 1981, based on similar events. Regarding the scram discharge volume drain valve failure, the NRC had, in July 1980, requested all operating BWR licensees to propose technical specification surveillance requirements for the existing scram discharge volume vent and drain valves. The NRC determined in December 1980, that long term hardware improvements in the isolation arrangements for the scram discharge volume system would also be required. On June 24, 1983, the NRC issued a confirmatory order regarding the surveillance requirements. The same order confirmed the Hatch licensee's commitment to install permanent scram discharge system modifications (including redundant vent and drain valves) by December 31, 1983. These modifications were developed by the BWR Owners Subgroup.

Improper Control Rod Manipulations. Events involving improper control rod insertions and other violations at two separate boiling water reactors demonstrated breakdowns in plant management control systems designed to control operations activities and ensure safe operation of the facilities. The first event occurred on March 10 and 11, 1983 at Quad Cities Unit 1; the second event occurred on July 14, 1983, at Hatch Unit 2. For both events, the cause was a weakness in the plant management control systems, as evidenced by the number of procedural violations, the number and types of personnel involved, the poor judgment exercised by the control room staff, and insufficient guidance provided by management.



NRC's Office of Analysis and Evaluation of Operational Data, in 1984, found that procedural violations and poor management guidance were among the causes of two 1983 events involving improper control rod insertions and other violations at the Quad Cities (Ill.) and Hatch (Ga.) nuclear plants. At Quad Cities, shown here, civil penalties of $150,000 were assessed by the Commission; the Hatch licensee was assessed $100,000.

At Quad Cities Unit 1, the plant was being shut down for a scheduled maintenance outage. During the day shift on March 10, the nuclear engineer requested to have the rod worth minimizer (RWM) bypassed so that he could load a new shutdown control rod sequence into the RWM computer. (The RWM system prevents rod movements, if the existing control rod pattern deviates from a specific sequence developed by the plant nuclear engineers and loaded into the RWM computer memory.) The nuclear engineer loaded the sequence into the RWM computer and gave the unit operator the new shutdown control rod sequence procedure and an RWM rod sequence computer printout of a rod withdrawal sequence that was the reverse of the approved rod insertion sequence. Following shift change, the operator mistakenly concluded that the rods should be inserted in the sequence listed on the RWM computer printout. This sequence was the reverse of the proper new sequence.

At Hatch Unit 2, during normal startup activities from a refueling outage, the plant was operating at about 25 percent power. Problems with main condenser vacuum had occurred and air ejector troubleshooting had been in progress. Condenser vacuum began to decrease and the turbine was unloaded and tripped. Control rods were inserted in an attempt to reduce reactor power to within the limit of the mechanical vacuum pump so that it could be placed in service in order to maintain vacuum above the trip setpoint of the reactor feed pumps. To reduce power more quickly, the licensee bypassed the RWM and assigned a second licensed operator to verify control rod movement as permitted by the technical specifications.

When the operator reached groups of low worth peripheral rods in the sequence, a collective discussion among the licensed operators and the supervision in the control room resulted in a decision to scram individual rods by using the individual scram switches at the scram timing panel which was already set up for scram time testing. This was not an approved procedure and resulted in the insertion of rods in an out of sequence manner. While the plant operator continued inserting rods at the front panel, two other operators began to insert rods at the scram timing panel with the individual scram switches. When the front panel operator observed those rods going in, he stopped inserting and verified further insertions from the scram panel. A process computer printout indicated that several rods were not fully inserted. These rods were subsequently rescrammed. One rod was also found in a position which was not expected based upon the rod manipulations performed by the operators. Because the one rod was improperly positioned, the reactor was scrammed as required by procedure.

Several corrective actions were taken at both Quad Cities Unit 1 and Hatch Unit 2, including modifying procedures and training techniques, and counseling individuals on the improper actions taken during the event. The NRC proposed civil penalties of $150,000 for Quad Cities and $100,000 for Hatch. These were subsequently paid by the licensees.

Emergency Diesel Generator Problems. On August 12, 1983, emergency diesel generator (EDG)-102 at Shoreham (99 percent construction completion) failed due to a fractured crankshaft. The failure occurred after 1.75 hours of testing at the two-hour overload rating (3900 kW). At the time of failure, EDG-102 had accumulated about 718 operating hours and about 12.5 hours at the two-hour overload rating. The test in progress when the crankshaft fractured was being performed to demonstrate EDG load carrying ability following replacement of all eight cylinder heads with a newer design (originally supplied cylinder heads had developed leaks from the cooling water area). There are three EDG units at Shoreham. Examination of the other two EDGs identified cracks similar in location and orientation to the one which developed into a fracture on EDG-102. In addition, four of 24 connecting rod bearings were found to contain cracks in the bearing shells.

The EDGs are TDI Model DSR-48 diesels. These EDGs are the only DSR-48 diesels manufactured with a crankshaft assembly having an 11-inch crank pin diameter and 13-inch crankshaft diameter (11 x 13). On November 13, 1983, the applicant and its technical consultant reported that the crankshaft failures were definitely caused by a basic design inadequacy. Independent analysis by the contractor established that the crankshaft was overstressed relative to industry standards. The licensee has replaced the three 11 x 13 crankshaft assemblies like those reportedly installed in all other DSR-48 diesels. In addition, the connecting rod bearings were replaced with bearings designed to accommodate the new 12-inch pin diameter and to address the factors which caused the earlier bearings to develop cracks.

The NRC staff continues to gather information regarding problems concerning TDI units, reviewing specifics of the problems, and developing a course of action to assure that the affected plants have reliable EDG capability. The staff believes that before additional licensing action is taken to authorize the operation of a nuclear power plant with TDI engines, issues relating to quality assurance, operating experience, and the ability of the machines to reliably perform their intended function, must be addressed.

Inoperable Containment Spray System. On November 29, 1983, while performing a containment spray surveillance test with the plant at 100 percent power, Consolidated Edison of New York discovered that two motor operated spray header discharge valves at Indian Point Unit 2 were found in the locked-closed, deenergized position. This condition would have prevented automatic actuation of the containment spray system during the safety injection phase of an accident.

During a cold shutdown for unscheduled plant maintenance, the spray header discharge valves were closed and tagged out of service. Following the maintenance, personnel were assigned to perform a check-off procedure which should have returned the valves to their proper position prior to heating the reactor coolant system above 350°F and subsequent core criticality. However, due to personnel errors in completing the check-off procedure, this was not done.

The safety function of the containment spray system is to spray borated water into the containment to limit the maximum pressure in the containment to less than the design pressure following certain steam line breaks or loss of coolant accidents (LOCAs) and to reduce the pressure and temperature to minimize containment leakage. The system is also designed to spray sodium hydroxide into the containment to remove radioactive iodine which would limit iodine doses to less than 10 CFR 100 limits should a LOCA occur. The plant also has a containment fan cooler system which is used during normal operation to recirculate and cool the containment atmosphere. Following a LOCA or steam line break accident, the system acts in conjunction with the containment spray system to reduce containment temperature and pressure.

During the time in question, automatic actuation of the containment spray system would not have been possible. However, there are indications in the control room which could inform the reactor operator that spray injection is not taking place. The operators then have various options to manually initiate containment spray. However, if no operator action would be taken, calculations predict iodine doses at the exclusion area boundary which exceed the 10 CFR Part 100 guidelines, if a LOCA occurs.

The licensee's investigation included interviews with cognizant personnel and review of pertinent procedures, qualification programs, technical specifications, and other reference documentation. Immediate corrective action steps included verifying correct valve positions of similarly deenergized safeguards valves found on check-off lists. In addition, the licensee determined that improvements could be made in the training/qualification program of nuclear plant operators to place new emphasis on equipment status identification. The licensee also undertook other long term corrective actions.

On March 13, 1984, NRC Region I forwarded a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $40,000. In addition, the NRC will monitor the actions taken by the licensee to prevent recurrence.

There have been several events at various nuclear power plants which involved inadvertent isolation of either the containment spray system or the chemical (sodium hydroxide) addition tanks while the plants were at power. On May 25, 1984, the NRC issued Information Notice 84-39 to all facilities holding an operating license or construction permit, which described these events.

Through Wall Crack in Vent Header Inside BWR Containment Torus. On February 3, 1984, a through wall crack was discovered in the vent header within the containment torus which degraded the containment pressure suppression capability of Georgia Power Company's Hatch Unit 2. The event raised a possible generic concern for other BWR plants which utilize similar containment and inserting system designs.

Hatch Unit 2 was shut down on January 13, 1984, for an extended outage to replace recirculation piping. On February 3, 1984, during a routine visual inspection of the torus interior, the licensee discovered the circumferential crack in the 54-inch diameter torus vent header. The ends of the pipe on either side of the crack were displaced about 1/2 inch. Further inspection showed that the through wall crack extended about 330 degrees around the header, which has a wall thickness of 0.25 inch.

The containment system is designed such that in the event of a LOCA, pressurized steam and water is released into the drywell. Drywell pressure quickly increases and forces the steam flow through the vents into the vent header. The vent header directs the steam through the downcomer pipes into the torus water resulting in condensation of the steam. The condensation of steam serves to limit the maximum pressure the containment structure will experience. However, as a result of the large through wall crack in the vent header, the amount of steam condensed by the torus would be reduced because some steam would bypass the vent header and reduce the differential in pressure used to drive the steam into the water. This increases the possibility of overpressurizing the primary containment, allowing for a release into the secondary containment. This condition has not been specifically analyzed in the plant's final safety analysis report (FSAR), thus leading to a serious safety concern.

The location of the crack was directly below a nitrogen discharge outlet to the torus. The nitrogen line is 20 inches in diameter with the outlet about seven feet above the vent header. The licensee stated that there have been problems with operation of the nitrogen evaporators and heaters, and that the low temperature isolation provisions had also malfunctioned. The crack was determined to be a brittle-fracture type of failure. The primary contributor to the cracking was attributed to impingement of low temperature nitrogen onto the vent header.

The licensee verified that the same condition did not exist on Hatch Unit 1. Unit 1 was then restarted. (The Unit 1 nitrogen line discharge is not located directly above the vent headers, as it is in Unit 2.) For Unit 2, repairs to the vent header commenced.

General Electric issued a service information letter (SIL) which contained recommended actions to be taken by all BWR owners with Mark I or Mark II containment systems. The actions involve evaluations of inerting system design and operation, performance of a leakage test to confirm the integrity of the vent system, inspection of the nitrogen injection line, and inspection of containment components and equipment.

On March 14, 1984, the NRC forwarded to the licensee a notice of violations based on inspections performed at Hatch Units 1 and 2 between January 21 and February 20, 1984. The violation germane to the vent header problem pertained to procedural inadequacies in not properly implementing a procedure.

NRC issued Bulletin 84-01 on February 3, 1984 to all BWRs with operating licenses or construction permits.

The bulletin requested that facilities with operating licenses in cold shutdown and with primary containment similar to the Hatch containment (Mark I) perform inspections as to the condition of their vent headers. It was also recommended that, for BWRs with Mark I containment that were in operation, the licensees review plant data on differential pressure between the drywell and the torus for anomalies that could be indicative of cracks.

On March 5, 1983, NRC Information Notice 84-17 was sent to all reactor facilities with operating licenses or contruction permits to alert them to possible problems associated with cooling components to below their nil ductility temperatures with liquid nitrogen.

Serious Degradation of Reactor Depressurization System. On February 22, 1984, the NRC was notified by Consumers Power Company that three of four reactor depressurization system (RDS) isolation valves failed to open at Big Rock Point during a surveillance test. At the time of the event, the plant was in hot standby condition (reactor shut down, system at reduced pressure and temperature - approximately 50 psig and 265°F, respectively). The plant had been shut down since February 19, 1984, for various maintenance activities.

The RDS is a set of piping and valves which was installed at Big Rock Point in the mid-1970s. One large pipe from the steam drum feeds four parallel lines; each line contains an isolation valve and a depressurization valve (both normally closed). Both valves must open to allow flow through the line. The purpose of the RDS is to provide a method of rapidly depressurizing the reactor in the event of a small break loss of coolant accident (SB-LOCA). In such an accident the reactor would lose cooling water while the system pressure would remain high. Since Big Rock Point does not have a high pressure injection system, the RDS reduces the system pressure to the point where the core spray system (a low pressure system) can deliver cooling water to the reactor. The plant technical specifications require that three of the four lines be operable whenever the reactor is not in cold shutdown. If the RDS does not operate properly in the event of a SB-LOCA, use of the core spray system could be delayed and the core could become uncovered and damaged.

The licensee determined that the cause of the valves' failing to open was a combination of thermal binding and the increased force holding the valves closed due to the recently installed air amplifier system. The increased force holding the valves closed resulting from the installation of the air amplifier further heightened the effects of thermal binding to the point that the springs were not strong enough to open the valves. Prior to the installation of the air amplifier, there had been no instances of valves failing to open because of thermal binding.

The licensee removed the air amplifier system from service, and returned to the closing air pressure used previously. The licensee disassembled one valve for inspection with no defects found. The valves were then cycled at operating temperature and retested during a partial unit cooldown and depressurization. All valves functioned properly during these tests. Having satisfactorily completed the testing and inspections required by the Confirmatory Action Letter, the licensee was given permission to resume normal operations.

Some abnormal occurrences involving other NRC and Agreement State licensees included the following:

- During the fourth quarter of 1982 and the first quarter of 1983, several foundry workers employed by Nuclear Metals, Inc., of Concord, Mass., received exposures to their hands estimated at 125 rems each quarter. The licensee estimated that some workers had received between 1,000 and 2,200 rems to the hands over a six-year period.
- On August 24, 1983, the NRC Region I office was notified by Thomas Jefferson University Hospital, Philadelphia, Pa., that a patient had been orally administered 100 millicuries of technetium-99m DTPA (diethylene-triaminepentacetic acid) for the purpose of evaluating gastric emptying. The dose prescribed for this procedure was 100 microcuries of technetium-99m, which is 1,000 times less than the dose actually administered.
- On September 13, 1983, a sealed radiation source containing cesium-137 was damaged at the Shelwell Services, Inc., facility in Hebron, Ohio. The cesium contamination was spread about the Shelwell facility and subsequently carried to employees' houses and other locations in the Hebron area.
- On February 20, 1984, an industrial radiographer and his assistant, employed by Industrial NDT, Inc., North Charleston, S.C. (a licensee of the State), received hand exposures estimated to be about 3,000 rads and 5,300 rads, respectively. The whole body exposures were about nine rems and 63 rems to the two individuals.
- On March 6, 1984, a representative of Henry Ford Hospital, Detroit, Mich., reported that a 26-year old female patient had received a therapeutic radiation dose to the head which was 45 percent in excess of that prescribed. The misadministration had occurred in a radiation treatment program which began on January 30, 1984, and was terminated on March 5, 1984, when the excessive radiation dose was discovered.

Nuclear Materials

CHAPTER 5

The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) administers the regulation of nuclear materials. 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 discussed in this chapter include licensing and other regulatory activities concerned with (1) conversion of uranium ore concentrates (after mining and milling) to uranium hexafluoride; (2) conversion of enriched uranium hexafluoride to ceramic uranium dioxide pellets and subsequent fabrication into light water reactor fuel; (3) production of naval reactor fuel; (4) storage of spent reactor fuel; (5) transportation of nuclear materials; and (6) production and use of reactor-produced radioisotopes ("byproduct material").

Highlights of actions taken during fiscal year 1984 include:

- Completion of 29 major and 69 minor licensing actions dealing with fuel cycle plants and facilities.
- Completion of 107 design certification reviews for transportation packages.
- Completion of nearly 5,900 licensing actions on applications for new byproduct materials licenses and amendments and renewals of existing licenses. Over 4,300 of these actions were completed by the five Regional Offices; the remainder were completed at Headquarters.
- Transfer of additional categories of materials licensing from NMSS Headquarters to the five Regions in April 1984.

FUEL CYCLE ACTIONS

Licensing Actions

Licensing actions associated with the possession and use of source and special nuclear material continue to consume a large portion of staff effort. Special nuclear material licenses were issued at reactor sites to allow early receipt of fuel for the purpose of inspection prior to receipt of the Operating License. Currently under review are additional major amendments for fuel cycle facilities, including an automated dry conversion line at Westinghouse Electric's facility in Columbia, S.C. This new production line will have the advantage of increasing the production capability of the plant while decreasing the effluents from the facility. General Electric Corporation, Wilmington, N.C., has a major amendment application under review. This change, entitled "Uranium Process Management Project," includes expanded uranium recovery operations and an improved liquid waste treatment system. When completed, this system will decrease the quantity of uranium in liquid effluents. A new solid waste treatment facility is also being installed at General Electric to allow the recovery of uranium from solids currently being stored at the site.

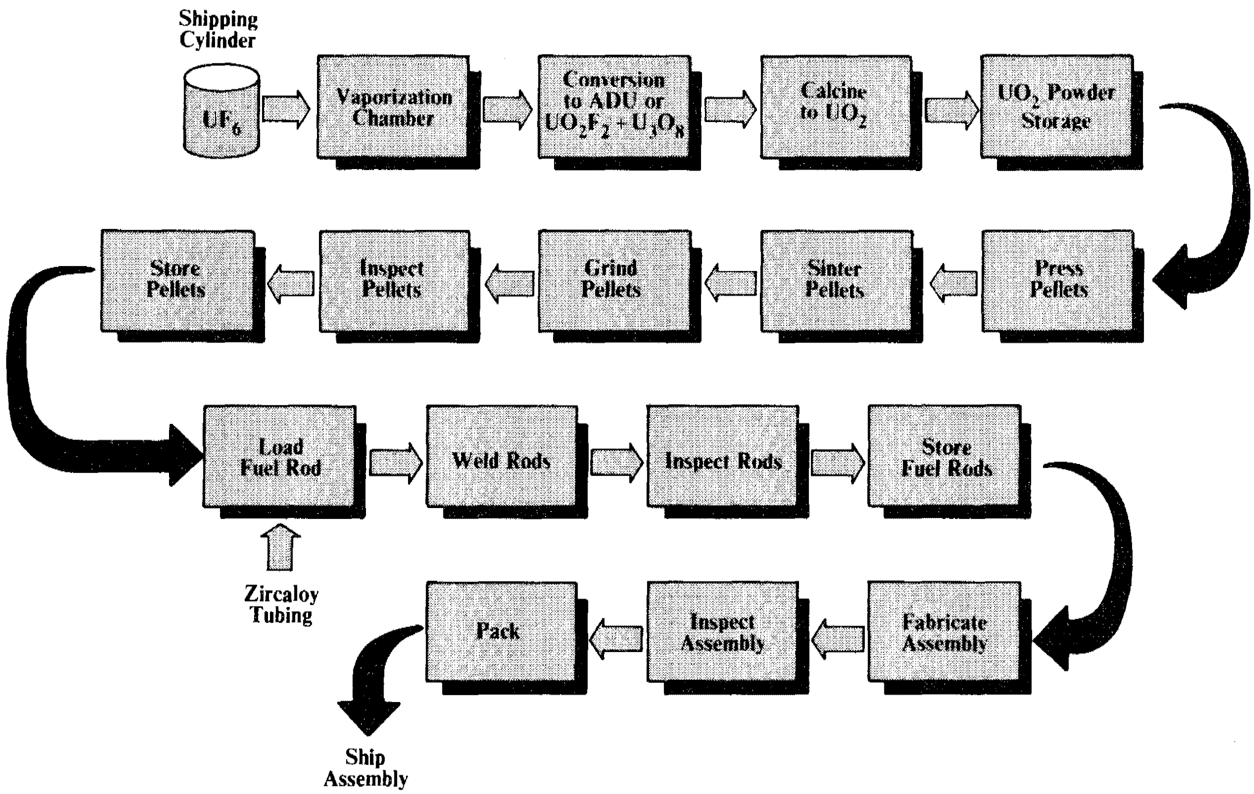
Decommissioning and Decontamination

Decommissioning and decontamination of fuel cycle facilities continue to take up a large amount of staff time. These activities are summarized below:

Decommissioning and decontamination activities are underway at five former plutonium fuel facilities, four uranium fuel fabrication plants and a number of facilities possessing source material. Four plutonium facilities are presently in the process of decontamination, and a fifth facility has essentially been decontaminated. Two additional plutonium facilities have been decontaminated and are now being used for other nuclear purposes. Of the uranium fuel fabrication facilities, the decontamination of two plants has been essentially completed. Licenses for the plants where decontamination is completed will be terminated, or modified if other nuclear activities are planned there. At the remaining plants, equipment is being removed and shipped to disposal sites.

Decommissioning and decontamination activities are also under way at a number of source material facilities. At the end of 1984, there were approximately six licensed facilities undergoing decontamination, with several actions nearing completion. At several sites, decommissioning awaits a decision on where low-level waste, such as process slags, may be disposed.

Kerr-McGee. The Attorney General of Illinois and the Chamber of Commerce of West Chicago, Ill., requested a hearing on the decommissioning of the Kerr-McGee Rare Earths Facility in West Chicago, Ill. In response to the



Fuel Manufacture Flow Diagram

requests, the Commission, by Order dated November 3, 1983, conveyed the requests to the Atomic Safety and Licensing Board Panel. On November 9, 1983 a hearing board was appointed pursuant to the Commission Order. A pre-hearing conference was held on February 2, 1984. Following that conference, the Attorney General was admitted to the proceedings and a number of his contentions were accepted. The Chamber of Commerce withdrew its request for a hearing in favor of a limited appearance statement. A second pre-hearing conference was held on August 22, 1984. The hearing is now expected to be held in March 1985.

On March 2, 1984, the staff issued an Order to Show Cause to Kerr-McGee Chemical Corporation that sought to require the company, among other things, to prepare and carryout a plan for the cleanup of radiologically contaminated areas in and along Kress Creek and the West Branch of the DuPage River.

On March 19, 1984 in response to the Order, Kerr-McGee demanded a hearing. The staff determined not to rescind or vacate the Order and the Commission on June 28, 1984, ordered that a hearing board be established. The Attorney General of Illinois and the Nichiren Shoshu Temple petitioned for leave to intervene in the proceedings. A pre-hearing conference was held on August 22, 1984. The board, in a September 7, 1984 pre-hearing conference Memorandum and Order, admitted (1) the people of the State of Illinois and the Illinois Department of Nuclear Safety and (2) the Nichiren Shoshu Temple as parties to the proceeding. The hearing is scheduled for March 1985.

DOE "UMTRCA" Site. The staff has continued to work with the Department of Energy (DOE) on the remedial action required under Title I of the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978, for the Canonsburg, Pa., site. On-site remedial actions started in the Fall of 1983 and have continued through the summer of 1984. DOE expects the remedial action to be completed in 1985. Once remedial action is complete, the Nuclear Regulatory Commission (NRC) will issue a license to DOE, as site owner, for the care, maintenance and monitoring of the radioactive material stabilized at the site.

Special Sites. Under the "Special Sites" Section (Section 151(c)) of the Nuclear Waste Policy Act of 1982, title to low-level waste generated as a result of recovering zirconium, hafnium or rare earths from source material, and the land upon which the wastes are disposed, shall be transferred to DOE upon request of the owner. However, such transfer can occur only after the site has been decontaminated and stabilized in accordance with NRC requirements, and after the owner has made adequate financial arrangements, approved by NRC, for long term maintenance and monitoring.

NRC staff are developing decontamination and stabilization criteria, and long-term financial arrangement requirements, to apply to these "Special Sites." This work is being closely coordinated with DOE to ensure that the final criteria and arrangements meet its needs.

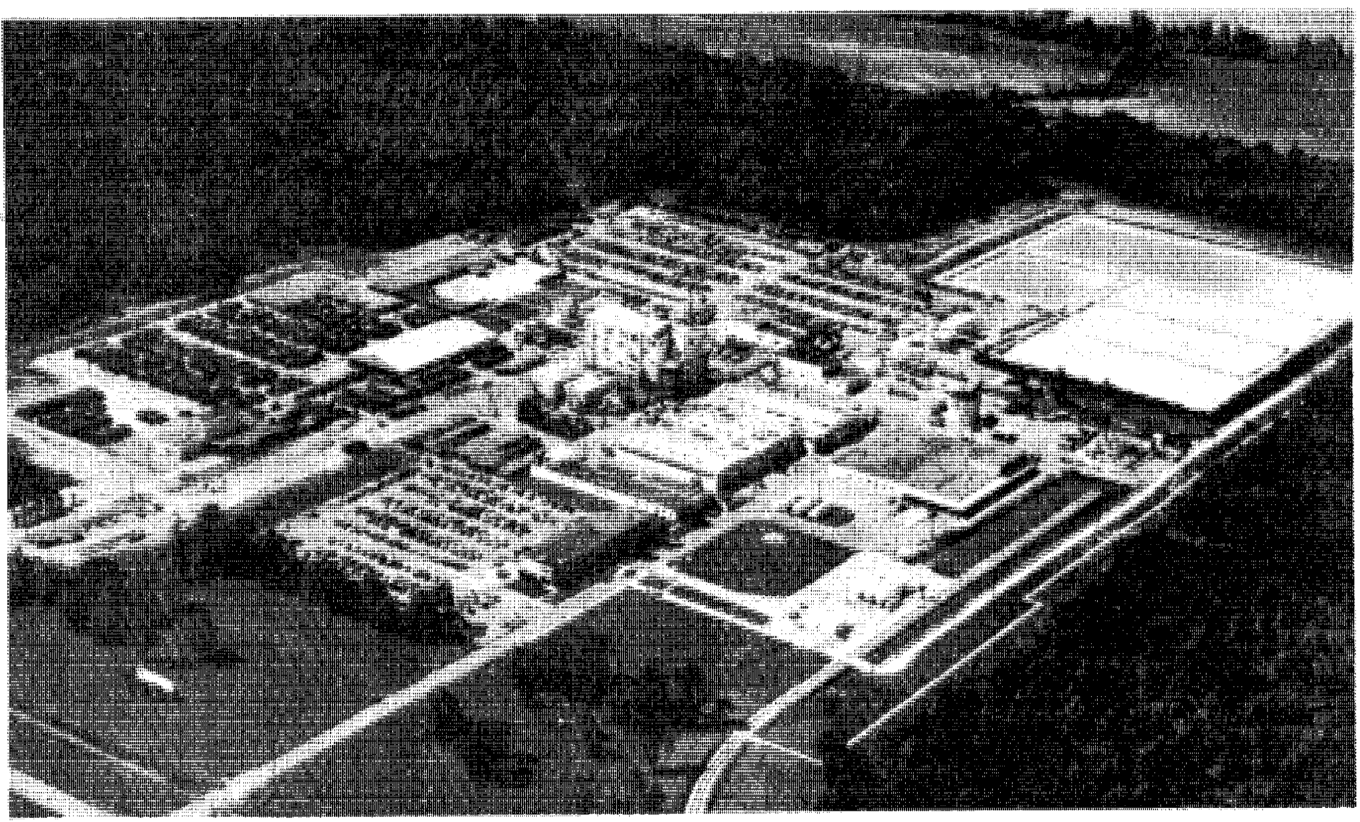
West Valley Demonstration Project. The West Valley Demonstration Project Act of 1980 directed the Department of Energy (DOE) to carry out a demonstration of the solidification of the high-level radioactive wastes at West Valley, N.Y. These wastes were produced as a result of the reprocessing of commercial and Federal nuclear fuel from 1966 to 1972. Under the same act, the Commission was given a safety oversight role to further assure that the project was carried out with due regard for public health and safety.

On February 25, 1982 DOE took possession of the West Valley site to begin the demonstration project. NRC has been advised continually by DOE of all activities at the site which could affect public health and safety. The Commission has been observing and commenting on these activities since DOE took possession of the site.

In 1983 DOE, in consultation with the NRC, selected borosilicate glass as the waste form for the solidification process. The process to be used is similar to the process selected for the vitrification of the high-level wastes at the Savannah River Project in South Carolina. The heart of the process is a slurry-fed ceramic melter.

In 1984, DOE continued with the design and planning for the project, obtaining detailed information on the radiochemical characteristics of the waste. A component test stand was constructed for installation and test of key vitrification components. DOE hopes to begin producing nonradioactive test specimens of the borosilicate glass by the end of 1984. A test facility for sludge mobilization was completed and hydraulic testing initiated. Cell decontamination activities continued in the West Valley facility to prepare for actual waste solidification operations now projected to begin in 1988.

The geohydrological investigations of the facility disposal area used for low-level radioactive plant wastes is continuing. The need for these continued studies was emphasized by the previous detection of contaminated organic solvent in a shallow well adjacent to the disposal area. DOE has determined that no solvent has migrated to nearby surfaces outside of the disposal area boundaries and has initiated actions to limit the source of contamination. Recommendations for further investigation of the groundwater flow regimen were made by NRC to determine what further corrective actions may be appropriate.



This photo shows the uranium hexafluoride (UF6) conversion plant operated by the Allied Chemical Company near Metropolis, Ill. The conversion of UF6 is one of five steps in the "front end" of the fuel cycle, in which the nuclear fuel for reactors is produced. Except for uranium mining and uranium enrichment, the NRC regulates the entire process.

Interim Spent Fuel Storage

The Nuclear Waste Policy Act of 1982 (NWPA) clearly established that utilities have the primary responsibility for interim storage of their spent fuel until a Federal repository or monitored retrievable storage (MRS) installation is available, projected for 1998. Although some contingency storage is available from DOE under NWPA, this Federal interim storage is available only as a last resort under NWPA criteria and NRC implementing regulations (10 CFR Part 53). Thus, utilities continue to develop plans for providing necessary additional storage capacity as they approach current storage limits of their reactor pools.

When possible, utilities continue to re-rack spent fuel pools, a measure that has extended storage capacity for most reactors into the 1990's. Beyond re-racking, rod consolidation is being considered by some utilities as a means of increasing pool storage capacity. On-site dry storage of aged spent fuel in modular units also is of high interest for meeting storage needs.

Four topical safety reports for dry storage cask designs are presently being considered by the NRC staff. If found acceptable by the staff, a topical report can be referenced in the license application by a utility to expedite the review of a proposed dry storage system. Final evaluation is being made of the topical report on the Castor IC cask design submitted by Gesellschaft fur Nuklear Service (GNS) of West Germany. This metal cask has a capacity of 16 BWR fuel assemblies. Based on staff comments on the initial report, General Nuclear Systems, a partnership of GNS and Chem-Nuclear Corp., is preparing a revised report on the Castor V cask design for submittal in early 1985. This cask has a capacity of 21 PWR assemblies and is proposed for use by the Virginia Electric and Power Company (VEPCO) at its Surry Nuclear Power Station under a license application under review by the NRC staff. Transport certification by the NRC for the Castor V cask also has been requested by General Nuclear Systems. Staff review of two topical reports filed by Ridihalgh, Eggers and Associates (REA) is on hold pending response by REA on staff comments. The REA dry cask designs would provide storage for 24 PWR or 52 BWR fuel assemblies. Three other firms have informed the NRC that topical reports for metal dry cask storage designs will be submitted in fiscal year 1985. It is expected that transportation certification will also be requested for these casks to establish them as "dual purpose" casks. Another firm plans to submit a topical report on a modular concrete storage design that is planned for a licensed demonstration at the Carolina Power and Light Robinson reactor site in South Carolina.

IMPACT OF THE NUCLEAR WASTE POLICY ACT ON INTERIM STORAGE OF SPENT FUEL

The Nuclear Waste Policy Act (NMPA) of 1982 (PL-97-425) defines the Federal Government's overall program for the management of spent fuels and high-level waste from commercial nuclear power operations.

The NWPA specifies both policy and action on interim spent fuel storage, pending development of a repository or "monitored retrievable storage." The salient policy provisions are:

(1) Utilities have the primary responsibility to provide interim storage, by maximizing use of existing facilities and by adding new on-site storage capacity in a timely manner.

(2) DOE and NRC should take the actions necessary to encourage and expedite use of available storage and necessary construction of additional storage at each reactor site, consistent with safety, economic considerations, and the views of adjacent population; and

(3) DOE should provide limited Federal storage (not more than 1,900 tonnes) when reactors cannot reasonably provide the required storage for continued, orderly operations.

An important feature of the Federal interim storage program is that before DOE may enter into a contract with a utility to provide storage of any spent fuel, the Commission must determine that the utility cannot provide the necessary storage in a timely manner for continued orderly reactor operation. Within 90 days after enactment, the Commission was required to propose a rule specifying the criteria and procedures to be followed in making this determination. This proposed rule was issued April 29, 1983 (48 FR 19382). Under the limitations noted, DOE may enter into contracts with utilities until January 1, 1990, to provide Federal storage of spent fuel, not to exceed 1,900 tonnes. DOE takes title to the fuel at the reactor and provides transportation, subject to NRC regulations.

DOE may not establish Federal interim spent fuel storage capacity at any candidate site for a repository, and must prepare an Environmental Impact Statement if 300 of more tonnes capacity is to be provided at any one site. A State or tribal council may veto plans for storage of 300 tonnes or more at any site, and both houses of Congress must override the veto for DOE to proceed. As of the effective date of NWPA, DOE is also prohibited from using any away-from-reactor storage facility not owned by the government.

Under Title II of the NWPA, which deals with DOE research and development activities, DOE is directed to establish a demonstration program, in cooperation with the industry, for dry storage of spent fuel at reactor sites. The objective of this program is to establish dry storage technologies that the NRC may approve for use by rule, without, to the extent practicable, the need for addition site-specific approvals. Within one year, DOE is to select at least one, but not more than three, reactor sites for demonstration. These demonstrations would be subject to NRC licensing.

Monitored Retrievable Storage

Under the Nuclear Waste Policy Act (NWPA), DOE is directed to submit a proposal for one or more monitored retrievable storage (MRS) facilities for storage of commercial spent fuel and high-level wastes. MRS is considered as a backup to repository development and would be licensed by the NRC if DOE is authorized by Congress to proceed with an MRS facility. For the proposal, due to Congress by June 1, 1985, DOE has chosen as the preferred design a sealed concrete storage cask concept with a dry well concept as the alternate. Both designs would employ a large hot cell complex for receipt, handling, and packaging operations. DOE has consulted with NRC who, as specified by the Act, must prepare comments on the proposal for submittal with the DOE package to Congress.

In preparation for licensing activities—should Congress authorize DOE to proceed with MRS—the staff is preparing minor modifications to its regulation, 10 CFR Part 72, for Commission consideration and publication for comment. These changes would make the rule applicable to both interim storage of spent fuel outside reactor pools and to monitored retrievable storage of spent fuel and high-level waste.

MATERIALS LICENSING

The NRC currently administers approximately 8900 licenses for the possession and use of nuclear materials in applications other than the generation of electricity, or operation of a research reactor. Of these, about 300 are academic, 2800 are medical, and 5800 are industrial licenses. The NRC's licensing program is designed to ensure that activities involving such uses of radionuclides do not endanger the public health and safety. The agency took more than 5800 licensing actions during fiscal year 1984. Of these, 800 were on applications for new licenses, 3600 concerned amendments, 1200 were license renewals, and 200 were sealed source reviews. In addition to these NRC licenses, the 27 Agreement States administer approximately 13,000 licenses. These Agreement States have authority over such materials under regulatory agreements with the NRC (see Chapter 9).

Regionalization of these licensing functions continued in 1984 (see *1982 NRC Annual Report*, p. 66 and *1983 NRC Annual Report*, p. 58). Additional categories of licensing functions were delegated to the Regions on April 2, 1984. Prior to the transfer, license reviewers in the

MATERIALS LICENSES ADMINISTERED BY NRC*
(THROUGH SEPTEMBER 1984)

Types of Licenses

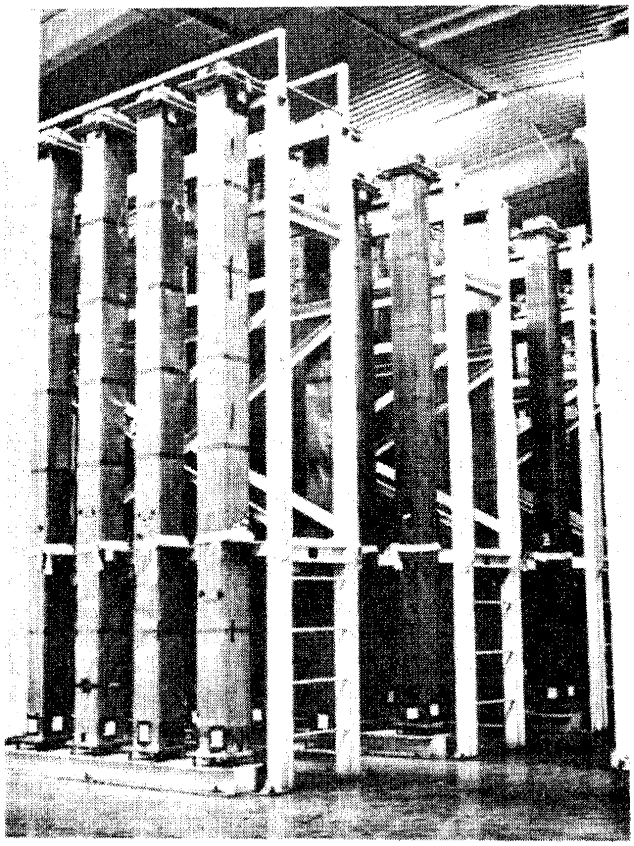
Academic	300
Medical	2800
Industrial	5800
	8900**

Licensing Actions Taken in FY 1984

New Licenses	800
License Amendments	3600
License Renewals	1200
Sealed Source Reviews	200
	5800**

*In addition to the NRC licenses, some 13,000 licenses were administered by 27 states which have authority over certain materials under regulatory agreements with the NRC.

**Totals are approximate due to almost daily fluctuation in numbers.



Fuel assemblies are stored at the fabrication facility until needed at reactor plants. The assembly storage area shown here is typical of those at the 11 operating fabrication plants in the United States.

Regions were trained to process cases which were somewhat more complex than those they had previously completed. A final phase of the regionalization process is scheduled for April 1985, when over 95 percent of the materials licensing program will have been regionalized. As a part of this training for regionalization of the material licensing program, the licensing personnel were introduced to, and made significant comments on, new and revised guides and standard review plans. As a result of these comments, and additional review by the Office of the Executive Legal Director, the guides and standard review plans have been redrafted and sent to the Office of Nuclear Regulatory Research for publication.

Headquarters and Regional staffs worked together to develop a feedback mechanism to identify and resolve any licensing problems at an early stage. This process, known as the National Program Review, was developed to assure technical adequacy, timeliness and consistency in the licensing program. It is an ongoing process that consists of day-to-day interface between the Headquarters and Regional staff, biweekly conference calls, management seminars, reviewer workshops, and an annual visit to each Region. This program helps to assess and upgrade each Region's licensing program and improve Headquarters ability to provide technical assistance.

Industrial Licensing

NRC-licensed radioactive materials are used by industry—in such areas as industrial radiography, manufacture of gauging devices, gas chromatography, and well-logging—and also by members of the general public, in various consumer products. (A more detailed description of the activities covered by NRC industrial licensing may be found in the *1981 NRC Annual Report*, pp. 63 and 64.)

General Licenses. There are two types of NRC licenses for byproduct, source, and special nuclear materials - specific and general. Specific licenses are issued only to specifically named persons following application and NRC review. General licenses are effective without the filing of an application with the NRC for the issuance of license documents to particular persons. However, the manufacturer of products to be distributed to general licensees must apply to 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 generally licensed product and the regulations contained in 10 CFR 32, 40 and 70, the applicant is granted a specific license for distribution of the product to general licensees.

In April 1984, NRC initiated a study to evaluate the adequacy of the existing policy pertaining to the distribution of gauges containing byproduct, source and special nuclear materials to the public under a general license issued by the NRC. This ongoing study combines the efforts of NRC Headquarters, NRC Regional Offices, and the Agreement States. Data collection will be completed by February 1985. This will be followed by the detailed analysis needed that will help determine whether the general license policy should be changed.

Medical and Academic Licensing

Physicians use NRC-licensed radioactive materials in their private offices and in medical institutions for the diagnosis and treatment of patients. In universities, colleges and other academic institutions, instructors and other staff use radioisotopes as part of their teaching and research programs. A more detailed description of these activities may be found in the *1982 NRC Annual Report*, pp. 67 and 68. During the report period, 173 registration sheets were issued for radioactive sources and containment devices. These registration documents require a detailed safety review of the sources and devices, and the preparation of a safety analysis for use by NRC and Agreement State reviewers in the licensing process. A computerized registry system for approved sealed sources and devices is updated twice a year, using 500 reports to NRC Regional offices and Agreement States. During the report period, 100 special reports were produced for both NRC and other governmental users.

Policy Matters

Medical licensing staff has developed a program for conducting a statistical analysis of inspection reports. This will provide a quantitative measure of non-compliance and will highlight problem areas that require the special attention of licensing and inspection staff.

NRC staff met with Food and Drug Administration's National Center for Devices and Radiological Health to discuss the roles of the two agencies in regulating the production, distribution, and use of medical devices that contain byproduct material.

Staff has also met with individual licensees, groups of licensees and representatives of professional organizations to explain NRC's role in regulating the use of byproduct materials and to assure that NRC requirements are not duplicative of other regulatory agency requirements or industry standards.

Oversight Matters

To assure uniform implementation of the program to regionalize licensing, NMSS has provided seminars for regional management, developed standard review plans for use by regional reviewers, and developed a quality assurance program for reviewing regional licensing actions. The staff has also presented licensing workshops to meet individual regional needs.

Headquarters staff has monitored the regional staff's handling of incidents, potential generic problems, and cases where escalated enforcement action was considered, and has offered policy and technical guidance.

EVENT RESPONSE

Plan for NRC Response To Materials Contamination Incidents

In January 1984, the NRC became aware of products imported from Mexico (reinforcing bar and table pedestals) made from steel that had been accidentally contaminated with cobalt-60. The presence of these products in the United States had the potential for creating a radiation safety hazard and resulted in an extensive NRC response to recover and return the contaminated products to Mexico.

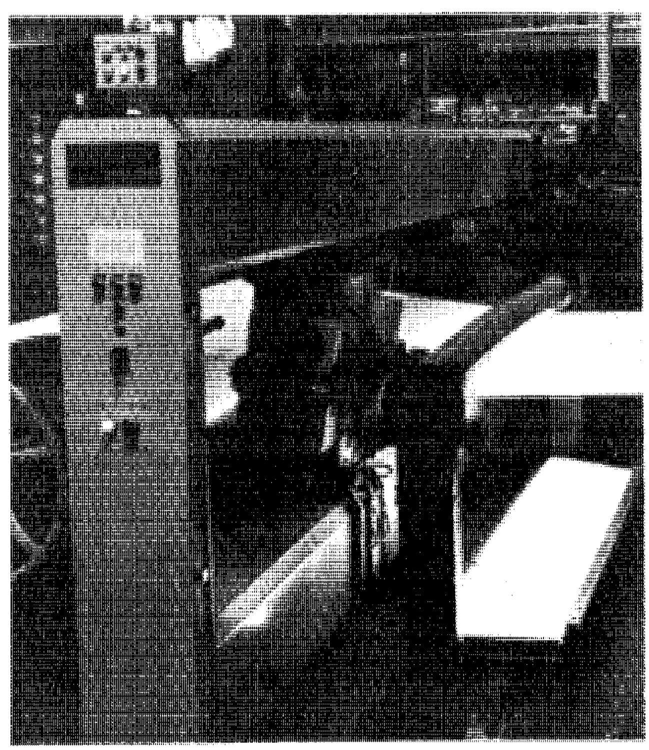
As a result of this incident, NRC prepared an action plan for responding to events involving significant off-site distribution of radioactive materials below the emergency threshold. The interim plan ensures a unified NRC response with a single point of contact within the agency. The plan has been augmented with implementing procedures that ensure each headquarters and regional office understands its role and responsibility.

NRC Policy in Responding to Transportation Accidents

On March 29, 1984, the NRC issued a General Statement of Policy, "NRC Response to Accidents Occurring During the Transportation of Radioactive Material (49 FR 12335)." The stated policy acknowledges the primacy of State and local government officials in taking charge at the accident scene, but assures that the NRC will provide technical assistance in the form of information, advice, and evaluations if requested. The NRC will maintain awareness of the situation until normal conditions are restored, provide information on packaging characteristics for NRC-approved packages, and ensure that the shipper (if an NRC licensee) provides complete and accurate information concerning the radioactive material and details of the shipment. The NRC will also assure that other appropriate Federal agencies are aware of the incident.

TRANSPORTATION OF RADIOACTIVE MATERIALS

The Federal Government regulates the transportation of radioactive materials primarily through the NRC and



Shown above is a paper-thickness gauge in use at a paper mill in Maine. The gauge incorporates an americium-241 radioactive source which, when scanning the paper sheet, automatically and swiftly gives computer readouts of sheet thickness with great accuracy. Paper companies employ this method to produce the exact thickness specified by a customer. The system has largely replaced the mechanical caliper for making these measurements.

the Department of Transportation (DOT). These two agencies have divided their regulatory responsibilities, and documented them in 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 the International Atomic Energy Agency (IAEA) standards. NRC advises DOT on technical matters.

NRC staff worked on several tasks during fiscal year 1984 designed to address transportation safety issues or to provide stability to regulatory requirements regarding the transportation of radioactive materials.

Highlights of Transportation Safety Efforts

NRC staff is working on a systematic study to evaluate transportation accident environments and to develop conditions which more clearly represent the reality of high severity accidents for each mode of transport. During the year, NRC received from its contractor, the Lawrence Livermore National Laboratory, an Air and Marine Report and a Road and Rail Report. Now, in its final phase, the study will attempt to quantify the stress forces associated with hundreds of actual accident cases in terms that can be compared with the hypothetical conditions in 10 CFR 71.73. An independent peer review is scheduled in fiscal year 1985 to address those actual accident cases which cannot be demonstrated to fall within the hypothetical accident conditions.

Interagency Responsibilities

An NRC/DOE Transportation Procedural Agreement was published in the *Federal Register* (48 FR 51875) on November 14, 1983, and is now in force. This agreement is the prologue to the important task of exchanging information and identifying transport packaging issues at the earliest opportunity in the new cask development process. Interagency meetings have begun to discuss how the two agencies will implement the transportation issues from the Nuclear Waste Policy Act of 1982.

A study was begun in November 1983 to analyze the institutional relationships with respect to spent fuel shipments. This Aerospace Corporation Project began with interviews of a number of groups involved in this issue and will conclude in May 1986 with publication of a NUREG/CR report summarizing the study's findings, assessments, and recommendations regarding future spent fuel and high-level radioactive waste shipments.

Spent Fuel Shipments

Shipments of spent reactor fuel began in late July 1983 from a General Electric facility in Illinois to the Point Beach Nuclear Station in Wisconsin. Shipments from the West Valley facility in New York to Point Beach and to the Dresden Nuclear Station in Illinois began in October 1983. Shipments from West Valley to the Oyster Creek Nuclear Station in New Jersey, and the Ginna Nuclear Station in New York are scheduled to begin in the Fall of 1984 and the Spring of 1985, respectively. Rail shipments of spent fuel from the Cooper Station reactor in Nebraska to the General Electric Facility in Illinois began in August 1984, and rail shipments from the Monticello Nuclear Station to the General Electric Facility are scheduled to begin by the end of 1984. These shipments will continue over a period of five years. The return of the spent fuel from these locations to the nuclear utilities that own the fuel is being carried out to reduce storage costs, or, in the case of the West Valley shipments, as a result of a Federal court decision. The return of the spent fuel to Point Beach and Dresden is scheduled for completion in 1984. Site operating personnel inspect each shipment before the transport vehicle leaves the site; the NRC makes audit inspections to ensure that its requirements are being met; and DOT and State agencies make still other safety checks. There have been no significant security or health and safety problems resulting from these shipments.

Safeguards

CHAPTER 6

The Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 direct the NRC to regulate the safeguards provided by certain nuclear facilities and activities to assure protection of the public health and safety and national defense and security. To accomplish this, NRC ensures that measures 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 certain nuclear facilities. In general, safeguards for fuel cycle facilities and non-power reactors emphasize protection against theft or diversion of special nuclear material (SNM), while those for power reactors stress protection against radiological sabotage. (SNM and Strategic Special Nuclear Materials, or SSNM, are shorthand for technical definitions of various kinds of nuclear materials, different quantities thereof, and different degrees of enrichment. In general, SSNM is highly enriched uranium or plutonium.)

During fiscal year 1984, NRC safeguards measures were applied to 87 power reactors, 67 non-power reactors, and 28 fuel cycle facilities. They were also applied to 294 shipments of spent fuel, 19 shipments of SNM involving more than one but less than five kilograms of highly enriched uranium, and three shipments of SNM involving five or more kilograms of highly enriched uranium.

NRC/IAEA Interaction. During 1984, the International Atomic Energy Agency (IAEA) carried out routine inspections of the Combustion Engineering Corporation's low-enriched uranium fuel fabrication plant in Connecticut, the Arkansas-2 reactor in Arkansas, and the San Onofre-2 reactor in California. Also, the NRC submitted accounting data to the IAEA on a monthly basis throughout the year for these facilities as well as for the low-enriched uranium plants of Babcock & Wilcox at Lynchburg, Va., of Exxon at Richland, Wash., and of Westinghouse at Columbia, S. C.

During June 1984, the NRC submitted an update of the eligible facility list for application of IAEA safeguards at licensed facilities to the Executive Branch for review and transmittal to the IAEA.

STATUS OF SAFEGUARDS IN 1984

Reactor Safeguards

Power Reactors. The NRC continued to accelerate its reviews of physical security plans received from applicants for licenses to operate power reactors. The staff expanded the scope of these reviews to provide more comprehensive safeguards statements for the Safety Evaluation Reports. Eight such statements were completed during fiscal year 1984.

The Regulatory Effectiveness Review (RER) program continued to evaluate the effectiveness of safeguards systems and regulations at licensed nuclear facilities and to validate the proper identification and protective measures for vital equipment at power reactors. These reviews are conducted independently of NRC's regular inspection and enforcement activities and are intended to assure that safeguards programs, as implemented by the licensees, are effective against the design basis threats defined in 10 CFR 73.1. During fiscal year 1984, reviews were conducted at five power reactors. These reviews identified some deficiencies in implemented safeguards programs that caused the level of protection to be significantly less than intended by the NRC. The NRC is working to resolve any generic regulatory concerns and to assure timely correction of any licensee program inadequacies identified by the RER teams. It is planned that this RER program will be accelerated in fiscal year 1985.

Non-power Reactors. While currently available information contains no indication of a specific threat aimed at a domestic nuclear facility, acts by international terrorists have repeatedly shown that a threat can materialize without sufficient warning from intelligence sources. Thus, as a matter of prudence, the NRC is considering increased security measures at non-power reactors possessing high enriched uranium. These measures include requiring the removal of all fresh high enriched uranium from those few sites which still have some small quantity stored on-site, and tightening the accountability procedures for maintaining the required 100 rem/hour at three feet irradiation level required when material is exempted from security regulations.

Inspection and Enforcement at Reactors. During fiscal year 1984, the physical security inspection for power reactors was revised to: (1) permit flexibility in management of the program to accommodate contingencies or unavailability of resources, (2) emphasize determinations of adequacy and feedback into the regulatory process, (3) emphasize independent observations of performance and de-emphasize records checks, and (4) encourage early on-site examination of security features prior to completion of construction or installation to preclude later difficulty in changing completed work. (See Table 1 for a summary of inspection activity at reactors.)

Fuel Cycle Facilities

In 1984, safeguards requirements were in force at 28 licensed fuel facilities. The safeguards at 20 of these consisted of physical security and material control and accounting systems. Four of the 20 facilities had actual holdings of formula quantities of SSNM, requiring implementation of extensive physical security and material accountability measures. The remaining eight facilities were not required to have detailed material control and accounting systems, but were required to implement a moderate level of physical security. The activities associated with SNM at these 28 fuel facilities include full scale production, pilot plant operations, decommissioning efforts, and the storage of sealed items.

NRC licensing activity associated with these 28 facilities consisted of review and approval of changes to the in-place physical security and material control and accounting systems. The NRC received and completed action on approximately 160 licensing matters associated with these facilities during 1984.

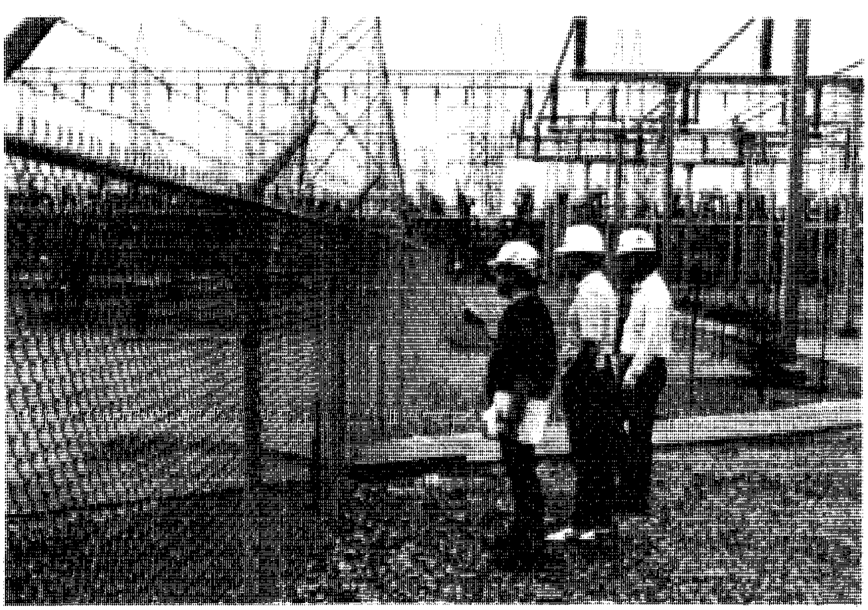
An RER was conducted at one fuel cycle facility during fiscal year 1984. Measures are being taken to correct the deficiencies that were identified.

In May 1983, the Commission placed in abeyance a hearing proceeding that the Natural Resources Defense Council had requested in 1980 regarding the Nuclear Fuel Services high-enriched uranium facility at Erwin, Tenn. This action was taken after all parties involved had submitted a joint motion to the Commission requesting a tightening of the inventory difference limits that require reinventory at the facility. The Commission specified performance criteria for the facility's inventory differences over a two year period by order of May 11, 1983, and by NRC letter to NFS dated May 17, 1983. The licensee met the performance criteria for the first year of operation. In accordance with the Commission Order and letter, the licensee is now following the more restrictive performance criteria for the second year.

Inspection and Enforcement at Fuel Facilities. In fiscal year 1984, the fuel facility safeguards inspection program manual chapters were revised and consolidated, and most of the associated inspection procedures were substantially modified, resulting in a new manual chapter providing updated and expanded overall program guidance for safeguards inspections. Revisions to the material control and accounting safeguards inspection procedures eliminated redundancies, combined similar inspection activities into single inspection procedures, and deleted questionnaires, checklists and worksheets so as to allow greater freedom in tailoring detailed inspection efforts to specific facilities. Changes to the physical security safeguards inspection procedures emphasize independent observations of performance, determinations of adequacy, and feedback into the regulatory process, and de-emphasize records checks. The new manual chapter and procedures were to be implemented on or before the beginning of fiscal year 1985. (See Table 1 for a summary of inspection activity at fuel facilities.)

Transportation

Spent Fuel Shipments. During fiscal year 1984, NRC approved 45 transport routes from the perspective of protection against sabotage. Two hundred ninety-four spent fuel shipments were made over these routes. In



The NRC expanded the scope of its review program for physical security in 1984, as more members of its Regulatory Effectiveness Review staff visited the field. Here a staff member discusses a perimeter intrusion detection system with representatives of a utility at a nuclear power plant site.

Table 1. Summary of Safeguards Inspections Visits—FY 1984

	Number of Licensee Sites Inspected	*Number of Inspection Visits*	*Number of Violations*	*Manhours of Inspection Effort*
FUEL FACILITIES				
Strategic (Formula Quantity)	5	65	17	3,220
Nonstrategic (Less than Formula Quantity)	16	54	21	2,498
TOTAL	21	119	38	5,718
POWER REACTORS				
Operating	79	185	85	3,927
Pre-Operating	19	68	1	2,475
TOTAL	98	253	86	6,402
NON-POWER REACTORS				
TOTAL	36	41	9	585
SHIPMENTS				
Formula Quantity	1	2	0	30
Irradiated Fuel	5	81	0	528
Other	2	3		9
TOTAL	8	86	0	567
OTHER	7	9	0	37
GRAND TOTAL	170	508	133	13,309

N.B. Some data estimated. Because of the multi-disciplinary nature of most inspections and documentation, the safeguards portion of these inspections can only be estimated.

As the NRC intensified its research and planning activities concerning the growing terrorist threats at nuclear facilities overseas, security personnel at plants in this country continued to train for measured response to hypothetical hostile situations. In this photo, several nuclear power plant security officers engage in training for crowd control.

conjunction with these route approvals, to facilitate the disclosure of spent fuel shipment information, NRC publishes a document (NUREG-0725, Revision 4) entitled "Public Information Circular for Shipments of Irradiated Reactor Fuel," which contains all approved routes. The latest revision of this circular was published in June 1984.

SSNM Shipments. Three export shipments, each involving five or more kilograms of highly enriched uranium, were made during the reporting period. Ten export and nine domestic shipments, each involving less than five but more than one kilogram of highly enriched uranium, were made also during the reporting period.

Shipment Route Surveys. In fiscal year 1984, NRC safeguards teams, each composed of an NRC HQ/SG representative and a regional representative from the region concerned, conducted field surveys of routes proposed for the shipment of spent nuclear fuel or of SSNM, working with more than 185 local law enforcement agencies. These teams analyzed 45 routes through 38 States, traveling approximately 3,200 total route miles. The NRC brochure entitled "Information Package on Spent Nuclear Fuel Shipments for Law Enforcement Agencies" was distributed to local officials and agencies during these surveys.

Transport Inspection and Enforcement. During fiscal year 1984, the NRC continued to inspect selected domestic shipments and the domestic segments of import and export shipments of SSNM. These shipments were inspected at points of origin, in transit, during intermodal transfer and temporary storage, and at destinations. No items of noncompliance were noted. (See Table 1 for a summary of transportation inspection activity.)

Contingency Planning and Threat Assessment

Safeguards contingency plans deal with threats, thefts, and sabotage relating to licensed SNM and nuclear facilities. In August 1984, the NRC staff completed a review and revision of its headquarters contingency plan in accordance with "Agency Procedures for the NRC Incident Response Plan" (NUREG-0845).

In light of terrorist activity abroad, in February 1984, the NRC notified licensees of its concerns regarding the potential use of vehicle bombs by terrorists against nuclear activities or facilities. This notice suggested that power reactors and certain nuclear fuel facilities review their vehicular access control equipment and procedures as well as contingency plans and security measures to protect against entry by unauthorized vehicles. During fiscal year 1984, NRC threat analysis of available information indicated no change to the domestic threat environment as it relates to a vehicular explosive threat. As a matter of prudence, however, the NRC elected to pursue research in order to obtain a clear and comprehensive understanding of the technical aspects of the issue so as to be prepared if future changes in the domestic threat environment indicate a need for upgraded protection against a vehicular explosive threat.

The staff updated its "Safeguards Summary Event List" (NUREG-0525) again in June 1984 (Rev. 9). This list provides data on safeguards-related events involving licensed nuclear material and facilities. The staff also completed semiannual design basis threat reviews in January and July 1984, as well as a review of the coordination between NRC and DOE safeguards programs and hypothetical design basis threats. No change in the NRC design basis threat was made as a result of these reviews.

The "Communicated Threat Credibility Project" continued to provide support in the form of guidance to the NRC, DOE, the FBI and other concerned agencies for investigation of communicated threats.

SAFEGUARDS REGULATORY ACTIVITIES AND ISSUES

Reactor Safeguards

Power Reactors. On August 1, 1984, the NRC published for public comment a package of three proposed rules designed to help safeguard power reactors against an insider threat (see *1982 NRC Annual Report*, p. 75). The cornerstone of the package is the proposed Access Authorization Rule which would provide for a screening program for persons seeking unescorted access to the protected areas and the vital areas of power reactors. As proposed, screening requirements consist of three major industry-run components: background investigation, psychological assessment, and continual behavioral observation programs. The other two rules of the "insider rule" package would clarify and refine requirements for the designation and protection of vital locations containing safety-related equipment, and requirements for physical pat-down searches of employees at protected area portals. The staff expects to begin analysis of public comments on the rules in early 1985, with subsequent development of final requirements later in the year.

Fuel Facilities
Material Control and Accounting

Strategic Special Nuclear Material (SSNM). In February 1984, NRC published a proposed rule, inviting public comment on a regulatory approach to: (1) provide for timely indication of possible loss of SSNM (e.g., highly enriched uranium and plutonium); (2) facilitate the recovery of lost material; and (3) provide long-term assurance that no significant loss has occurred. The public comment period on the proposed rulemaking ended in September 1984. The NRC staff is analyzing public comment and preparing a final rule for consideration by the Commission. Reactors are not affected by this rule, since it applies only to fuel cycle facilities.

Low-Enriched Uranium (LEU). The NRC has been evaluating appropriate ways to allow for the difference in

This is a typical entry control point at a nuclear power plant. Here a security guard carries out a "pat-down" search as a condition for entry into the plant.

safeguards significance between SSNM and LEU (less than 20 percent enrichment), and to develop more cost-effective accountability requirements for LEU facilities by permitting licensees greater flexibility in designing site-specific measures to comply with regulations. The Commission is now considering a final amendment to the regulations and the associated acceptance criteria required for licensing actions.

Transportation

Convention on Physical Protection. The NRC staff is developing a final rule to implement the Convention on the Physical Protection of Nuclear Material, a part of the IAEA agreements originally proposed by the Secretary of State in 1974, and signed in 1980. The Convention, which provides for the security of international shipments of significant quantities of source or special nuclear material, was ratified by the Senate on July 30, 1981. The NRC amendments call for: (1) the physical protection of transient shipments of SSNM of moderate and low strategic significance, irradiated reactor fuel and natural uranium; (2) advance notification to the NRC of the export of Convention-defined nuclear materials; (3) advance notification, and assurance of protection, to the NRC on transient shipments of Convention-defined nuclear materials between countries that are not parties to the Convention; and (4) advance notification, and assurance of protection, to the NRC on the importation of Convention-defined nuclear materials from countries that are not parties to the

Convention. By adopting these amendments, the United States will have implemented the provisions of the Convention, resulting in improved security for Convention-defined nuclear materials during international transport.

Spent Fuel Transportation. On June 8, 1984, the Commission issued for public comment a proposed rule for physical protection of irradiated reactor fuel shipments to replace the interim requirements issued in 1979 and amended in 1980. Research projects completed in 1981 and 1982 show that the quantity of radioactive material likely to be released as a result of sabotage is much less than was supposed when the interim rule was issued. Public comments are being analyzed by the staff and a revised final rule is being developed which would eliminate overly conservative requirements now applicable to spent fuel shipments.

SAFEGUARDS RESEARCH, STANDARDS AND TECHNICAL ASSISTANCE

Approximately $5.4 million was spent in fiscal year 1984 on safeguards technical assistance and research contractual projects. Of this total, approximately $4.3 million was spent on technical assistance projects, and the remaining $1.1 million on research projects. Some of these projects are discussed below.

Technical Assistance

- *Development of Acceptance Criteria*. Acceptance criteria provide a framework to help ensure a systematic, technically sound review of the significant aspects of a safeguards plan, and to document the decision-making process with identification of the specific regulations, regulatory guidance, staff technical positions and professional judgments which are the bases for findings of acceptability. During fiscal year 1984, the project provided proposed acceptance criteria for use by licensees and license reviewers in the development and subsequent review of safeguards plans to be submitted by license applicants in complying with a new proposed regulation on material control and accounting of low enriched uranium.
- *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 explosion threats. The assessment methodology evaluates a number of different factors associated with threat messages. This program enables NRC, in coordination with other federal agencies, to perform a unique assessment of nuclear-related threats and provides vital inputs to joint NRC-DOE contingency planning in the area of threat evaluation. It also supports NRC's capability to respond to threats, such as theft of nuclear material, sabotage of a nuclear facility or a nuclear-related hostage situation, in a coordinated and timely manner.
- *Nuclear Materials Management and Safeguards System*. The system supported by this project is the national data base and information support system for managing and safeguarding nuclear materials. The system is jointly funded with DOE and is based on the premise that organizations with mutual or related interests can collectively create a more efficient and effective information support system than they could individually. The system has the following objectives: (1) the accounting for nuclear materials flowing through government and commercial facilities, and (2) the fulfillment of international commitments derived from bilateral agreements, IAEA requirements for export/import reporting, and IAEA requirements under the US/IAEA Safeguards Agreement.

Safeguards Research

- *Human Factors*. During fiscal year 1984, the Office of Nuclear Regulatory Research (RES) undertook a study to analyze safety/ safeguards interaction during safety-related emergencies at nuclear power reactor facilities. The results provided a technical basis for guidance to licensees on assigning appropriate duties and responsibilities of safety/safeguards personnel during safety-related emergencies.
- *Research in Support of Licensing*. Two RES studies were initiated in fiscal year 1984 to improve the technical bases for safeguards licensing. These were: (1) research to provide guidance for licensees in defining, developing, implementing, and maintaining computer-managed physical security systems to meet regulatory requirements; and (2) research to provide guidance to licensees for developing response systems for recurring loss from testing a site-specific set of material loss alarm resolution procedures for a segment of an existing nuclear material processing plant. Three multi-year research projects were initiated in prior years to: (1) quantify experimentally the magnitude and chemical/physical form of any released radioactive material which may result from sabotage of shipments of spent fuel from a High Temperature Gas Reactor or a non-power reactor, (2) assess and upgrade reactor vital equipment determination techniques; and (3) evaluate selected intrusion detection sensors under harsh environmental conditions.
- *Standards Development*. The Handbook of Nuclear Safeguards Measurement Methods, NUREG/CR-2078, was published. Work continued on two

reference documents needed for implementation of the MC&A Reform Amendment: the "Structural Handbook of Nuclear Material Accountability," and "Handbook of Passive Non-Destructive Assay of Nuclear Material." Also, the RES staff initiated an effort to review the existing MC&A regulatory guides to ensure that all guides are relevant to the current regulatory requirements.

SAFEGUARDS DECENTRALIZATION

Licensing functions involving review of safeguards system changes that do not decrease the effectiveness of the program, as defined in 10 CFR 50.54(p) and 10 CFR 70.32(c), (d), (e), and (g), have been transferred to the NRC Regional Offices. The transfer of responsibility for the conduct of transportation route surveys to Regions I, II, IV, and V on October 1, 1984, completed the decentralization of that safeguards activity as well.

Safeguards decentralization activities in 1984 focused on four general areas: (1) assessment of Regional Office performance of delegated review functions; (2) the revision and update of the Regional Guidance Documents based on items identified in the annual assessment and day-to-day contact with the Regions; (3) conduct of an annual workshop for the regional reviewers and section chiefs to clarify technical aspects of review activities, insure a consistent approach to policy areas, and establish a common approach to licensing issues; and (4) performance of day-to-day oversight activities of the Regions by providing technical advice and assistance.

Waste Management

CHAPTER 7

The NRC's nuclear waste management program is conducted and coordinated through the Office of Nuclear Material Safety and Safeguards (NMSS). NMSS activities in this area relate mainly to regulation of the disposal of all nuclear waste derived from NRC-licensed source, byproduct and special nuclear material, including uranium mill tailings.

NRC waste management comprises these basic functions:

- Developing the criteria and framework for high-level waste regulation, including the technical bases for the licensing of high-level waste repositories.
- Licensing and regulating low-level waste disposal facilities and providing the technical support for such regulation.
- Providing national program management for the licensing and regulating of uranium recovery facilities and associated mill tailings. These operations include uranium mills, heap-leach facilities, ore-buying stations, solution mining, and byproduct uranium recovery.
- Reviewing and concurring in significant decisions of the Department of Energy (DOE) related to inactive uranium mill tailings, remedial action program sites, and licensing of long-term monitoring and maintenance programs for stabilized tailings piles.

Highlights of 1984

In fiscal year 1984, NRC staff continued its work on developing, improving and implementing regulations for the safe management and disposal of radioactive wastes. During this period, NRC staff initiated two rulemaking proceedings to amend certain procedural and technical portions of 10 CFR Part 60 regulations for the disposal of high-level waste to bring them into conformity with requirements of the Nuclear Waste Policy Act of 1982. On August 31, 1984, the Commission published a final decision on its waste confidence rulemaking. The Commission found reasonable assurance that the safe disposal of high-level radioactive waste and spent fuel is technically feasible, that there will be sufficient disposal capacity available within 30 years after the expiration of any reactor operating license, and that the waste can be safely managed until a repository is available.

In the area of low-level waste disposal, licensees began to comply with changes in 10 CFR Part 20 regarding waste classification, waste form and manifests. The NRC issued inspection guidance on the Part 20 requirements related to 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." An inspection program was also initiated to confirm licensee compliance.

The Congressionally mandated suspension of portions of 10 CFR Part 40—concerning the implementation of the Uranium Mill Tailings Radiation Control Act of 1978—ended on April 1, 1984, leaving NRC rules in full force (See *1983 NRC Annual Report* p. 74). The Environmental Protection Agency (EPA) issued standards for uranium and thorium mill tailings at licensed commercial processing sites in October 1983. These standards became effective for NRC and Agreement State licensees on December 6, 1983. An NRC task group is currently working to conform the NRC rule to the EPA standards.

The NRC continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) for inactive sites. This activity is required by the Uranium Mill Tailings Radiation Control Act of 1978.

Congressional enactment of the Nuclear Waste Policy Act of 1982 (Public Law 97-425) on January 7, 1983, created a need for additional resources within the Division of Waste Management of NMSS, principally to carry out expanded NRC responsibilities in the high-level waste program area. Consequently, NMSS effected a major reorganization to better enable the Division to meet its responsibilities under the Act.

The NRC Waste Management Review Group (see *1980 NRC Annual Report* pp. 127-8), which is responsible for coordinating technical assistance and research projects, approved descriptive summaries and statements of work for 40 projects during the report period.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development

The staff initiated two rulemaking proceedings to amend the procedural and technical provisions of 10 CFR Part 60, "Disposal of High-Level Radioactive Waste in Geologic Repositories". The procedural amendments deal with the control of DOE's site characterization plan and the participation of States and Indian tribes in the process of siting, licensing, and developing high-level radioactive waste disposal facilities. The staff found that these procedural amendments were necessary so that 10 CFR Part 60 would conform to requirements of the Nuclear Waste Policy Act of 1982. The proposed procedural amendments were sent to the Commission on June 26, 1984, and with the Commission's approval, they will be released for public comment.

The technical amendments to 10 CFR Part 60 would make this regulation applicable to geologic repositories that may be excavated within the unsaturated zone. The proposed amendments were published for public comments on February 16, 1984. After considering these comments, the staff prepared the final amendments which were awaiting Commission approval at the close of the report period.

Regulatory Guidance

During 1984, the NRC staff continued to develop technical positions and other guidance documents by which to provide DOE with acceptable methods which that agency may adopt to satisfy the NRC's 10 CFR Part 60 requirements for high-level waste disposal. The NRC staff published three such technical positions in draft during the report period:

- "Hydrologic Testing Strategy for the Basalt Waste Isolation Project (BWIP)."
- "Licensing Assessment Methodology for HLW Geologic Repositories."
- "In-Situ Testing during Site Characterization for High-Level Nuclear Waste Repositories."

The staff also prepared five issues-oriented site technical positions (STP) directed at the three geologic media under investigation for a repository. The STPs cover potential licensing issues peculiar to sites in each of the three media under investigation for the first repository (i.e., salt, basalt and tuff). The staff expects that these STPs will be released as drafts in late 1984.

The staff has also continued its work in the area of quality assurance (QA). During 1984, the staff released a review plan—"Quality Assurance Programs for Site Characterization of High Level Nuclear Waste Repositories—which, along with other studies already under way, will inform and facilitate the staff's assessment of DOE's quality assurance program.

Site Investigations

As noted in the *1983 NRC Annual Report* (p.70), the Nuclear Waste Policy Act of 1982 (NWPA) requires DOE to issue an environmental assessment (EA) for each site that DOE intends to nominate for site characterization. DOE may issue as many as nine EAs near the end of calendar year 1984 and will allow the public 90 days to comment.

The NRC staff intends to comment on each EA and has devoted considerable effort to preparing itself for the EA review. Staff members have been organized into teams, each of which is assigned to one of the geologic media (salt, tuff and basalt) that DOE is investigating. Team members frequently visit their assigned site to review data, identify licensing issues, and consult with DOE on methods and approaches for resolving these issues. The teams meet once a week to discuss the data and technical reports they have reviewed. The staff is also preparing an

NRC geologists and other scientists have formed teams to investigate potential high-level waste sites. Each team specializes in a geologic medium (e.g., salt, tuff, basalt) and makes frequent visits to appropriate sites, usually in conjunction with experts from other Federal agencies. This September 1984 photo shows NRC, DOE and USGS staff members during a visit to the Nevada test site examining evidence of faulting in a trench cut. The gird laid out on the face of the trench aids in mapping this sedimentary rock.

EA review plan which should be completed before the EA's are released.

DOE submitted its siting guidelines, as required by § 112(a) of the NWPA, to the Commission on November 22, 1983. The Commission concurred in the guidelines on June 22, 1984.

Prior to receiving the DOE guidelines, the Commission received two related petitions for rulemaking: one from the Yakima Indian nation, and one from the States of Texas, Wisconsin, Minnesota, Nevada, and Utah. The States' petition requested that the NRC undertake a rulemaking to establish notice and comment procedures for Commission concurrence in the DOE guidelines, thus providing an additional opportunity to ensure that the States' concerns on the guidelines would be considered.

The Commission decided that concurrence procedures were not amenable to rulemaking under the Administrative Procedures Act. However, for policy reasons, the Commission held three public meetings, considered public comments on the guidelines, considered public comment on its preliminary concurrence decision, assisted DOE in revising the guidelines, and issued a final concurrence decision. The guidelines explain how DOE will select two sites for development as repositories.

Work with Other Agencies

As noted in the *1983 NRC Annual Report* (p. 72), the NRC has been working with the DOE and the EPA in the development of EPA standards for the high-level waste management program. The NRC has continued to receive technical support from the U.S Bureau of Mines and the U.S. Corps of Engineers in conducting site-specific reviews of the Basalt Waste Isolation Project (BWIP) and the Nevada Test Site (NTS). The staff also consulted with the U.S. Geological Survey regarding site characterization.

Waste Confidence Rulemaking

Throughout 1984, the NRC staff continued work on the generic rulemaking proceeding to reassess the Commission's confidence that radioactive waste produced by nuclear facilities will be safely disposed of. This determination involves a judgment as to when adequate disposal capacity will be available, and whether such wastes can be safely stored until final disposal. The Commission initiated its rulemaking proceeding on October 25, 1979 (44 *FR* 61372), and published a draft decision on the proceedings on May 16, 1983. After considering public comments on the draft decision, the Commission released a final decision on August 31, 1984 (49 *FR* 34658). In the final decision, the Commission found that there was reasonable assurance that:

1) Safe disposal of high-level radioactive waste and spent fuel in a mined geologic repository is technically feasible.
2) One or more repositories would be available by the years 2007 and that sufficient repository capacity will be available within 30 years beyond the expiration of any reactor operating license to dispose of existing commercial high-level radioactive waste and spent fuel originating in such reactors and generated up to that time.
3) The radioactive waste can be safely managed until a repository is available.
4) Spent fuel can be safely stored at the reactor for at least 30 years beyond the reactor's expiration date.
5) Safe on-site or off-site storage for spent fuel will be available, if needed.

The Commission also amended its rules, 10 CFR Parts 50 and 51, providing procedures for considering environmental effects of extended on-site storage of spent fuel in licensing proceedings.

REGULATING LOW-LEVEL WASTE

Regulatory Development

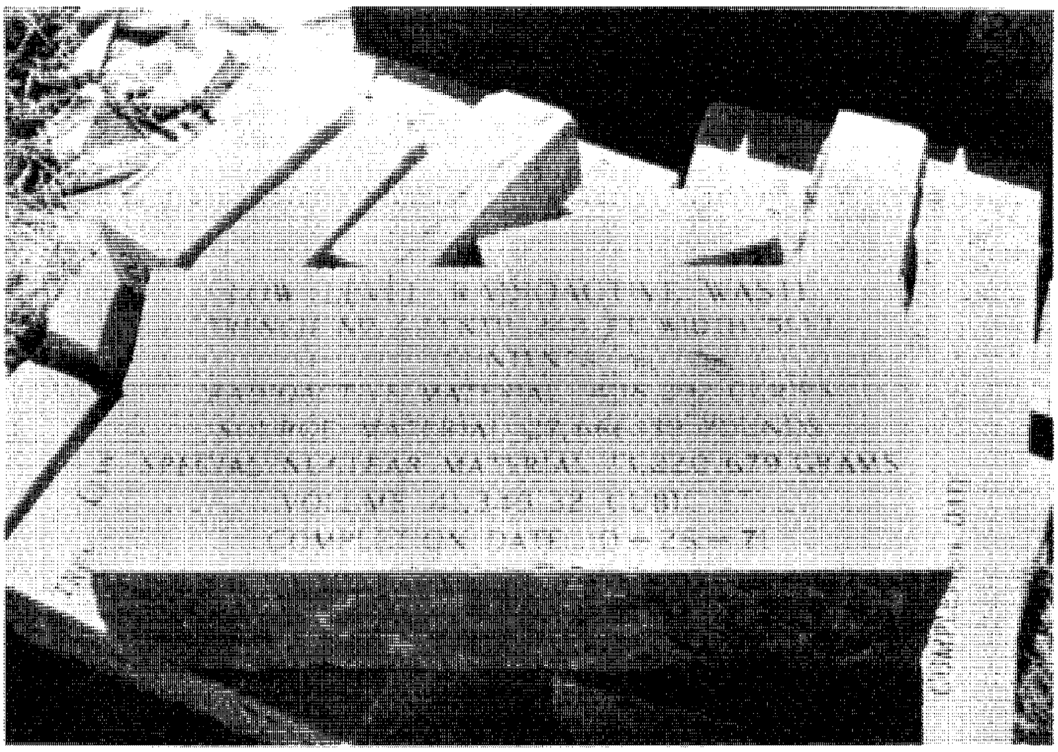
In 1983, the NRC made significant progress in the development of low-level waste regulations with the issuance of the final 10 CFR Part 61 rule, "Licensing Requirements for Land Disposal of Radioactive Waste" and related changes to the 10 CFR Part 20.311 requirements for waste classification form and manifests. (See *1983 NRC Annual Report*, p. 72, for discussion.) Throughout fiscal year 1984, considerable staff effort was given to assisting licensees in complying with these changes. The NRC also issued inspection guidance on these requirements and initiated an inspection program to confirm licensee compliance.

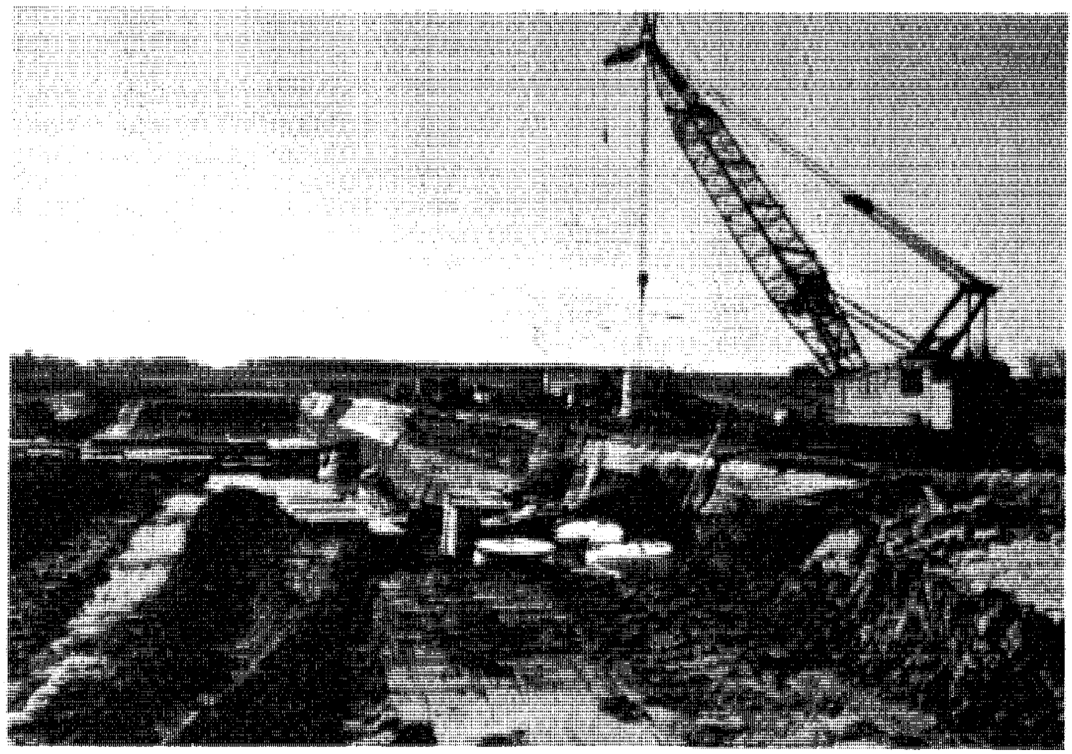
In fiscal year 1984, the NRC staff prepared two draft regulatory guides, one on waste classification, and one on waste form. It is expected that both of these guides will be released for public comment during fiscal year 1985. The NRC staff is also preparing a regulatory guide entitled "Standard Format and Content Guide for License Applications for Low-Level Waste Disposal Facilities."

Low-Level Waste Licensing

During the current report period, the NRC issued amended Special Nuclear Material (SNM) licenses for the low-level waste disposal facilities at Hanford, Wash., and Barnwell, S.C. The amended licenses implement the waste classification, waste characteristics, and waste manifest criteria of 10 CFR Part 61.

As previously reported (see *1982 NRC Annual Report* p. 82), the NRC, the State of Washington, and U.S. Ecology—the licensed operator of the low-level waste





At top is a typical granite parapegm used to mark permanently the location and contents of low-level waste trenches at the Barnwell facility in South Carolina. The lower photo shows a crane at the facility lowering a high-integrity low-level waste container into a burial trench. Barnwell received more than a million cubic feet of such waste in 1984.

disposal facility at Hanford, Wash.—have resolved the terms under which U.S. Ecology may accept SNM at Hanford. The NRC staff amended the SNM license in early 1983 and minor quantities of SNM continued to be buried at the site during 1984.

The Barnwell facility accepted 1.2 million cubic feet of low-level waste in 1984. Approximately 10 percent of the waste received was SNM.

There were no new licensing activities at the Sheffield, Ill., site. Nevertheless, the NRC continued to analyze the technical aspects of the operator's (U.S. Ecology) plans for site closure. As reported in the *1983 NRC Annual Report*, p. 72, low levels of tritium were detected off-site in January 1982 (approximately 3 per cent of the Maximum Permissible Concentration). Since that time, an interagency technical working group comprising representatives of the U.S. Geological Survey, the NRC, the State of Illinois and U.S. Ecology determined that off-site tritium levels have not increased. Additional monitoring during 1984 showed no change in tritium levels. A meeting between the technical working group and the DOE was scheduled after the close of the report period to seek at least tentative agreement on the tritium issue. The NRC staff is also working with the site owner and the DOE to examine the feasibility of site transfer, pursuant to Section 151 of the Nuclear Waste Policy Act.

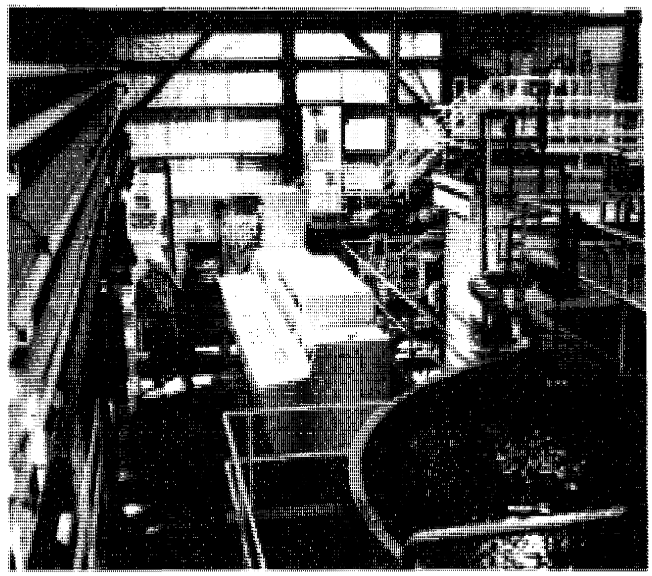
Assistance to Agreement States

Throughout 1984, the NRC continued to provide technical assistance to the NRC Agreement States (see Chapter 9). Technical assistance was given to the States of Nevada, California, Washington and New Hampshire. The staff completed one review of high integrity containers for Three Mile Island (Pa.) waste and is currently reviewing a second request from the State of Washington.

Other Activities

During 1984, the NRC worked with the U.S. Corps of Engineers in examining the technical criteria for alternatives to near surface disposal from the civil engineering perspective. The Corps is expected to provide the NRC with recommendations for potential additions to current 10 CFR Part 61 technical requirements. The Corps will also be developing suggested license application review procedures for use use by the NRC and Agreement States.

The NRC staff also reviewed 18 topical reports related to 10 CFR Part 61 requirements. These reports covered such topics as solidification agents, high integrity containers and waste classification computer codes.



This photo of the interior of the Bear Creek Uranium Mill in Converse County, Wyo., shows the central control panel (left center) and a thickener tank (foreground). Representatives of the NRC's new Uranium Recovery Field Office in Denver made some 20 inspections of mills and related facilities in 1984.

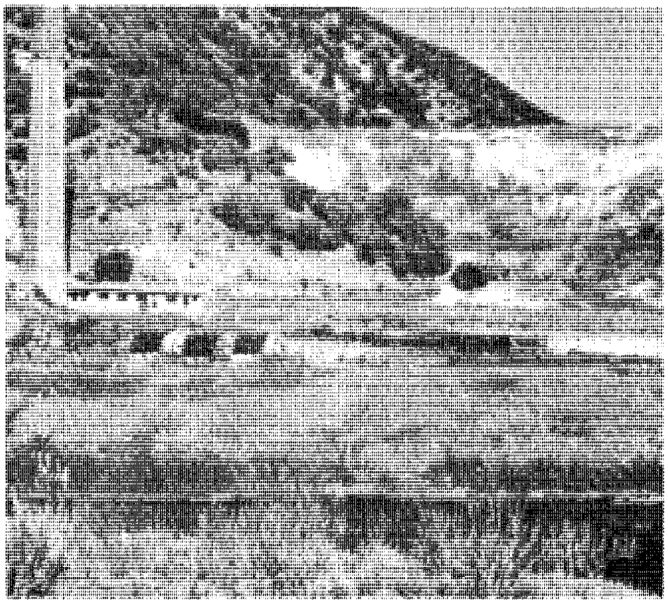
URANIUM RECOVERY AND MILL TAILINGS

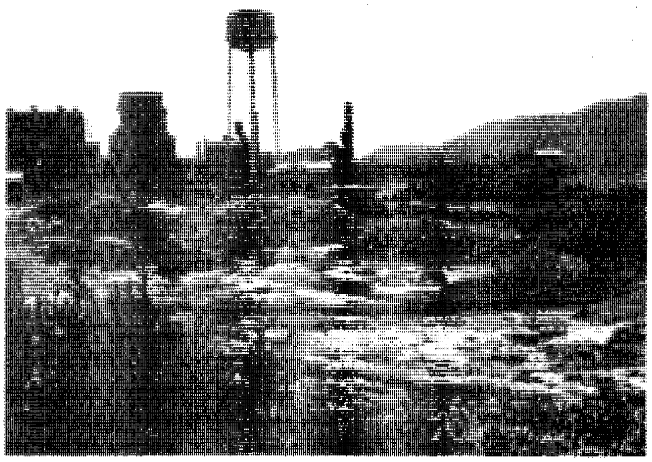
The NRC is responsible for assuring that uranium recovery facilities are constructed, operated, and decommissioned in a manner that will protect the public health and safety and the environment. The NRC Uranium Recovery Field Office (URFO) was opened in Denver, Colo., to improve the responsiveness of the NRC to problems of uranium recovery regulation in the Western States. Since October 1983, URFO has been fully operational. The office is responsible for implementing the NMSS policies for uranium recovery licensing.

Regulatory Development

Throughout 1984, the uranium recovery licensing program was affected by Congressional prohibitions on the use of certain portions of NRC's 10 CFR Part 40 regulations on source materials during the fiscal year (see *1982 NRC Annual Report*, p. 83, for background). During this time, Congress mandated that the NRC temporarily suspend portions of its regulations on uranium milling and tailings disposal until the EPA promulgated its final standards under the Uranium Mill Tailings Radiation Control Act of 1978 (UMTRCA).

The EPA issued its final standards in October 1983, which became effective for NRC and Agreement State licensees in December, 1983. Under the Authorization Act of FY 1983, the NRC must conform its rule to the EPA





The NRC, often in conjunction with other Federal agencies took a variety of actions under the Uranium Mill Trailings Remedial Action Program in 1984 and had other remedial actions under way at year's end.
At top left is the large tailings pile at a Durango, Colo., processing site, which might have to be moved because of the erosion of tailings into the Animas river (foreground).
Top right, an NRC inspector taking readings at the Pinal, Ariz., heap-leach uranium recovery site.
Lower left is a mill at Rifle, Colo., which once processed both uranium and vanadium and now does vanadium only. Cleanup of the raffinate evaporation pond (foreground) will entail stabilization of radioactive materials or, possibly, its relocation.
Lower right an NRC inspector observing maintenance of a tailings pond liner at the Sweetwater uranium mill in Wyoming.

standards. The Commission is currently considering staff proposals on the matter, and an NRC task group is working out the implications of the conformance mandate. The NRC has taken steps to implement the EPA standard in the interim until conforming and implementing rule changes are in place.

During 1984, NRC staff continued work on regulatory guides, dealing with such topics as: long-term stabilization and erosion protection, bioassay at uranium mills, tailings liner requirements, meteorological measurement programs, financial sureties, tailings pile cover material, pollution control devices, and estimating techniques for determining radioactive and toxic material releases.

Licensing and Inspection Activities

Responsibility for the inspection of uranium recovery facilities was transferred to the URFO (see above) in fiscal year 1984. Since that time, the URFO has performed or assisted in 20 inspections of uranium milling or in-situ facilities.

In 1984, the URFO licensing staff began to review one license application, completed eight license renewals, completed nine major license amendments, reviewed eight additional amendments, and reviewed approximately 205 operating facility safety and environmental data reports.

Of the 39 uranium recovery facilities licensed at the end of the report period, 14 were uranium mills, eight were heap-leach/ore buying stations or byproduct recovery facilities, 13 were research and development solution mining operations, three were commercial solution mining facilities, and one was a facility with both uranium milling and commercial solution mining activities at the same site.

Technical Assistance to Agreement States On Uranium Recovery

In 1983, the Uranium Recovery Field Office (URFO) was given responsibility for providing technical assistance to Agreement States on their licensing actions.

As part of the periodic review of Agreement State regulatory programs, NRC Headquarters and URFO staff participated in the review of the uranium recovery licensing programs of New Mexico in November 1983, and Colorado in June 1984. These reviews examined the States' programs for mills, commercial solution mining facilities, and research and development solution mining facilities.

Remedial Action at Inactive Sites

The NRC has 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 of 1978 (UMTRCA). Staff resources allocated to this program steadily increased over the past year in order to keep up with the increasing volume of DOE submissions of Environmental Impact Statements (EIS), Environmental Assessments (EA), Remedial Action Plans (RAP), Radiologic Engineering Assessments (REA) and other documents for NRC review. The NRC must review and concur in the selection and performance of remedial actions as described in these documents for 24 processing sites and possibly as many as several thousand contaminated vicinity properties.

Among the documents NRC must review and concur in are Cooperative Agreements between the Department of Energy and each State or Indian Tribe involved in remedial actions. During fiscal year 1984, NRC has reviewed and concurred in the South Dakota, Wyoming and Navajo Nation Cooperative Agreements and a major revision to the Utah Agreement which governs remedial action work at the Salt Lake City site.

There are potentially several thousand vicinity properties which will need to be decontaminated under the UMTRAP. To speed up the process, NRC staff reviewed and concurred in the basic procedural arrangements for these actions as described in "Summary Protocol for Designation and Inclusion of Vicinity Properties" and the "Vicinity Property Management and Implementation Manual." The NRC also concurred in 54 vicinity property clean-up plans for properties located in Edgemont, S.D.; Salt Lake City, Utah; and Grand Junction, Colo.

Documentation satisfying procedures specified in the National Environmental Policy Act (NEPA) is prepared for each processing site. During fiscal year 1984, the Salt Lake City EIS, and Shiprock EA were reviewed and concurred in by NRC. Draft NEPA documents reviewed during 1984 include the Grand Junction, Rifle, and Durango EIS's and the Gunnison EA

The Remedial Action Plan (RAP) for the Canonsburg, Pa., site was concurred in by NRC and remedial action work began at the site. Draft RAPs for Salt Lake City, Shiprock and Gunnison were also reviewed.

Inspection, Enforcement, Quality Assurance and Emergency Preparedness

CHAPTER 8

During fiscal year 1984, the Office of Inspection and Enforcement (IE) and the Regional Offices continued to carry out important activities in the areas of inspection, incident response, emergency preparedness, technical training, and quality assurance. Responsibility for the agency's vendor inspection program was transferred to the Office of Inspection and Enforcement, and the NRC Operations Center was upgraded and moved to a new location. The Regions continued their inspection activities and IE continued the Performance Appraisal Team (PAT) inspections, Construction Appraisal Team (CAT) inspections, and Independent Design Inspection (IDI) efforts. These subjects, and other activities, are covered in this chapter under the following major subject headings: Quality Assurance; the Inspection Programs, including vendor inspections, fuel facilities and materials licensees inspections, and reactor plant inspections; the Appraisal Programs, including PAT and CAT activities; the Enforcement Program; Incident Response facilities and activities; and Emergency Preparedness.

QUALITY ASSURANCE

QA Report To Congress

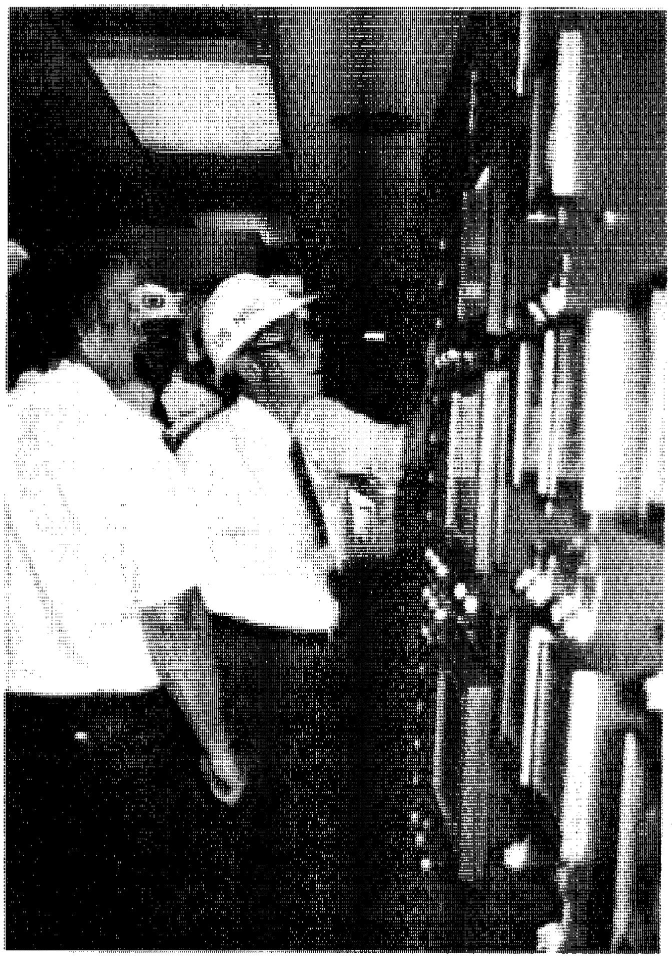
In April 1984, the NRC completed a Congressionally-mandated study of existing and alternative methods and programs for improving the quality of, and the assurance of quality in the design and construction of commercial nuclear power plants. A primary focus of the study was to determine the underlying causes of major quality-related problems in the construction of some nuclear power plants and the lack of timely detection and correction of these problems. The study concluded that the root cause for major quality-related problems was the failure or inability of some utility management to effectively implement a management system that ensured adequate control over all aspects of the project. These management shortcomings arose in part from inexperience on the part of some project teams in the construction of nuclear power plants. The NRC's past licensing and inspection practices did not adequately screen construction permit applicants for their overall capability to manage or provide effective management oversight for the construction project.

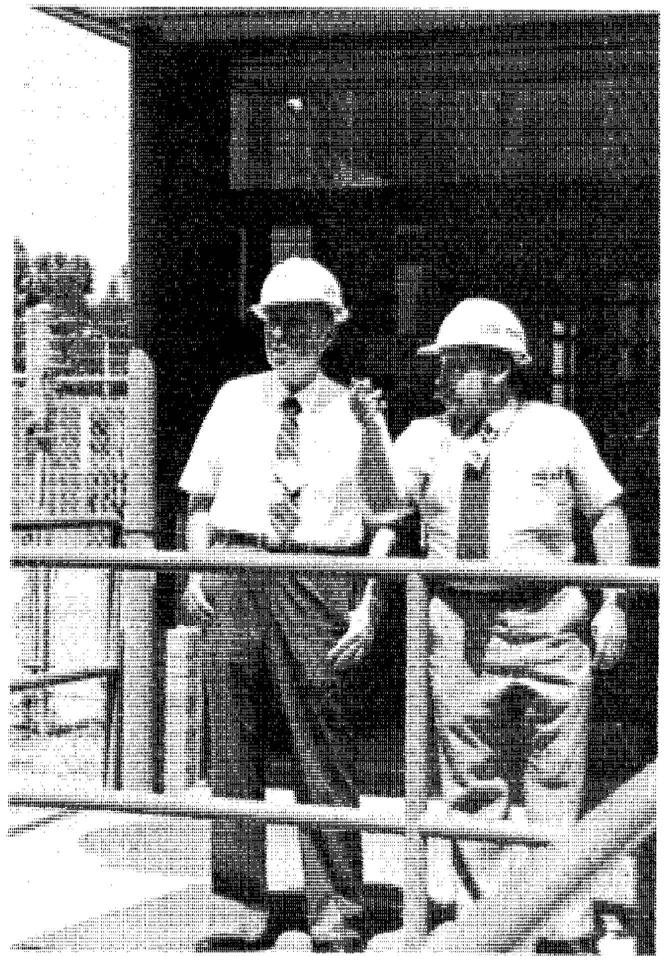
The study put forward a number of recommendations for improving both industry and NRC programs. For industry, the study recommended self-imposed, rising standards of excellence; a concept of quality assurance as a management tool, rather than as a substitute for management; improved trend analysis and identification of root causes of quality problems; and a program of comprehensive third party audits of present and future construction projects. To improve NRC programs, it recommended increased emphasis on team inspections and resident inspectors, an enhanced review of new applicants' capability to construct commercial nuclear power plants, more attention to management issues, improved diagnostic and trending capabilities, improved programs for quality and quality assurance for operating reactors, and development of guidance to give priority to quality assurance measures commensurate with the safety significance of plant structures, systems, and components. The report has been the subject of considerable attention by the public and the nuclear industry. The American Society for Quality Control convened a special topical meeting to discuss the report and its implications for the future of quality and quality assurance in the nuclear industry.

QA Program Plan

The NRC prepared a quality assurance program plan which incorporates the lessons learned in the QA study, reflects public comments on the QA Report to Congress, and includes further analysis of issues and concepts and recent input of the Advisory Committee on Reactor Safeguards on quality goals and objectives. The plan addresses all aspects of NRC's QA program, including operating reactors and reactors under construction, future reactors, and such non-reactor activities as radioactive waste storage, transportation, and nuclear fuel facilities.

The revised QA program presented in the plan consists of a number of interlocking features designed to address the root causes of problems identified in the QA Report to Congress and in ongoing NRC inspections. The main thrust of the revised program is to provide a framework in which: (1) deflections of management attention from achieving and assuring plant quality are minimized so that management can focus its energies on activities of safety significance (i.e., create a stable, predictable regulatory environment), and (2) management authorities and responsibilities are clearly defined, and management is clearly accountable for quality-related shortcomings and failures as well as successes.





While concentrated in the Office of Inspection and Enforcement and the Regional Offices, NRC inspections may involve personnel from many offices at all levels. In the photos at top, Chairman Nunzio J. Palladino is shown during a July 1984 inspection visit to the Grand Gulf nuclear plant in Mississippi. At left, Commissioner Lando W. Zech is being briefed by NRC Regional Administrator James P. O'Reilly (Region II, in Atlanta) during an inspection visit there in September 1984.

Pilot Program

One of the activities described in the QA program plan has progressed to the point that a pilot program is being planned to test the feasibility, practicality, and benefits to be achieved through its implementation. The pilot program, an operational readiness review program at a plant under construction, tests many of the key features of the QA program plan including the development of master inspection plans, readiness reviews, and incremental NRC approval of completed work. The readiness review pilot program being tested at Vogtle Unit 1 in Georgia is a formal assessment of the licensee's implementation of programs, procedures, and actions to determine: (1) the preparedness of the personnel, plant and hardware, and management systems, (2) the utility's conformance with requirements and licensee commitments, and (3) the readiness of the licensee to proceed based on the current state of implementation and performance to date. The pilot program will run for approximately two years.

Regulatory Guide Development

Appendix B of 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," sets forth the quality assurance requirements for the design, construction, and operation of those structures, systems, and components. Current guidance on controls the NRC staff considers acceptable for complying with those criteria is provided in a number of regulatory guides that endorse industry standards such as ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities." Efforts are under way to revise and reissue NRC Regulatory Guides for nuclear power plant design, construction, and operation to reflect recent industry changes and to reflect the lessons learned in the QA Report to Congress and in recent events and NRC inspections. Revisions to the Regulatory Guide for design and construction are expected to be completed in fiscal year 1985 and implemented in fiscal year 1986.

Integrated Design Inspection Program

As part of the program to improve the assurance of design quality and its implementation at nuclear power plants, the NRC has developed and implemented Integrated Design Inspections (IDIs). The IDI provides a comprehensive examination of the design development and implementation for a selected system on a given project. It encompasses the total design process from formulation of principal design and architectural criteria through the development and translation of the design and its revisions. The program includes inspection at licensee and architect-engineer offices as well as on-site verification of the design. The results of the IDI are conveyed to the appropriate Regional and Headquarters Offices and are used as input to the overall NRC assessment prior to issuance of an operating license.

IDIs have been performed to date at the Callaway (Mo.), Byron (Ill.), Seabrook (N.H.), River Bend (La.) and Perry (Ohio) Nuclear Power Plants.

Independent Design Verification Program

The Independent Design Verification Program (IDVP) was introduced into the process of reviewing nuclear power plant operating licenses after a significant problem in the design control process was discovered at Diablo Canyon (Cal.) subsequent to low power licensing. The IDVP has normally involved a review of the design process, including a sample of design details, performed by an independent contractor hired by the applicant. The IDVP often includes elements of on-site verification in selected areas. On January 1, 1984, IE assumed responsibility for the IDVP.

INSPECTION PROGRAMS

Nearly one-third of the NRC's current resources are used to develop and carry out inspection programs and procedures designed to verify and evaluate the adequacy and effectiveness of licensee compliance with NRC rules and regulations. The headquarters IE staff is charged with developing and promulgating comprehensive and uniform inspection procedures and policies, monitoring and assessing the effectiveness and uniformity of inspection programs carried out by the five NRC Regional Offices. IE also conducts inspections on a national basis as described later in this chapter. The inspection resources are focused on those licensee activities which are most significant in terms of protection of the public health and safety. The inspection program is also structured in a manner that focuses increased inspection attention outside the routine, planned program in those cases where licensee performance indicates the need for additional NRC oversight.

The majority of NRC's inspection activities are carried out by personnel located in the five Regional Offices and at reactor sites. The program basically comprises three kinds of activities. First, routine or planned inspections are conducted at all facilities in order to ensure that the safety programs established by licensees are in fact being routinely implemented and managed in a manner which will prevent a nuclear accident or unsafe condition. Second, NRC conducts reactive inspections in response to events or conditions at individual sites; here, the emphasis is placed upon determining the root cause of the condition, evaluating the adequacy of licensee management's response and long term corrective action to preclude recurrence, and generic application to other facili-

Table 1. Inspections Conducted During FY 1984

Program	*Number of Licensees Inspected*	*Number of Inspections*
Power Reactor Construction	50	1,386
Operating Power Reactors	85	2,272
Other Reactors	42	66
Fuel Facilities	295	464
Materials	1,685	1,751
Vendors	90	160
Shipments of Spent Fuel	55	475

ties. Finally, the program affords the opportunity for each inspector to spend approximately 20 percent of available inspection time in independently pursuing and evaluating licensee programs which affect nuclear safety. Notwithstanding the specific instructions contained in established inspection procedures, inspectors are encouraged to go beyond this guidance in areas where they, as professionals, believe additional information is needed. Table 1 reflects inspection activity in fiscal year 1984.

Reactor Inspection Program

The reactor inspection program is carried out by a corps of NRC resident inspectors and region-based inspection specialists. Resident inspectors are at the heart of the inspection program, with one or more assigned to every site with reactors under active construction or in operation. They live near the sites, and their offices and duty stations are on-site. While they serve in a variety of inspection functions as NRC representatives, their primary job is to observe, evaluate and report on the adequacy of licensee nuclear safety activities on a day-to-day basis. In the event of an emergency or unsafe condition, resident inspectors report to the site to assist in the collection and communication of information to NRC region and headquarters response teams. The region-based corps of inspection specialists supplement the basic activities of the resident inspectors through a variety of programmatic inspections which afford an in-depth look at licensee programs.

The operating reactor inspection program is performed by both region-based and resident inspectors. Region-based inspectors are specialists whose efforts include detailed inspections in such areas as plant operations, systems surveillance, maintenance, modifications, inservice inspection, fire protection, nondestructive testing, training, refueling, radiation protection, quality assurance, emergency planning, environmental protection, management systems, and security/safeguards. Resident inspectors are generalists who concentrate on day-to-day operations, event followup, licensee management and staff performance. Their work includes close monitoring of control room activities and of maintenance and testing carried out by the licensee, with periodic auditing of the correctness of system line-ups for nuclear systems that are important to safe operation. In addition, resident inspectors coordinate on-site activities of various NRC offices and participate in emergency exercises. Resident inspectors also serve as the NRC contact with local officials, the press, and the public.

In 1984, the NRC monitored a number of the full-scale emergency preparedness exercises. The exercises demonstrated that significant progress had been made in upgrading emergency preparedness.

The Emergency Preparedness Inspection Program is developed by IE and implemented by the Regional Offices. The program employs a standardized methodology to evaluate the adequacy and effectiveness of licensee emergency plan implementation and the overall state of emergency preparedness at each reactor facility. The program is accomplished through routine inspection and exercise observation.

Routine inspection is aimed at ensuring that adequate equipment, instrumentation, facilities, supplies, procedures and trained personnel are readily available to detect and assess an accident and its potential severity; that the licensee's emergency organization, appropriate government authorities and the general public will be notified promptly; that appropriate mitigating actions and protective measures will be taken in response to the emergency, and that regulatory requirements and licensee commitments are met.

Emergency preparedness exercises provide licensees and all emergency response organizations the opportunity to test their emergency response capabilities and identify impediments to effective response and coordination. Exercises typically involve the simulation of a major accident leading to core degradation and ultimately to a radiological release to the environment. In response to this simulated accident, the aspects of emergency response that are demonstrated are accident recognition and classification, prompt alert and notification of emergency response organizations, emergency response activation and staffing, organizational interfaces and communications, prompt decisionmaking, accident investigation and mitigation techniques, prompt notification and instruction for the general public, protective measures for emergency workers and the general public, and provisions for timely and accurate distribution of information to the news media. The NRC observes and evaluates licensee performance at these exercises. The exercises are performed initially within one year prior to operation above 5 percent rated power. The on-site portions of the exercises are carried out on an annual basis thereafter. In years when off-site participation is required, the Federal Emergency Management Agency evaluates the off-site response capability.

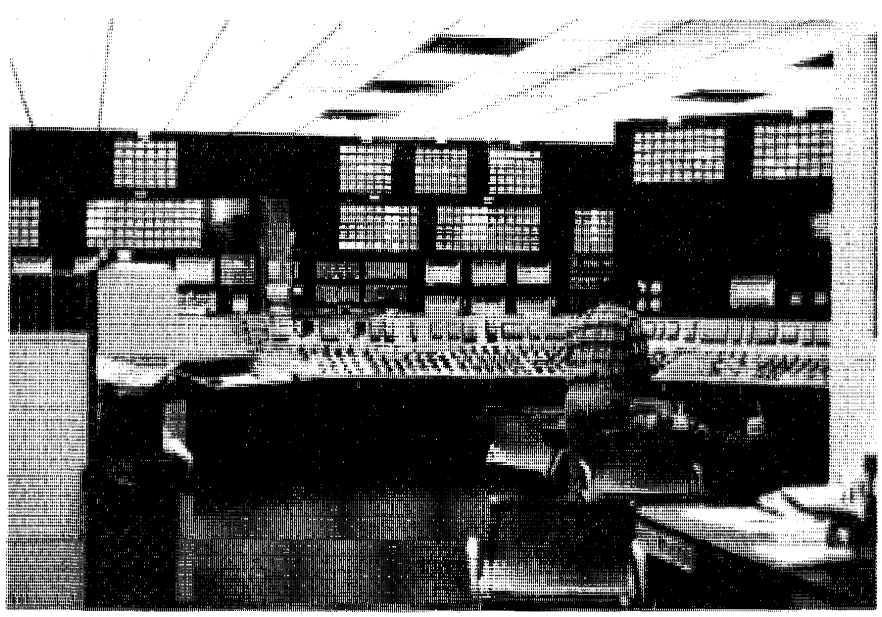
Radiological safety inspection efforts involved the implementation, beginning in January 1984, of completely revised inspection procedures that reflected lessons learned from the Health Physics Appraisals (see the *1981 NRC Annual Report*, p. 90) that were performed at operating reactors following the accident at the Three Mile Island plant in Pennsylvania.

Another NRC program is the direct radiation monitoring network. Radiation detectors, called thermoluminescent dosimeters (TLDs), have been placed in the vicinity of all operating power reactors and those nearing construction completion. The TLDs are periodically replaced and analyzed to measure radiation present at that location.

For reactors under construction, the region-based specialists and resident inspectors address such things as welding and nondestructive examination; civil, mechanical, electrical and instrumentation engineering; preoperational testing; emergency preparedness; and environmental protection. The resident inspector applies more general experience in construction activities to assure that installation of equipment and structures is accomplished in accordance with design and quality assurance requirements. The resident inspector has frequent contact with construction management personnel from the utility, architect-engineer, constructor, vendors, and contractors. He reviews procedures, observes the work, and audits quality control. He may also participate in NRC hearings, licensing meetings and public discussions.

Supporting the region-based and resident inspectors, NRC maintains a specially equipped mobile nondestructive examination laboratory at its Region I (Philadelphia) office.

While NRC inspection programs for reactor construction and operations cover the spectrum of activities that are important to nuclear safety, available resources permit only a limited sampling of licensee activities to be examined in each functional area reviewed. When deficiencies are identified through the inspection program, the NRC expects licensees to examine the deficiency in terms of all of their activities to determine whether or not an isolated deficiency is symptomatic of a more widespread problem. Followup inspections by the NRC inspectors are designed to determine the adequacy of the licensee management program in this regard. Table 1 shows the number and types of licensees inspected and the number of inspections performed during fiscal year 1984.



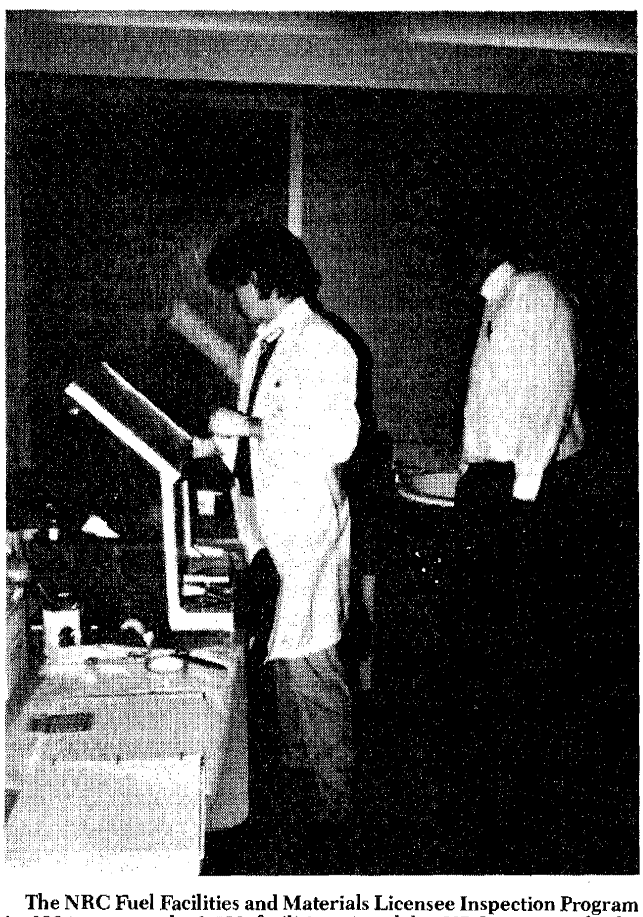
NRC emphasis on complete emergency preparedness programs at each licensed nuclear power facility has resulted in extensive training with elaborate equipment. This control room simulator is part of the emergency crisis center at the Oconee nuclear station at Seneca, S.C.

During fiscal year 1984, special task forces investigated allegations of poor workmanship and inadequate quality assurance programs at the Waterford and Comanche Peak construction sites. Extensive follow-up was performed relative to the concerns of the allegers, including review of the utilities' corrective actions and independent inspection of installed systems and components.

Vendor Inspection Program

The Vendor Inspection Program covers inspection of the non-licensed organizations that provide products and services for licensed activities to ensure they meet applicable industry and NRC requirements. These non-licensed organizations include nuclear steam system supply and architect engineer firms, suppliers of products and/or services, testing laboratories and facilities performing equipment qualification tests, and third party inspection organizations performing activities associated with reactor licensees.

These inspections have proved to be an efficient way to assess the quality assurance programs of vendors and also to assure that the generic aspects of discovered deficiencies are examined by the NRC.



The NRC Fuel Facilities and Materials Licensee Inspection Program in 1984 saw nearly 2,000 facilities visited by NRC personnel who conducted a total of 2,215 such inspections. The photo shows NRC radiation specialist John Miller observing a nuclear medical technician during a dose-calibrator assay of radiopharmaceuticals.

In January 1984, the Vendor Program Branch was transferred from the Region IV office in Arlington, Tex., to the Office of Inspection and Enforcement. The transfer was made to increase the efficiency of this nation-wide program from a central headquarters perspective and to enhance the implementation of Commission-level policy and guidance on vendor-related issues. The transfer will also promote increased interaction between the vendor inspection program and other headquarters programs on significant reactor safety issues.

Fuel Facilities and Materials Licensees Inspection Program

The fuel facilities and materials licensees inspection program covers all safety and safeguards-related activities at licensed fuel facilities--uranium mills, uranium conversion facilities, fuel production plants, and materials licensees' activities such as nuclear medicine, radiography, industrial testing, well-logging and academic and other purposes, including handling and storage of radioactive wastes. Through State agreements, the NRC has delegated similar responsibility to the States (see Chapter 9, "Cooperation with the States.") The program also involves inspections of nuclear fuel shipments, and shipments of other radioactive materials, as well as inspections of nuclear material exported from or imported into the United States (See Chapter 10, "International Cooperation").

During 1984, the materials inspection program included routine inspections and special inspections concerning incidents and allegations. Examples of incidents include the spread of cesium-137 contamination to a number of homes in Hebron, Ohio as a result of a ruptured well-logging source, and 13 therapeutic medical misadministrations, primarily from teletherapy radiation using cobalt-60 for cancer treatment. Other special inspections were conducted in connection with enforcement cases involving issuance of orders. Inspections were conducted to determine whether the affected licensee was complying with the provisions of the order. Overexposures of workers occurred in a few cases during the performance of radiography and well-logging. None of these cases involved an observable physical injury as a result of the radiation exposure.

During 1984, the fuel facility inspection program was completed on a routine schedule. Upgrading of radiological safety inspection procedures was completed during the report period. The fuel facility safeguards inspection program manual chapters were revised and consolidated, and most of the associated inspection procedures were substantially modified. The resultant new manual chapter provides updated and expanded overall program guidance for safeguards inspections.

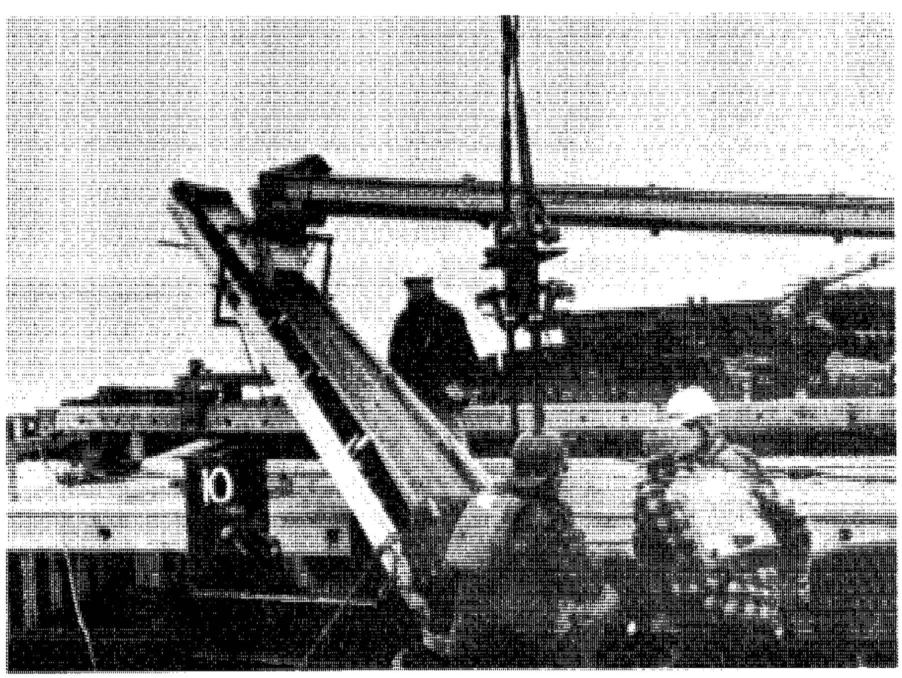
The number of operating fuel facilities remained the same except for a few uranium milling operations that were placed on standby basis. Facilities on standby basis and those being decommissioned continue to be inspected, with a level of effort appropriate to the status of the facility.

In late 1983, 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," became fully effective (see *1983 NRC Annual Report*, p. 72). These requirements involve related changes to 10 CFR 20.311, applicable to the generators of low-level waste to be transferred to licensed land disposal sites. Essentially, through the waste manifest requirements of 10 CFR 20.311, waste generators are required to classify their waste and determine certain characteristics of the waste form. The waste classification system establishes three categories for waste acceptable for near-surface burial. This classification system is based on the concentrations of radionuclides important to disposal. Under this system, wastes having greater radiologic hazards are required to be disposed of with greater protection. The classification system also is used as a basis for determining appropriate waste form requirements.

In early 1984, the Regional Offices incorporated into their routine inspections of waste generators the verification of these 10 CFR Part 61 requirements. The Office of Nuclear Materials Safety and Safeguards issued two Branch Technical Positions which provide guidance to licensees on acceptable methods for demonstrating compliance with the waste classification and waste form requirements of 10 CFR Part 61. Thus far, the inspections of waste generators during 1984 have revealed no major instances of noncompliance with the new rule. Reasonable efforts are being made by licensees to properly classify and characterize process waste streams, to properly label waste packages, and to prepare and track waste shipments.

During the year, the Regional Office inspection staff continued routine inspection of highway shipments of spent fuel from the General Electric storage facility in Morris, Ill., to the Point Beach Nuclear Station in Wisconsin, and similar shipments from the DOE West Valley facility in New York to the Point Beach facility and to the Dresden station in Illinois. (See *1983 NRC Annual Report*, p. 60). Also in 1984, movements of spent fuel by highway commenced from West Valley to the Oyster Creek facility in New Jersey and rail shipments commenced from the Cooper Power Station in Nebraska to the G.E. Morris facility. Region I inspectors routinely inspect, on a sampling basis, the shipments originating from West Valley. Region III inspectors inspect shipments, on a sampling basis, both at origin in Morris and at destination in Point Beach and Morris. During these inspections, the adequacy of compliance with both safety (packaging) and security (safeguards) requirements is verified.

Since the inception of the increased number of spent fuel shipments in mid-1983, no major problems or noncompliance have been noted by NRC inspectors, although public and media interest in the shipments remains high. Based on this favorable inspection experience, the Regional Offices have been able, in the past year, to scale down the inspection frequency from the initial 100 percent coverage to a more modest sampling frequency. The Region III staff conducted training in late 1983 for members of the Wisconsin Health Department to enable that group to carry out inspections of spent fuel casks being shipped from the Point Beach location.



The systematic assessment of licensee performance carried out by the NRC includes the work of both Performance Appraisal Teams (PAT) and Construction Appraisal Teams (CAT) to supplement normal regulatory programs to monitor plant construction and operation. A CAT inspector, as shown, is typically concerned with actual construction activity and his assessments may lead to improvements in NRC regional and headquarters programs.

APPRAISAL PROGRAMS

Systematic Assessment of Licensee Performance

The program for Systematic Assessment of Licensee Performance (SALP) is an integrated NRC effort to collect available observations on a periodic basis and evaluate the performance of each nuclear power facility in construction and operation based on those observations. The SALP process is a comprehensive review of the manner in which licensee management directs, guides and provides resources for assuring plant safety. The goal of a SALP review is to direct NRC and licensee attention toward areas affecting nuclear safety that need improvement.

Part of the input to a SALP assessment consists of the past year's Licensee Event Reports, inspection reports, enforcement history and licensing issues. Another important input consists of evaluations by resident inspectors, licensing project manager and senior regional managers, all of whom are familiar with the facility's performance. No new data are specifically obtained as an input to a SALP assessment.

The product of a SALP assessment consists of performance evaluations in a number of functional areas such as plant operations, maintenance, surveillance, emergency preparedness, security and licensing issues.

The SALP program supplements the normal regulatory processes and is intended to be sufficiently diagnostic to provide meaningful guidance to utility management as to NRC concerns regarding quality and safety in plant construction and operation. The results of the program are also used by regional managers to allocate inspection resources.

Appraisal Teams

The Performance Appraisal Team (PAT) is a group of experienced inspectors who conduct comprehensive inspections of operating reactor facilities' management control systems and related performance to determine their adequacy. The team focuses on such selected areas of plant activities as operations, maintenance, design change and modification, and training. The PAT inspections of operating reactors provide an independent check on regional inspection effectiveness, assess the adequacy of headquarters program guidance, and judge the effectiveness of the nuclear industry's Institute of Nuclear Power Operations (INPO). Three PAT inspections were conducted during 1984. Members of the PAT also accompanied INPO personnel during plant and corporate evaluations, and several meetings were held to keep NRC abreast of INPO activities.

In 1984, the Construction Appraisal Team (CAT) inspection program, was continued, and the goal of conducting four CAT inspections per year was achieved. The primary purpose of the CAT inspections is to evaluate the design controls, construction practices, and as-built conditions at nuclear plants under construction. The CAT also assesses Regional Office implementation of the IE inspection program and monitors the progress of the INPO construction project evaluation program.

The CAT inspection findings included fabrication, installation and testing deficiencies. Followup of the corrective actions are accomplished by Regional Offices with assistance, as requested, from IE.

THE ENFORCEMENT PROGRAM

The purpose of the NRC's enforcement program is to protect public health and safety by ensuring that licensees comply with regulatory requirements. The program is carried out under the revised enforcement policy published this year (10 CFR Part 2, Appendix C, 49 FR 8583, March 8, 1984). The policy calls for strong enforcement measures to encourage compliance and prohibits operations by licensees who fail to achieve adequate levels of protection.

The NRC uses three types of enforcement actions, described in detail in earlier annual reports (see the *1980 NRC Annual Report,* p. 144). Generally, Notices of Violations are issued for all instances of noncompliance with NRC requirements. Civil penalties are issued in cases of significant or repetitive noncompliance or when a Notice of Violation has not been effective. Orders to cease and desist operations, or to suspend, modify or revoke licenses are issued in extremely serious cases.

Certain headquarters enforcement functions have been regionalized. The regional administrators have always been authorized to issue Notices of Violation not involving civil penalties. They are also authorized to issue proposed civil penalties, with the concurrence of the Director of the Office of Inspection and Enforcement. The Director, however, is responsible for all enforcement decisions and issues orders, including those imposing or proposing civil penalties.

Table 2 provides a listing and brief summary of the 73 civil penalty actions during fiscal year 1984. The proposed penalties totalled over $2.3 million. With some cases still pending and some of the penalties remitted or mitigated, a total of $1,501,675 in penalties had been collected at the close of the report period. Some of these were civil penalties proposed in fiscal year 1983.

Table 3 provides a description of the 19 enforcement orders issued during fiscal year 1984.

Table 2. Civil Penalty Actions During FY 1984

Licensee	*Amount*	*Reason*
GPU Nuclear Corporation (Three Mile Island) EA 82-124	$140,000 proposed in FY 83; $40,000 paid in FY 84; $100,000 Pending	Inadequacies in reactor operator retraining program. Material false statement (MFS) in licensed operator application for recertification. The maximum civil penalty was imposed for the MFS because it was submitted willfully.
Arizona Public Service Company (Palo Verde) EA 83-30 & 83-130	$80,000 proposed for two violations in FY 84. $20,000 imposed and paid in FY 84 for one violation. The other violation is under review. Pending	Failure to control construction quality assurance for records and correcting deficiencies. Civil penalty for this violation was mitigated for prompt and extensive corrective action. Failure to follow procedures involving electrical terminations.
Baltimore Gas and Electric Co. (Calvert Cliffs) EA 83-58	$60,000 proposed and paid in FY 84	Two violations of technical specification limiting conditions for operation. One concerned the inoperability of both emergency core cooling system pump room air coolers. The other violation involved a diesel generator that stopped running during a surveillance test because of a lack of fuel. The second violation was mitigated 50% because of extensive corrective action.
Dairyland Power Coop. (La Crosse) EA 83-61	$40,000 proposed in FY 83; $10,000 imposed and paid in FY 84	Violation of technical specification limiting condition for operation involving inoperability of safety-related equipment when a containment pressure sensing line was capped. The violation was mitigated after consideration of the duration of the violation, size of the facility, and the enforcement actions that resulted from similar violations at other facilities.
Texas utilities Generating Co. (Comanche Peak) EA 83-64	$40,000 proposed in FY 83; Pending	Discrimination against a member of the Quality Assurance/Quality Control organization.
U.S. Testing Company Hoboken, NJ EA 83-81	$8,000 proposed, imposed and paid in FY 84	Violations based on a Severity Level I overexposure event that occurred during licensed radiographic activities conducted by the licensee.
Niagara Mohawk Power Corp. (Nine Mile Point) EA 83-84	$40,000 proposed and paid in FY 84	Violation based on the licensee's failure to place a main steam line high radiation trip system in a tripped condition as required by a technical specification limiting condition for operation once sufficient information existed to indicate that both channels in that system were inoperable.
Portland General Electric Co. (Trojan) EA 83-85	$100,000 proposed in FY 83; $50,000 imposed and paid in FY 84	Failure to comply with several fire protection requirements relating to separation of redundant trains of equipment. Civil Penalty was mitigated for prompt and extensive corrective action.
Georgia Power Company (Hatch) EA 83-86	$100,000 proposed and paid in FY 84	Violations involved improper reactor shutdown which resulted in an unanalyzed control rod configuration. The penalty was escalated because of the seriousness of the event.
Carolina Power and Light Co. (Brunswick) EA 83-88	$40,000 proposed and paid in FY 84	One violation involved fire protection requirements. Another involved a material false statement, but no civil penalty.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
University of Virginia Charlottesville, VA EA 83-90	$1,000 proposed and paid in FY 84	Violations based on the licensee's alteration of the core configuration without making the required control rod worth measurements and subsequent operation of the reactor without the required minimum reactor shutdown margin.
Carolina Power & Light Co. (Robinson) EA 83-94	$20,000 proposed and paid in FY 84	Failure to control personnel access into a vital area from inside the protected area. The civil penalty was mitigated due to prompt response and extensive corrective action.
Terre Haute Regional Hospital Terre Haute, IN EA 83-95	$2,500 proposed, imposed and paid in FY 84	Programmatic breakdown as indicated by twelve violations, five of which were similar to previous violations. These similar violations were use of byproduct material by unauthorized individuals, failure to leak test sealed sources at required intervals, failure to provide personnel monitoring devices, failure to calibrate survey meters at required intervals, and failure to post required documents.
Brigham and Women's Hospital Boston, MA EA 83-97	$1,875 proposed and paid in FY 84	Failure to comply with radiation level limits for shipping packages and failure to follow DOT regulations. The civil penalty was increased due to multiple examples of the violation.
Professional Service Industries Oak Brook, IL EA 83-102	$2,000 proposed, imposed and paid in FY 84	Programmatic breakdown as indicated by eight violations involving loss of control of licensed material, use of licensed material by a technically unqualified employee, failure to issue a film badge to an employee using licensed material, failure to conduct semiannual inventories of sealed sources, two whole body overexposures to employees, failure to report overexposures to the NRC, failure to use shipping containers and proper shipping papers, and possessing unauthorized sealed sources.
Commonwealth Edison Company (Dresden) EA 83-103	$50,000 proposed, imposed and paid in FY 84	Failure to classify the torus-to-drywell vacuum breaker shaft seals as "Q"-items. Installed non "Q" seals resulted in seal leakage that failed a primary containment leak test. The civil penalty was increased due to lack of prompt and complete corrective actions.
Union Carbide Corporation Grand Junction, CO EA 83-108	$4,000 proposed, imposed and paid in FY 84	Programmatic breakdown as indicated by eight violations involving failure to perform daily tests of an audible alarm system, failure to calibrate radon and portable survey instrumentation, failure to establish and approve various written radiation safety procedures, failure to survey for fixed alpha contamination in office areas, failure to submit plan for seepage and surface water collection, failure to perform monthly ultrasonic testing of tailing lines, failure to perform stock sampling each quarter and to perform semiannual audits of the environmental program, and failure to document results and remedial actions associated with fire drills.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
Arkansas Power and Light Company (ANO) EA 83-117	$40,000 proposed and paid in FY 84	Failure of the 125-volt DC battery system to comply with technical specification operability requirements.
Lehigh Testing Laboratory, Inc. W. Boylston, MA EA 83-121	$6,400 proposed and paid in FY 84	Programmatic breakdown as indicated by numerous violations involving failure to provide adequate training, failure to adequately control licensed material, failure to control personnel exposures, and failure to maintain required records.
Northern States Power (Monticello) EA 83-125	$2,500 proposed and paid in FY 84	Failure to properly package radioactive material as required by the Department of Transportation regulations. The shipment had external radiation levels in excess of regulatory requirements when it arrived at the South Carolina burial site.
Southern California Edison Co. (San Onofre) EA 83-126	$40,000 proposed and paid in FY 84	Violations involving failure to meet a technical specification limiting condition for operation by isolating both charging pumps.
Automation Industries, Inc. Danbury, CT EA 83-128	$5,625 proposed in FY 84; Pending	Violations involving shipments of licensed material with surface radiation levels in excess of regulatory limits.
Texas Utilities Generating Co. (Comanche Peak) EA 83-132	$40,000 proposed FY 84; Pending	Violations regulated to intimidation of QA inspectors. The civil penalty was mitigated due to prompt and extensive corrective action.
Mississippi Power and Light Co. (Grand Gulf) EA 83-133	$12,000 proposed and paid in FY 84	Failure to control temporary alterations to equipment and failure to follow approved procedures.
Commonwealth Edison Co. (LaSalle) EA 83-134	$10,000 proposed and paid FY 84	Failure to control personnel access into a vital area from within the protected area. The civil penalty was mitigated due to prompt identification and reporting, and prompt and extensive corrective action.
Niagara Mohawk Power Corporation (Nine Mile Point) EA 83-137	$180,000 proposed and paid in FY 84	The Unit 1 violations involved failure to maintain containment integrity for approximately 3½ months in violation of technical specifications and failure to properly conduct surveillance tests on an isolation valve required by technical specifications for five operating cycles, a period of approximately 10 years. The Unit 2 violations involved construction problems related to ASME code radiographs and related violations in the plant quality assurance program. The penalty was increased because of the many examples of Quality Assurance program violations. Orders were issued in conjunction with the penalty.
Florida Power & Light Co. (Turkey Point) EA 83-138	$40,000 proposed in FY 84; Pending	Failure to have procedures which adequately implement the plant's technical specifications for entry into a locked high radiation area.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
Consolidated Edison Co. (Indian Point) EA 83-139	$40,000 proposed and paid in FY 84	Violation of technical specification limiting condition for operation involving an engineered safety features system. The containment spray system was inoperable for approximately 1 month.
GPU Nuclear Corporation (Three Mile Island) EA 83-140	$40,000 proposed, imposed and paid in FY 84	Violations involved two examples of nonautomatic containment isolation valves being left open, four instances in which procedures important to safety were not followed, failure to properly classify an event in accordance with the Emergency Plan Implementing Procedure, improper revision of a procedure used to check injection of radioactive tracer gas into the reactor coolant system and failure to complete a required report or to notify the NRC of an unplanned release.
Tennessee Valley Authority (Browns Ferry) EA 83-142	$40,000 proposed and paid FY 84	Failure to control access into a vital area due to a guard leaving his post.
Pennsylvania Power & light Co. (Susquehanna) EA 84-5	$75,000 proposed and paid FY 84	Violation involving the inoperability of a source range monitor during initial fuel loading and the movement of control rods in the Unit 2 reactor vessel.
Pittsburgh Testing Laboratory Pittsburgh, PA EA 84-6	$8,000 proposed and paid in FY 84	Failure to equip two radiography rooms with audible and visible alarms as required. In one of these rooms an employee received an exposure of 3400 rems to his thumb from an x-ray device that is regulated by the Commonwealth of Pennsylvania.
Energy Fuels Nuclear, Inc. Denver, CO EA 84-8	$6,000 proposed and paid in FY 84	Programmatic breakdown as indicated by numerous violations involving a radiological occurrence in which a potential existed for radiation exposures in excess of NRC regulatory limits.
Iowa Electric Company (Duane Arnold) EA 84-9	$20,000 proposed, imposed and paid FY 84	Failure to control access to a vital area.
Veterans Administration Hospital Indianapolis, IN EA 84-10	$2,000 proposed and $1,833 imposed in FY 84; Pending	Programmatic breakdown in the licensee's radiation safety program. After review of the licensee's response, one violation was remitted, and the civil penalty was imposed.
Lixiscope of America, Inc. Northbrook, IL EA 84-11	$2,500 proposed and imposed FY 84; Pending	Violations involved unauthorized transfer of licensed material, failure to perform required leak tests, failure to perform safety reviews, and failure to maintain records of the receipt and transfer of licensed material.
Gearhart Industries, Inc. Ft. Worth, TX EA 84-12	$3,000 proposed and paid in FY 84	Violations involving the loss of a well logging source.
Carolina Power and Light Co. (H.B. Robinson) EA 84-13	$30,000 proposed and mitigated completely in FY 84	Failure to follow procedures that implement the licensee's technical specifications for entry into a locked high radiation area. The violation was mitigated for unusually prompt and extensive corrective action taken.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
Edlow International Company Washington, DC EA 84-17	$1,600 proposed and paid FY 84	Failure to control access to SNM of low strategic significance and failure to properly maintain the sprinkler system.
Inspection and Testing, Inc. Chubbuck, ID EA 84-18	$4,800 proposed and $1,000 imposed in FY 84; Pending	Violations included failure of licensee personnel to perform adequate radiation surveys after each radiographic source had been returned to the fully retracted and shielded position. As a result, a radiographer received a whole body radiation dose of 8.2 rems.
Prillaman & Pace, Inc. Martinsville, VA EA 84-19	$1,000 proposed, imposed and paid in FY 84	Inadequate management of the licensed program by persons who were unfamiliar with NRC requirements and provisions of the NRC license.
U.S. Testing Company, Inc. Hoboken, NJ EA 84-20	$10,000 proposed, imposed and paid in FY 84	Violations of NRC requirements associated with an exposure in excess of regulatory limits to the hand of a U.S. Testing employee during the performance of licensed activities. The penalty was increased because of similar violations in 1983.
Nuclear Fuel Services, Inc. Rockville, MD EA 84-22	$18,750 proposed and paid in FY 84	Failures to comply with NRC requirements for the handling of special nuclear material in the high-enriched uranium processing and storage area. The penalty was increased because of multiple violations.
Commonwealth Edison Co. (Dresden) EA 84-24	$140,000 proposed FY 84; Pending	Failure to use effective compensatory measures to control access into the protected area, and actions of licensee and contractor personnel in deliberately circumventing plant procedures in an attempt to expedite repair activities on the radwaste solidification system. The civil penalty was increased due to careless disregard for requirements evidenced by the failure of supervisory employees to notify the security organization of the violation.
Tennessee Valley Authority (Browns Ferry) EA 84-25	$120,000 proposed and paid in FY 84	Violations involved failures to promptly identify and correct conditions adverse to quality, failure to make required reports to the NRC, and failure to perform a functional surveillance test as required by plant technical specifications.
Geo-Cim, Inc. Hato Rey, PR EA 84-27	$800 proposed and paid in FY 84	Violations involved the use of two soil density gauges containing byproduct material after expiration of the license.
Alaska Welding Center Fairbanks, AL EA 84-28	$2,000 proposed and paid in FY 84	Violations involved a radiation overexposure of an employee while performing radiographic operations at Prudhoe Bay. Other violations included a deficiency in the training of a radiographer and lack of control over a field operation. The penalty was mitigated by 50% for prior good enforcement history.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
Boston Edison Company (Pilgrim) EA 84-29	$40,000 proposed and paid in FY 84	Violations involved an incident in which a worker received an unplanned occupational radiation exposure of approximately 4.5 rems to his hand. Although the exposure was not in excess of regulatory limits, a substantial potential for such an exposure did exist.
Retina Foundation Boston, MA EA 84-30	$2,000 proposed and paid in FY 84	Violations involved a failure to make surveys to determine that individuals who handled significant quantities of iodine-125 were not exposed to airborne concentrations exceeding the limits specified, failure to limit the use of licensed material to named individuals, and failure to use licensed material in accordance with statements, representations and procedures contained in the license applications.
Tennessee Valley Authority (Browns Ferry) EA 84-32	$60,000 proposed and paid in FY 84	Violations involved improper reactor shutdown in violation of the technical specifications and station procedures. The penalty was increased by 50% because the licensee failed to take preventive steps suggested by an information Notice and because of the lack of adequate long-term corrective action.
Southern California Edison Co. (San Onofre) EA 84-34	$250,000 proposed; $125,000 imposed in FY 84; Paid in FY 85	Violations involved exceeding a technical specification limiting condition for operation requirement involving an Engineered Safety Feature System.
Duke Power Company (McGuire) EA 84-37	$40,000 proposed and imposed in FY 84; Paid in FY 85	Violation involved a failure to implement adequate independent verification which resulted in a mispositioned valve.
Philadelphia Electric Co. (Peach Bottom) EA 84-39	$30,000 proposed and paid in FY 84	Violations involved several instances of technical specification violations. The violation was mitigated because of the unusually prompt and extensive corrective actions taken.
Florida Power and Light Co. (Turkey Point) EA 84-41	$150,000 proposed and paid in FY 84	Violations involved inoperability of the auxiliary feedwater system, numerous examples of failures to follow procedures, and failure to conduct an adequate review of a design change that led to the degradation of electrical equipment.
Pacific Gas and Electric Co. (Diablo Canyon) EA 84-42	$50,000 proposed and paid in FY 84	Violation involved the inoperability of an emergency core cooling system for a period in excess of 15 hours. The boron injection tank inlet and outlet valves were closed, which blocked the flow path of the emergency core cooling system between both charging pumps and the reactor primary cooling system.
Triad Engineering Consultants, Inc Morgantown, WV EA 84-43	$250.00 proposed and paid in FY 84	Violation involved the unauthorized use of moisture-density gauges by several university students who had not been trained in accordance with specifications contained in the NRC license. The penalty was mitigated by 50% because of the licensee's prompt and effective corrective action.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
University of Pennsylvania Philadelphia, PA EA 84-50	$4,000 proposed and imposed in FY 84; paid in FY 85	Violations involved a programmatic breakdown in management oversight and control of the radiation safety program as evidenced by an exposure to a licensee employee in excess of regulatory limits, failure to maintain control of licensed material, failure to perform thyroid bioassays, failure to use syringe shields, failure to perform adequate evaluations of airborne effluents, and excessive radiation levels in unrestricted areas.
Virginia Electric and Power Co. (Surry) EA 84-52	$40,000 proposed and paid in FY 84	Failure to implement an adequate snubber service life program as required by technical specifications.
Caribe Shell and Tube, Inc. Ponce, PR EA 84-56	$1,000 proposed and paid in FY 84	Failure to assure adequate management oversight and control of the radiation safety program resulting in unnecessary radiation exposure to licensee employees and members of the public. The civil penalty was mitigated because of the small size of the licensee's operation relative to most radiography licensees and because it is not the NRC's intention that the economic impact of the civil penalty be such that it puts a licensee out of business.
Georgia Power Company (Hatch) EA 84-59	$60,000 proposed FY 84; Pending	Failure to adequately control access to the protected area. The civil penalty was increased because the violation represented a second failure to control access at the plant within the past year.
Nuclear Fuel Services, Inc. Erwin, TN EA 84-60	$100,000 proposed FY 84; Pending	Failure to maintain Material Access Area barriers in an effective and reliable condition. The civil penalty was increased because of multiple examples of the violation.
Mid-states Logging and Perforating Co. Fairfield, IL EA 84-61	$500 proposed and paid in FY 84	Programmatic breakdown as indicated by ten violations which involved licensed material being used and stored in an unauthorized location, unauthorized personnel using licensed material, inadequate records, failure to perform monthly vehicle surveys or quarterly storage area surveys, failure to properly post radiation areas, and job log sheets not being maintained.
International Wireline Services Newton, IL EA 84-62	$500 proposed and paid in FY 84	Programmatic breakdown as indicated by eight violations including failure to block, brace and/or secure radioactive packages during transportation, failure to prepare shipping papers, failure to label a radioactive material package, failure to post a radiation area, failure to perform leak tests, permitting unauthorized employees to use licensed material, operating without a Radiation Safety Officer, and storing licensed material in an unauthorized storage area.
Community Hospital of Anderson Anderson, IN EA 84-65	$4,000 proposed in FY 84; Pending	Violation involved the licensee's failure to implement effective management control over the radiation safety program and the falsification of records that NRC requires to be maintained.

Table 2. Civil Penalty Actions During FY 1984
(continued)

Licensee	*Amount*	*Reason*
Arkansas Power and Light Company (ANO) EA 84-66	$40,000 proposed in FY 84; Pending	Failure to conduct an adequate quality assurance program relating to receipt inspections involving procurement of fasteners to ASME code requirements.
Syncor International Corp. Sylmar, CA EA 84-73	$8,500 proposed in FY 84; Pending	Distribution of radiopharmaceuticals contaminated with molybdenum-99 resulting in at least sixteen patients receiving contaminated doses of technetium-99m in excess of regulatory limits.
Kraft, Incorporated Glenview, IL EA 84-74	$500 proposed and paid FY 84	Storage of licensed materials in an unrestricted area, removal from storage by unauthorized individuals, failure to monitor laboratory areas, and use of licensed materials without the approval of the Radiation Safety Committee.
Reich Geo-Physical, Inc. Billings, MO EA 84-78	$1,600 proposed FY 84; Pending	Violations involving use of unauthorized material and failure to calibrate survey meters at the required intervals.The penalty was increased after considering the licensee's poor enforcement history and the length of time the violations were allowed to continue.
Miami Fort Station Cincinnati, OH EA 84-79	$500 proposed and paid in FY 84	Violations including unauthorized individuals removing an Ohmart Model SHRM-PA source holder containing a 10 millicurie cesium-137 sealed source and storing it in an unrestricted area without having secured it against unauthorized removal.
University of Connecticut Storrs, CT EA 84-80	$2,500 proposed FY 84; Pending	Violations involving failure to properly secure licensed materials. This violation was previously identified in an inspection and corrective actions were not sufficient to preclude its recurrence.
Union Carbide Corporation Grand Junction, CO EA 84-84	$5,000 proposed FY 84; Pending	Multiple violations representing a breakdown in management oversight and control of licensed activities. Although the civil penalty could have been increased due to two previous, similar violations, the licensee's prompt and extensive corrective actions were balanced against the potential increases and offset them.
Kansas Gas & Electric Co. (Wolf Creek) EA 84-87	$64,000 proposed FY 84; Pending	Discrimination against a member of the Quality Assurance/Quality control organization.
Minnesota Mining and Mfg. Co. St. Paul, MN EA 84-90	$250 proposed FY 84; paid FY 85	Storage of licensed materials in an unrestricted area resulting in loss of the materials. The civil penalty was mitigated due to the licensee's prompt and extensive corrective action.

Table 3. IE Orders Issued During FY 1984

Licensee	*Amount*	*Reason*
Consumers Power Company (Midland Plant) EA 83-109	October 6, 1983	Confirmatory Order for Modification of Construction Permits (Effective Immediately) **Reason:** To confirm that the licensee shall adhere to the Construction Completion Program dated August 26, 1983.
International Nutronics, Inc. Dover, New Jersey EA 83-122	November 1, 1983	Order Modifying License **Reason:** To confirm commitments identified in a Confirmatory Action Letter dated October 4, 1983.
Shelwell Services, Inc. Hebron, Ohio EA 83-96	November 7, 1983	Rescission of Suspension and Order Modifying License **Reason:** Based on licensee's response to an Order Temporarily Suspending License, Effective Immediately and an Order to Show Cause issued on September 20, 1983.
American Testing Laboratories, Inc. Salt Lake City, Utah EA 83-47	December 10, 1983	Order Revoking License **Reason:** Based on licensee's response to an Order to Show Cause and Order Temporarily Suspending License issued on June 10, 1983.
Roof Auditing Services Oreland, Pennsylvania EA 83-112	December 27, 1983	Decision on Order to Show Cause **Reason:** Maintains the Order to Show Cause and Order Temporarily Suspending License, Effective Immediately issued on October 13, 1983.
Consumers Power Co. (Midland Plant)	January 12, 1984	Confirmatory Order **Reason:** To confirm licensee's commitment by Order to have an independent appraisal of site and corporate management organizations and functions.
International Nutronics, Inc. Dover, New Jersey EA 83-122	January 30, 1984	Order Modifying License, Effective Immediately **Reason:** Licensee's failure to fully meet the terms of the November 1, 1983 Order.
Perforating Services, Inc. Casper, Wyoming EA 83-110	February 28, 1984	Rescission of Suspension and Order Modifying License **Reason:** Based on the licensee's response to an Order to Show Cause and Order Temporarily Suspending License, Effective Immediately, issued on October 13, 1983.
Niagara Mohawk Power Corp. (Nine Mile Point Station) EA 83-137	March 20, 1984	Order Modifying License Effective Immediately **Reason:** Lack of management attention to the control of safety-related activities at Unit 1.
Niagara Mohawk Power Corp. (Nine Mile Point Station) EA 83-137	March 20, 1984	Order **Reason:** Significant deficiencies in the Quality Assurance Program identified by the Construction Appraisal Team at Unit 2.
Superior Production Logging, Inc. Snyder, Texas EA 84-51	June 7, 1984	Order to Show Cause and Order Temporarily Suspending License (Effective Immediately) **Reason:** Licensee's neglect and careless disregard of the Commissions requirements and lack of control of its licensed operations.

Table 3. IE Orders Issued During FY 1984
(continued)

Licensee	*Amount*	*Reason*
International Nutronics, Inc. Dover, New Jersey EA 83-122	June 11, 1984	Modification of Order Dated January 30, 1984 **Reason:** Revised decontamination plan submitted by the licensee.
Philadelphia Electric Co. (Peach Bottom) EA 84-39	June 18, 1984	Order Modifying License Effective Immediately **Reason:** Violations involving control rod patterns.
John C. Haynes Co. Newark, Ohio EA 84-48	June 19, 1984	Order to Show Cause **Reason:** Contamination in excess of NRC's "Guidelines for Decontamination of Facilities and Equipment Prior to Release for unrestricted Use or Termination of Licenses for Byproduct Source or Special Nuclear Material."
Florida Power and Light Co. (Turkey Point) EA 84-55	July 11, 1984	Confirmatory Order **Reason:** Licensee weaknesses in controlling plant activities.
Tennessee Valley Authority (Browns Ferry) EA 84-54	July 12, 1984	Confirmatory Order **Reason:** Licensee's poor history of regulatory compliance.
Henry Ford Hospital Detroit, Michigan EA 84-67	July 17, 1984	Confirmatory Order **Reason:** Misadministration of a prescribed therapy dose.
Pennsylvania Power and Light Co. (Susquehanna) EA 84-77	July 27, 1984	Confirmatory Order **Reason:** Loss of AC power on Unit 2.
Inspection and Testing, Inc. Chubbuck, Idaho EA 84-18	August 31, 1984	Order to Show Cause and Order Suspending License Effective Immediately **Reason:** Overexposure of personnel.

BULLETINS AND INFORMATION NOTICES

The NRC Office of Inspection and Enforcement issues Bulletins and Information Notices to licensees, including construction permit holders, to inform them of events that may have generic implications. Each of these issuances is based on events reported by licensees, NRC inspectors, Agreement States, or others, where a preliminary evaluation indicates that the event may affect other licensees. A total of 99 NRC Information Notices were issued in fiscal year 1984, including five updates of previously issued Information Notices. (Table 4 lists all Information Notices issued in fiscal year 1984). Information Notices provide information but do not require specific actions. They are transmittals of information which may not yet have been completely analyzed by the NRC, but which licensees should be aware of. Licensees receiving an Information Notice are expected to review the information for applicability to their facilities, and consider actions, if appropriate, to preclude a similar problem occurring at their facilities. The NRC then follows up through inspections and regional assessment to assure that the actions are being taken.

NRC Bulletins provide information about one or more similar events of significance and require that licensees take specific actions. Bulletins usually require the licensee to submit a report to the NRC that describes actions taken or to be taken, and to provide information the NRC may need to assess the need for further action. Prompt response by licensees is usually required and failure to respond will normally result in NRC enforcement action. Prior to issuing a Bulletin, the NRC may seek comments from the nuclear industry. This technique has proven effective in generating faster and more informed responses from affected licensees. However, the nature of the problem and a need for timely action may limit such prior consultation. NRC Bulletins generally

Table 4. IE Information Notices Issued in FY 1984

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
83-65	Surveillance of Flow in RTD Bypass Loops Used in Westinghouse Plants	10/7/83	All Westinghouse power reactor facilities holding an operating license (OL) or construction permit (CP)
83-66	Fatality at Argentine Critical Facility	10/7/83	All power reactor facilities holding an OL or CP; non-power reactor, critical facility and fuel cycle licensees
83-67	Emergency-Use Respirator Material Defect Causes Production of Noxious Gases	10/11/83	All power reactor facilities holding an OL or CP; research and test reactor licensees, fuel facilities; Priority I material licensees
83-68	Respirator User Warning: Defective Self-Contained Breathing Apparatus Air Cylinders	10/11/83	All power reactor facilities holding an OL or CP; research and test reactor licensees; fuel facilities; Priority I material licensees
83-69	Improperly Installed Fire Dampers at Nuclear Power Plants	10/21/83	All power reactor facilities holding an OL or CP
83-70	Vibration-Induced Valve Failures	10/25/83	All power reactor facilities holding an OL or CP
83-71	Defects in Load-Bearing Welds on Lifting Devices for Vessel Head and Internals	10/27/83	All power reactor facilities holding an OL or CP
83-72	Environmental Qualification Testing	10/28/23	All power reactor facilities holding an OL or CP
83-73	Radiation Exposure from Gloves Contaminated with Uranium Daughter Products	10/31/83	All licensees authorized to process uranium as source material and metal producers of alloys except uranium mills, uranium fuel fabrication plants and nuclear power plants
83-74	Rupture of Cesium—B7 Source Used In Well-Logging Operations	11/3/83	All licensees authorized to possess and use sealed sources containing by product or special nuclear material in well-logging operations
83-75	Improper Control Rod Manipulation	11/3/83	All power reactor facilities holding an OL or CP
83-76	Reactor Trip Breaker Malfunctions (Undervoltage Trip Devices on GE Type AK-2-25 Breakers)	11/2/83	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1984
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
83-77	Air/Gas Entrainment Events Resulting in System Failures	11/14/83	All power reactor facilities holding an OL or CP
83-78	Apparent Improper Modification of a Component Affecting Plant Safety	11/17/83	All power reactor facilities holding an OL or CP
83-79	Apparently Improper Use of Commercial Grade Components in Safety-Related Systems	11/23/83	All power reactor facilities holding an OL or CP
83-80	Use of Specialized "Stiff" Pipe Clamps	11/23/83	All power reactor facilities holding an OL or CP; nuclear steam system suppliers and architect-engineers
83-81	Entry into High Radiation Areas from Areas Which are Not Under Direct Surveillance	12/7/83	All licensees authorized to use portable radiography devices in radiography programs
83-82	Failure of Safety/Relief Valves to Open at BWR-Final Report	12/20/83	All power reactor facilities holding an OL or CP
83-83	Use of Portable Radio Transmitters Inside Nuclear Power Plants	12/19/83	All power reactor facilities holding an OL or CP
83-84	Cracked and Broken Piston Rods in Brown Boveri Electric Type 5HK Breakers	12/30/83	All power reactor facilities holding an OL or CP
84-01	Excess Lubricant in Electric Cable Sheaths	1/10/84	All power reactor facilities holding an OL or CP
84-02	Operating a Nuclear Power Plant at Voltage Levels Lower than Analyzed	1/10/84	All power reactor facilities holding an OL or CP
84-03	Compliance with Conditions and Notification of Disability by Licensed Operators	1/18/84	Licensed operators & facility licensees
84-04	Failure of Elastomer Seated Butterfly Valves Used Only During Cold Shutdowns	1/18/84	All power reactor facilities holding an OL or CP
84-05	Exercise Frequency	1/16/84	All power reactor facilities holding an OL or CP
84-06	Steam Binding of Auxiliary Feedwater Pumps	1/25/84	All power reactor facilities holding an OL or CP
84-07	Design-Basis Threat and Review of Vehicular Access	2/3/84	All power reactor facilities holding an OL or CP; and certain fuel fabrication & processing facilities using or possessing a formula quantity of SNM
84-08	10 CFR 50.7, "Employee Protection"	2/14/84	All power reactor facilities holding an OL or CP; and NSSS & AE
83-63 Supp 1	Potential Failures of Westinghouse Electric Corporation Type SA-1	2/15/84	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1984
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-09	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)	2/13/84	All power reactor facilities holding an OL or CP
84-10	Motor-Operated Valve Torque Switches Set Below the Manufacturer's Recommended Value	2/21/84	All power reactor facilities holding an OL or CP
84-11	Training Program Deficiencies	2/24/84	All power reactor facilities holding an OL or CP
84-12	Failure of Soft Seat Valve Seals	2/27/84	All power reactor facilities holding an OL or CP
84-13	Potential Deficiency in Motor-Operated Valve Control Circuits and Annunciation	2/28/84	All power reactor facilities holding an OL or CP
84-14	Highlights of Recent Transport Regulatory Revisions by DOT and NRC	3/2/84	All NRC licensees
84-15	Reporting of Radiological Releases	3/2/84	All power reactor facilities holding an OL or CP
84-16	Failure of Automatic Sprinkler System Valves to Operate	3/2/84	All power reactor facilities holding an OL or CP
84-17	Problems with Liquid Nitrogen Cooling Components Below the Nil Ductility Temperature	3/5/84	All power reactor facilities holding an OL or CP
84-09 Rev 1	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10 CFR 50, Appendix R)	3/7/84	All power reactor facilities holding an OL or CP
84-18	Stress Corrosion Cracking in Pressurized Water Reactor Systems	3/7/84	All power reactor facilities holding an OL or CP
84-19	Two Events Involving Unauthorized Entries into PWR Reactor Cavities	3/21/84	All power reactor facilities holding an OL or CP
84-20	Service Life in Safety-Related Systems	3/21/84	All power reactor facilities holding an OL or CP
84-05 Rev 1	Exercise Frequency	3/28/84	All power reactor facilities holding an OL
84-21	Inadequate Shutdown Margin	3/28/84	All power reactor facilities holding an OL or CP
84-22	Deficiency in COMSIP, Inc. Standard Bed Catalyst	3/29/84	All power reactor facilities holding an OL or CP
84-23	Results of the NRC-Sponsored Qualification Methodology Research Test on ASCO Solenoid Valves	4/5/84	All power reactor facilities holding an OL or CP
84-24	Physical Qualification of Individuals to Use Respiratory Protective Devices	4/5/84	All power reactor facilities holding an OL or CP; research and test reactors, fuel facilities and Priority I material licensees

Table 4. IE Information Notices Issued in FY 1984
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-25	Recent Serious Violations of NRC Requirements by Radiography Licensees	4/16/84	Radiography Licensees and Manufacturers who distribute devices that incorporate sealed sources
84-26	Recent Serious Violations of NRC Requirements by moisture Density Gauge Licensees	4/16/84	Byproduct materials licensees who possess and use byproduct materials in moisture density gauges and manufacturers who distribute devices
84-27	Recent Serious Violations of NRC Requirements by Medical Licensees	4/17/84	Medical Licensees
84-28	Recent Serious Violations of NRC Requirements by Well-Logging Licensees	4/17/84	Byproduct materials licensees who possess and use byproduct materials in well-logging devices & manufacturers who distribute sealed source devices
84-29	General Electric Magne-Blast Circuit Breaker Problems	4/17/84	All power reactor facilities holding an OL or CP
84-30	Discrepancies in Record Keeping and Material Defects in Bahnson Heating, Ventilation, and Air Conditioning Units	4/18/84	All power reactor facilities holding an OL or CP
84-31	Increased Stroking Time of Bettis Actuators Because of Swollen Ethylene-Propylene Seals and Seal Set	4/18/84	All power reactor facilities holding an OL or CP
84-32	Auxiliary Feedwater Sparger Pipe Hanger Damage	4/18/82	All power reactor facilities holding an OL or CP for PWRs
84-33	Main Steam Safety Valve Failures Caused By Failed Cotter Pins	4/20/84	All power reactor facilities holding an OL or CP
84-34	Respirator Users Warning: Defective Self-Contained Breathing Apparatus Air Cylinders	4/23/84	All power reactor facilities holding an OL or CP; research and test; fuel cycle; and Priority 1
84-35	BWR Post Scram Drywell Pressurization	4/23/84	All power reactor facilities holding an OL or CP
84-36	Loosening of Locking Nut on Limitorque Operator	5/01/84	All power reactor facilities holding an OL or CP
84-37	Use of Lifted Leads and Jumpers During Maintenance or Surveillance Testing	5/10/84	All power reactor facilities holding an OL or CP
84-38	Problems With Design, Maintenance, and Operation of OffSitė Power Systems	5/17/84	All power reactor facilities holding an OL or CP
84-39	Inadvertent Isolation of Spray Systems	5/25/84	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1984
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
83-66 Rev 1	Fatality at Argentine Critical Facility	5/25/84	All power reactor facilities holding an OL or CP; nonpower reactor, critical facility, & fuel cycle licensees
84-40	Emergency Worker Doses	5/30/84	All power reactor facilities holding an OL or CP; research and test reactor and fuel cycle licensees
84-41	IGSCC in BWR Plants	6/1/84	All BWR reactor facilities holding an OL or CP
84-42	Equipment Availability for Conditions During Outages Not Covered by Technical Specifications	6/5/84	All power reactor facilities holding an OL or CP
84-43	Storage and Handling of Opthalmic Beta Radiation Applicators	6/7/84	All medical licensees
84-44	Environmental Qualification Testing of Rockbestos Cables	6/8/84	All power reactor facilities holding an OL or CP
84-45	Reversed Differential Pressure Instrument Sensing Lines	6/11/84	All power reactor facilities holding an OL or CP
84-46	Circuit Breaker Position Verification	6/13/84	All power reactor facilities holding an OL or CP
84-47	Environmental Qualification Tests of Electrical Terminal Blocks	6/15/84	All power reactor facilities holding an OL or CP
84-48	Failures of Rockwell International Globe Valves	6/18/84	All power reactor facilities holding an OL or CP
84-49	Intergranular Stress Corrosion Cracking Leading to Steam Generator Tube Failure	6/18/84	All power reactor facilities holding an OL or CP
84-50	Clarification of Scope of Quality Assurance Programs Pursuant to 10 CFR 50 Appendix B	6/21/84	All power reactor facilities holding an OL or CP
84-51	Independent Verification	6/26/84	All power reactor facilities holding an OL or CP
84-52	Inadequate Material Procurement Controls on the Part of Licensees and Vendors	6/29/84	All power reactor facilities holding an OL or CP
84-53	Information Concerning the Use of Loctite 242 and Other Anerobic Adhesive	7/5/84	All power reactor facilities holding an OL or CP
84-54	Deficiencies in Design Base Documentation and Calculations Supporting Nuclear Power Plant Design	7/5/84	All power reactor facilities holding an OL or CP
84-55	Seal Table Leaks at PWRs	7/6/84	All power reactor facilities holding an OL or CP
84-56	Respirator Users Notice for Certain 5-Minute Emergency Escape Self-Contained Breathing Apparatus	7/10/84	All power reactor facilities holding an OL or CP

Table 4. IE Information Notices Issued in FY 1984
(continued)

Information Notice No.	*Subject*	*Date of Issue*	*Issued to*
84-57	Operating Experience Related to Moisture Intrusion in Safety-Related Electrical Equipment at Commercial Power Plants	7/27/84	All power reactor facilities holding an OL or CP
84-58	Inadvertent Defeat of Safety Function Caused By Human Error Involving Wrong Unit, Wrong Train, or Wrong System	7/25/84	All power reactor facilities holding an OL or CP
84-59	Deliberate Circumventing of Station Health Physics Procedures	8/6/84	All power reactor facilities holding an OL or CP
84-60	Failure of Air-Purifying Respirator Filters To Meet Efficiency Requirement	8/6/84	All power reactor facilities holding an OL or CP
84-61	Overexposure of Diver in Pressurized Water Reactor (PWR) Refueling Cavity	8/8/84	All power reactor facilities holding an OL or CP
84-62	Therapy Misadministrations To Patients Undergoing Cobalt-60 Teletherapy Treatments	8/10/84	All NRC licensees authorized to possess and use sealed sources in teletherapy units
84-63	Defective RHR Replacement Piping	8/13/84	All power reactor facilities holding an OL or CP
84-64	BWR High-Pressure Coolant Injection (HPCI) Initiation Seal-In and Indication	8/15/84	All BWR licensees and applicants for an OL
84-65	Underrated Fuses Which May Adversely Affect Operation of Essential Electrical Equipment	8/16/84	All power reactor facilities holding an OL or CP
84-66	Undetected Unavailability of the Turbine-Driven Auxiliary Feedwater Train	8/17/84	All power reactor facilities holding an OL or CP
84-67	Recent Snubber Inservice Testing with High Failure Rates	8/17/84	All power reactor facilities holding an OL or CP
84-68	Potential Deficiency in Improperly Rated Field Wiring to Solenoid Valves	8/21/84	All power reactor facilities holding an OL or CP
84-69	Operation of Emergency Diesel Generators	8/29/84	All power reactor facilities holding an OL or CP
84-70	Reliance on Water Level Instrumentation with a Common Reference Leg	9/4/84	All power reactor facilities holding an OL or CP
84-71	Graphitic Corrosion of Cast Iron in Salt Water	9/6/84	All holders of a nuclear power reactor OL or CP
84-72	Clarification of Conditions For Waste Shipments Subject to Hydrogen Gas Generation	9/10/84	All power reactor facilities holding an OL or CP and users of transport packages
84-36 Supp 1	Loosening of Locking Nut on Limitorque Operator	9/11/84	All power reactor facilities holding an OL or CP
84-73	Downrating of Self-aligning Ball Bushings used in Snubbers	9/14/84	All power reactor facilities holding an OL or CP
84-74	Isolation of Reactor Coolant System from Low-Pressure Systems Outside Containment	9/28/84	All power reactor facilities holding an OL or CP

require one-time action and are not intended as substitutes for formally issued regulations or for imposed license amendments. However, the NRC does followup through inspections and regional assessments to assure that the actions are being taken. In fiscal year 1984, IE issued four Bulletins and two supplements to a previously issued Bulletin. The subject of each of the Bulletins and the required licensee actions are summarized below.

(1) IE Bulletin 83-07, Supplement 1 (issued on October 26, 1983), and Supplement 2 (issued on December 9, 1983), informed nuclear power reactor, fuel facility and Category B material licensees (processors and distributors) of additional information with regard to apparently fraudulent products sold by Ray Miller, Inc. IE Bulletin 83-07, issued on July 22, 1983, had provided a comprehensive list of apparently fraudulent material provided to approximately 450 customers of the Charleston branch of Ray Miller, Inc., during the period 1975 through 1979. On July 29, 1983, the NRC sent a letter to each non-licensee company on that list asking the company to identify nuclear facilities that may have been supplied fraudulent material. Those companies were also asked to identify the customer to whom they sold the material if they themselves were not the end-user. The two supplements to IE Bulletin 83-07 that were issued in fiscal year 1984 provided information collected from the responses to the July 29 letter. Specifically, a list was provided that included companies that had been identified as end-users and secondary recipients, and on NRC licensees identified as secondary recipients. No additional actions beyond those specified in IE Bulletin 83-07 were requested by these supplements.

(2) IE Bulletin 83-08, issued December 28, 1983, informed nuclear power reactor licensees and CP holders of findings involving circuit breakers with undervoltage trip attachments (UVTA). The bulletin gave examples of problems identified as causes for failure of the circuit breakers to trip, including: improper lubrication of linkages and other moving parts within either the UVTA or the circuit breaker trip bar latch assembly; inadequate adjustment of spring tension of the UVTA; excessive torque required to trip the circuit breaker because of hardening and contamination of the grease in the trip shaft bearings; and excessive wear of moving parts within either the UVTA or the trip bar latch assembly because of infrequent lubrication of these moving parts or improper adjustments of the spring tension of the UVTA. In addition to the failures of circuit breakers to trip on demand, the bulletin identified failure of breakers to close on demand as a safety concern. Holders of construction permits and operating licenses were asked to: identify the safety-related applications of the breakers and the systems in which they were used; review the adequacy of the design, testing and maintenance of the breakers in light of their operating experience and information conveyed in the bulletin; and evaluate the need to take corrective measures to ensure proper operation of the breakers.

(3) IE Bulletin 84-01, issued on February 3, 1984, informed all boiling water power reactor licensees and CP holders of a reported through wall crack which appeared to be 360° around the vent header within the containment torus at the Hatch Unit 2 plant. All boiling water reactor facilities having a Mark I containment that were in cold shutdown at the time the bulletin was issued were required to: (a) visually inspect for cracks in the entire vent header and in the main vents in the region near the intersection with the vent header; (b) declare the containment inoperable if cracks were found; and (c) report the results of the inspection by telephone to the NRC Operations Center, followed by a written report to the appropriate NRC Regional Administrator. In addition, although it was not required by the bulletin, boiling water reactor plants with a Mark I containment that were operating at the time the bulletin was issued were requested to review their plant data on differential pressure between the wetwell and drywell for anomalies that could be indicative of cracks. Any such anomalies were then to be reported to the NRC in accordance with 10 CFR 50.72 and 10 CFR 50.73.

(4) IE Bulletin 84-02, issued March 12, 1984, informed power reactor licensees and CP holders of failures of General Electric HFA type relays. Failures at several plants involved relays that were continuously energized in ac circuits and failed to open when de-energized. General Electric identified the cause of the failures as deterioration of the coil wire insulation as a result of the effects of aging with an estimated age at failure of 10 to 12 years. Because many facilities are now approaching this age, the likelihood of concurrent failures is increased. This potential for concurrent failure may be considered a precursor of ATWS (anticipated transient without scram), since concurrent failure of certain safety-related relays at nuclear power plants could result in failure of the reactor trip function. The bulletin required facilities holding an operating license to: develop plans and schedules for replacing HFA relays used in normally energized and normally de-energized safety-related applications; during the period before relay replacement, develop and implement surveillance plans; provide a basis for continuing operation for the period of time until the normally energized relays are replaced; and provide a written report of these actions. Holders of construction permits were required to: provide plans and schedules for replacing both normally energized and normally de-energized HFA relays

used in safety-related systems; provide a written report of these actions; and revise the appropriate administrative controls to ensure that the specified HFA relay coils are not used as replacement parts in safety-related applications.

(5) IE Bulletin 84-03, issued August 24, 1984, informed power reactor licensees and CP holders of a failure of the refueling cavity water seal at the Haddam Neck plant. The refueling cavity had been flooded in preparation for refueling. The seal assembly, which had been redesigned by the licensee and used only once previously, was subject to a gross failure due to lack of an interference between the width of the seal annulus and the width of the opening. This allowed the seal to be significantly displaced. Had fuel been in transfer at the time of the seal failure, it could have been partially or completely uncovered with possibly high radiation levels, fuel cladding failure and release of radioactivity. In addition, if the fuel transfer tube had been open, the spent fuel pool could have drained to a level which would have uncovered the top of the fuel. Licensees were required to evaluate the potential for and consequences of a refueling cavity seal failure and provide a written summary report to the appropriate Regional Administrator.

TRACKING SYSTEM FOR REPORTS OF DEFECTS

During fiscal year 1984, the staff initiated a system to track notifications made to the NRC in accordance with 10 CFR Parts 21 and 50.55(e). Part 21 requires vendors who supply components for nuclear power plants to report any defects or non-compliance with NRC requirements that could create a substantial safety hazard. Part 50.55(e) requires holders of construction permits (CP) for nuclear power plants to make similar reports. These requirements result in several thousand reports to the NRC each year. The Office of Inspection and Enforcement reviews the reports to determine whether the corrective action taken by the vendor or CP holder is sufficient, and to determine whether an information notice or bulletin should be issued, or if other actions may be required (e.g., an inspection conducted, a temporary instruction issued, or further correspondence with the vendor or licensee). When the appropriate corrective action has been completed the item is closed out by IE. The new tracking system will computerize all of the reports, allowing automated searches of the information on file and tracking of the actions taken by vendors and the NRC, and will assist in communication of the information between NRC Headquarters and the Regions.

INCIDENT RESPONSE

Events Analysis

The Nuclear Regulatory Commission has maintained a 24-hour-a-day, 365-days-a-year, manned Operations Center since the emergency incident at the Three Mile Island Nuclear Power Plant in 1979. The Center, which is located in Bethesda, Md., functions as the NRC's point of direct communication through dedicated telephone lines for reports of significant events at licensed nuclear power plants and certain fuel cycle facilities.

The staff at the Operations Center evaluate the telephone notifications and, depending on the safety significance of the particular event, notify other appropriate NRC personnel and other Federal agencies. In all cases, the NRC Regional Office responsible for the facility 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 teams within Headquarters and the appropriate NRC Region. Events are monitored by the NRC while they are in progress from the standpoint of action to protect the health and safety of the public.

Each event called into the Operations Center or reported by a Regional Office is evaluated to determine any generic implications for similar facilities. Event reports are screened during the first hours of the first working day following the receipt of a notification. Events that may be significant from a generic standpoint then receive additional in-depth evaluation. For events found to have significant generic implications, the NRC issues either an Information Notice or a Bulletin to the appropriate licensees and construction permit holders. Some examples of events that received significant attention within the NRC during fiscal year 1984 because of their generic implications are provided below.

On February 3, 1984, a through-wall crack that went almost completely around the vent header within the containment torus was discovered at Hatch Unit 2 located in Georgia. Since Hatch Unit 2 was in cold shutdown at the time, there was no immediate safety problem at that plant. However, the crack represented a failure of a major piece of safety-related equipment required to limit containment pressure during a loss-of-coolant accident and, thus, had significant generic implications. A Bulletin was issued within hours after discovery of the crack. (See IE Bulletin 84-01, above.) Further evaluation revealed that the crack in the Hatch 2 vent header resulted from brittle fracture caused by the injection of cold nitrogen into the torus during containment inerting. An Information Notice (IN 84-17) was issued to inform licensees and applicants of the potentially significant problems with the use of liquid nitrogen that may cool vital components below the nil ductility temperature of associated materials and, thereby, precipitate brittle fracture.

While the Palisades plant in Michigan was shut down for refueling, an inspection revealed damage to several components associated with the auxiliary feedwater system. There was a cracked thermal sleeve and a missing hold-down clamp for an auxiliary feedwater sparger in one steam generator. There was a broken weld and a broken clamp on piping to the auxiliary feedwater sparger in the second steam generator. During the same refueling shutdown, eight hangers had previously been found to be either loose or damaged on the auxiliary feedwater piping external to the steam generators. An Information Notice (IN 84-32) was issued describing the findings at Palisades and suggesting continued vigilance on the part of plant staffs with respect to investigating symptoms indicative of water hammer.

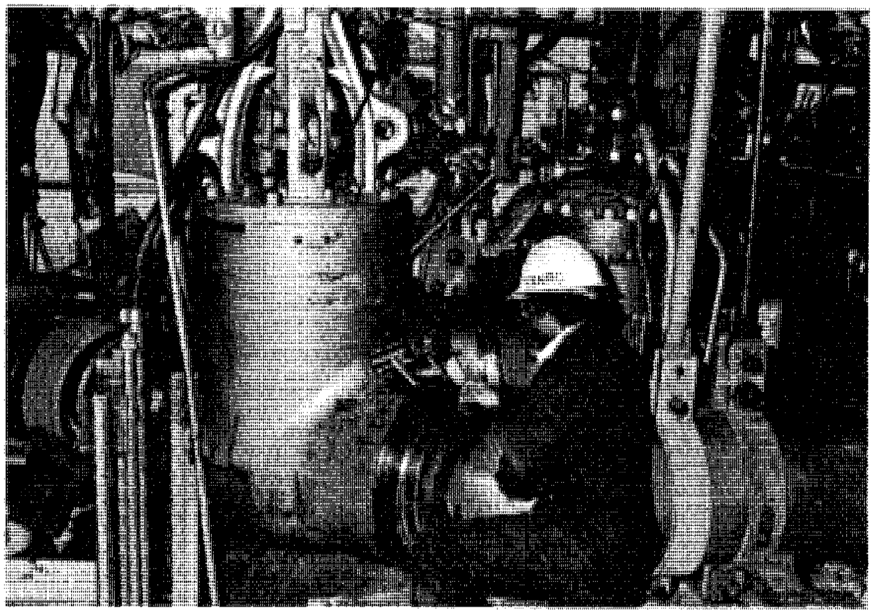
On March 2, 1984, the Davis Besse plant in Ohio reported a reactor trip from 99 percent power due to an inadvertent closure of a main steam line isolation valve. Following the plant trip, the steam generator safety valves opened and one of nine safety valves associated with steam generator number two failed to close. The feedwater to this steam generator was isolated, as it should have been in the case of an uncontrolled continuous steam release from that steam generator. The steam generator then went dry in approximately 15 minutes. It was 5:00 a.m. the following morning before the malfunctioning safety valve could be replaced and 7:30 a.m. before the empty steam generator was refilled.

The Davis Besse event had the potential for progressing into a much more serious transient, and, thus, the NRC followed it closely until the residual heat removal system was placed into operation to complete the cooldown. The NRC went to a Standby mode for the first several hours of the event on the first day and then kept additional technical professionals both in the headquarters Operations Center and the Region III Incident Response Center until the event was over. Region III also sent people to the Davis Besse site. As in all cases, the licensee had the responsibility for operating the plant and limiting the consequences of the event. The NRC followed the transient and evaluated the possibilities for further degradation in safety. The NRC remained cognizant of the licensee's plans throughout the event. The status of the plant with respect to technical specification requirements, design limits, and procedures was continually followed. At the completion of the plant cooldown, the licensee determined the analyses and corrective actions to be taken prior to plant restart. These were delineated in a Confirmatory Action Letter issued by NRC Region III.

The event was caused by the release nut turning on the spindle to a lower position while the main steam safety valve was lifted. When the main steam pressure was reduced to the reseat pressure, the valve could not reseat because the release nut was holding the spindle up. This turning of the release nut was apparently caused by the failure of the release nut cotter pin and the vibration of the spindle while the main steam safety valve was listed. Evaluation of the safety valve problem indicated that the cotter pin release nut design is common to most steam safety valves and many primary safety valves on nuclear power plants. For this reason, an Information Notice (IN 84-33) was issued to all power reactor licensees.

Of the over 100 generic communications issued by NRC in fiscal year 1984, approximately two-thirds discussed the generic implications of events such as those described above. The NRC Operations Center receives telephonic notification of between 2,000 and 3,000 events each year. Although all such events are evaluated by the NRC, a relatively small percentage of them have generic implications.

Aging nuclear facilities must be checked frequently and thoroughly to ensure that normal wear of parts and equipment is not creating a dangerous or otherwise unacceptable condition. Thor Oberg, an NRC radiation specialist, is shown here checking the radiation level of a licensee's liquid holding tank, as part of such efforts.



NRC inspector Richard Harris of the Region 1 office is shown performing ultrasonic testing on valves during an inspection of the Millstone nuclear power plant near Waterford, Conn.

Operations Center Upgrade

The past six years of operational experience by the Center has resulted in a definitive policy on the NRC's role in response to a nuclear emergency involving the licensed industry and the development of a fixed set of operational procedures for incident response personnel. In 1982, the "Design Basis for the NRC Operations Center" was published and a major project was undertaken to build a new and dedicated Operations Center. The considerable effort in 1983 to upgrade personnel procedures and resources enhanced the old Operations Center to a significant extent, given the constraints of the existing space.

The year 1984 was spent constructing a new facility in the below-ground-level space located in the Maryland National Bank Building. The contract to undertake physical construction was awarded by the General Services Administration at mid-year and construction of the new Center was expected to be completed by December 1984. The letting of two contracts to integrate an improved telecommunications system and provide an electronically sophisticated video display wall in the Executive Team Room were vigorously pursued. The results of these contractual efforts will provide facilities for a more efficient and timely response by the Commission to any future nuclear incidents.

Regional Response Capability

The regional-office level of response is based on predetermined classification of events and NRC response modes. For a more significant event, a regional base team and a regional site team are assembled. The base team monitors licensee performance, supports NRC headquarters incident management, when appropriate, and coordinates response effort until the site team arrives at the site of the event and is operational. The site team goes to the site and is responsible for coordinating the NRC's incident response activities there.

To assure an adequate agency-wide response capability, a program to develop and assess regional response capabilities has been undertaken. The year 1984 was the second year for assessing regional response capabilities. The principal area of concentration this assessment year was the evaluation of the Regional Director of Site Operations and overall interface performance during exercises. The site exercises observed were:

Region I	Millstone
Region II	Hatch
Region III	Prairie Island
Region IV	Ft. Calhoun
Region V	San Onofre

Assessment criteria for assessment year 3 will concentrate on Federal agency interface.

Emergency Response Procedures

More extensive and detailed emergency response procedures were developed during fiscal year 1984. Using NUREG-0728, Rev. 2, the "NRC Incident Response Plan" and NUREG-0845, "Agency Procedures for the NRC Incident Response Plan" as a foundation, technical procedures for reactor analysis, and protective action decisionmaking have been developed and successfully tested through the Federal Field Exercise at St. Lucie in Florida in March of 1984. In addition, interface procedures for technical and liaison team members have been developed. Site team composition and organization has also been reviewed and modified to reflect current and projected analysis needs.

Emergency Response Training

In 1984, the response training efforts concentrated on basic emergency response roles and principals. Training was conducted for all headquarters and regional response personnel. Workshops were also held in Regions II and III with NRC, State, and Federal response personnel to assure all parties understood each others roles. Efforts were also started to develop a formal training manual for all NRC response personnel.

Federal Field Exercise

NRC staff was heavily involved in the development of the Federal Radiological Emergency Response Plan (FRERP) which provides the overall direction and guidance as to how the Federal government would respond to a civilian radiological emergency. A test of this plan, the FRERP Field Exercises (FFE), was held in conjunction with a full-scale exercise at the St. Lucie Nuclear Power Plant site on March 6, 7, and 8, 1984. A total of about 150 NRC staffers participated in the exercise.

EMERGENCY PREPAREDNESS

Support to Licensing Activities

During the report period, IE staff continued to evaluate the adequacy of the on-site plans to be included in the Safety Evaluation Report, and supplements thereto, for each plant in a near-term licensing status. The staff also took part in licensing hearings before the Atomic Safety and Licensing Board panels and served on inspection teams appraising applicants' implementation of emergency preparedness programs and their full-scale exercises. NTOL's appraised during fiscal year 1984 include Comanche Peak, Fermi 2, Catawba, Byron, Calloway, Watts Bar, Limerick and Wolf Creek. The staff reviewed, in addition, evaluations by the Federal Emergency Management Agency (FEMA) of off-site emergency plans for these facilities, as well as FEMA reports on State and local government performance during emergency preparedness exercises.

Exercise Frequency Rule Changes

On July 6, 1984, the NRC published in the *Federal Register* a final rule amending 10 CFR Part 50 with regard to emergency planning and preparedness (49 FR 27733). This rule amends sections of the Commission's regulations which require licensees to conduct emergency preparedness exercises (10 CFR 50.47 and Part 50, Appendix E). Taking into account the experience gained from emergency preparedness exercises conducted since 1980--as well as public comments received on the proposed revision of this rule--the Commission concluded that the requirements related to frequency of participation by State and local authorities in these exercises at nuclear power plant sites could be relaxed. The amended regulation continues to require licensees to conduct an annual on-site emergency preparedness exercise, but State and local governments are required to participate in such an exercise only every two years, returning to any given site every seven years. The amendment also provides for remedial exercises to assure that significant deficiencies identified during an exercise have been corrected. FEMA, in consultation with the NRC, determines the need for and extent of these remedial exercises.

Emergency Response Facilities

During fiscal year 1984, a program for the appraisal of emergency response facilities (ERFs) was initiated. The adequacy of these facilities, which are support facilities for nuclear power plants, is being appraised against requirements of Supplement 1 to NUREG-0737, as issued in generic letter 82-33. A training program and an appraisal training document with inspection procedure were developed for the teams making the appraisals. Two ERF appraisals were completed during the period.

The program will extend over the next several years, and the remaining appraisals will be made as ERFs are completed at each plant.

Reviews of Non-Power Reactor Emergency Plans

Regulations revised in 1980 required research and test reactor licensees to submit emergency plans for their facilities by November, 1982. During fiscal year 1983, the staff initiated its review of these plans. In October, 1983, NRC published the Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors (NUREG-0849). Using NUREG-0849, the staff completed essentially all reviews of emergency plans for 65 research and test reactor licensees during the report period. Evaluations of the adequacy of implementation of these plans were furnished by the staff for inclusion in the Safety Evaluation Report for each facility.

In addition, the staff developed and issued an emergency preparedness implementation inspection procedure for non-power reactors rated at greater than two megawatts. Six inspections using the new procedure were completed during fiscal year 1984. The remaining five are scheduled for fiscal year 1985.

TECHNICAL TRAINING PROGRAM

NRC's Technical Training Center (TTC), located in Chattanooga, Tenn., has primary responsibility for the training of NRC employees in specialized technology areas related to regulation, inspection, and enforcement. The TTC currently offers 70 different highly specialized technical training courses designed to give NRC inspectors the appropriate background to perform inspections at commercial nuclear power plants, fuel fabrication and byproduct utilization facilities, test and research reactors and vendor facilities. A program for newly employed, recent college graduates provides practical engineering training for relatively inexperienced employees.

Although the courses were designed to provide specialized training to meet specific job requirements of NRC inspectors and engineers, participants come from all NRC offices. Additionally, representatives of other government agencies, NRC contractors and foreign nationals attend when priorities permit. Reactor technology courses are also taught in other countries in support of their regulatory agencies.

In fiscal year 1984, the TTC presented a total of 1,825 student weeks of instruction. Courses are presented by members of the TTC staff and various contractors. Training is conducted in conventional classrooms, scientific laboratories, nuclear power plants, and reactor control room simulators at the NRC Technical Training Center and contractor locations throughout the United States.

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 NRC's Office of State Programs. (Certain aspects of NRC's State programs are being implemented by the Regional Offices under policies and procedures established by the Office of State Programs.) This chapter reports on activities in three major areas: the State Agreements Program; various liaison and cooperative programs; and financial protection and related concerns.

STATE AGREEMENTS PROGRAM

The Nuclear Regulatory Commission has agreements with 27 States by which those States have assumed regulatory responsibility over byproduct and source materials and small quantities of special nuclear material. At the end of 1984, there were about 13,100 radioactive material licenses in Agreement States; these represent about 60 percent of all the radioactive materials licenses in the United States. An Agreement with Utah became effective April 1, 1984. The NRC State Agreements Program is implemented by the NRC Regional Offices in accordance with policies and procedures established by the Office of State Programs.

Review of State Regulatory Programs

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 published in the Federal Register, December 4, 1981. 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 one follow-up review were conducted in 1984. 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. One follow-up review of problem areas identified in a routine review was conducted in California in 1984 to assess the State's corrective actions.

The overall results of the NRC reviews conducted during the report period indicate that the Agreement States continue to conduct effective regulatory programs. Periodic meetings are held with U.S. Department of Labor officials to exchange information and to keep them apprised of the status of Agreement State radiation control programs.

NRC Technical Assistance to States

The NRC provided technical assistance to Agreement States during 1984 in the areas of licensing, inspection, enforcement and proposed statutes and regulations. Examples include assistance provided to Georgia and North Carolina in their evaluation of license applications for large irradiators, to South Carolina in the inspection of a depleted uranium processing facility and a nuclear laundry, to Kentucky in the inspection of a uranium oxide catalyst facility, and to Alabama in its evaluation of a license application for use of cobalt 60 for animal studies. Assistance was also provided to Arizona in its evaluation of a broad license amendment for a backup laboratory to a nuclear power plant, and to California in its evaluation of licensee designates for a low-level waste disposal site. In addition, New York was provided guidance on acceptable americium-241 decontamination limits.

Training Offered by NRC

State radiation control personnel regularly attend NRC-sponsored courses to upgrade their technical and administrative skills and, thus, their ability to maintain high quality regulatory programs. In 1984, the NRC sponsored 17 short-term training courses, attended by 257 State personnel. Applications for the training courses exceeded availabilities. Courses included health physics, industrial radiography safety, nuclear medicine procedures, orientation in licensing practices, inspection procedures, well logging, uranium mill inspection, teletherapy calibration, transportation inspections and low-level waste manifest systems. On-the-job training in licensing and compliance was provided to individual staff members in Washington and Georgia.

Utah became the 27th NRC Agreement State on April 1, 1984, when Governor Scott M. Matheson and NRC Chairman Nunzio J. Palladino signed the formal document.

Annual Agreement State Meeting

The annual meeting of Agreement State radiation control program directors, held in October 1984 in the NRC Region I area, covered a wide range of regulatory issues being faced by State personnel, including low-level waste management, transportation, materials licensing and compliance, revision of medical and radiation safety regulations, and cases involving abnormal occurrences and incidents.

Regulation of Low-Level Waste

NRC is continuing to provide technical assistance to the Agreement States in implementing programs for the regulation of low-level radioactive waste. Nevada, Texas, New Hampshire, North Carolina and California have adopted compatible regulations patterned after 10 CFR 61. Other Agreement States are expected to adopt similar regulations during their regular process for revising and updating regulations.

COOPERATIVE ACTIVITIES

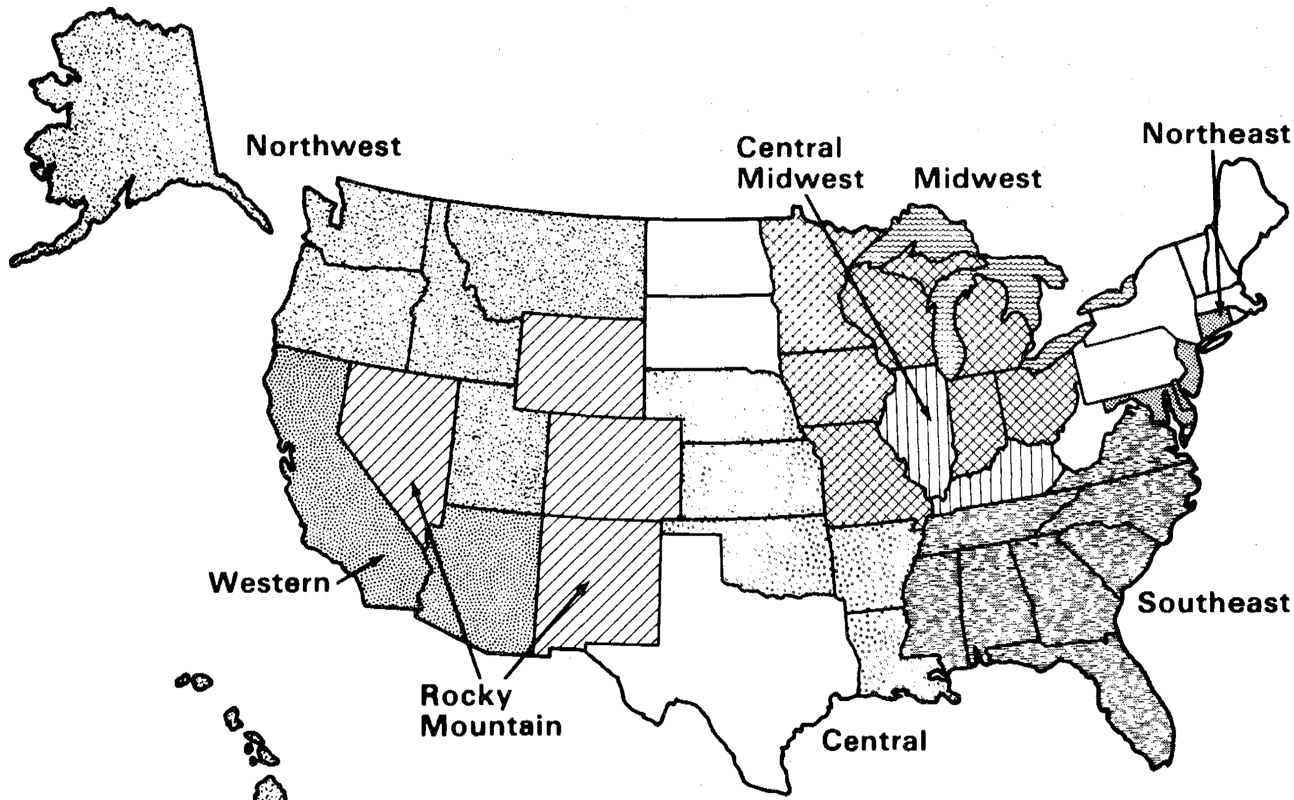
Low-Level Waste Compacts

In response to the Low-Level Radioactive Waste Policy Act, enacted in December 1980, the States continued their efforts to bring about workable interstate compacts that would provide for regional low-level waste disposal sites. The compacts and their member States are depicted on the map. Five compacts were introduced into Congress for Congressional consent—the Northwest, Central States, Rocky Mountain, Midwest and Southeast.

Congressional hearings, at which NRC testified, have been held on the compacts. Provisions of NRC-proposed Congressional consent language include preservation of existing NRC and Agreement State authority, jurisdiction and discretion; the assertion that the States and compact commissions are not authorized by the compacts to regulate in specified areas where the compacts have the greatest potential for conflict with NRC and the Department of Transportation regulatory schemes; the need for uniformity in the definition of low-level radioactive waste; and reservations on compact commission restrictions on the export of waste from a region. Congress took no consent action on the compacts this year.

In the course of development of low-level waste compacts, questions related to the method of disposal were raised in a number of States. Consequently, shallow land burial and alternative low-level radioactive waste disposal concepts were the subjects of a State workshop sponsored by NRC. Alternative disposal concepts considered were augered holes with liners, below-ground vaults, earth mounded concrete bunkers, above-ground vaults and mined cavities. Some workshop participants noted that the public appears to place greater confidence in disposal methods that incorporate man-made engineered barriers because of a number of past problems at shallow land burial facilities no longer in operation. Public acceptance of risk was considered to be a "critical" factor by State officials in selecting a disposal technology.

NRC representatives emphasized that Part 61 provides new regulatory requirements for shallow land burial as well as for the licensing of alternative land disposal tech-



LOW-LEVEL WASTE COMPACT GROUPS

nologies. NRC urged the States to take the lead in evaluating alternatives as part of the States' responsibilities under the Low-Level Radioactive Waste Policy Act since detailed concept engineering and economic feasibility studies depend in major part on factors specific to individual sites and regional characteristics.

Additional discussion by NRC, DOE, and the States is available in "Proceedings of the State Workshop on Shallow Land Burial and Alternative Disposal Concepts, May 2-3, 1984, Bethesda, Maryland." NUREG/CP-0055.

State Liaison Officers

There are 51 Governor-appointed State Liaison Officers, representing all 50 States and the Commonwealth of Puerto Rico, who provide a contact for communication between the States and the NRC.

On a periodic basis, regional and national State Liaison Officers' meetings are conducted to keep the State Liaison Officers updated on major aspects of NRC's programs.

A national meeting was held in Washington, D.C., in December 1983. Subjects discussed included NRC regionalization, emergency preparedness, waste management, including low- and high-level waste, transportation, and other items of mutual regulatory interest.

Liaison with American Indian Tribes

The NRC, and its predecessor, the AEC, carried out for three decades programs to maintain and enhance intergovernmental relations. The major thrust, as provided in section 274 of the Atomic Energy Act, has been cooperation with States as described in this chapter.

The Nuclear Waste Policy Act of 1982 provides for Federal agency consultation and cooperation not only with States, but also with affected Indian Tribes in decisionmaking regarding high-level waste management. Relationships have been established not only with individually affected tribes, but also with the National Congress of American Indians (NCAI, which represents over 200 tribes) and with the Council of Energy Resource Tribes (CERT, representing 39 tribes with significant energy resources on their reservations).

Memoranda of Understanding

In April 1984, Illinois entered into a broad Memorandum of Understanding (MOU) with NRC which provided principles of cooperation between the State and NRC in areas of concern to the State (49 FR 20587, May 15, 1984). This MOU also provided the basis for subsequent subagreements.

A subagreement on low-level waste inspection developed by NRC would allow States to inspect waste packaging and shipping procedures on the premises of certain NRC licensees. The inspections would review compliance with State laws and regulations as well as compliance with NRC's rules and regulations regarding packaging and transportation of low-level waste destined for disposal at a commercial low-level radioactive waste disposal site. This subagreement was developed in response to State requests during formulation of low-level waste compacts.

The NRC considers its program for reviewing licensees' procedures for quality assurance, packaging, marking, labeling and loading of vehicles adequate to ensure licensee compliance with Commission regulations, without the inspection of each individual shipment. Although States may exercise authority available to them under their own laws while supplementing this program review, the NRC remains the primary responsible party for undertaking enforcement actions under the subagreement. Coordination and cooperation on the part of the NRC and the States will be agreed to, to avoid duplicative enforcement actions.

The first such subagreement was signed by NRC and the State of Illinois in June 1984 (49 FR 27861, July 6, 1984). In it, both parties commit to mutually agreeable procedures whereby the State may perform inspection functions for and on behalf of the Commission at certain reactor and materials licensees' facilities which generate low-level radioactive waste. Negotiations are underway for similar subagreements with other States.

INDEMNITY, FINANCIAL PROTECTION AND PROPERTY INSURANCE

The Price-Anderson System

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

The first layer requires all licensees of commercial nuclear power plants rated at 100 electrical megawatts 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 a mechanism—payment of a retrospective premium—whereby the utility industry would share liability for any damages exceeding $160 million that result from a nuclear incident. 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. Presently, the secondary financial protection layer is $445 million (i.e., 89 power reactors rated in excess of 100 MW(e) licensed to operate x $5 million/reactor).

The third layer—Government indemnity—had equaled 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

NRC State Programs officials traveled widely to assist State and local officials in addressing public problems associated with nuclear power, waste disposal, etc. Here, Governor William J. Janklow of South Dakota presides over a public TV forum on low-level radioactive waste in Rapid City, with local and NRC experts participating. G. Wayne Kerr, Director of the NRC Office of State Programs, is in foreground.

power reactors, however, on November 15, 1982 when the sum of the first and second layers totaled $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 Study

In December 1983, the Commission transmitted to the Congress a detailed report entitled "The Price-Anderson Act—The Third Decade" concerning the need for renewal or modification of the Price-Anderson Act, which will expire on August 1, 1987. (For background, see the *1983 NRC Annual Report* p. 104.) On June 11, 1984, Chairman Palladino testified in a one-day hearing before the House Subcommittee on Energy and the Environment on the Price-Anderson Act and on the Commission's Price-Anderson Report. Others testifying at the hearing included representatives of States, the insurance and utility industries and public interest groups. The NRC proposal for modification of the Act which interested parties most frequently discuss is to replace the absolute limitation on liability with an annual limitation on liability.

Indemnity Operations

As of September 30, 1984, 136 indemnity agreements with NRC were in effect. Indemnity fees collected by the NRC from October 1, 1983 through September 30, 1984 totaled $100,662. Fees collected since the inception of the program total $23,023,766. 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 27 years of the program's existence.

Insurance Premium Refunds

The two private nuclear energy liability insurance pools—American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid to policyholders the 18th 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 for either payment of losses or ultimate return 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 1984 totaled $5,014,105—approximately 43.6 percent of all premiums paid on the nuclear liability insurance policies issued in 1974 and covering the period 1974-1984. The refunds represent 59.1 percent of the premiums placed in reserve in 1974.

Financial Qualifications Reviews Of Electric Utilities

In September 1984, the NRC published a final rule (see 49 *FR* 35747) which eliminated NRC staff review and licensing board consideration of the financial qualifications of electric utilities applying for power reactor operating licenses. As a result of a Federal appeals court ruling, the Commission reinstated a previous requirement for financial qualifications reviews and findings regarding electric utilities applying for power reactor construction permits. In support of this rulemaking the staff conducted a comprehensive study of the ratemaking practice and authority of State public utility commissions and Federal agencies relying partly on information developed by the National Association of Regulatory Utility Commissioners. The study included a national survey of, and visits to, State PUC's, Federal ratemaking agencies, and publicly-owned nuclear utilities to assess the ability of utilities to recover nuclear plant operating costs through the ratemaking process. Extensive public comments on the proposed rule were also considered. The study and the comments led the staff and the Commission to conclude that the ratemaking process allows utilities to recover all prudent expenditures for nuclear plant operation from customers. Accordingly, an independent NRC review of utilities' financial qualifications at the OL stage was deemed unnecessary.

Property Insurance

The NRC staff has prepared a revised property insurance rule based on public comments received in response to an advance notice of proposed rulemaking. The revised rule proposes to increase the required amount of on-site property damage insurance to $1.02 billion with further increases to be made as determined to be necessary to protect public health and safety.

As indicated by the second annual property insurance reports submitted by commercial reactor licensees, 58 plants are insured for $1.02 billion, the amount of insurance available on April 1, 1984 when the reports were due. Another 16 plants carry at least $935 million. As of August 15, 1984, Nuclear Electric Insurance Limited (NEIL II) announced that an additional $15 million of excess property insurance would be available, thus bringing the total available to $1.035 billion.

STATUS OF TMI-2 FACILITY

Financial Aspects of TMI-2 Cleanup

Funding by GPU. (For background, see the *1983 NRC Annual Report*, p. 105.) Use of property insurance pro-

Governor Richard Thornburgh of Pennsylvania met on numerous occasions with senior officials of the NRC regarding the financial and technical problems associated with the cleanup of Unit 2 of the Three Mile Island nuclear facility, as well as those involved in the startup of Unit 1. Shown above, the Governor (right foreground) addressed the Commission on Auguest 15, 1984, at its Washington headquarters. Left to right are Commissioner Lando Zech, Chairman Palladino, Commissioner Frederick Bernthal, and Commissioner James Asselstine. Commissioner Thomas Roberts was not in attendance at this session.

ceeds during 1983 for cleanup of the damaged reactor at Three Mile Island Unit 2 (Pa.) was slightly below the projection for the year such that $17 million (as opposed to $14 million) remained unused at the beginning of 1984. Based on the rate of insurance usage during 1984, it is estimated that the proceeds will be fully exhausted by the end of 1984.

Revenues collected by General Public Utilities Corporation's (GPU) three operating subsidiaries continue to be expended on cleanup at the total annual rate of $34 million. New Jersey ratepayer funds that were previously held in escrow have been released and are being expended on the cleanup. GPU informed NRC during 1984 that it was following a policy (approved by the GPU Board of Directors) of advancing cash from GPU internal sources to alleviate any cash flow problems in the effort to keep cleanup on schedule. Cleanup funding projections provided to NRC indicate a continuing GPU commitment to provide such cash advances through the end of the cleanup activity. GPU's financial condition and cash flow position have continued to improve such that the cash advances can be made to cleanup.

Cost Sharing Plan. During 1984, GPU announced that by early 1985 all elements of the TMI-2 cleanup cost sharing plan proposed by Pennsylvania Governor Richard Thornburgh in July 1981 would be in place. In September 1984, the Edison Electric Institute (EEI) informed NRC that its industry cost sharing program would contribute $25 million per year for six years beginning in 1985. The EEI program is being financed by EEI member investor-owned utilities and is to be supplemented by grants from Pennsylvania and New Jersey utilities such that a total of $150 million will be provided over the six years. The Pennsylvania and New Jersey utility commitment to the EEI program was made in June 1984. This industry commitment completes the suggested elements of the Thornburgh Plan. Contributions under the plan continued to flow in 1984 from the State governments of Pennsylvania and New Jersey, from the Federal government through the Department of Energy, from a Babcock and Wilcox legal settlement, from a Japanese industry consortium, and from GPU customers and GPU internal sources as discussed above. In 1984, the Electric Power Research Institute provided about $3 million in support of research related to the TMI-2 cleanup.

During 1984, commitments from all sources of funds originally contemplated by Governor Thornburgh were in place and at a level close to what was envisioned. The staff expressed a degree of optimism regarding the TMI-2 funding situation in an October 1984 report to the Commission.

The NRC continues to monitor the financial condition of the GPU companies as well as their efforts to secure TMI-2 cleanup funds from a variety of sources.

CHAPTER 10

International Programs

NRC's international activities during fiscal year 1984 continued to focus on efforts to improve worldwide nuclear safety cooperation and to ensure against further nuclear explosives proliferation.

During the fiscal year, the NRC:

- Renewed bilateral arrangements with France, Greece and Spain—three of the Commission's 21 partners in international exchange of reactor safety information and regulatory cooperation. Under the terms of one such arrangement, the agency cooperated with the Mexican National Nuclear Safety and Safeguards Commission in Mexico in cleaning up radioactive contamination traced to discarded cobalt-60 pellets. (See discussion under "Bilateral Cooperation," below.)
- Arranged and held meetings with visitors from 28 countries and four international organizations.
- Provided on-the-job training for 25 regulatory staff members from 11 foreign countries.
- Developed automated systems for summarizing reactor operating information from foreign countries and for cataloguing the foreign document collection.
- Issued 343 export licenses and 83 amendments of existing licenses.
- Continued to support domestic and international efforts to ensure that the risk of nuclear proliferation is minimized in the development and operation of the nuclear fuel cycle.
- Worked closely with the Executive Branch to assist the International Atomic Energy Agency in strengthening international safeguards.

BILATERAL COOPERATION

Bilateral Arrangements

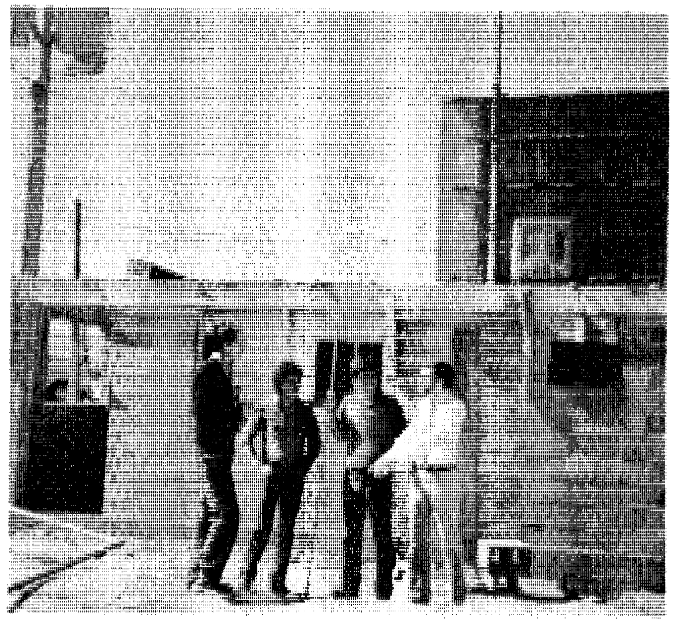
In May 1974, the NRC initiated a program for the exchange of technical information and for cooperation in nuclear safety affairs with other countries. The program was aimed first at those countries which had made major commitments to light water reactor technology, but was soon expanded to include both countries with developing nuclear power options and countries with firm plans to enter the field. These bilateral arrangements establish a mechanism for the timely exchange of significant reactor safety information and set up official communications channels for the sharing of information on broad reactor safety problems and other matters of mutual interest. They also serve as the conduit for most of the nuclear safety assistance that the NRC is able to provide to developing countries, particularly to those importing U.S. reactors and other equipment.

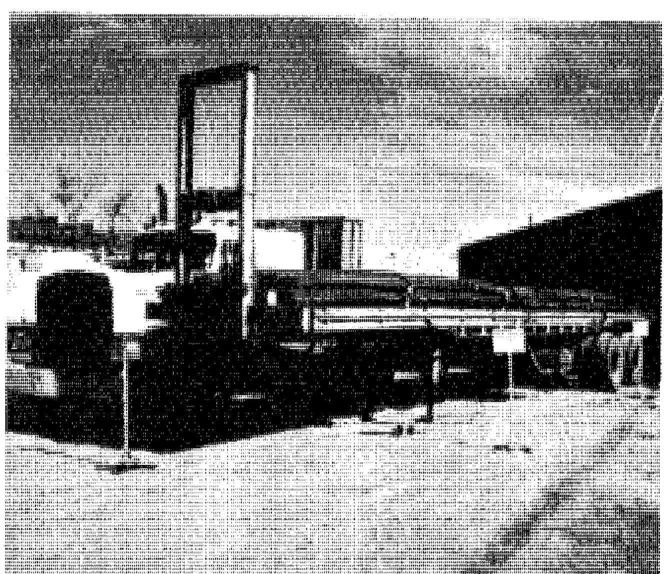
The NRC has 21 such arrangements currently in effect, with the regulatory authorities of Belgium, Brazil, China, Denmark, Egypt, Finland, France, the Federal Republic of Germany, Greece, Israel, Italy, Japan, Korea, Mexico, the Netherlands, the Philippines, Spain, Sweden, Switzerland, Taiwan, and the United Kingdom. Three of these—those with France, Greece, and Spain—were renewed in 1984. The NRC is also currently involved in negotiating an arrangement with the Federal Committee for Energy and industry in Yugoslavia.

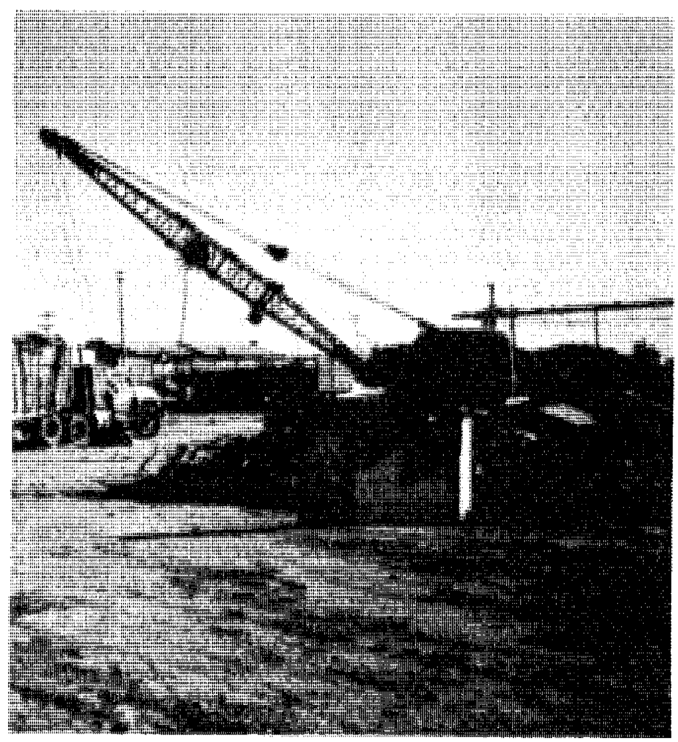
In addition, the NRC has a network of agreements for research cooperation, including funding and personnel exchanges on specific U.S. and foreign projects (see Chapter 11).

Technical Cooperation With Mexico

In 1984, NRC worked closely with the Mexican National Nuclear Safety and Safeguards Commission (CNSNS) in dealing with radioactively contaminated steel products discovered in the U.S. and traced to Mexican manufacturers using recycled materials containing pellets of cobalt-60. The pellets came from a radiotherapy device improperly discarded in a scrapyard in Ciudad Juarez, Mexico. The NRC dispatched a staff expert to Mexico to advise the CNSNS on the cleanup of the Co-60 contaminated scrapyard. Under the terms of the NRC-CNSNS Arrangement for Cooperation, the NRC was able to expedite the approval of procedures to return steel products to Mexico and to facilitate the securing of official permission for American technical personnel to aid the Mexican recovery effort with an aerial survey of the main areas contaminated by the Co-60 pellets. On almost a daily basis, the NRC informed the Department of Energy, the State Department and the Pan American Health Organization of significant developments in the case. This close contact, enabling daily communication, greatly facilitated U.S. response to the threat of the unauthorized import of these contaminated products.







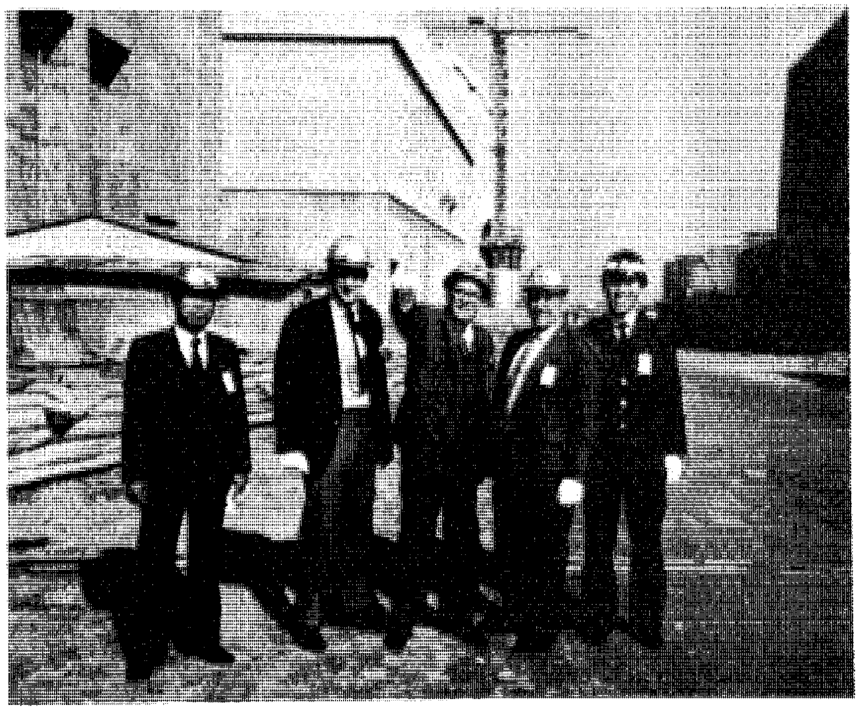
Cooperation between NRC and the Mexican National Nuclear Safety and Safeguards Commission (CNSNS) intensified during 1984 as both countries sought to resolve problems caused by radioactively contaminated steel traced to a Mexican source. The problem came about with the recycling of material in the Yonke Fenix scrapyard in Ciudad Juarez, Chihuahua, Mexico, where a radiotherapy device had been discarded allowing cobalt-60 pellets to scatter. Shown in this photo sequence are:

—radiation experts Greg Yuhas of NRC's Region V office (San Francisco) and Raul Ortiz Magana of the CNSNS at work during cleanup of the contaminated scrapyard (top left);

—Mr. Yuhas briefing cleanup crew members (top right);

—a three-sided wood screen shielding the source of radioactive contamination (lower right, in foreground); and

—truck loaded with the contaminated steel rebar waiting at El Paso, Tex., for clearance to cross the bridge into Ciudad Juarez to return the steel.



NRC Commissioner Thomas M. Roberts (center) and other NRC officials visited the Kori nuclear plant in South Korea in March 1984. The NRC provided technical expertise during an emergency preparedness exercise at the plant.

Foreign Visitors and Training Assignees

Delegations and individuals from 28 countries and four international organizations visited NRC in 1984 for discussions and, on occasion, visits to nuclear facilities and the Department of Energy's national laboratories. The discussions covered safety and policy questions of concern both in the U.S. and abroad, including intergranular stress-corrosion cracking, emergency preparedness, source term assessments, probabilistic risk assessment, safety goals, and the evaluation of operational data.

On-the-job work and training experience continued to be of interest to foreign regulatory organizations during the report period. Assigned to work with NRC staff members were 25 staff members from 11 foreign regulatory organizations: Finland, France, Israel, Italy, Korea, Mexico, People's Republic of China, the Philippines, Portugal, Spain, and Taiwan.

COOPERATION WITH INTERNATIONAL ORGANIZATIONS

Activities in the OECD

The NRC participates in the 24-nation Organization for Economic Cooperation and Development (OECD) through its membership in the Nuclear Energy Agency (NEA), one of the OECD's specialized agencies. This agency brings together specialists from Western Europe, the United Kingdom, Canada, Japan, and the U.S. to exchange information and coordinate joint activities on nuclear technology and safety-related issues.

Two of the major committees of the NEA were chaired by NRC representatives in 1984: William J. Dircks, Executive Director for Operations, served his second year as Chairman of the Committee on the Safety of Nuclear Installations (CSNI) and Richard E. Cunningham served his first year as Chairman of the Committee on Radiation Protection and Public Health. In addition to the usual technical meetings sponsored by the two committees, the NEA sponsored special NRC-supported meetings in 1984 on the basis for regulatory action on pipe cracking in boiling water reactors and on general safety criteria for advanced light water reactor designs.

Safety Assistance in the IAEA

In 1984, the NRC continued to offer safety advice and assistance—in cooperation with the International Atomic Energy Agency (IAEA)— to developing countries initiating nuclear programs. NRC supplied an advisor to Korea for two months to assist with radiation protection inspection and enforcement activities there and another advisor to Mexico to help with its quality assurance program. The NRC also dispatched staff members on short-term IAEA technical cooperation trips to Brazil in order to advise on fire protection safety, to Mexico in order to advise on radiation protection, and to Turkey in order to do a meteorology data review. Two-week courses on PWR Fundamentals, designed to train reactor inspectors, were given by NRC instructors in Brazil and in Yugoslavia. NRC staff members also lectured at an IAEA-sponsored training course in Seoul, Korea, to instruct utility and licensing personnel in performing safety analysis reviews on nuclear power plants. Foreign nationals from a number of countries visited the NRC for technical discussions and to participate in safety-related training courses at the Technical Training Center in Chattanooga, Tenn.

IAEA General Conference

NRC Chairman Nunzio J. Palladino served on the U.S. Delegation to the 28th IAEA General Conference in Vienna, Austria in September. He represented the United States in a special technical session involving the senior regulatory officials of 23 countries. The officials discussed such current regulatory issues as radioactive source terms, the use of probabilistic risk assessment in licensing, and the safety aspects of station blackout. Over 100 observers attended the question and answer period at the conclusion of the session.

International Emergency Preparedness Cooperation

During the year, NRC provided technical experts to join an IAEA special assistance mission to Korea to help observe and evaluate an emergency preparedness exercise at the Kori nuclear plant. This was one of several efforts by the NRC to help foreign safety officials of other countries to be prepared to respond effectively to major radiological emergencies at U.S.-supplied nuclear facilities.

CENTRALIZATION AND AUTOMATION OF FOREIGN INFORMATION

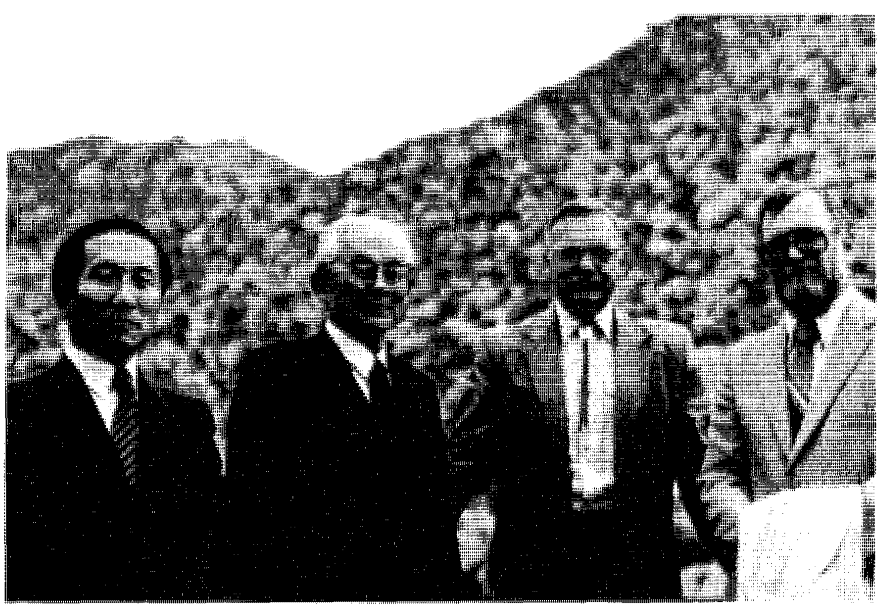
In the past, foreign documents were received at the NRC in both the Office of International Programs (IP) and the Office of Nuclear Regulatory Research. During the report period, the receipt, cataloguing, announcing and distributing of all foreign information was centralized in IP. This move will result in more uniform and efficient handling of the entire foreign document collection. Separately, the growing foreign document collection has been indexed on an automated data base system.

For the past several years, the NRC has been reviewing and evaluating all foreign incident information for possible relevance to the domestic safety program. During the past year, a program was effected by which each important event from all available foreign information was recorded. The summaries were placed on RECON, an automated data management system sponsored by the Department of Energy, which provides rapid access to the information for NRC headquarters and regional staff, as well as NRC contractors.

EXPORT-IMPORT ACTIONS

NRC Export License Summary For Fiscal Year 1984

During the fiscal year ending September 30, 1984, the NRC issued 343 export licenses and 83 amendments to



Commissioner Hideo Uchida (second from left) of the Japanese Nuclear Safety Commission toured the Sandia National Laboratories near Albuquerque, N.M., to observe NRC-sponsored safety research.

existing licenses. Of the licenses issued, 68 were "major" licenses in three categories: special nuclear material, source material, and reactors. The remaining 275 export licenses included 35 for small quantities of special nuclear materials, 18 for source material, 21 for byproduct material and 146 for components and materials. Eleven nations received shipments of special nuclear material under major export licenses during the year. The EURATOM consortium of nations (European Atomic Energy Community) was approved for a major quantity of source material. One license was issued during the period for export of two kilograms of plutonium to Japan.

Nuclear Export-Related Matters

The NRC reviewed several requests involving retransfers of U.S.-supplied nuclear material to other countries for reprocessing. During fiscal year 1094, the NRC reviewed two cases involving retransfers for reprocessing from Switzerland and 15 cases involving such retransfers from Japan. The Commission also reviewed a request involving the transfer of 253 kilograms of separated plutonium from France to Japan for use in the Joyo reactor.

In addition to the foregoing cases, the NRC reviewed 58 cases involving the retransfer of U.S.-origin nuclear material for various end-uses and involving the transfer of U.S.-origin technology abroad, as well as 175 Department of Commerce nuclear-related export license applications. The NRC also reviewed and provided comments to the Executive Branch on the proposed agreements for cooperation between the U.S. and the People's Republic of China and between the U.S. and Finland.

INTERNATIONAL SAFEGUARDS AND PHYSICAL SECURITY

Besides reviewing the implementation of international safeguards and physical security in countries receiving U.S. exports, the NRC continued its active participation in domestic efforts to improve nuclear safeguards. The NRC staff took part in the U.S. Program of Technical Assistance to IAEA Safeguards and in the U.S. Action Plan Working Group to strengthen IAEA Safeguards.

Throughout 1984, the NRC and other U.S. agencies continued to assist the IAEA in the implementation of IAEA safeguards at U.S. facilities, pursuant to the U.S./IAEA Safeguards Agreement. (See Chapter 6 for discussion of domestic safeguards.)

Nuclear Regulatory Research

CHAPTER 11

The NRC's Office of Nuclear Regulatory Research (RES) provides research information needed as part of the technical basis for rulemaking and regulatory decisions to support licensing and inspection activities, to assess the feasibility and effectiveness of safety improvements, and to increase our understanding of phenomena affecting regulatory safety.

The office also has responsibility for developing and coordinating NRC standards—the regulations and guides governing licensed activities of the United States nuclear industry. Regulations are set forth in Title 10, Chapter 1, of the Code of Federal Regulations and are published in the *Federal Register.* Those produced by the NRC in 1984 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which also contains a listing of those issued, revised, or withdrawn during fiscal year 1984.

OPERATING REACTOR INSPECTION, MAINTENANCE, AND REPAIR

Reactor Pressure Vessels

Pressurized Thermal Shock. Under certain postulated accident conditions such as small-break loss-of-coolant accidents, main steam line breaks, steam generator overfilling scenarios, and associated instrument and component failures, a pressurized water reactor (PWR) pressure vessel could undergo a cooling rate nearly as severe as that caused by a large break, but without loss of the internal pressure. This combination of thermal stressing and the action of the internal pressure, called pressurized thermal shock (PTS), could pose a serious challenge to the integrity of the reactor pressure vessel.

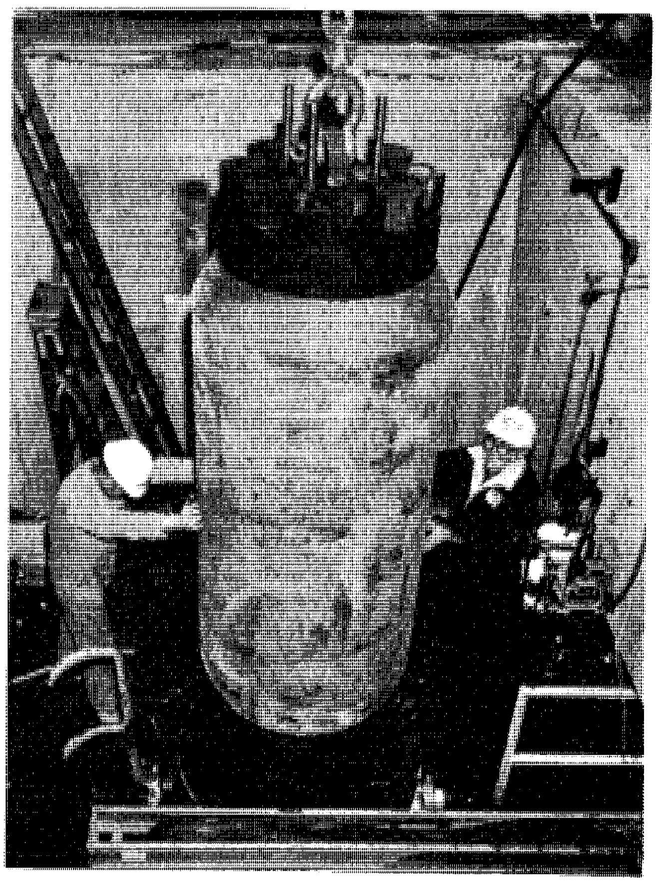
To help resolve this safety issue, the Oak Ridge National Laboratory (ORNL) has completed the first PTS experiment ever performed, on a flawed vessel with a thickness approaching that of a full-scale reactor pressure vessel (see figure). This first of a series of complex fracture tests succeeded in generating and arresting rapidly propagating fractures during PWR overcooling accident-type transients. The experiment confirmed the theoretically predicted fracture behavior and demonstrated the salutary effect of warm prestressing of cracks, a phenomenon that is conservatively neglected in the Commission's criteria for evaluating PWRs. These criteria were also shown to be conservative in other respects, inasmuch as the test vessel survived two episodes of fracture for which the criteria had predicted ruptures. The first experimental vessel was made of reactor vessel steel especially prepared to give it toughness representative of moderately embrittled steel. Preparations are being made for the second experiment in 1985-1986 in which the vessel steel will be highly embrittled, as is projected for some PWRs. Further experiments will be concerned with an important premise of the Commission's current criteria, i.e., that short cracks in clad vessels will always become very long.

Studies that complement the PTS tests are also under way to examine the fundamentals of fracture mechanics for wider ranges of materials and loading conditions. These studies include analyzing large-plate specimens that are deliberately flawed and tested to produce long crack propagations and crack arrest. These crack arrest tests require use of the largest available loading machines in the nation. The first test was conducted at the National Bureau of Standards (NBS) facilities in Gaithersburg, Md., in September 1984. Improved analytical models are providing a basis for better understanding of the margins that current design criteria provide against fracture. Although only unirradiated material can be used in these exceptionally large test specimens, properties within the specimens will be altered to study a wide range of material characteristics from low upper-shelf energy to the gradient imposed by neutron irradiation.

Radiation Embrittlement and Dosimetry. Normal operation of reactors produces excess neutrons from fuel fissioning that can hit the pressure vessel wall, causing it to gradually become more brittle during its lifetime. This process, called radiation embrittlement, has been studied for many years by a number of research teams with the results mainly showing the limits of embrittlement that can be expected for steels and welds during their operating lives. Studies this year at ORNL have concentrated on the more typical and improved versions of reactor vessel steel to ensure that the newer practices have indeed resulted in improvements in performance. The Fourth HSST (Heavy-Section Steel Technology) Irradiation Series was essentially completed this year, and it demonstrated that welds fabricated with low copper content and current-practice procedures experience only slight loss of fracture toughness due to irradiation. This is in contrast to results of previous studies of welds with higher copper contents (as used in some early welds) where severe loss of fracture toughness was observed. Sufficient specimens were included in the Fourth Series for statistical analyses of results, a factor that had been lacking in the past.

Additional irradiations have been started, to be concluded in 1985 and 1986, to quantify the Code-designated trend behavior for the irradiation-induced change in fracture properties that is used to evaluate vessel safety for subsequent plant operation. Finally, as additional support for PTS experiments concerned with the effect of cladding on crack extension, initial test results show that good-practice stainless steel reactor vessel cladding is highly resistant to irradiation damage while poor-practice cladding suffers embrittlement damage.

A critical factor in radiation embrittlement studies is to ensure that the laboratory tests performed and the irradiations conducted in test reactors actually represent the fracture properties of the reactor vessels themselves. One step in the process is to study material removed from the pressure vessel of a retired reactor (see *1983 NRC Annual Report*, p. 112). This has now been accomplished, and 15 four-inch-diameter full-vessel-wall-thickness trepans are awaiting machining for testing to measure the properties. Following several years of negotiations, the NRC has gained the cooperation of the Materialpruefungsanstalt (MPA) at the University of Stuttgart to do the machining



This steel test vessel is being lowered into the pressurized-thermal-shock test facility where it will be subjected to a sudden thermal shock as the temperature drops from 190° C to 14° C (about 550° F to 59° F) and simultaneous pressurization to 94 MPa (13,600 pounds-per-square-inch), which is 1.4 times the design pressure.

and testing as a part of their programs at no cost to NRC. Because the material and operating conditions of the Gundremmingen reactor closely represent those of currently operating reactors, the United States expects to gain much understanding of the properties of vessel wall embrittlement in operating reactors from this study.

Until such confirmatory results are available from retired vessels, reliance has been placed on calculations and predictions of embrittlement and the fluence of neutrons that cause the embrittlement. This discipline is called neutron dosimetry. NRC has been conducting an international neutron dosimetry program for several years with contractor efforts from the Hanford Engineering Development Laboratory, ORNL and NBS, supplemented by top personnel and laboratories from Belgium, France, the United Kingdom, and the Federal Republic of Germany. Researchers from this program have already developed a benchmark for the calculation of neutron fluence in reactor vessel walls, and in 1984 completed analyses of an experiment that allowed for irradiation of test specimens inside a steel block that was built to simulate the wall of an operating reactor pressure vessel. The results of the simulation experiment help confirm the methods used to predict the amount of embrittlement all the way through the thickness of the vessel wall. These results will be reduced to a benchmark case in 1985, and an entire series of standard methods and procedures for prediction and measurement of neutron dosimetry and embrittlement will be completed in the 1986-1987 period.

The result of this analysis was that the embrittlement does not fall off to essentially nothing at the outer vessel surface, but rather that a small but significant amount of embrittlement occurs that must be accounted for in vessel analyses. This finding had an unexpected application in 1984 when a flaw was discovered on the outside surface of the reactor pressure vessel at Indian Point 2. The licensing staff analyzed the effect of the flaw coupled with the embrittlement-induced fracture toughness conditions and concluded that, in this case, there was no safety problem because the embrittlement was sufficiently low and the crack was small. This was the first time that such an analysis was made for an outside surface flaw.

Steam Generators

The Steam Generator Group Project (SGGP) at Battelle-Pacific Northwest Laboratories (PNL) uses a retired-from-service PWR steam generator as a test bed for research on a number of licensing, safety, and reliability issues (see *1983 NRC Annual Report*, pp. 112-113). This steam generator (removed from the Surry nuclear station) represents a unique resource of service-degraded specimens for study. The NRC program has been joined by cosponsoring consortiums from France, Japan, Italy, and the Electric Power Research Institute (EPRI). Principal programmatic goals relate to how reliably steam generators can be nondestructively inspected during service to detect defects and accurately size these defects. The safety issue involved is to what extent (what portion of each steam generator) and how frequently should generators be inspected to ensure safe operation without failures. The program integrates nondestructive examination (NDE) reliability and accuracy with the remaining strength of degraded tubing to establish at what degradation level tubes should be removed from service by plugging.

The SGGP has accomplished several important milestones during its 2 years of operation: (1) the first long-distance transport of a large radioactive component, establishing a mechanism for disposal of other such components in the future; (2) demonstration of dilute chemical decontaminations for reducing worker exposure; (3) the largest effort to date in tube plug removal (which may become desirable to repair tubes or to replace plugs as they degrade), providing guidance on minimizing exposure and downtime; and (4) recent conclusions on staff radiation dosimetry indicating where on their persons work-limiting exposures are received by the staff. During 1984, the program has concentrated on a series of NDE examinations of the generator tubes to determine reliability and repeatability of examinations and also to ascertain the best current methods available for detection and characterization of the many types of cracking, denting, and other degradation that can occur in a steam generator. The validation of these methods began in 1984 with the start of tube removal for burst and leak rate testing, to be completed in 1985. Based on the correlations obtained from the NDE signals and the destructive examinations, improved tube-plugging criteria and tube inspection plans will be proposed in 1986 for use by the NRC staff.

Piping

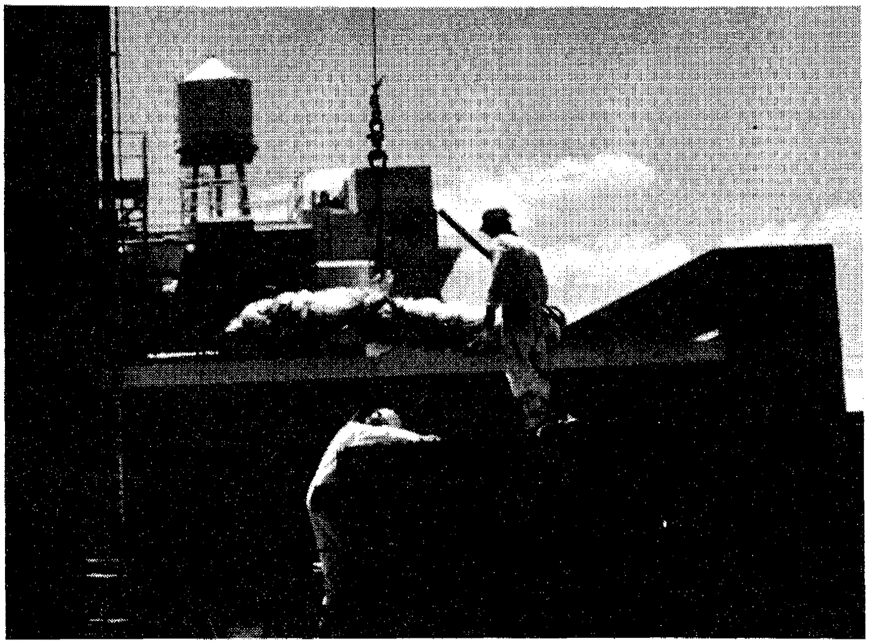
Environmentally Assisted Pipe Cracking. Cracks in the heat-affected zones of weldments in austenitic stainless steel piping in boiling water reactors (BWRs) have been observed since the mid-1960s. Since that time, indications have been found in all parts of the recirculation system, including large-diameter lines. Industry-proposed remedies include procedures that produce a more favorable compressive residual stress state at the inner surface of the pipe, replacement with materials that are more resistant to intergranular stress corrosion cracking (SCC), and changes in the reactor coolant environment that decrease susceptibility to cracking. A program has been under way at Argonne National Laboratory (ANL) (see *1983 NRC Annual Report*, p. 113) to provide an independent evaluation of these remedies. The main areas of investigation during this year have been (1) the effects of water chemistry on the SCC susceptibility of conventional and nuclear grades of austenitic stainless steel, (2) crack growth rate measurements, (3) finite-element studies and experimental measurements of residual stresses in weldments with weld overlays, and (4) the effects of long-term aging at normal reactor operating temperatures.

The effect of impurities and dissolved oxygen level in the reactor coolant on SCC has been studied by means of constant extension rate tests (CERTs) and fracture mechanics crack growth tests. Sulfur species (sulfate, sulfite, thiosulfate, and sulfide) were found to be the most deleterious of the 12 anions studied. Qualitatively, the effects of dissolved oxygen level and sulfate additions in fracture mechanics crack growth rate tests are consistent with those observed in CERTs. However, a much larger relative reduction in crack growth rate is observed in the fracture mechanics tests than in the corresponding CERT when the dissolved oxygen is reduced to very low levels.

The most widely used alternative piping material is Type 316 NG (Nuclear Grade) stainless steel with controlled carbon and nitrogen levels. Previous ANL work suggested that Type 316 NG stainless steel is subject to transgranular SCC (TGSCC) in the presence of impurities. Additional tests have shown that lowering the impurity level decreases both the transgranular crack growth rate and the critical strain rate required to produce TGSCC. In conjunction with the absence of TGSCC in high-purity water, these results confirm that TGSCC in Type 316 NG is directly related to the impurity level and suggest that significant benefits can be achieved by control of the coolant chemistry.

An important new aspect of the work this year has been the use of pipe and components removed from service and replaced with the new NG pipe and components. Materials have already been received from the Hatch 2 plant and from Monticello (see figure); they are being used in metallurgical studies at ANL to help validate the remedies being proposed by industry for the BWR pipe-cracking problem and by PNL for NDE studies. Measurements have been made this year by ANL of throughwall residual stresses on mini- and standard-weld overlays from Hatch 2 materials, as well as from NUTECH (one of the prime contractors for pipe repairs) through a cooperative venture. The weld overlays were prepared by procedures identical to those used to repair reactor piping at Hatch 2. A third mock-up weldment, fabricated using Last Pass Heat Sink Welding, was also examined. The residual stresses on the inner surface of the weldments were very compressive, and the throughwall distributions were in general agreement with those predicted by finite-element calculations.

Laboratory ultrasonic examination, dye penetrant examination, residual stress measurements, metallographic examination, and sensitization measurements were performed on two pipe-to-elbow weldments with overlays from the Hatch 2 reactor. The weldments were removed following about one year of service with the overlay applied. Although 360°indications on the pipe side were reported in both weldments during inservice inspection,



The entire recirculation piping system from the Monticello nuclear power plant (Minn.) is shown as a 12-inch riser is being unloaded to undergo both non-destructive and destructive testing at the Pacific Northwest Laboratory.

ANL examinations showed much more limited cracking on the elbow side in one weldment and no cracking at all in the other weldment. Blunting of the crack tips, presumably due to the application of the overlay, was observed for deep cracks. There was no evidence in any case of mechanical tearing or extension of the crack beyond the blunted region.

Piping Fracture Mechanics. NRC's ongoing programs in the fracture toughness assessments of piping were called on to quickly respond to the needs of the NRC staff for evaluations of the adequacy of the rules being used for evaluation of flaws being found in BWR stainless steel pipe. Early in 1984, the industry proposed new rules for the American Society of Mechanical Engineers (ASME) Code for flaw evaluation of BWR piping; however, the validation of these rules were felt to be inadequate. So a series of tests was started at the David Taylor Naval Ship Research and Development Center (DTNSRDC) at Annapolis, Md., (see *1983 NRC Annual Report*, p. 113) using eight-inch-diameter stainless steel pipe that included flaws typical of those being found in the field and for which the safety analyses were in question. The flaws were actually placed in the pipes by other NRC contractors at PNL using techniques developed earlier on other NRC programs. The tests conducted to date have been extremely helpful in defining for the NRC staff the limits of applicability of the new rules. In fact, partly based on the DTNSRDC results, NRC has declined to accept those rules for adequate evaluation of flaws in BWR piping. These few tests are the forerunner of many more tests on even larger-diameter piping and under more stringent conditions simulating full operation and accident loading. These tests will be conducted in the Degraded Piping II program at Battelle Columbus Laboratories. The first test series is getting under way to validate elements of elastic-plastic fracture mechanics for piping. The contractor is gathering pieces of pipe removed from service for use as test specimen material for subsequent proof tests of the margin against fracture of inservice degraded piping. Because of the anticipated expense of these tests, the NRC has enlisted the participation of EPRI into a follow-on Degraded Piping III program and both intend to jointly propose the entry of overseas participants into this program.

Electric and Mechanical Components

Nuclear Plant Aging Research. The program goals are to identify and characterize aging and service wear of equipment in operating nuclear power plants and to recommend methods for inspection, surveillance, and monitoring of aging and service wear effects prior to loss of safety function so that proper maintenance and timely repair or replacement can be implemented.

A general plan on nuclear plant aging research has been prepared with the research to date focusing on understanding the aging of motor-operated valves and electric cables located inside containment. Age-related data, components, and structures have been identified for further studies. Results of workshops and a survey of aged power plant facilities have been published, and recommendations from the studies have been considered in establishing research priorities.

Decommissioning. The NRC continued to develop an information base for decommissioning LWRs and other nuclear facilities, with five reports published during the year. They cover (1) long-lived activation products in reactor materials (NUREG/CR-3474), (2) utility financial stability and the availability of funds for decommissioning (NUREG/CR-3899), (3) decommissioning a reference independent spent fuel storage installation (NUREG/CR-2210), (4) decommissioning a reference BWR power station (Addendum 2 to NUREG/CR-0672), and (5) decommissioning a reference PWR power station (Addendum 3 to NUREG/CR-0130). A regulation on decommissioning was submitted to the Commission in September 1984.

NRC research to help develop decommissioning standards and guides resulted in topical reports of analyses of measurements of radioactive contamination at Dresden 1, Monticello, Turkey Point, and Rancho Seco. A report summarizing these and earlier topical reports on this subject will be published in 1985. Data needed to evaluate methods, radiation exposure, and costs of decommissioning nuclear facilities are still being collected. Four reports were published during the year.

Spent Fuel Storage. Research continued at the Idaho National Engineering Laboratory (INEL) on the effects of storing irradiated LWR fuel in a dry environment at low temperatures. Both defective and intact BWR and PWR assemblies stored in air and in non-oxidizing atmospheres are being used. Three reports published during the year discussed (1) considerations relevant to dry storage of LWR fuel rods containing water (NUREG/CR-3658), (2) a dry spent fuel storage test plan for final nondestructive fuel rod examination (NUREG/CR-3921), and (3) LWR spent fuel dry storage behavior at 229° C (NUREG/CR-3708). The final nondestructive examination of LWR spent fuel stored in a dry environment was completed. Destructive examination of this fuel was to begin in October 1984. Proceedings were published on the international workshop held in August 1983 on fuel and cladding oxidation during dry storage (NUREG/CP-0049).

The NRC is revising 10 CFR Part 72 to adapt it to the licensing of both short- and long-term storage of spent fuel and high-level waste (HLW) in a monitored retrievable storage (MRS) installation, which are options established in the Nuclear Waste Policy Act of 1982 for management of such materials. An environmental assessment was prepared to support this rulemaking action (NUREG-1092). Regulatory Guide 3.54 on spent fuel heat generation in an independent storage installation was published in September 1984.

Nondestructive Examination

This program includes studies of improved methods for the detection and sizing of flaws during inservice inspection in wrought and centrifugally cast stainless steel, and studies of leak detection methods.

Flaw Inspection by Ultrasonic Methods. The new method for vastly improved detection and sizing of flaws in BWR stainless steel piping called SAFT-UT (synthetic aperture focusing technique for ultrasonic testing; see *1983 NRC Annual Report,* p. 114) is a computer-based testing method that produces high-resolution, three-dimensional images of cracks and other material flaws. It also can distinguish between cracks and geometrical reflectors that might otherwise be considered flaws. PNL developed this method from earlier work done at the University of Michigan and demonstrated it in laboratory tests before putting it into practice this year in a real field test. During 1984, the laboratory system was taken to both the Dresden 3 and the Vermont Yankee reactors to help the NRC staff decide on the validity of conflicting results from two different inspection teams. In both cases, discrepancies were found between two ultrasonic examinations performed according to the requirements of Inspection and Enforcement Bulletins IEB 82-03 and 83-02, which laid out requirements for qualification of the inspectors and equipment to be used to perform inspections such as those done on these two reactors. The SAFT-UT inspections of the same areas revealed excellent images, making it possible to distinguish the cracked areas from the non-cracked areas, thus permitting the NRC staff to make an accurate safety analysis based on accurate flaw sizes.

Flaw Inspection of Centrifugally Cast Stainless Steel. Although the ASME Code requires that cast stainless steel pipe be inspected, the current inspection techniques have not been shown to be adequate. Studies under way on this topic at both PNL and ANL have shown that, for the near term, improvements that may increase the reliability of ultrasonic inspection include (1) the development of methods to establish the microstructure of the material (to help optimize the inspection technique), (2) calibration standards that are more representative of the material to be inspected, and (3) training that uses cast stainless steel (CSS) samples with cracks. For the long term, it will be necessary to establish (1) the variability of the microstructure of CSS, (2) the effect of microstructure on inspection reliability, (3) the improvements possible with electronics, techniques, and transducers designed for CSS, e.g., focused transducers and lower frequencies than those used conventionally, and (4) qualification of requirements for CSS inspections. ANL has demonstrated this year that frequencies of the pulse-echo probe inspection unit, which are lower than normally used in an ultrasonic inspection, improve flaw detection sensitivity. Experiments have also been carried out in the laboratory, as well as in the field, showing that microstructure may be characterized by measurements of ultrasonic pulse-echo transit times and signal amplitudes. After the specific microstructure of a component is known, the proper inspection parameters can be used to help ensure reliable flaw detection and sizing during inservice inspections.

Continuous Monitoring Leak Detection. Leaking in reactor systems is usually associated with the failure of the packing in pumps, valves, or seals. However, leaks can also develop from throughwall cracks and, as such, are the key element in the leak-before-break accident prevention scenario now being actively studied, especially for piping. Thus, the prompt detection, location, and characterization of leaks has recently taken on new importance. A review of the current practice in leak detection carried out by ANL has shown that, for the 74 reactors studied, the current leak detection systems are adequate to ensure a leak-before-break scenario in most situations. However, no leak-location information is available with existing systems. Furthermore, simply tightening current leakage limits may produce an unacceptably large number of unnecessary shutdowns. An evaluation of moisture-sensitive tape has also been completed, suggesting that under the right conditions the tapes can detect leaks on the order of 0.01 gal/min when reflective insulation is used. Despite their sensitivity, the tapes do not provide quantitative information, and a large leak at a long distance can result in the same response as a small leak near the tape.

EQUIPMENT QUALIFICATION

Qualification of Electrical Equipment

Research was completed at the Sandia National Laboratories (SNL), in cooperation with the French, on the evaluation of accelerated aging sequences and loss-of-coolant-accident (LOCA) simulation methods for qualifying safety-related electrical equipment for survival in a harsh environment. An accelerated aging sequence employing irradiation before thermal aging was found to lead generally to polymer degradation most similar to that found in natural aging. Including air (oxygen) in the LOCA test chamber to simulate the in-containment atmosphere was found to increase the degradation of some polymers such as ethylene propylene rubber, but had no effect on other polymers. A follow-on new cooperative research study, again in cooperation with the French, will determine the gamma radiation exposure during qualification that simulates beta radiation damage expected from a LOCA event. SNL also completed an evaluation of the use of terminal blocks and pressure transducers in nuclear power plants, as part of a component assessment program, to identify potential failure modes of instrument and electrical hardware.

Revision 1 to Regulatory Guide 1.89, on environmental qualification of certain electric equipment important to safety for nuclear power plants, was issued in June 1984.

Ten- and twelve-year-old nuclear battery cells (representing the three major manufacturers) were obtained from Fitzpatrick, North Anna, and Calvert Cliffs nuclear power plants and subjected to seismic fragility tests at the Ontario Hydro Research Center. All battery cells withstood a zero peak acceleration of up to one "g" without significant change in electrical capacity. This research was conducted to determine the capability of aged station batteries to withstand design basis seismic events.

Control Systems. The continuing ORNL study of the safety implications of control and associated support systems is concluding a failure modes and effects analysis for a Babcock and Wilcox plant (Oconee Unit 1). A similar study has been initiated for a Combustion Engineering plant (Calvert Cliffs Unit 1). These studies are being performed to support resolution of an important unresolved safety issue (USI A-47, "Safety Implications of Control Systems").

Brookhaven Laboratory has developed a preliminary set of criteria and a methodology to assist the NRC in preparing regulatory guidance on the graded classification of instrument and control systems important to safety.

Instrumentation and Control. SNL completed studies to assess the state of the art of LWR alarm and annunciator systems, including analysis of methods for upgrading annunciator systems.

Idaho National Engineering Laboratory (INEL) completed studies related to evaluating the implementation of Regulatory Guide 1.97, on instrumentation for LWRs to assess plant and environs conditions during and following an accident. INEL also completed an evaluation of existing and improved instrumentation for early steam generator tube leak detection. Work continued at the laboratory on evaluation guidelines for computer-based systems important to safety and on analog and digital devices isolating safety-related from non-safety-related systems.

ORNL completed an assessment of noise diagnostic methods for monitoring and analyzing operational anomalies in LWRs. At Lawrence Livermore National Laboratory, effort continued to evaluate the adequacy of protection of solid state devices against electromagnetic interference. At ANL, an assessment of solid state motor controllers for use in nuclear power plants was completed.

Fire Protection. Research began a phase to augment knowledge of the energy and effluent release characteristics of potential fires in plant areas and of the consequence environment and response of critical safety-related equipment. The tests are designed to improve the capability for fire risk assessment by reducing uncertainties in the determination of plant vulnerability. Research has been started on the effects of potential fires in control rooms and on the capabilities of alternative remote shutdown panels.

SNL has completed reports on cable fire suppression tests indicating water spraying as the most effective method; on cabinet fire tests; and on a systematic study of the occurrence and types of transient combustibles in the plant.

Environmental Qualification of Mechanical Equipment

This research program deals with the qualification of mechanical equipment when subjected to temperature, pressure, humidity, and radiation-type loads. Since the program is relatively new, the results have not progressed to the point where guidelines can be incorporated in the regulatory process. However, the results have provided the NRC licensing staff with important technical information concerning the problems related to equipment qualification under environmental conditions. Some of the areas for which technical information has been provided include methods for qualifying mechanical equipment, effects of PWR internal environments on main coolant pump seal integrity, effects of accident environment on specific containment penetration, seal integrity, and effects of containment accident environment on purge and vent valve leakage.

Dynamic Qualification of Equipment

This research program deals with the dynamic (including seismic) qualification of mechanical and electrical equipment. Since this program is also new, the results have not progressed to the point where guidelines can be incorporated in the regulatory process. However, the results have provided information on methods for qualifying equipment and knowledge of the effects of flow and inlet configurations on purge and vent valve performance. The results have also identified areas toward which dynamic research should be directed. These areas include the identification of component failure modes, quantification factors affecting the fragility of components, and the determination of the influence of foundations on component response.

SEISMIC RESEARCH

Seismic Hazard

The NRC research program in geology and seismology continued along the lines established during the major redirection and refocusing in 1982. The objective of this refocused effort is to better define seismic hazards in the United States east of the Rocky Mountains, to quantify these hazards, using probabilistic techniques where appropriate, and to develop dating techniques to reduce uncertainties in the estimation. Three items that contribute significantly to the uncertainty in seismic hazard estimations are seismic zonation, attenuation of seismic waves, and site-specific response.

Seismographic networks and geological/geophysical studies are being used to establish seismic zonation and to define relationships between crustal features and deep-seated tectonics. Emphasis is being given to developing an understanding of earthquake source parameters, propagation characteristics, and site-specific studies. The NRC continued to broaden the base of support for the seismographic networks among the user community while upgrading them by replacing older station equipment with digital instruments and by deploying additional strong-motion seismographs. A small earthquake (magnitude of 4.1) occurred on April 22, 1984, near Lancaster, Pa. It was felt by utility personnel at both Peach Bottom (about 10 km from the epicenter) and Three Mile Island (about 63 km) and by people as far away as northern Virginia (about 160 km). (See figures: an isoseismal map and a map of the well-located aftershocks. The two studies that produced these figures are improving the knowledge of earthquake source properties and seismic energy propagation in the Eastern United States.)

Recent studies in Oklahoma may have discovered the first capable fault in the United States east of the Rocky Mountains with surface exposure. The Meers fault, located in southwestern Oklahoma, has been found to show signs of geologically recent movement. Although the fault has been known for a long time, it was previously assumed to be of Paleozoic age like many of the faults in the midcontinent area. The NRC is supporting studies that will provide definitive data on the age of the fault's most recent movement. These studies include mapping, trenching, age-dating of organic materials, and low sun-angle photography. Seismographs placed near the faults have not, so far, detected any seismicity on the fault. However, it is not unusual for certain faults to be intermittently active without resulting seismic activity at a given time. Results obtained from these investigations will have an important bearing on considerations related to the seismicity of the midcontinent region.

As part of the effort to determine the cause of seismicity in the United States east of the Rocky Mountains, the NRC has undertaken cooperative programs, including those with the National Science Foundation and the U.S. Geological Survey, to conduct *in situ* stress measurements.

Measurements were conducted in the Moodus Seismic Zone in Connecticut and along the Ramapo fault system in New Jersey and southern New York State. The purpose of the measurements is to determine the direction and magnitude of stress in those areas. This information, combined with knowledge of the structure of the earth at the depths where the earthquakes occur and the distribution of earthquakes, will help clarify the cause of seismicity in those areas.

Preliminary data from the Moodus Seismic Zone suggest that high stresses are limited to the upper 3-5 km of the crust. This is consistent with the local seismicity since almost all earthquakes in the area are shallower than 5 km. Many seismologists believe that the small shallow earthquakes east of the Rocky Mountains result from causes different from those associated with deep ones and that their occurrence does not necessarily indicate a hazard from deeper, larger earthquakes.

Seismic Research Methods Developed

Seismic Safety Margins Research Program. The Seismic Safety Margins Research Program (SSMRP), completed in 1984, has developed a rigorous and thorough method of analyzing seismic risk. Guidelines and procedures for simplified seismic risk analysis methods for PWR nuclear plants were also developed. These methods are to serve as the seismic probabilistic risk assessment model for the Integrated Safety Assessment Program.

The method of analysis developed in the SSMRP involves certain assumptions that must be validated, especially those related to the estimated plant response and fragilities. A program to evaluate methods of estimating seismic risk is now being conducted.

BWR Seismic Safety Margins. A research program is under way to investigate the seismic safety margins contained in BWR nuclear plants. After the LaSalle County Station Unit 2 was selected as the reference plant, the seismic risk analysis of this BWR continued during 1984. Differences among the structure, piping, and system analysis (event tree and fault tree) models are being identified and incorporated into the simplified SSMRP methodology, making it applicable to either a BWR or a PWR.

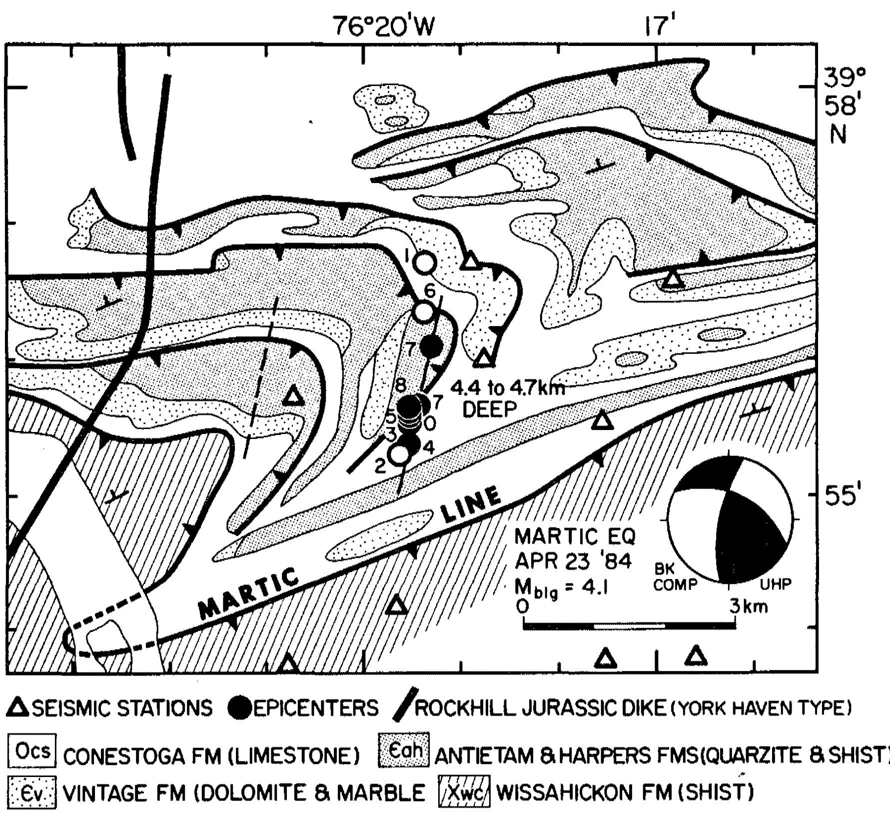
Seismic Design Margins. In 1984, the NRC established a panel of experts in the area of seismic design margins. It will identify regulatory and research needs and outline an approach for future NRC actions in evaluating seismic design margins. The panel met several times in 1984 and interacted with the internal NRC seismic margins working group.

Validation of Seismic PRA Calculations. The SSMRP and other methods currently used to estimate the seismic risks to nuclear plants may vary considerably in their details and sophistication. However, all the methods contain basically the following three elements: (1) an estimate of the seismic hazard, (2) estimates of plant response and fragilities, and (3) a system risk analysis that identifies failure events and estimates the probabilities of undesirable consequences. The methods for estimating plant response and fragilities can most readily be validated through the use of experimental data.

Results from past and ongoing experimental efforts will be evaluated for their applicability in achieving the objectives of this study as outlined above. Among others, the tests conducted at the HDR (Heissdampfreaktor) facility in the Federal Republic of Germany will be used. Significant seismic PRA validation information may also be forthcoming from the one-quarter scale containment model to be located and monitored in a seismically active region of Taiwan. These data will be made available through cooperation between NRC and EPRI.

Piping Response Investigations

Stiff Versus Flexible Piping. A confirmatory piping reliability assessment program being conducted to evaluate stiff versus flexible piping systems indicated that re-



A geologic map of the features in the vicinity of the 1984 earthquake near Lancaster, Pa., shows the preliminary distribution of after-shocks (epicenters on map) and focal mechanism.

moving rigid supports tends, in general, to reduce thermal stress but to increase seismic stress in the pipe. As a result, piping design can be made more reliable by some reduction of rigid supports. It was observed that piping design using snubbers among support devices may not give the intended reliability because snubbers often fail to perform the desired function. It was demonstrated that certain piping systems with snubbers removed actually exhibit higher reliability than the original design. Recommendations as to when snubbers may be safely removed from nuclear reactor piping were developed. This effort was concluded in 1984.

Pipe Whip and Impact. Additional pipe-to-pipe impact testing that simulated more representative pipe spacing in typical nuclear plants was started. In addition, a study on the effects of pipe-to-restraint impact was completed. The results to date of these two programs are consistent with some of the acceptance criteria currently in Section 3.6.2 of the standard review plan (SRP). Also, the study will provide NRC licensing staff with data that can be used for evaluating restraint designs in nuclear plant piping systems.

Pipe Damping Studies. A series of vibrational tests on 3- and 8-inch-diameter carbon steel piping was conducted to determine the changes in structural damping due to various parametric effects. Excitation was provided by low-level hammer impacts, a hydraulic shaker, and a 50-ton acceleration, strain, and displacement time histories.

Further tests planned for this year would involve a more complicated two-or three-dimensional piping system. Tests at high-strain levels will be analyzed, and data at frequencies above 33 Hz will be recorded to assess the effect of damping at higher frequencies.

Substantial literature-searching and regressional analysis have been performed to develop a revised regulatory position on pipe damping and to support the Pressure Vessel Research Committee efforts in this area.

Load Combinations for Piping Systems. The work on Westinghouse and Combustion Engineering PWR pipe leak and rupture probability estimates was extended to Babcock and Wilcox and General Electric reactor systems. Substantial modeling of intergranular stress corrosion cracking was needed to handle BWR piping problems. Results of this work have been used to resolve Unresolved Safety Issue (USI) A-2 and are being cited by the NRC Piping Review Committee. Additionally, resolution of USI B-6, expected in 1985, depends wholly on the outcome of these efforts.

NRC/EPRI Cooperative Pipe Tests. Results of these now completed experiments were reported in NUREG/CR-3893 and indicated that full-scale ASME-Code-designed piping can sustain more than four times the safe shutdown earthquake without any apparent damage. Moreover, in one case, a piping system withstood 9 safe shutdown earthquakes, 5 operating basis earthquakes, and 30 severe shocks without any loss of function or damage.

Mechanical Piping Benchmarks. A major objective of this work is to develop analytical, as well as physical, model-test solutions, which will be used as specified benchmark problems for piping structures subjected to deadweight, internal pressure, and dynamic loads resulting from seismic and non-seismic events. The piping structures deform elastically and may contain gapped supports or snubbers and any other supports used in the nuclear industry.

An additional objective of this work is to fully develop and investigate analytical methods suitable for the evaluation of piping systems excited by multiple independent support motions. Both time history and response spectrum methods and the combinational procedure between the pseudostatic and the inertial components of response are being investigated.

This program provides support in the evaluation and verification of various structural computer programs and analytical procedures presently used in the nuclear industry for the design of nuclear piping systems and components.

The developed benchmarks will be used to determine the acceptability of applicant methods and solutions for piping system analysis. The developed analytical methodology will also be used for determining the levels of conservatism associated with the inertial and pseudostatic responses inherent in multiple-supported piping systems subjected to distinct inputs and for establishing a revised SRP position for their evaluation.

Structural Loading and Response

Characterization of Ground Motion. This research program involved an investigative study aimed at providing guidance and the development of procedures for characterizing earthquake ground motion used in designing nuclear power plant structures. The effort was divided into two separate tasks:

(1) The development of a basis for selecting design response spectra based on free-field motion.

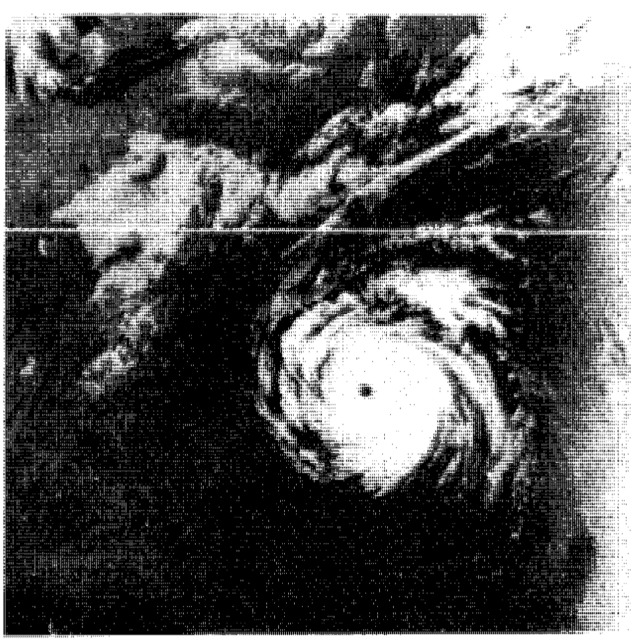
(2) The development of recommendations for methods of selecting design response spectra and time histories to be used as input motions at the foundation level.

Task 1 results demonstrated that both the elastic and inelastic response of stiff structures to free-field ground motion can be adequately approximated by Regulatory Guide 1.60 response spectra anchored to an "effective" peak acceleration for earthquake ground motion of relatively long duration. However, actual plant site conditions often are significantly different from free-field assumptions, and use of design spectra based on free-field motion may be inappropriate. Variations in the site soil shear moduli may cause significant impedance mismatches resulting in reflection of radiation energy dissipated by the structure. In addition, kinematic interaction of the foundation with the surrounding soil for a deeply embedded structure results in wave-scattering of the ground motion.

For these reasons, a consistent approach to the development of foundation-level input design motion should consider the importance of effects such as kinematic and inertial interaction of the structure and soil, structure embedment, soil layering and high-strain non-linearity, earthquake duration and frequency content, and structural non-linearities on overall response. Inadequate consideration of these effects in developing foundation-level design input motion results in earthquake design criteria for nuclear facilities with uneven conservatism.

Seismic Category 1 Structures. The static and dynamic testing of small-scale (1-inch-thick wall) one- and two-floor rectangular reinforced concrete buildings and the dynamic testing of an intermediate-scale (3-inch-thick wall) two-floor building continued during the year. This current series of tests will continue through 1985 in order to demonstrate the applicability of extrapolating scale-model test results to actual nuclear power plant buildings. The overall goal is to assess the ability of Category I structures other than the containment to sustain earthquake loads in excess of their original design bases.

Probability-Based Load Combinations for Nuclear Structures. The project on developing national load combination criteria for design of nuclear structures has been going on for 3-1/2 years. The necessary background work, including the research data base for the statistical description of load-limit states and the development of a computer code on reliability analysis of structures, has been completed. Nine NUREG/CR reports describing the results of the background work are available. The first draft report recommending the probability-based load combination criteria for designing concrete containment structures was issued for comment in 1984. Subsequent reports will address the criteria for other Category I nuclear structures.



The NRC is conducting research on hurricane-operated storm surge. This September 12, 1984 satellite photo from the National Oceanic and Atmospheric Administration shows Hurricane Diana off the North Carolina coast near the Brunswick nuclear power plant.

Standard Problems for Structural Computer Codes. A review of the methods currently used by applicants and licensees to perform soil-structure-interaction (SSI) analysis has been completed. Uncertainties in the SSI analytical methods have been identified and evaluated against experimental and actual earthquake data, including the EPRI SIMQUAKE tests and the Miyagi-Ken-Oki earthquake recorded at the Fukushima nuclear power plant. An effort was started this year to determine the limitations and applicability related to the SSI methods evaluated.

Other External Hazard Research

Severe Weather. Most of NRC's severe weather research was completed in 1984. Technical reports related to near-ground tornado wind fields (NUREG/CR-3874 issued July 1984), the experimental investigation of unsteady tornadic wind loads on structures (NUREG/CR-3848 issued June 1984), and a compilation of violent tornado occurrences between 1880 and 1982 describing tornado intensity based on descriptions of damage (NUREG/CR-3670 issued May 1984) were results of this work. NUREG/CR-3759, concerning lightning strike probability for the contiguous United States, based on records of thunderstorm duration, was issued in May 1984.

Surface-Water Hydrology. The collection of meteorological and oceanographic information related to hurricane activity along the east and west coasts of Florida continued this fiscal year. These data are being collected to test and evaluate the ability of models to predict wave and storm surge heights that may result from hurricanes.

Work continued on the evaluation of models for the ultimate heat sinks for nuclear power plants and the development of guidelines to determine appropriate models for application to specific problems related to several generic designs. These evaluations will be used to assess the anticipated model performance and determine the sensitivity of model predictions to uncertainties in the input data.

Ground-Water Hydrology. A study to investigate and evaluate mitigative techniques and examine generic site conditions for the control or reduction of radionuclide contamination of ground water resulting from a postulated core-melt accident is continuing. The assessment of generic hydrogeologic characteristics is attempting to identify factors that would be expected to affect potential offsite releases and identify possible interdictive options. NUREG/CR-3681, issued in April 1984, reports on the study.

REACTOR OPERATIONS AND RISK

Risk Analysis

Risk Assessment Methods Development. In 1984, work began on the Risk Methods Integration and Evaluation Program (RMIEP). Through the mechanism of a full-scale probabilistic risk assessment (PRA) on the LaSalle 2 nuclear plant, RMIEP is using existing and newly developed internal, external, and common cause risk methods coupled with improved human factor models to develop an integrated logic model for internal and external events, including dependent failures. Portions of the methods involved require expert judgments where data are lacking. Reports were issued in 1984 on methods for quantification of informed opinion and on some experimental results gained from eliciting subjective judgments.

The NRC continued to collect and analyze data from selected power plants, including failure data reports on pumps, diesel generators, batteries, and inverters. A plan for an integrated risk assessment data acquisition program was developed. By building on existing data programs, this plan will lead to the acquisition of data necessary to meet current and future regulatory needs in the area of risk assessment.

Methods Development for Risk Reduction. As part of the Severe Accident Risk Reduction Program, NRC has studies under way at Sandia National Laboratories to determine the cost effectiveness of a number of reactor design alternatives such as filtered-vented containments that have been proposed as offering the potential for a significant reduction in accident risk. These studies are focusing on those severe accident sequences that dominate the risk for a particular plant or design and are evaluating the effectiveness of proposed risk reduction alternatives in mitigating consequences for all current United States plant designs. A draft study to examine the effectiveness of a filtered-vented system for a BWR with a Mark III containment has been completed. This study concluded that the effectiveness of the filtered vent depends on the magnitude of the accident source terms. For this reason, this program has been integrated with continuing NRC research into accident source terms. An initial reassessment of accident source terms is expected in 1985.

Risk Assessment Application for Acceptable Risk Level Maintenance. During 1984, work continued to develop a methodology and a data base that could be used to relate information on the contributors to plant risk gained in probabilistic risk analyses to NRC inspection decisionmaking. The objectives of this work are to aid inspection personnel in setting priorities for their limited resources by identifying where their activities may have the greatest potential for reducing the failure probabilities of the systems and components that are most important to plant safety. A methodology involving the use of systems and component importance measures coupled with root-cause failure data has been developed that directly relates risk to an inspection function. Work is presently being undertaken to develop and interpret importance measures that can be used in this program as well as to develop methods and a data base that will permit the use of root-cause information. The program is presently being studied by NRC's regional personnel to determine how it should be focused to assist the Regions in making decisions concerning priorities in their inspection activities.

Human Factors

Human Reliability. This program provides the research necessary to support human reliability evaluations of nuclear power plants, especially those evaluations employing PRA methodologies. The products of this research support resolution of human reliability issues raised in the TMI Action Plan (NUREG-0660) and in the NRC Human Factors Program Plan (NUREG-0985, Revision 1). Major products of 1984 research included methods for assessing human error probabilities using psychological scaling, computer modeling, and multiple-failure sequencing approaches; human reliability data bank specification; a method for integrating human reliability analysis more fully into the PRA process; and a method for systematically using PRA results to resolve pertinent human reliability safety issues. Ten publications reporting research completed under this program were issued during 1984.

Organization and Staffing. This program provides the research necessary to establish objective safety-related performance measures for assessing organizational effectiveness at utilities with operating and near-operating nuclear power plants and to establish suitable personnel staffing requirements to ensure safe operation and maintenance during normal and abnormal conditions. The products of this research also support resolution of safety issues raised in the NRC Human Factors Program Plan. Major products of 1984 research included an initial set of safety-related performance measures for assessing plant organizational effectiveness and methods for comparing safety-related performance of different control room staffing configurations in responding to normal and abnormal plant conditions. Three publications dealing with research completed under this program were issued during 1984.

Operational Readiness. This program provides the research necessary to support upgrading of NRC operator training and licensing requirements and nuclear power plant operating and maintenance procedures to ensure safe and timely responses to normal and abnormal plant conditions. The products of this research support resolution of safety issues raised in the NRC Human Factors Program Plan. Major products of 1984 research included methods for identifying operator training requirements to respond to normal and abnormal conditions, criteria for specifying the appropriate role of training simulators in the NRC licensing process, and methods for establishing criteria to assess the adequacy of symptom-based operating procedures. Five publications dealing with research completed under this program were issued in 1984.

Man-Machine Interface. This program provides the research needed to develop a technical basis for NRC evaluation of man-machine relationships in central control rooms or other control stations. Research is being conducted to assess and recommend human factors standards and guidelines for new or improved designs that may be introduced into existing control stations in order to improve the operator and maintenance personnel man-machine interface. Significant accomplishments include a series of experiments on operator fault diagnosis and problem solving using artificial intelligence as an aid and parametric displays for identification of system status. The results of the control room crew task analysis were analyzed to identify supervisor and operator skill and knowledge requirements, the differences between function-based and event-based procedures, and operator use of alarms and annunciators. A review of operator aids for control room personnel was completed, and a classification structure of operator decisionmaking tasks was developed. Six publications from this program were issued in 1984.

Accident Management. Accident management research provides a technical basis to evaluate the contribution of operating plant personnel in the mitigation and arrest of the accident sequence, to establish how emergency operating procedures could be improved to reduce the likelihood of off-normal conditions degrading to the level of a severe accident, and to identify what unique man-machine interfaces and operational personnel training requirements reside in accident management. The products of this research support resolution of safety issues raised in the Nuclear Power Plant Severe Accident Research Plan (NUREG-0900). In 1984, the anticipated transient without scram (ATWS) sequence was analyzed from a human factor perspective to identify the procedures, the man-machine interactions, and training that could affect significantly the sequence's outcome. Reliabilities of critical operator actions were estimated, and the displays and alarms used in accident management were assessed. A review of existing NRC regulations related to accident management and the industry response to these regulations was performed to provide a framework for considering the NRC's role in accident management. Two reports were published under this program in 1984.

Emergency Preparedness

This program provides research to develop a technical basis for monitoring, assessing, developing, upgrading, or clarifying emergency preparedness for nuclear power plants and certain fuel cycle and material licensees. Research included evaluation of protective action strategies, emergency-action-level approaches used by nuclear power plant licensees. A rule change relating to the frequency of full participation by State and local governments in emergency preparedness exercises was issued in 1984.

Atmospheric Dispersion. Research efforts pertaining to atmospheric dispersion have concentrated on completing the analysis and documentation of the meteorological and tracer data collected during previous field tests (see *1983 NRC Annual Report*, p. 127) to evaluate atmospheric dispersion models. This is being done to identify those capable of, and suitable for, real-time predictions of the atmospheric transport and diffusion of effluents through the airborne pathway during and immediately following an accidental radioactive release from a nuclear power plant, and thus to aid in emergency response decisions. NUREG/CR-3488, Volumes 1 and 2 (October 1983 and April 1984, respectively), provides information related to the Idaho field experiment, and NUREG/CR-3456 (November 1983) summarizes the Hanford Field tests conducted in the 1960s. A report on modeling the mesoscale diffusion and transport processes for releases within coastal zones during land/sea breezes (NUREG/CR-3542) and a report addressing the variation of dispersion properties with height (NUREG/CR-3773) were issued in December 1983 and May 1984, respectively.

Studies at the Idaho National Engineering Laboratory are continuing to measure the washout and wet depositive factors for the chemical forms of airborne radioiodines released to the environment during periods of precipitation and fog. Laboratory studies involving methyliodine, including retention of methyliodine on several vegetation forms, have been completed and a topical report is being published.

Transportation Safety Research

Data from licensees were collected and computerized during 1984 on numbers and characteristics of radioactive material shipments in the United States. The updated shipment information will be used to reevaluate the impact of radioactive material shipping on transport workers and the general public in a revision of the NRC's generic environmental impact statement on transportation, originally published in 1977.

Efforts to develop information on the interactions between explosives and LWR spent fuel shipments were completed. The goal of this effort was to understand the magnitude of the potential radiological consequences if these shipments should be subjected to specified explosive threats. The research results, including the findings of a peer review effort, were summarized for use by the NRC licensing staff in formulating appropriate safeguards measures for spent fuel shipments.

Fuel Cycle Risk Analysis

A series of tests on the release of radioactive material as the result of a fire was completed, and the results were incorporated into updated models describing the amount and characteristics of radioactive aerosols generated by the combustion of contaminated fuels.

The development of methods for analyzing accidents involving uranium hexafluoride was completed. The methods are being applied to support the rulemaking on the need for emergency response at nuclear fuel cycle facilities.

Materials Safety

In September 1984, the NRC amended its regulations to delete an exemption from licensing requirements for the receipt, possession, use, and transfer of glass enamel and glass enamel frit containing source material and any products containing these materials such as radioactive cloisonne jewelry. Because non-radioactive alternatives exist to produce the colors produced by using uranium and because there is no benefit associated with the exemption, the radiation exposure from the radioactive glass enamel and glass enamel frit cannot be justified.

THERMAL-HYDRAULIC TRANSIENTS

Best-estimate systems codes and evaluation model computer codes are two basic computer tools for analyzing nuclear power plant safety. Best-estimate systems codes offer a way to apply the results from reactor safety research to evaluations of accidents because they encompass the whole reactor coolant (RC) system. Evaluation model codes provide conservative analyses for use in independent audits of licensing calculations.

RC experimental programs comprise the separate effects and integral systems tests needed to support the improvement and assessment of these computer codes. These experiments and computer codes assist the licensing staff in resolving licensing and safety issues, which arise with regularity. During 1984, work was performed to improve the usability of the codes through the nuclear plant analyzer and data bank programs, while programs for the assessment of the codes using experimental data were expanded to include several international partners. Application of the codes continues in support of licensing issues such as pressurized thermal shock, revision to Appendix K, and core liquid depression during a small-break loss-of-coolant accident (LOCA).

Separate Effects Experiments

FLECHT SEASET. Testing and data analysis on single-phase natural circulation, two-phase natural circulation, and reflux condensation were completed and a final report published. Flow blockage model development using the COBRA-TF code to analyze the blocked bundle data was completed, and a final report on the program will be published in early 1985. (This program is jointly sponsored by the NRC, the Electric Power Research Institute (EPRI), and Westinghouse Corp.)

Model Development. Most NRC model development occurs at universities and is aimed at supplementing separate effects experiments, helping to interpret data from larger test programs, and developing correlations based on a new understanding of the phenomenology (see the *1981 NRC Annual Report,* p. 124). A small experimental loop facility at the University of Maryland was constructed to achieve a better understanding of various system transients (e.g., loop oscillations, natural circulation interruption) common to Babcock and Wilcox (B&W) reactors. This new facility will support the larger Multiloop Integral Systems Test (MIST) program (see Integral Systems Tests below).

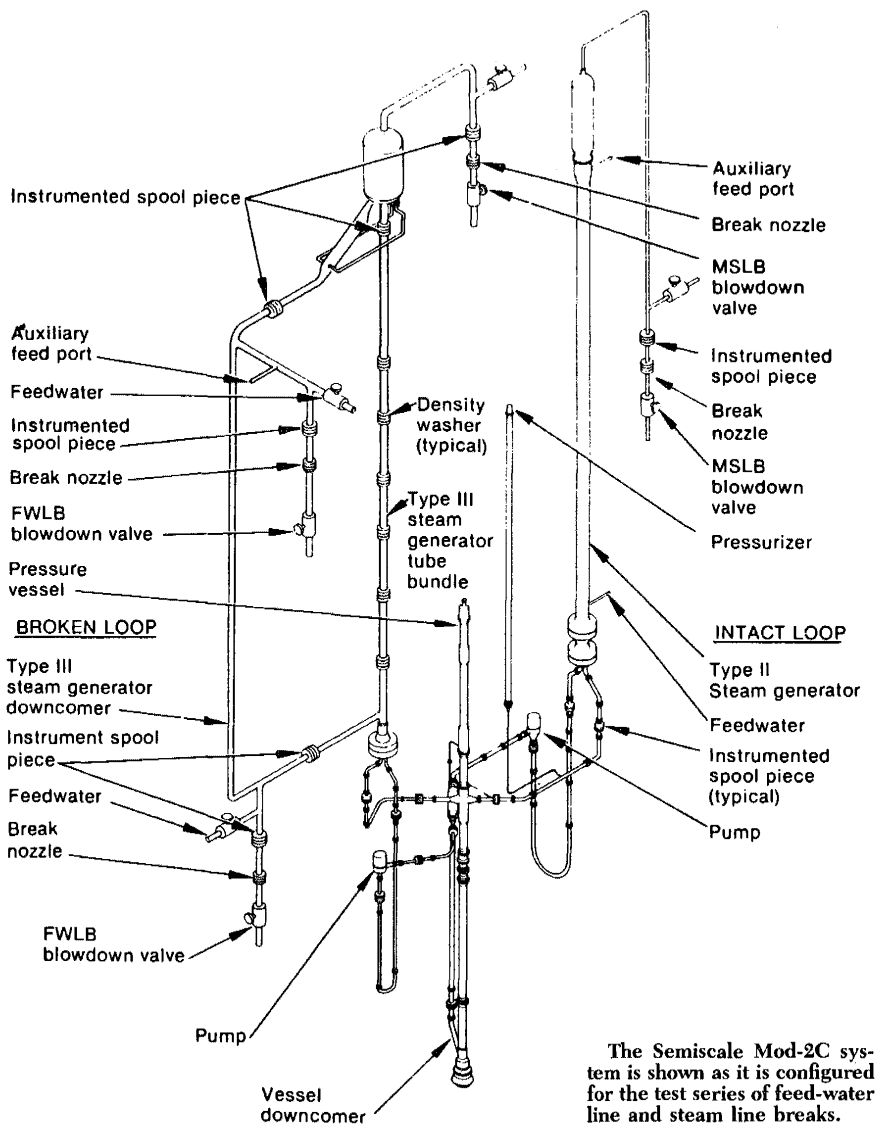
THL/Critical Flow Experiments. Codes used for loss-of-coolant analyses by NRC do not always correctly model break flows in small piping connected to the primary coolant loops in an LWR when two-phase, stratified flow is present. Experiments were completed in the Thermal Hydraulic Loop at INEL, jointly funded by NRC and EPRI, to provide critical flow data for a variety of representative conditions, including stratified flow and broken pipe orientation.

MB-2. A steam generator test program, Model Boiler-2 (MB-2), operated jointly by Westinghouse, EPRI, and NRC, produced data simulating accident conditions resulting from steam generator tube rupture and steam line break.

Integral Systems Tests

The NRC has been the major source of support for the Loss-of-Fluid Test (LOFT) and Semiscale PWR test facilities at INEL, although approximately 10 percent of LOFT support has come from foreign countries. Since early 1983, the LOFT facility has been operated by the Department of Energy for a consortium of which NRC is a member. Other United States integral facilities include the Full Integral Simulation Test (FIST), a BWR test facility supported almost equally by the NRC, EPRI, and the General Electric Co. (GE); and the Integral Systems Test (IST) program sponsored by B&W plant owners, B&W, EPRI, and NRC. In addition, the NRC participates through international agreements in the 2D/3D facilities in Germany and Japan and the ROSA-IV facility in Japan.

LOFT. This large-scale integral systems test facility, used to simulate reactor accidents, performed two tests in 1984. The first was a large-break loss-of-coolant accident simulation that was initiated at PWR licensing limits and simultaneously simulated a loss of off-site power. In addition, minimum United Kingdom safeguard assumptions for emergency core cooling (ECC) injection and rapid primary coolant pump coastdown were used. A peak clad temperature of 1,810°F was reached during blowdown and 1,803° F during refill/reflood. No fuel failures oc-



The Semiscale Mod-2C system is shown as it is configured for the test series of feed-water line and steam line breaks.

curred. A comparison of this test with three previously conducted experiments showed differences in the primary system thermal-hydraulic responses. These differences are largely due to differences in primary coolant pump operation and, to a lesser extent, differences in ECC injection and initial core power. The second test was a small-break loss-of-coolant experiment that simulated a 4.67-cm-diameter single-ended break in the cold leg of a three-loop PWR. Coincidentally, failure of the high-pressure-injection system was assumed. The reactor coolant pumps (RCPs) were left running for a period of time to increase the mass depletion rate. The RCPs were tripped after a period of time, and the coolant system continued to lose mass at a slower rate resulting in gradual core uncovering. The break was closed, and, when core temperature reached a pre-selected value, steam generator feed and bleed was begun. This lowered the primary system pressure allowing accumulator injection to occur and reflooding the core.

Semiscale. During 1984, the steam generator tube rupture test series was completed. A total of nine tests were performed to investigate the effects of several parameters, e.g., number of tubes ruptured, operator recovery procedures, and additional simultaneous failures. Tube rupture was simulated by injecting primary coolant water into the secondary side of the steam generator. After the tube rupture tests were completed, the facility design was modified to perform experiments on the loss of secondary coolant by steam line or feedwater line breaks. (See figure for an isometric view of the modified facility.) Loss of secondary coolant is important to light-water-reactor safety because it may result in pressurized thermal shock (steam line break) or primary coolant system overpressurization (feedwater line break). System checkout was under way at the end of this year and testing will begin in January 1985.

BWR FIST Facility. The second phase of testing in the FIST facility (see the *1983 NRC Annual Report*, p. 121) was completed. This marks the conclusion of over a decade of BWR transient research sponsored by NRC, EPRI, and GE. Other programs that have been part of this jointly sponsored research are the Two-Loop Test Apparatus (see *1980 NRC Annual Report*, p. 199), the BWR Countercurrent Flow Limit Refill/Reflood Program (see *1980 NRC Annual Report*, p. 199), and the TRAC-BWR computer code (see Code Assessment and Applications below). This research has significantly improved our understanding of the performance of the ECC systems during LOCAs in BWRs and has permitted NRC approval of a more accurate BWR evaluation model, SAFER, submitted by GE. Use of SAFER by individual licensees will result in a significant reduction in unnecessary plant operating restrictions and has the potential to result in a reduction of plant operating costs.

IST Program. The Integral Systems Test (IST) program was initiated in 1983 to conduct integral tests representative of Babcock & Wilcox (B&W) plants. The program includes the Once-Through Integral Systems (OTIS) test facility, which will simulate raised loop B&W plants, and the Multiloop Integral Systems Test (MIST), which will represent lowered loop B&W plants. During 1984, the entire series of planned OTIS tests was successfully completed, and data from tests simulating small-break LOCAs were provided for the verification of analytical models used in the advanced systems codes. The effects of break size, break location, leak isolation, steam generator characteristics, feed and bleed cooling, and natural circulation cooldown were investigated by the 15 OTIS tests. Following completion of the OTIS testing phase, facility upgrade to the 2x4 MIST facility was initiated. Testing in MIST is scheduled to begin in October 1985.

2D/3D Program. Under this joint research program with Germany and Japan to study PWR LOCAs, the Japan Atomic Energy Research Institute (JAERI) completed the Core II test series in the Cylindrical Core Test Facility. JAERI also completed nine of the 20 tests planned in the Slab Core II Test Facility. Both facilities have 2,000 electrically heated rods, although their configurations are different. These two facilities are the largest large-break LOCA research facilities in operation today. Data obtained to date indicate that the uncovered reactor core is effectively cooled by the two-phase mixture flowing up through the core. The liquid carryover to the region above the quench front is significant, and a substantial degree of cross flow exists, resulting in a fairly uniform progression of the quench front in the vertical direction, in spite of power density differences in the lateral direction.

The Federal Republic of Germany has almost completed the construction of the Upper Plenum Test Facility at Mannheim, and is planning to start the shakedown tests in April 1985 and the main tests in October 1985. This facility will offer the opportunity to study, in full scale, deentrainment of liquid in the upper plenum and the ECC bypass around the downcomer, as well as the countercurrent-flow-limitation phenomenon in hot legs, which is a concern in small-break LOCAs.

Construction of the ROSA-IV facility in Japan continued in 1984. This large facility will study small-break LOCAs and operational transients in PWRs. The NRC provided advanced instrumentation to help improve understanding of test results and experimental phenomena.

Code Assessment and Applications

Code Improvement. Work continued on several best-estimate codes during 1984: (1) Further improvements were made to TRAC-PF1/MOD1, used to analyze system transients that require a complete simulation of PWR plant controls and balance-of-plant systems. This code is also capable of analyzing loss-of-coolant accidents (LOCAs) since it contains models similar to its predecessor codes, i.e., TRAC-PD2 and TRAC-PF1. (2) TRAC-BD1/MOD1, used to analyze the same aspects of boiling water reactors, was completed, and development of an interim version of the fast-running version, TRAC-BF1, was also completed. (3) The COBRA-TF code to analyze flow blockage and rod swelling effects upon the cooling of a fuel assembly was completed.

Code Assessment. Independent assessment of TRAC-PF1/MOD1 and TRAC-BD1/ MOD1 continued and that of RELAP5/MOD2 was started. Important contributions in the assessment program will be made through bilateral agreements, as discussed below.

Code Applications. These computer codes continued to be used to address licensing concerns. Calculations in support of evaluation of pressurized thermal shock were completed using TRAC-PF1/MOD1 and RELAP5. These codes are also being used for calculations of experimental systems and large PWRs to support experimental programs such as MIST and OTIS. Best-estimate calculations for large-break LOCAs were performed using TRAC-PF1 and TRAC-BD1 to support potential revisions of Appendix K to 10 CFR Part 50.

Pressurized Thermal Shock. In order to develop licensing guidance on the pressurized thermal shock (PTS) issue, detailed thermal-hydraulic calculations have been performed for three specific nuclear reactor systems using TRAC and RELAP5. Design information for these analyses was provided by Duke Power Co. (B&W), Baltimore Gas and Electric Co. (Combustion Engineering), and Carolina Power and Light Co. (Westinghouse). These calculations were supported by thermal-fluid-mixing experiments performed at Purdue University and Creare.

The results were used to determine the likelihood of cracking a reactor pressure vessel by inadvertent overcooling. Because of the international interest generated by this program, bilateral agreements have been signed with Imatran Voima OY of Finland and Kernforschungszentrum—Karlsruhe of Germany to exchange information on PTS and materials research.

Plant Analyzer and Data Bank. The plant analyzer includes calculational tools to easily and accurately analyze plant transients. This is, in effect, the end product of this research area. Three concepts are currently being pursued: (1) to make use of existing codes such as TRAC and RELAP5 but make them faster and easier to use, (2) to investigate new computing techniques to speed calculations, and (3) to harness personal computers to make simplified calculations of the complex interactions calculated during the transient. Speed, ease of use, and ease of interpretation of results are essential if the calculational tools developed under this research area are to provide maximum benefit.

During 1984, the data bank software for plant geometries, control systems, and operating data was largely completed. The data bank provides automated input data describing specific plants for a user's interactive solution of nuclear reactor accident problems using TRAC and RELAP5. Vectorized versions of TRAC and RELAP5 allowing the use of the fastest existing computers coupled with improved new two-step numerics provide new speed and efficiency. An interactive version of TRAC was created. A common workstation was selected and procured at Los Alamos National Laboratories, INEL, NRR, and RES so that each user sees the same display with the same interpretation of symbols, colors, commands, and messages independent of which code or on which mainframe computer it is running.

International Thermal-Hydraulic Agreements. Several cooperative bilateral thermal-hydraulic research agreements were concluded. In these cooperative programs, the NRC provides computer codes and necessary training tailored to individual countries. In return, these countries contribute to nuclear safety by providing NRC with unique test data and code assessment studies. Results of the code assessment work from these cooperative programs will be made available worldwide, enhancing the safety of nuclear reactor operation in all countries.

SEVERE ACCIDENTS

Accident Likelihood Evaluation

In 1984, accident likelihood evaluation continued to develop LWR accident sequence information to support the source term reassessment, the implementation of the Severe Accident Policy Statement, and other regulatory issues. The accident sequence evaluation program with support from the accident sequence precursor analysis program continued to develop a reliable source of information usable in probabilistic risk assessments (PRAs), to update accident sequence likelihood estimates, and to expand on the accident sequence likelihood insights previously developed for operating and near-term operating LWRs. A draft report summarizing the dominant accident sequence information from 12 PRAs and identifying the factors that drive the sequence likelihoods was prepared. The accident sequence likelihood information on the six Severe Accident Research Program (SARP) reference plants was updated based on insights from recent safety studies, operating experience, and post-TMI fixes. Several reports were published summarizing plant-specific design information and fault tree/event tree model development that will be used to further analyze reliability and risk features of the various LWR designs. A report was published summarizing the analysis of the most recent precursors (1980-1981) to potential severe core damage accidents.

Severe Accident Sequence Analysis Program

The Severe Accident Sequence Analysis (SASA) research program is continuing to assess power reactor response to possible sequences of events beyond the design basis accidents. In addition to ongoing studies at four of the national laboratories—Los Alamos, Idaho, Oak Ridge, and Sandia—the Brookhaven laboratory has undertaken studies of BWR events using the RAMONA-3B code during 1984.

PWR "front end" (transients up to the start of core damage) responses to postulated event sequences have been analyzed by Los Alamos and Idaho, while the "back end" (transients following the start of core damage through containment damage) responses have been analyzed by Sandia. BWR "front end" responses are being analyzed at Oak Ridge, Idaho, and Brookhaven, while "back end" responses are being analyzed at Oak Ridge.

The Los Alamos program for 1984 included:

- Feed-and-bleed analyses in support of Unresolved Safety Issue A-45, "Shutdown Decay Heat Removal Requirements," were completed and documented.
- The analysis of unmitigated boron dilution events for B&W and Combustion Engineering (CE) plants was completed and documented.
- An analysis of core damage sequences in PWRs using TRAC/MIMAS was initiated.

The Idaho program included:

- Detailed analyses of two risk-dominant core damage transients on the Bellefonte PWR, using the RELAP5 and Severe Core Damage Analysis Package (SCDAP) codes to analyze station blackout and small breaks with failure of emergency injection.

- Analyses of the Browns Ferry Unit 1 BWR anticipated transient without scram (ATWS), using the RELAP5 and SCDAP codes to determine reactor system and core response and the CONTEMPT-LT/028 code to compute containment thermal-hydraulic and structural loads response.

- Analyses of the Seabrook PWR station blackout transient with emergency system failure.

The Oak Ridge program included analyses of dominant severe accident sequences for the Browns Ferry Unit 1 plant. Studies were reported on accident sequence analyses for ATWS and fission product transport analyses for the loss of decay heat removal accident sequence on the Browns Ferry Unit 1 plant, MK-1 pressure suppression pool modeling, and the MARCH BWR code modifications for containment analyses in ATWS studies. The user manual is in preparation and a test of HEPA filters for plugging and subsequent tearing and collapse is in progress.

The Sandia program included analyses of pressure-temperature response for a variety of severe accidents. A report was issued on structural analyses of deformation-level loads at Yankee, Watts Bar, and Bellefonte. Calculations were performed for several Containment Loads Working Group standard problems. Coding links were produced to connect MARCON source terms to HECTR and CONTAIN. Currently a summary is being prepared of the work performed on the standard problems for the Containment Loads Working Group with emphasis on the potential impact of this work on severe accident studies.

The Brookhaven program in 1984 provided best-estimate calculations for the Browns Ferry Unit 1 ATWS transients with the RAMONA-3B code. Browns Ferry Cycle 5 nuclear data or cross sections that will be used in the final RAMONA-3B calculations for ATWS transients were generated at Brookhaven.

Behavior of Damaged Fuel

Severe Fuel Damage Test. A Severe Fuel Damage Test (SFD1-3) was successfully performed in the Power Burst Facility (PBF) at the INEL in August 1984. The test fuel bundle and experiment procedures were very similar to the previous experiment (see the *1983 NRC Annual Report*, p. 117) except that the fuel rods were irradiated rods provided by Belgium (one of the foreign partners in the SFD program).

ACRR Experiment on Debris Formation. The debris formation (DF-1) experiment was successfully performed in the Annular Core Research Reactor (ACRR) at Sandia in March 1984. DF-1 is the first in a short series of separate effects experiments which examine processes that occur during fuel damage development and hydrogen generation under core-uncovery accident conditions. The heat associated with rapid oxidation of the zircaloy cladding by steam produced extensive liquefied fuel (fuel dissolved in unoxidized molten zircaloy) with attendant relocation (slumping).

Coolant Boil-Away and Damage Progression Test. The Materials Test 6B (MT-6B) was successfully performed in the National Research Universal (NRU) reactor in Canada by PNL in June 1984. The test bundle contained 12 full-length (three-meter) fuel rods. A peak temperature of 1,600° Kelvin was reached, verifying the bundle design and operation for subsequent tests.

Hydrogen Generation and Control

In this program, ways of preventing deflagrations and detonations and schemes for mitigating the effects of hydrogen burns in LWR plants are assessed. Two reports were published during the year covering (1) a feasibility study on the use of deliberate flaring to rid the primary system of hydrogen (NUREG/CR-3638), and (2) experiments on the effectiveness of igniters in the Variable Geometry Experimental System (VGES) facility (NUREG/CR-3273). Information was obtained on the operability of Tayco and GM igniters in a water spray environment and coordinated with NRC licensing activities.

Tests exposing electric equipment and cables to both premixed global hydrogen burns and continuous hydrogen injection burns were completed at the Nevada Test Site (1/20th-scale facility) under EPRI program management with NRC participation. The equipment degradation results from the tests are being evaluated by EPRI and its contractor. The NRC is planning to evaluate the ability of the Sandia National Laboratories (SNL) HECTOR/HYBER hydrogen burn code to accurately describe equipment thermal response against the Nevada Test Site data. (The HECTOR code is designed to analyze hydrogen burning in a full range of containment types, assessing the resulting temperatures and pressures. HYBER is a more of a special purpose code designed to model the combustion processes in nuclear containments and to estimate the thermal response of safety-related equipment subject to combustion and post-combustion environments.) Separate effects tests are to be made this coming year at the SNL Central Receiver Test Facility in which equipment degradation and associated thermal response will be measured for simulated hydrogen burns. The separate effects test results will be integrated with code calculations to establish criteria for evaluating equipment survival for a hydrogen burn from a 75 percent core-melt/water reaction in a large, dry, full-scale pressurized water reactor containment.

Fuel-Structure Interaction

Comparison of the concrete attack by molten steel and by molten uranium was made. The effect of the delayed addition of water in mitigating the concrete attack is being investigated. A method for simulating decay heat is being applied in sustained and hot solid tests to study the long-term effects of the core-debris/concrete interaction. High-pressure melt ejection tests were successfully carried out in an open environment, and plans to conduct similar tests in a scaled containment are under way.

Containment Analysis

The CONTAIN 1.0 computer code is an integrated systems level analytical tool for computing the abnormal loads imposed on reactor containment structures during severe accident conditions. Evolution of the potential radiological source term is tracked and the information made available for environmental-consequence computation in the event of containment failure. The first version, CONTAIN 1.0, together with documentation and user information, was subjected to formal technical peer review August 7, 1984. The code continues to be made available to qualified foreign and domestic research organizations.

The CORCON MOD2 code computes the source terms for containment load calculations that arise from the interactions between molten core debris and the structural concrete in the reactor cavity. Improvements over the MOD1 version include the freezing of debris and interactions with overlying water that permit extended application to long-term severe accident consequences. CORCON MOD2, after intensive technical review, has been released to the reactor safety community and is incorporated into experimental validation programs in the United States and in the Federal Republic of Germany.

Fission Product Release and Transport

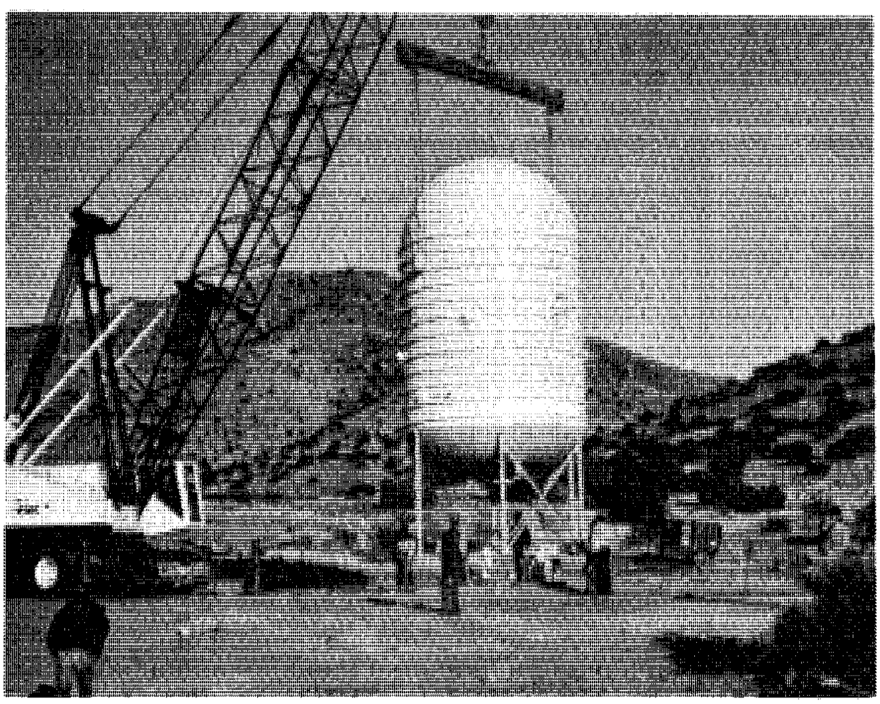
This program develops computer models and obtains experimental data to determine the radiological source term that might be released from nuclear power plants during severe accidents. The research is used in developing reactor siting policy, emergency planning and response requirements, PRA consequence calculational methods, and equipment qualification standards.

Fission Product Experiments. At ORNL, three high-temperature fission product release tests were run with fuel containing non-radioactive fission product isotopes provided by the Federal Republic of Germany. This "simulant" fuel offers a less expensive method of testing and preliminary results showed comparable fission product releases to those from tests conducted at the SASCHA facility in Germany. However, the releases of volatile fission products are much higher than those from real fuel for the same time-at-temperature conditions.

Aerosol Transport Test. The NRC is a participant in an internationally sponsored project called Aerosol Transport Tests being conducted in Sweden at the Marviken facility to provide a large-scale demonstration of the transport and behavior of aerosols in primary systems. The first series of tests has been completed.

Containment Failure Mode

Research during 1984 on containment shells and major penetrations included the development of a pressure testing facility for containment models, the testing of small steel models, and construction and instrumentation of a large steel model. A matrix has been developed for full-



A large steel model containment (1/8 scale) is shown being set in place at Sandia National Laboratories for use in internal pressure and leakage tests.

scale tests of electrical penetrations, and an existing facility has been modified for these tests. Also, preliminary tests on seal and gasket materials used in valves and electrical penetrations were completed. Pressure tests have been performed in the valve test facility on three purge valves.

The largest single effort involved a large steel model, about 1/8 the size of a steel containment, designed and built by Chicago Bridge and Iron Company to the ASME Code. The model is built of A516 steel to a design pressure of 40 psig. A516 steel was chosen because it is used extensively in steel containments and steel-lined reinforced concrete containments. Features incorporated into the model include operable equipment hatches, pipe penetrations, a constrained pipe penetration, personnel lock representation (inoperable), stiffening rings, and thickened sections around penetrations. (See figure showing the model being erected at the test facility at Sandia National Laboratories, Albuquerque, N.M.)

Test facilities at Sandia capable of duplicating the steam temperature and pressure profiles expected in severe accidents were completed. They will be used in the coming year to measure the leak rate and failure mode of electric penetration assemblies in BWR MKI, BWR MKIII, and large, dry PWR containments when subjected to a severe accident environment.

Fission Product Control

Most engineered-safety-feature (ESF) systems are likely to be operational during postulated accidents substantially more severe than current design basis accidents. However, there may be a substantial variation in the effectiveness of fission product removal of various ESF systems under conditions exceeding their design basis. A program is in progress to facilitate review and evaluation of ESF-system behavior under severe accident conditions. Three reports were published during the year covering (1) fission product removal in ESF systems (NUREG/CR-3727), (2) chemical forms of fission products released following an accident at a plutonium recycle test reactor in 1965 (NUREG/CR-3669), and (3) effectiveness of ESF systems in retaining fission products (NUREG/CR-3787).

Risk Code Development

Risk code development generally encompasses all the analytical studies being conducted by the NRC in an attempt to quantify reactor risk. The focus here is on the narrow area of research involving the development of codes (computer programs) to integrate all aspects of the physical processes that can occur in severe accidents to allow radioactive materials to be released from the fuel (reactor core) into the atmosphere. This research is being performed under the NRC's MELCOR program in order to (1) develop a second generation accident analysis code for use in risk analysis to replace the current generation MARCH, CORRAL, and CRAC codes; and (2) develop and integrally implement methods by which MELCOR could be used to perform quantitative uncertainty analyses. An initial working and transferable version of the code was provided in 1984. Plans were developed to lay out methods by which MELCOR will be fully tested, validated, provided to the user community, and applied to support regulatory issues such as severe accident assessment, backfitting issues, ESF assessment, NRC's safety goal, and source term estimates.

Accident Consequences and Risk Reevaluation

Public health accident risks are functions of the probability of reactor accident source terms (the inventory of radioactive releases into the atmosphere) and the probabilities of off-site consequences given the nature of these source terms. The magnitude of the consequences depends on the weather, atmospheric transport conditions, distance from the reactor, and emergency response by the public. Development of an improved computer code for consequence estimation was initiated in 1984 to support the MELCOR program. Initial analyses were performed on the existing CRAC 2 code to exercise the uncertainty analysis techniques being developed under MELCOR as a precursor to the broad application of the uncertainty analysis techniques. In addition to the code development, a major effort to develop a revised health effects model was continued and was expected to be completed in early 1985. Applications of these consequence codes were used in support of NRC staff testimony at the Indian Point (N.Y.) and Catawba (S.C.) hearings, as well as for analyzing the licensing implications of the proposed NRC safety goal.

Value-Impact Analysis

The NRC Office of Nuclear Regulatory Research has as one of its concerns the development and dissemination of systematic methods that facilitate NRC decisionmaking. To date the program has provided insights into NRC perceptions and requirements for risk-related decisionmaking and conducted specific research tasks in support of decisionmaking processes.

During the past year, the Commission has published a value-impact handbook for use by NRC staff and industry in evaluating the need for and effectiveness of a variety of regulatory actions such as rulemaking, standards development, and backfitting safety improvements on operating plants. This document should provide a unifying frame of reference for estimating the cost-benefit relationships in the regulatory process.

ADVANCED REACTORS

NRC's advanced reactor safety technology research program (see *1983 NRC Annual Report*, p. 122) on liquid-metal fast-breeder reactors (LMFBRs) and high-temperature gas-cooled reactors (HTGRs) produced the following results in 1984.

Transient-over-power (TOP) tests performed on single-pin fast breeder fuel in the CABRI experiment in Cadarche, France, were analyzed by the Los Alamos National Laboratory using the SAS-3D and LAFM codes to assess their predictive capability for clad failure. Results of these analyses show that the codes over-predict cladding strain by significant margins, but that the discrepancies are due to model assumptions concerning axial fuel expansion during the TOP. SIMMER-II code modifications are in progress using new cross-sectional algorithms based on multi-isotope mixing techniques.

An agreement was concluded with the Commissariat a l'Energie Atomique of France to exchange the Argonne National Laboratories' COMMIX-1A computer code for operating data on the Rapsody reactor. Significant improvements were made in the natural convection models of the Brookhaven National Laboratory's Super Systems Code, which is being used in the licensing review of the MONJU reactor in Japan and the SNR-300 reactor in West Germany.

A series of 10 experiments in which uranium oxide samples were vaporized under sodium to generate aerosols was conducted in the Fuel Aerosol Simulant Test Facility at Oak Ridge National Laboratory in order to investigate the potential for transport of radioactive materials during a core-disruptive accident. Results indicate that no significant amount of material is released unless the vapor bubble breaks the surface of the sodium pool.

A series of 10 experiments in which uranium oxide samples were vaporized under sodium to generate aerosols was conducted in the Fuel Aerosol Simulant Test (FAST) facility at Oak Ridge National Laboratory, thereby completing the program.

Two separate-effects experiments were performed in the Annular Core Research Reactor (ACRR) at Sandia on the process of molten fuel removal from the core during the transition phase of an LMFBR core-disruptive accident. The removal of molten fuel from the core limits the energetics of possible fuel recriticalities in such accidents, and evaluation of this process was a key element in the Clinch River Breeder Reactor licensing review. One of these TRAN experiments checked modeling prediction on the effect of ablation of the steel surface by the molten fuel, and the other was the first experiment to use the slab geometry of the fuel-removal paths between adjacent steel subassembly canister walls. The Japanese are partners in the TRAN program and support much of the program cost.

Two separate-effects experiments were performed in the ACRR on the upward expulsion of molten steel cladding on fuel by sodium-vapor flow, and on freezing and blockage formation during the initiation phase of an LMFBR core-disruptive loss-of-flow accident. West Germany is a partner in this program and provides significant financial support.

RADIATION PROTECTION AND HEALTH EFFECTS

In 1984, efforts continued under the NRC radiation protection and health effects program to develop regulatory requirements that ensure that workers in licensed activities and individuals living in the areas near them are not subject to any significant risks of health damage from radiation exposure, and also to promote public understanding of such health risks through the use of risk assessment. This program consists of the development of radiation protection standards, support of related research, assessment of health risks, and cooperation with other agencies and organizations having similar interests.

Radiation Protection Standards

Revision of 10 CFR Part 20. The NRC staff continued development of a major revision of 10 CFR Part 20, "Standards for Protection Against Radiation" (see *1983 NRC Annual Report*, p. 130). The objective is an improved rule that provides better assurance of protection; establishes a clear health protection basis for limits; applies to all licensees in a consistent manner; and reflects current information on health risk, dosimetry, and radiation protection practices and experiences. In addition, issuance of the rule would provide the NRC with a health protection base from which it may consider other regulatory actions taken to protect public health.

Decommissioning. During 1984, work continued on the proposed rule changes to 10 CFR Part 20 to specify generic residual radioactivity limits for decommissioning. The proposed rule is intended to ensure that the radioactivity levels associated with buildings, structures, equipment, materials, and land used in NRC-licensed activities being decommissioned are low enough to protect public health.

Petitions. The NRC has been petitioned to modify the present requirements for the calibration of teletherapy units covered under 10 CFR Part 35. During 1984, a study to develop a less time-consuming and less expensive means to check teletherapy calibration was initiated to mitigate the current backlog in calibrations.

A petition has been filed to amend 10 CFR Part 20 to provide additional options for the disposal of very low concentrations of short-lived radionuclides similar to those for disposal of certain tritium and carbon-14 wastes.

A January 1984 *Federal Register* notice requested comments on the petition, on the use and inventories of short-lived radionuclides, and on disposal methods and costs.

In July 1984, the NRC was petitioned to issue a regulation governing the disposal of low-level radioactively contaminated oil from nuclear power plants by methods other than land burial at low-level radioactive disposal sites. Alternative disposal methods suggested by the petitioner include incineration, recycling, spraying on roadways to control dust, and disposal at ordinary landfill sites. A *Federal Register* notice requested public comments on the petition by November 19, 1984.

Radiation Protection Research

Metabolism and Internal Dosimetry. The problems of protecting uranium mill workers from occupational exposure to uranium through routine bioassay programs and assessing accidental worker exposures are being addressed in metabolic studies of inhaled refined uranium ore (yellowcake) in animals. Comparison of chemical properties and the biological behavior of yellowcake are being made to identify important properties that influence uranium distribution patterns among organs. An interim report (NUREG/CR-3745) on the biological characterization of radiation exposure and dose estimates for inhaled uranium milling effluents was published in May 1984.

In order to improve predictions of health consequences in humans from airborne radioactivity that might be released in normal operations or under accident conditions during production of nuclear fuel composed of mixed oxides of uranium and plutonium, metabolic studies of inhaled mixed oxides in animals were completed and dose-response studies in rats are continuing. A report (NUREG/CR-3870) on radiation dose estimates and hazards evaluations for inhaled airborne radionuclides was published in July 1984.

Work continued on research designed to resolve a long-standing practical problem in radiation protection, i.e., the relative biological effectiveness of low-level neutron irradiation. This value is important because it is used in establishing the maximum permissible occupational exposure limit for low dose rate and low total dose neutron exposures. (See *1982 NRC Annual Report*, p. 137.) Other continuing projects during 1984 included medical evaluation of workers occupationally exposed to thorium, gastrointestinal uptake of actinides in baboons, and biokinetics of actinides and rare earths in monkeys.

Health Effects and Risk Estimation. As part of the ongoing objective to improve assessments of health effects from exposure to low-level ionizing radiation, a study of the effectiveness of chronic versus acute radiation exposure in cancer induction was completed (final report to be published in 1985). Continuing projects during 1984 included the development of models for early mortality and morbidity resulting from inhalation of radionuclides that could be released in potential accidents, and a study of iodine-131 as a causative agent in inducing thyroid cancer in children.

Health Risk Assessments

Radon. Exposure to radon gas and radon progeny has been associated with increased lung cancer incidence. In order to improve health risk assessments in populations exposed to these carcinogens, a computer program for modeling the health risks of radon exposure was published (NUREG-1029). In addition, work continued on the study of radium dial painters who were exposed to radon daughters in early adult life.

Severe Accident Health Effects Model. Work continued on updating the health effects model that was used in the Reactor Safety Study (WASH-1400) to estimate the consequences of postulated severe accidents at nuclear reactors.

Cooperative Efforts

In 1984, the NRC health effects program was well coordinated with other Federal programs and with national and international scientific organizations concerned with radiation research and protection. Broader areas of mutual interest were coordinated through participation on the Committee on Interagency Radiation Research and Policy Coordination, which meets under the auspices of the Office of Science and Technology Policy. Radiation protection programs were coordinated with the National Council on Radiation Protection and Measurements and by participation on interagency working groups established by the Environmental Protection Agency (EPA) to coordinate its Federal guidance on radiation matters.

Specific research areas were coordinated through meetings and joint programs with the Departments of Defense and Energy, the EPA, the National Science Foundation, and the National Institutes of Health. A study by the National Academy of Sciences designed to develop a report on the biological effects of internally deposited alpha-emitting radionuclides and their decay products (BEIR 4, Part 1) has been initiated in cooperation with EPA. This study will be used by numerous NRC programs, such as the high-level-waste program, that require assessments of the genetic and carcinogenic effects of alpha and other radiation. Particular emphasis was also given to quantification of health risks of exposure to internal alpha emitters when NRC sponsored the American Statistical Association Conference on Radiation and Health to consider research problems in quantification of radiation health effects.

Occupational Radiation Protection

Health Physics Measurement Improvement. Research is continuing at Pacific Northwest Laboratory (PNL) to evaluate draft standard ANSI N42.17, "Performance Specifications for Health Physics Instrumentation," by testing a cross section of currently available commercial instruments. This research will allow the NRC staff to decide whether current health physics instrument performance is satisfactory for licensees conducting required surveys (10 CFR Part 20). This work has led to the formation of a government-industry task force to address problems involving radiation survey instruments and calibrations. PNL has also completed the work needed to assess the adequacy of current dosimetry systems and techniques for monitoring extremity exposures at NRC-licensed facilities. Results of this study will provide a technical basis for resolving current inconsistencies in health physics practices and a basis for guidance to NRC licensees for monitoring extremity exposures.

As part of a research program to evaluate personnel neutron dosimetry at commercial nuclear power plants, a report published this year (NUREG/CR-3610) describes a technique that can improve the accuracy of personnel neutron dosimetry measurements. Personnel routinely enter containment for maintenance and inspections while the reactor is operating and can thereby be exposed to neutron radiation. The low-energy neutron fields in reactor containment often require the use of energy correction factors for TLD-albedo dosimeters. NUREG/CR-3610 describes the use of a helium-3 neutron spectrometer system for taking reactor energy spectrum measurements and for determining dose equivalent rates.

Personnel Dosimetry. A proposed rule published in January 1984 would require NRC licensees to use the services of personnel dosimetry processors accredited by the National Bureau of Standards (NBS). The NBS has established and is operating a National Voluntary Laboratory Accreditation Program for personnel dosimetry processors under an interagency agreement with the NRC. Processors participate in proficiency testing and undergo on-site evaluation of quality control procedures. The first accreditations occurred in October 1984 with subsequent actions to be made on a quarterly basis. Participation in this program by all dosimetry processors will result in significant improvements in the measurement and recording of occupational dose.

A program jointly funded by the NRC and DOE to assist in the development of a consensus performance standard for bioassay laboratories is now providing data for consideration by the organization developing the standard (Health Physics Society). The second round of testing bioassay laboratories on their ability to analyze simulated in vitro samples for six radionuclides was completed. Results indicate that the bias and precision criteria chosen for the draft standard are effective in identifying laboratories having analytical problems.

A draft regulatory guide on instrument test and calibration criteria and methods, endorsing ANSI standard N323-1978 and providing methods of tracing the accuracy of radiation measurements to NBS reference standards, was issued in September 1984.

Research is continuing on ultrasensitive analysis procedures for improving detection capabilities for certain radionuclides by resonance ionization spectroscopy and the isotope dilution mass spectrometry techniques. Neither of these methods is yet developed to the point of commercial availability.

Occupational Exposure Data System. In 1969, the Atomic Energy Commission began requiring certain types of licensees to submit certain reports on occupational radiation doses received by their employees. 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).

Preliminary summaries of the annual statistical reports for 1982 reveal that the four categories of licensees monitored about 190,000 individuals of whom about 60 percent received a measurable dose. These workers received a collective dose of 59,000 person-rems, or an average dose of 0.5 rem per worker among those receiving a measurable dose (0.3 rem per monitored person). Seventy-one percent of the individuals monitored were in nuclear power facilities, and they incurred about 89 percent of the total annual collective dose. The average measurable dose received by nuclear power plant workers was about 0.6 rem.

A second kind of exposure report required of certain NRC licensees provides identification and dose data each time that a monitored individual terminates employment with the licensee. Such information is now maintained for some 300,000 individuals, most of whom were or are employed by nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories. The data are used to ensure that transient workers moving from plant to plant do not receive doses in excess of regulatory limits.

Chemical Decontamination. The NRC continued to develop an information base for assessing the safety and effectiveness of decontamination alternatives for reducing occupational dose in nuclear power plants. Observations and measurements were made during selected chemical decontamination activities at the Pilgrim, Millstone, Peach Bottom, Quad Cities, Dresden, Palisades, and Monticello nuclear power plants. A report analyzing these results and similar measurements to be conducted at other nuclear power stations will be published in 1985. A report published in 1984 described a computer program used to calculate doses associated with different tasks during decontamination phases of maintenance activity (NUREG/CR-3573).

A Small Business Innovation Research contract funded by NRC supported a feasibility study on the use of robotic devices such as that shown in performing radiation surveys and inspections in nuclear power plants. Remotec, Inc., will construct a mobile survey and inspection robot from commercially available components and investigate its capabilities in a TVA operating reactor.

Optimization of Public and Worker Dose. A review was published in September 1984 (NUREG/CR-3665) of the extent to which occupational dose is factored into cost-benefit analyses and probabilistic risk assessments, provides recommendations as to how worker dose should be considered in establishing inspection and maintenance requirements, and provides a new model to assist NRC decisionmakers in evaluating proposals for inspection and maintenance requirements that entail occupational dose.

Robotics in Radiation Surveys. Phase I of a Small Business Innovative Research study on robotics in reactor inspection demonstrated the feasibility of using robots to replace workers for inspecting and monitoring radiation in areas of potentially high exposure (NUREG/CR-3717). Phase II, initiated in October 1984, is a continuation of Phase I and includes the design, construction, and demonstration testing of an inspection robotic system at the Browns Ferry nuclear plant. The robotic system will be capable of replacing human workers in performing visual and audible inspections and radiation mapping within controlled areas. The objective is to obtain operating experience and actual cost/benefit data to demonstrate that robotics can reduce occupational radiation exposure and be cost effective.

Radiation Safety at Advanced Reactors. An evaluation of radiation protection program requirements unique to LMFBRs was completed. The report, expected to be published early in 1985, will provide the technical basis for developing a standard review plan in the area of radiation safety to be used by NRC personnel in reviewing LMFBR license applications.

REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth in Title 10, Chapter 1, of the Code of Federal Regulations requirements that must be met.
- Regulatory guides, describing, primarily, methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations.

When NRC proposes new or amended regulations, they are normally published in the *Federal Register* to allow interested citizens 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 become 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 national organizations, often with NRC participation. NRC makes use of a national standard in the regulatory process only after an independent review by the NRC staff and after public comment on NRC's planned use of the standard has been reviewed.

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 indicate the objectives of each guide, along with its expected effectiveness and impact.

To reduce the burden on the taxpayer, the NRC has an arrangement with the U.S. Government Printing Office to act as a consigned sales agent for certain of its publications, including regulatory guides. Draft guides issued for public comment continue to receive free distribution, but the active guides are sold. NRC licensees receive pertinent draft and active guides at no cost.

United States and Foreign ALARA Programs. A study was completed to determine how doses to workers in nuclear power plants in the United States compare to doses to nuclear plant workers in other countries. It appears that most technically advanced countries have been more successful in controlling occupational dose than the United States. The results of this study suggest ways to improve the design, operation, and regulation of nuclear power plants in the United States to lower occupational radiation exposures.

WASTE MANAGEMENT

NRC's waste management research assesses, tests, and improves measurement and prediction methods; confirms data bases; provides technical support to the licensing staff in their interactions with the Department of Energy (DOE) and the States; and develops 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 has an active research program in geohydrology related to the management of high-level waste (HLW). This program is divided into two areas that cover the fundamental geohydrologic areas of greatest uncertainty with respect to HLW disposal: ground-water flow in fractured media and ground-water flow in the unsaturated zone. Both avenues of research combine theoretical study with laboratory and field experiments to identify the physical processes that control and determine ground-water movement in the types of geologic media found at sites currently under investigation by DOE. The ultimate goal of the NRC's waste management research into geohydrology is to provide the technical base upon which the licensing staff will build an independent judgment as to the appropriateness and adequacy of DOE's description of the geohydrology at a proposed repository site.

In 1984, research on ground-water flow in fractured rock produced several reports that will be of direct use in assessing DOE's site characterization activities. The first (NUREG/CR-3213) deals with crosshole geophysical methods used to investigate the near vicinity of HLW repositories. The second (NUREG/CR-3612) describes a way to analyze statistical values of locally measured hydraulic conductivities to obtain far-field dispersion coefficients. In a related study, NUREG/CR-3680 issued in April 1984, information is presented about the relationship between the gas conductivity and geometry of a natural rock fracture.

Other research being conducted by NRC addresses the suitability of available geophysical measurement methods for characterizing and monitoring potential geologic repository sites. This research produced a final topical report on geotomography (NUREG/CR-3758 issued in August 1984) from the Lawrence Livermore National Laboratories. These geotomographic techniques can be used to characterize large rock masses from boreholes for HLW repositories. Of particular importance to siting HLW repositories is the use of the method for remote detection of fracture zones. This work will be completed in 1985.

A major achievement in the borehole sealing research program was the completion of a report (NUREG/CR-3473) on borehole seal performance. Among the significant findings were:

- Cement plugs installed in horizontal drill holes have shown very good sealing performance. The data gathered thus far demonstrate the feasibility and effectiveness of sealing horizontal holes.

- Dynamic effects (e.g., earthquake loading) are unlikely to impair seal (or engineered barrier) performance.

- Drilling damage studies on basalt confirm results previously obtained in granite, i.e., that drilling damage of borehole walls is unlikely to result in a significant preferential migration path.

In addition, it was found that drying of cement seals has a significant detrimental effect on seal performance. Testing is in progress to find out if changing the cement mixes would improve performance under drying conditions.

The performance that can be expected from the waste form and waste package is a major area of NRC HLW research. Several programs that are investigating the mechanisms of waste package/waste form performance under expected repository conditions are taking place at national laboratories and in private corporations. A research project was initiated during 1984 to study the relationship between nuclear waste glass and comparable naturally occurring glasses that have aged in appropriate geologic environments. The results will be useful for evaluating DOE's application regarding waste form. During 1984, the corrosion research groups under contract to NRC focused on localized corrosion in carbon steel, which was perceived to be the most critical corrosion issue for the basalt and salt repositories.

In February 1984, NRC published for public comment proposed amendments to Part 60 related to the disposal of HLW in geologic repositories within the unsaturated zone. These amendments will ensure that NRC regulations are applicable to all geologic repositories, whether sited in the saturated or unsaturated zone. A report issued in February 1984 (NUREG-1046) identifies positive aspects and potential concerns associated with the disposal of HLW within unsaturated geologic media that were considered by NRC during the preparation of the proposed amendments to 10 CFR Part 60.

Studies continue in meteorology and hydrology that investigate the climatic characteristics of the last 10,000 years to better extrapolate future climatic trends of temperature and precipitation that could impact the performance of a waste disposal facility by causing changes in ground-water transport, sea level elevation, and glaciation. The climatic calibration of pollen data is addressed in NUREG/CR-3847, issued in June 1984.

Low-Level Waste

NRC research in support of licensing activities for low-level waste (LLW) disposal facilities is focused on (1) water entry into burial trenches, (2) performance of waste packages, (3) characterization of the LLW source term, (4) mechanisms for transport of radionuclides from the burial trenches, and (5) evaluation of the overall performance of the disposal system. This information will be useful not only to the NRC licensing staff but also to States facing similar regulatory efforts.

LLW research has produced two models and associated computer codes in 1984. The first, BIOPORT, models both biotic transport of radionuclides and the corresponding dose from shallow land burial of LLW. The other is a model with an interactive computer program that illustrates the complexities in developing a sampling program to monitor commercial LLW sites for performance or location and cleanup of radionuclide spills. In the materials area, research was completed on the effects of high-level irradiation on organic ion-exchange resin and the suitability of incineration ash and solidified ash for near-surface disposal.

Additional products included reports on geochemical investigations at the Maxey Flats LLW disposal site (NUREG/CR-3607) summarizing years of research on the radionuclide source term at that site and a report on LLW shallow land burial trench isolation (NUREG/CR-3570).

Two technical reports were issued in the area of meteorology and hydrology. NUREG/CR-3838 (issued in June 1984) presents an initial review of atmospheric dispersion models suitable for LLW disposal facilities, and geologic and hydrologic characterization of the West Valley site is described in NUREG/CR-3782 (also issued in June 1984).

STANDARDS PROGRAMS

IAEA Reactor Safety Standards

The NRC continued to coordinate U.S. technical activities associated with the IAEA Nuclear Safety Standards program to develop safety codes of practice and safety guides for nuclear power plants. The codes and guides provide a basis for national regulation by developing countries of the design, construction, and operation of these plants. In 1984, two safety guides were forwarded through the Senior Advisory Group and Technical Review Committees to the Director General of the IAEA. All the planned IAEA safety guides were undergoing review at year's end with the NRC research staff coordinating the reviews within the U.S. Revision of one safety guide is under way in response to user request that more information on the commissioning phase be included. (See *1980 NRC Annual Report*, p. 196).

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 so that regulatory views are known regarding the standards that can be most useful in protecting the public health and safety. NRC participation is based on the need for national standards to define acceptable ways of implementing the NRC's basic safety regulations.

Approximately 210 NRC staff members serve on working groups organized by technical and professional societies.

Proceedings and Litigation

CHAPTER 12

The first section of this chapter reports on proceedings involving the Atomic Safety and Licensing Board Panel and the Atomic Safety and Licensing Appeal Board; also included are some of the more significant decisions of the Commission itself (see "The Licensing Process," Chapter 2). The second part of the chapter is a judicial review of the report period covering noteworthy litigation involving the NRC, including cases pending and closed.

ATOMIC SAFETY AND LICENSING BOARD PANEL

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. 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 and single Administrative Law Judges hear and decide other matters. These hearings are the Commission's principal public forum for individuals and organizations to voice their interest in a particular licensing or enforcement issue and to have their concerns adjudicated by an independent tribunal.

On September 30, 1984, the panel included 24 permanent and 28 part-time administrative judges drawn from various professions. There were 20 lawyers, 16 environmental scientists, 7 engineers, 6 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 cases in which their professional expertise will assist the board in resolving the issues to be litigated. Generally, boards consist of a lawyer 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: health, safety, common defense and security aspects of the application, as required by the Atomic Energy Act; environmental considerations as required by the National Environmental Policy Act (NEPA); and emergency planning requirements. These matters, as well as especially complex technical issues, are frequently treated by boards in separate initial decisions.

Administration

As cases have become more intensely and actively litigated, and the issues to be decided have grown increasingly complex, the effective logistical management of the hearing process has become especially important. In this effort, the boards are supported by 22 full-time employees and one part-time—including management personnel, a legal counsel, law clerks, legal secretaries and docket personnel.

Administrative support for the boards and the panel is furnished by uniform word processing equipment, a joint ASLBP/ASLAP library, the LEXIS automated legal research system, docket room, and a computerized travel and timekeeping system. A computerized Hearing Status Report now has a virtually complete data base and is capable of generating valuable case management information.

The successful computerization of the *Indian Point* (N.Y.) hearing record was followed in fiscal year 1984 with an effort to obtain expert analysis and evaluation of the computer systems and software employed. The objectives of the study will be to quantify the benefits achieved through the availability of a computer-searchable record; to evaluate the benefits in relation to the costs of both the system utilized and alternative document management systems; to determine the feasibility and utility of expanding the availability of the system to other NRC offices involved in the hearing process; and to develop recommendations for future computerized management of hearing records. At the close of fiscal year 1984, proposals from three organizations specializing in computer support of litigation were under consideration by the Panel.

The Caseload

During the fiscal year ending September 30, 1984, Licensing Boards conducted 65 proceedings involving nuclear power plants and other nuclear facilities, with a construction value well in excess of $100 billion. Twenty-nine percent of the proceedings were completed. Some 406 days of hearings were held, comprising 307 days of trial and 99 days of prehearing conferences. Twenty-one proceedings were closed while 17 new cases were opened. The operation of three nuclear power plant units was authorized. Hearings on four additional units were completed during the fiscal year, and decisions on those cases were in preparation as of September 30, 1984.

Hearing Procedure

The heavy ASLBP caseload, combined with increasing public awareness and involvement in the licensing process, has made effective hearing management essential to the timely completion of licensing decisions. Using the procedural tools available under Commission regulations, Licensing Boards have increasingly endeavored to assure that issues for hearing are soundly based and well-defined. Prehearing 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 duplicative efforts and to assure the early resolution of potentially time-consuming disputes. As a result of this active management, nearly three-quarters 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.

Rules of Practice

At the request of the Chairman, the ASLBP undertook in fiscal year 1984 to produce the first comprehensive revision of the Commission's Rules of Practice in over a decade. The principal goals of the revision effort were to eliminate unnecessary and redundant verbiage; reorganize the regulations in a more logical order; present the rules in readable "plain English"; and incorporate procedures and standards established by NRC case law but as yet uncodified. After a six-month effort, with the participation of all NRC offices directly concerned with the Commission's adjudicatory processes, a proposal was submitted to the Commission in September 1984. Action on the proposal was pending at the end of the fiscal year.

Cases of Note

Indian Point. In October 1983, the *Indian Point* (N.Y.) special proceeding was concluded with the issuance by the Licensing Board of its findings and recommendations addressing seven questions originally presented by the Commission and related contentions proffered by intervenors. The board concluded that with the implementation of certain safety improvements recommended by it, Indian Point Units 2 and 3 could operate with reasonable assurance that the public health and safety would be protected. The board concluded that the risk of fatalities from an accident at Indian Point constituted a very small fraction of the competing non-nuclear background risks to which the population around Indian Point is exposed, but recommended certain plant and plant procedure modifications. The board found that off-site emergency planning at the facility was inadequate as of the close of the hearing record, but concluded that no convincing showing had been made for the need to extend the emergency planning zone beyond its existing 10-mile radius. Finally, the board recommended that in considering the risk posed by continued operation of the Indian Point plants, the Commission should consider the cumulative risk to the population in the vicinity that would result from operation for the remainder of the plants' license lifetimes, and also the potential consequences of a low-probability accident.

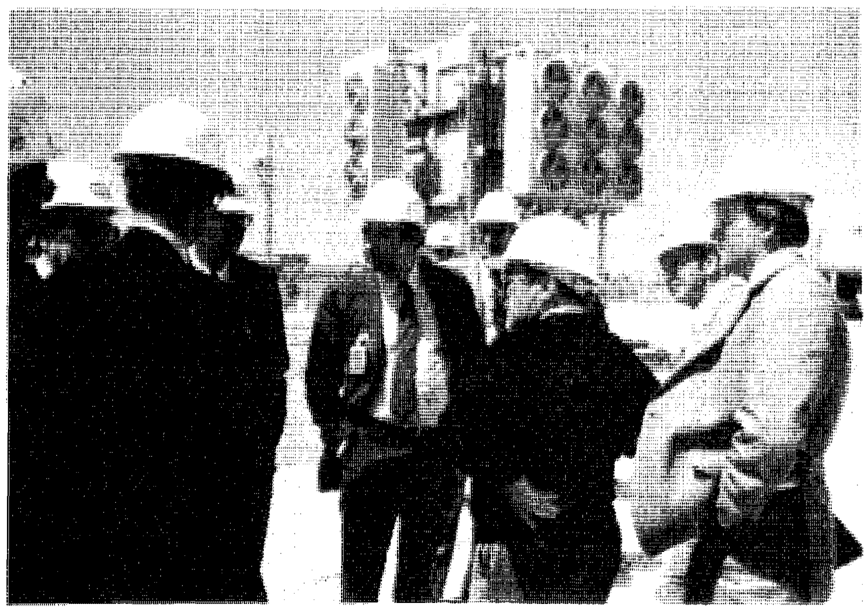
Shoreham. The extensively litigated *Shoreham* (N.Y.) operating license proceeding continued through fiscal year 1984. In order to facilitate the expeditious consideration of the diverse, complex issues pending, this case was divided in fiscal year 1983 between two Licensing Boards, with the original board continuing to hear health and safety issues and a second board considering contentions related to off-site emergency planning. In 1984, a third board was established to rule on the applicant's motion for a low-power license and its request for an exemption from the Commission's general design criteria governing plant electric power systems. At the close of the fiscal year, all three boards were nearing completion of their proceedings, but significant issues remained unresolved concerning emergency diesel generators and off-site emergency evacuation plans.

Callaway. In October 1983, the Licensing Board in *Callaway* (Mo.) issued an initial decision resolving all remaining emergency planning issues and authorized the issuance of a full-power license. The board found that the State of Missouri had exercised its responsibility reasonably in deciding against the distribution of potassium iodide to the general public and that existing arrangements for sheltering during an emergency met regulatory requirements.

Byron. In January 1984, the *Byron* (Ill.) Licensing Board issued an initial decision denying authorization for an operating license because of failures in the applicant's quality assurance program. Subject to certain conditions related to emergency planning and other commitments made by the applicant, all remaining issues were resolved in applicant's favor.

Following an appeal, the Byron case was remanded to the Licensing Board to permit the applicant to present

Late in 1983, NRC's Atomic Safety and Licensing Board issued an initial decision involving the controversial question of emergency planning at the Callaway nuclear plant at Fulton, Mo. Among other things, the board found that existing sheltering met emergency regulatory requirements. The photo, taken during a visit to Callaway by Harold R. Denton, NRC Director of Reactor Regulation, and other staff members. Dr. Denton (at center, facing camera) is conferring with licensee and NRC Region III (Chicago) personnel. James G. Keppler, Regional Administrator for Region III, is at center foreground (hand on hip).



additional evidence in order to demonstrate that the quality assurance deficiencies identified by the board had been corrected. Hearings on the remanded issue were completed in August 1984 and a decision was pending at the close of the fiscal year.

Decision authorizing operating license issuance were also rendered during the fiscal year in *Wolf Creek* (Kan.) following resolution of the remaining emergency planning issues, and in *Beaver Valley* (Pa.) as a consequence of dismissal of the proceedings for failure of intervenors to respond to a show cause order.

Other Proceedings

Licensing Boards concluded eight operating license amendment proceedings during fiscal year 1984. Six cases were dismissed after settlement or other prehearing resolution of the issues, and two others were decided in favor of authorization of the amendments requested after hearing.

In the companion cases of *Kress Creek* and *West Chicago Rare Earth Facility* (Ill.), a Licensing Board began consideration of issues related to the decommissioning of a thorium milling plant and the decontamination of adjoining property. Those cases remained in the early prehearing stages at the close of the fiscal year.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Atomic Safety and Licensing Appeal Boards, consisting of three members each, perform review functions for the Commission in facility licensing proceedings and others the Commission may specify. Unless the Commission decides to review an Appeal Board decision, that decision becomes the final agency order and is subject only to judicial review in a Federal court of appeals. The board for each proceeding is selected from among the members of the Atomic Safety and Licensing Appeal Panel (ASLAP) by the panel chairman. (See Appendix 2 for membership of the panel.)

Under Commission rules, the Appeal Boards hear appeals from Licensing Board decisions and certain rulings on intervention petitions by members of the public seeking to become parties to the proceeding. They also review Licensing Board decisions on their own initiative in the absence of any appeal. In limited circumstances, Appeal Boards also consider questions posed by the parties or the Licensing Board, or rulings referred by a Licensing Board, while the proceeding is still in progress. Appeal Boards also occasionally conduct evidentiary hearings either as part of their appellate review function or on direction from the Commission.

As in past years, the Appeal Boards were called upon to rule on a wide variety of matters affecting the public health and safety and the environment, as well as on numerous questions relating to the fair and efficient conduct of licensing proceedings. Appeal Board actions resulted in some 40 published decisions in the Nuclear Regulatory Commission Issuances, the permanent collection of NRC licensing and other decisions available to the bar and the general public, and a number of other memoranda and orders. The past year also saw the retirement of the Appeal Panel's long time Vice-Chairman, Dr. John H. Buck. Highlighted below are the more significant of the published Appeal Board decisions.

TMI-Restart Proceeding

Previous Appeal Board decisions in this proceeding to consider the restart of Unit 1 of the Three Mile Island (Pa.) facility dealt with the emergency planning, environmental, and design issues raised. This year, in addition to procedural matters, the Appeal Board was called upon to consider issues related to the ability of the licensee management to operate the unit in a competent, responsible and safe manner. Upon consideration of the appeals of three intervenor groups from the Licensing Board's determination that the licensee had demonstrated its managerial capability and technical resources to operate the Unit, the Appeal Board decided that the existing state of the record did not permit it to make an ultimate judgment on the licensee's competence. It therefore returned the proceeding to the Licensing Board for further hearing.

Diablo Canyon

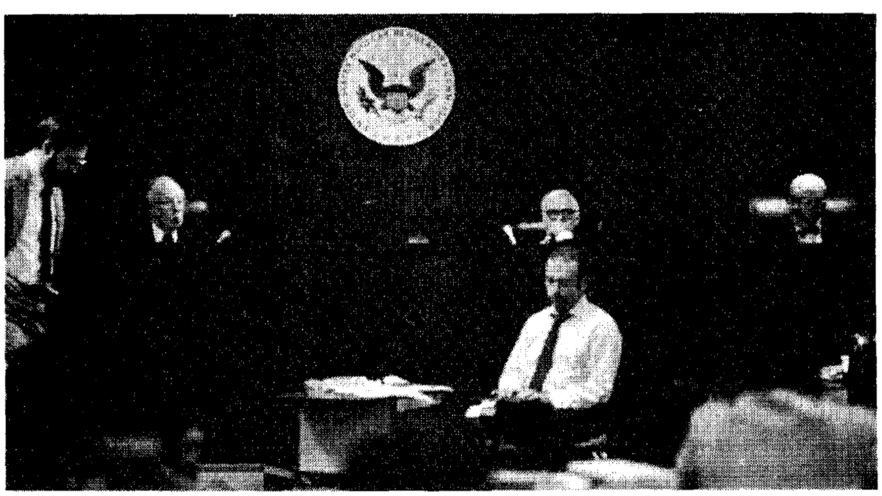
The *Diablo Canyon* (Cal.) operating license proceeding continued to occupy considerable time of the Appeal Board. This proceeding alone was the subject of five published decisions. One of these, concerning the efforts by the licensee to verify the adequacy of the design of the plant, followed the taking of evidence by the Appeal Board near the plant site. The Appeal Board decided that the actions taken by the applicant provided adequate evidence that Unit 1's structures, systems and components were designed to perform satisfactorily in service and that any significant design deficiencies resulting from defects in the applicant's design quality assurance program had been remedied. The board thus concluded that there is reasonable assurance that Unit 1 can be operated without endangering the health and safety of the public. In two other decisions, the Appeal Board disposed of the appeal of each party from various portions of the 1982 Licensing Board decision authorizing a full power license for the facility. The decisions allowed the Licensing Board's full power authorization to stand. Under Commission practice, however, the license could not issue pending Commission action.

Other Proceedings

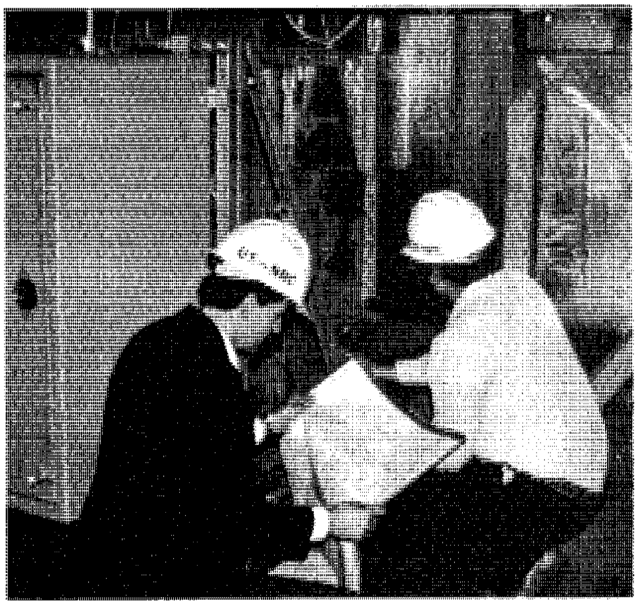
Health, Safety and Environmental Issues. The *Point Beach* (Wis.) operating license amendment proceeding presented the novel question of whether degraded steam generator tubes could be repaired by sleeving them. Under the plant's existing license, such tubes would have had to be plugged and removed from service. On the intervenor's objection to the Licensing Board decision authorizing the use of sleeving, the Appeal Board found that the Licensing Board had properly considered the safety aspects of sleeving. It thus affirmed the Licensing Board's decision.

Callaway (Mo.) involved the question of the adequacy of the quality assurance/control programs utilized in the construction of the facility. The Licensing Board had found on an examination of the matter that there had been no general breakdown in quality assurance procedures, that the various identified construction defects had been remedied and there was reasonable assurance that the plant could be operated safely. Upon review, the Appeal Board agreed with the Licensing Board. According to the Appeal Board, neither the Atomic Energy Act nor the Commission's regulations hinged the grant of an operating license upon a demonstration of error-free construction; rather, what was required was simply a finding of reasonable assurance that, as built, the facility can and will be operated without endangering the public health and safety.

The environmental impacts of the plant's supplementary cooling water system was a principal issue in the appeal in *Limerick* (Pa.). It involved, among other considerations, the Commission's responsibility *vis-a-vis* that of the Delaware River Basin Commission, which allocates the use of water from the Delaware River among compet-



A hearing before the Atomic Safety and Licensing Board dealing with the proposed restart of Three Mile Island Unit 1 (Pa.) was held in Bethesda, Md., on December 18, 1984. From left are William Jordan, an attorney for the Union of Concerned Scientists, and board members Sheldon J. Wolfe, Ivan W. Smith, and Gustave A. Linenberger.



An NRC Licensing Board addressed questions raised by environmentalists and other citizens living near the Limerick nuclear power station at Pottstown, Pa. The questions dealt with use of water from the Delaware River and the discharge of water back into the river. The photo, taken at Limerick in 1984, shows NRC Senior Resident Inspector James T. Wiggins and Senior Radiation Specialist Ronald L. Nimitz inspecting drain valves.

ing interests. The Licensing Board found that there would be no adverse environmental impact from use of Delaware River water for the plant. The Appeal Board affirmed the Licensing Board's decision on all but two issues; i.e., the impact of withdrawal of Delaware water on the salinity of the river and the effect on the neighboring Point Pleasant Historic District. In an earlier decision, the Appeal Board affirmed the Licensing Board's denial of an intervenor's request for a hearing on a proposed license authorizing the applicant to ship, receive and store new fuel at the station prior to receiving a plant operating license.

In *Cherokee* (S.C.), the Appeal Board terminated its appellate jurisdiction (and the sole remaining question concerning the health effects of radon) upon being advised by the applicant of the cancellation of all three units of the plant. The Appeal Board terminated its jurisdiction over two other proceedings as a result of facility cancellations: *Phipps Bend* (Tenn.) and *Hartsville* Units 1B and 2B (Tenn.).

Reopening of Hearing Record. Motions to reopen a closed record were the subject of several Appeal Board decisions. In *TMI-1* (Pa.), the Appeal Board granted a motion (but denied two others) to reopen the record for further hearing on certain allegations of falsification of leak rate data. In *Waterford* (La.) the Appeal Board denied a request to reopen the record based on discovery of hairline cracks in the foundation on which the facility rests. The Appeal Board concluded that the foundations cracks did not present an issue of safety significance. The subject of the reopening motion in *Callaway* (Mo.) was an alleged breakdown in the utility's quality assurance program at the plant. The Appeal Board, however, found that reopening was not justified.

Disqualification of Judges. Under Commission rules, a motion for recusal of a judge is first acted upon by the judge in question who must then refer his or her ruling to the Appeal Board. In *Seabrook* (N.H.), various parties to the proceeding sought the recusal of the judge on three separate occasions. In each instance, the Appeal Board agreed with the judge's denial of the motion. The subject of judge disqualification was also involved in the *Shoreham* (N.Y.) operating license proceeding. There, two intervenors sought the disqualification of all three judges serving on one of three Licensing Boards considering various issues in that proceeding. The judges' refusal to disqualify themselves was affirmed by the Appeal Board which found no basis for the requested action. In *Hope Creek* (N.J.), however, a judge had refused to disqualify himself after he had been asked to do so by a party on the ground that the judge had worked as a consultant for the operating license applicant years earlier (when he was a member of a University faculty) on matters involving the construction permit application for the same facility. In this instance, the Appeal Board disagreed with the judge's decision. In doing so, it ruled that the judge's prior association with the applicant might lead a fully informed, reasonable person to question his impartiality, and for this reason had to step aside.

Public Intervention. Licensing Board rulings on petitions by members of the public desiring to participate in licensing proceedings as parties were the subject of several Appeal Board decisions. In *WPPSS Nuclear Project No. 3* (Wash.), the Appeal Board disagreed with the Licensing Board's acceptance of a late petition to intervene for lack of a showing on one of the criteria for late admission and returned the matter to the Licensing Board. The intervenor was then offered a further opportunity to show why it should be admitted to the proceeding at that late date. The Licensing Board admitted the petition and the applicant again appealed but this time the Appeal Board affirmed. In *Seabrook* (N.H.), the Appeal Board affirmed the denial of a late intervention petition. There, the only issue the petitioner had sought to raise was one which had already been raised by an intervenor. In such circumstances, the Appeal Board saw no reason to permit late intervention. And in the *Wolf Creek* (Kans.) operating license proceeding, the dismissal of an intervenor that had sought only to challenge the financial qualifications of the applicant was affirmed by the Appeal Board following the promulgation by the Commission of a rule which eliminated financial qualification issues from operating license proceedings.

Shoreham (N.Y.) presented the unusual situation in which an organization sought to intervene late in the proceeding for the purpose of supporting, rather than opposing, the operating license application. The Appeal

Board agreed with the Licensing Board's denial of the petition for failure to meet the test for late intervention. It explored, but declined to decide, whether the organization's asserted interest in the outcome of the proceeding was of the type that allowed it to participate in the proceeding.

Production of Documents. The *Shoreham* (N.Y.) proceeding also produced an appeal of the Federal Emergency Management Agency (FEMA) from a Licensing Board's decision ordering it to produce various documents in connection with the ongoing litigation of emergency planning issues. FEMA had opposed an intervenor's request for production of the documents under the executive or deliberative process privilege. The Appeal Board decided that the privilege was validly invoked and that the intervenor had not made the requisite showing of need for the documents at this stage of the litigation.

Interlocutory Appeals. Under Commission rules, with limited exceptions, interlocutory rulings of Licensing Boards—i.e., rulings issued during the course of a proceeding, as contrasted with the decision at the end of the proceeding—are not immediately appealable. The Appeal Board will hear such an appeal as a matter of discretion only if the Licensing Board ruling either (1) threatens the party adversely affected by it with immediate and serious irreparable impact which, as a practical matter, could not be alleviated at a later appeal, or (2) affects the basic structure of the proceeding in a pervasive or unusual manner. During the course of the year, parties in a number of proceedings sought Appeal Board review of Licensing Board rulings with which they disagreed. In each instance the Appeal Board refused to hear the appeal.

Sua Sponte. Under Commission practice, Appeal Boards review on their own initiative (i.e., *sua sponte*) the Licensing Board decisions and the underlying record on every safety and environmental issue considered by the Licensing Board, even where no appeal has been taken on a particular issue. The Appeal Board completed its *sua sponte* review of the Licensing Board's decision in the *Rancho Seco* (Cal.) special proceeding to determine the adequacy of certain short-term actions and long-term requirements for continued reactor operation ordered by the Commission as a result of the TMI-2 accident. The Appeal Board affirmed the Licensing Board's decision supporting the plant's operation.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1984 are discussed below. The Commission's actions on export licensing cases are discussed in Chapter 10.

Separate EIS Not Required For Shoreham Low-Power Operation

In *Long Island Lighting Company* (Shoreham Nuclear Power Station, Unit 1), CLI-84-9, 19 NRC (323 (1984), the Commission held that where an environmental impact statement for full power operation has been prepared and adjudicated, the pendency of an adjudication of emergency planning issues material to full power operation does not constitute a significant changed circumstance triggering the obligation to prepare a supplemental environmental impact statement for low power operation. In addition, the Commission announced its intention to conduct a rulemaking proceeding to decide the relationship between the issues of "important to safety" and "safety related." Pending completion of the rulemaking, the Commission instructed the boards to follow current precedent which holds that the term "important to safety" applies to a larger class of equipment than the term "safety related" but does not mean that the class of equipment which is important to safety has been defined at every plant.

Commission Finds Confidence in Interim Waste Storage

In *Rulemaking on the Storage and Disposal of Nuclear Waste* (Waste Confidence Rulemaking), CLI-84-15, 20 NRC 288 (1984), the Commission concluded its "waste confidence" rulemaking. In general, the Commission found reasonable assurance that high level radioactive waste can be stored until permanent disposal is required and that disposal facilities will be available when needed. Specifically, the Commission found reasonable assurance that: (1) safe disposal of high level radioactive waste and spent fuel in a mined geologic repository is technically feasible: (2) one or more mined geologic repositories for commercial high level radioactive waste and spent fuel will be available by the years 2007-09, and that sufficient repository capacity will be available within 30 years beyond expiration of any reactor operating license to dispose of existing commercial high level radioactive waste and spent fuel originating in such reactor and generated up to that time; (3) high level radioactive waste and spent fuel will be managed in a safe manner until sufficient repository capacity is available to assure the safe disposal of all high level radioactive waste and spent fuel; (4) if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the expiration of that reactor's operating license at that reactor's spent fuel storage basis, or at either on-site or off-site independent spent fuel storage installations; and (5) safe independent on-site or off-site spent fuel storage will be made available if such storage capacity is needed.

TMI-1 Design and Procedures Issues Resolved or Narrowed

In *Metropolitan Edison Company et al.* (Three Mile Island Nuclear Station, Unit 1), CLI-84-11, 20 NRC 1 (1984), the Commission completed its review of five issues related to plant design and procedures, resolving four of those issues in favor of restart. In connection with the design of the emergency feed water (EFW) system the Commission—while noting that a board has the discretion to examine any system, to determine whether it poses an unacceptable risk—found nothing in the record that raised a reasonable question regarding the reliability of the TMI-1 EFW. The Commission also concluded that two functions of the PORV were backups to other systems, and hence that those functions did not require the PORV safety-grade. Similarly, in light of improvements in systems interactions at TMI-1, the Commission found reasonable assurances of safety on the issue of systems interaction and did not require a TMI-1 specific study. With respect to the main steam line rupture detection system (MSLRDS), the Commission agreed with both the Licensing and Appeal Boards' finding that MSLRDS modifications are not required prior to restart. Finally, the Commission concluded that the existence of generic rulemaking on environmental qualification of electrical equipment did not preclude plant specific challenges based on allegations of plant-specific deficiencies: thus, environmental qualifications remains an issue in the TMI-1 proceeding. However, because the TMI-1 proceeding is of limited scope (i.e, issues having a nexus to the TMI-2 accident), the relevant environmental qualifications issue is similarly limited. Because the record did not include environmental qualification information on the limited issue properly before the board, the Commission directed the staff to certify the status of qualification of TMI-1 electrical equipment for TMI-2 type accidents. If the staff could not certify the equipment, the Commission directed the licensee to provide specific justifications for interim operation.

NRC Concurrence Not Rulemaking

In *NRC Concurrence in High-Level Waste Repository Safety Guidelines Under the Nuclear Waste Policy Act of 1982,* CLI-83-26, 18 NRC 1139 (1983), the Commission rejected a petition for rulemaking in connection with its concurrence role under Section 112(a) of the Nuclear Waste Policy Act of 1982. In the Commission's view, where one agency is concurring in another agency's action, it is not the act of concurrence but the underlying substantive rule that is of interest to the public. Where the agency promulgating the substantive rule has itself provided for public notice and comment, a parallel formal opportunity to provide written comments by the public is not legally required. The Commission did, however, provide parties who had previously demonstrated an interest in the NRC's concurrence decision an opportunity to make oral presentations.

Full Power License Granted to Diablo Canyon

In a series of orders, the Commission lifted its suspension of the Diablo Canyon low power license and then granted a full power license to Pacific Gas & Electric Company (PG&E). In *Pacific Gas and Electric Co.* (Diablo Canyon Nuclear Power Plant, Units 1 and 2, CLI-81-30, 14 NRC 950 (1981)), the Commission suspended PG&E's low-power license on the basis of new seismic and quality assurance program information. In two decisions, the Commission lifted that suspension. In CLI-83-27, 18 NRC 1146 (1983), the Commission lifted that suspension, in part to permit fuel loading and precriticality testing at the Diablo Canyon facility. In reaching its decision, the Commission concluded that parties to the proceeding did not have a statutory right to such a prior hearing and the Commission did not, as a matter of its discretion, intend to grant such a right in its prior suspension order. Based on inspections, studies, evaluations and reviews undertaken subsequent to the suspension order, the Commission found that no significant safety issues existed, and the risk to the public health and safety was extremely low, with respect to fuel loading and pre-criticality testing, since no actions which could lead to a self-sustaining nuclear chain reaction would be authorized.

In CLI-84-12, 19 NRC ——— (1984), the Commission determined that there was no reason to depart from its previous decision in *San Onofre,* CLI-81-33, in which the Commission held that the NRC's regulations "do not require consideration of the impacts on emergency planning of earthquakes which cause or occur during an accidental release." However, the Commission decided to initiate a rulemaking to "address whether the potential for seismic impacts on emergency planning is a significant enough concern for large portions of the motion to warrant the amendment of the regulations to specifically consider those impacts." The Commission also determined that the issuance of a full-power license for Diablo Canyon need not be delayed until the conclusion of that rulemaking.

Finally, in CLI-84-13, 19 NRC ——— (1984), the Commission permitted the Atomic Safety and Licensing Board's fourth and final Partial Initial Decision, LBP-82-70, 16 NRC 756 (1982), authorizing the issuance of a full-power operating license to Pacific Gas and Electric Company (PG&E), to become effective. The Commission also considered several other issues, some of which arose as a result of the unique circumstances associated with this plant. The Commission found that Supplement 27 to the Safety Evaluation Report adequately addressed the uncontested full-power technical issues raised by the staff. The Commission also found that: Supplement 25 to the Safety Evaluation Report adequately addressed conditions in the low-power license regarding small and large bore piping and its supports; and that Supplement 24 to the Safety Evaluation Report adequately addressed all other items which had been identified by the NRC staff as requiring resolution prior to full-power operation.

In addition to passing on various plant-specific issues, the Commission determined that new seismic information concerning the character of the Hosgri Fault did not require a stay of the proceeding and that the regulations did not require the specific consideration of the effects of earthquakes on emergency planning. Similarly, the Commission found that full-power operation need not be deferred pending the conclusion of the investigation by the Office of Investigations regarding allegations of harassment of PG&E contractor quality inspectors.

Application of Attorney-Client Privilege To Applicant Employee-Witness

In *Duke Power Company, et al.* (Catawba Nuclear Power Station, Units 1 and 2 (S.C.)), CLI-83-31, 18 NRC 1303 (1983), the Commission denied the applicant's motion for a stay of an Appeal Board decision permitting counsel for the Intervenor, subject to some limitations, to approach the applicant's employee-witnesses in order to seek their cooperation. In denying the stay, the Commission rejected the applicant's assertion that employee-witnesses were "clients" of applicant's counsel within the meaning of the attorney-client privilege. In the Commission's view, not every employee of the applicant was a "client" simply because they were or may be called upon to testify at a licensing proceeding. Rather, in the absence of a showing that individuals are something more than employees and witnesses of the applicant, counsel for a party is free to contact a witness regarding the underlying facts at issue in a licensing proceeding. The Commission noted, however, that the Appeal Board decision at issue did prohibit any inquiry into the existence or nature of any communications between the employee-witness and counsel for the applicant.

JUDICIAL REVIEW

The more significant litigation involving the Commission either resolved during fiscal year 1984 or pending at the close of the fiscal year is summarized below.

Pending Cases

Cranston, et al. v. Reagan, et al. (D.D.C. Civil Action No. 84-1545)

In this action against the President, the Secretaries of State and Energy, the Director of the Arms Control and Disarmament Agency and the five NRC Commissioners,

NRC Chairman Nunzio J. Palladino (center) visited Duke Power Company's Catawba nuclear power plant in South Carolina in October 1984, prior to Commission action on the company's request for an operating license for Unit 1. See text for discussion of an earlier dispute over attorney-client privileges, involving this licensee.

three members of Congress and six environmental groups are challenging defendants' approval and implementation of certain "Agreed Minutes" to the Agreements for Cooperation with Sweden and Norway. Defendants claim that the provisions of the minutes, which provide for the advance, long-term consent of the United States to the transfer of spent reactor fuel subject to the Agreements to France and the United Kingdom for purposes of reprocessing, violate the Nuclear Nonproliferation Act. Plaintiffs argue that approval of reprocessing can only be done on a case-by-case basis. The Government is seeking to dismiss the case on jurisdictional grounds because the issues raised are non-justiciable political questions, petitioners lack standing and Congress did not contemplate judicial challenges such as the present action. Plaintiffs have filed a motion for summary judgment.

Deukmejian v. NRC (D.C. Cir. No. 82-1549)
In May 1982, California Governor Brown challenged the NRC's Appeal Board decision approving the seismic design bases for the Diablo Canyon nuclear facility. In July 1982, the court granted the NRC's motion to hold the case in abeyance pending the NRC's completion of administrative proceedings for either a low-power or full-power license for this facility. In the interim, the Commission advises the court at sixty day intervals as to the status of the administrative proceedings. The attorney for Governor Deukmejian, successor in interest to Governor Brown, has stated that he will move the court to dismiss this case.

General Electric v. NRC (D.C. Cir. No. 80-2496)
Prairie Alliance v. NRC(C.D. Ill. No. 80-2095)
General Electric v. NRC(C.D. Ill. No. 80-2244)
General Electric v . NRC(7th Cir. No. 84-2066)
In May, 1980 the Prairie Alliance sued the NRC under the Freedom of Information Act (FOIA) to compel disclosure of the General Electric Nuclear Reactor Study known as the "Reed Report." Upon reconsideration of its decision to withhold the report, the Commission, on a 2-2 vote, was unable to muster a majority to claim any FOIA exemption for the report and hence ordered its release. In response to that decision to release, the General Electric Co. in October 1980 filed an action in the District of Columbia to enjoin release of the report and to require its return to General Electric (GE). The District Court for the District of Columbia transferred the case to Illinois where the *Prairie Alliance* case had been filed, and enjoined the Commission from releasing the Reed Report pending disposition of the case by the court in Illinois. In November 1980, the Illinois District Court granted summary judgment in the Government's favor sustaining the Commission's decision to release the Reed Report. In June 1984, the court rejected GE's motion for reconsideration and reaffirmed its earlier decision. GE immediately appealed to the Seventh Circuit and obtained from that court a stay of the District Court's order pending appeal. The parties are waiting for the Seventh Circuit to decide the matter.

Guard v. NRC (9th Cir. No. 83-7844)
In November 1983, Guard sought review in the Ninth Circuit Court of Appeals of a Commission order of September 16, 1983, authorizing the full-power operation of San Onofre Unit 3, and deleting a condition on the operating license regarding off-site medical service arrangements. That condition on operation was deleted in the full-power license based on a Licensing Board decision which found that the arrangements for off-site medical services were consistent with 10 CFR § 50.47(b)(12), as interpreted by the Commission in CLI-83-10, 17 NRC 528 (1983).

In December 1983, the parties filed a joint motion to transfer the Guard proceeding to the D.C. Circuit where another then-pending case (*Carstens v. NRC*) was challenging the San Onofre operating licenses. The Ninth Circuit transferred *Guard* to the D.C. Circuit in February 1984. All briefs in this case have been filed and oral argument is scheduled for December 19, 1984.

Joseph W. Johnston v. NRC, et al. (7th Cir. No. 84-1583) (On appeal from N.D. Ill. No. 83-C-3615)

Rockford Newspapers, Inc.-v.-NRC, et al. (N.D. Ill. No. 83 C-20074)
In August 1983, the American Civil Liberties Union sought a declaratory judgment that the Government in the Sunshine Act, 5 U.S.C. 552b, applies to proceedings before NRC Licensing Boards. In November 1983, the Government requested the District Court to dismiss the case or, in the alternative, grant summary judgment to the Government. In February 1984, the District Court granted the summary judgment in the Government's favor. The matter is pending before the Seventh Circuit Court of Appeals.

Kerr-McGee Nuclear Corp. v. NRC (10th Cir. No. 80-2043)
Uranium Mining and Milling Council, et al. v. NRC (No. 80-2271)

Western Nuclear Corp. v. NRC (No. 80-2269)
United Nuclear Corp. v. NRC (No. 80-2043)
In October 1980, Kerr-McGee, later joined by a number of other uranium milling companies, petitioned the Tenth Circuit to review the Commission's Uranium Mill Licensing Requirements (see 45 *Fed. Reg.* 65521 (Oct. 3, 1981)). Petitioners challenge the Commission's regulations on a number of grounds, including alleged insignificance of the radon risk, asserted excessive cost of complying with the regulations and the NRC's failure to await promulgation of EPA standards. In March 1982, the 10th Circuit upheld the NRC's mill tailings regulations in their entirety (673 F.2d 1124). In May 1982, Kerr-McGee filed for rehearing. In October of that year, the Tenth Circuit vacated its prior judgment. Rehearing *en banc* was scheduled but subsequently deferred in response to several Commission requests. This case remains in abeyance pending the completion of administrative action.

Lorion v. NRC (D.C.Cir. No. 82-1132)
In February 1982, Joette Lorion sought review by the D.C. Circuit Court of Appeals of the NRC's decision denying her request that Turkey Point Unit 4 (Fla.) be shut down for a steam generator inspection. Lorion alleged that the Commission acted unlawfully (1) in treating her letter requesting such action as a petition under 10 CFR 2.206 and (2) in denying her request. In July 1983, the D.C. Circuit Court upheld the NRC's action in treating Lorion's letter under 10 CFR 2.206 but held *sua sponte* that the courts of appeals lack subject matter jurisdiction to review denials by the Nuclear Regulatory Commission of requests under 10 CFR 2.206 for enforcement action against NRC licensees (712 F.2d 1472). The court stated that jurisdiction to review such denials lies initially in the district court. The NRC's Petition for Rehearing with a Suggestion for Rehearing *en banc* was denied in September 1983. A petition for a writ of certiorari from the Supreme Court was filed by the U.S. Solicitor General in December 1983. In March 1984, the Supreme Court granted the petition and oral argument was held in October 1984.

New England Coalition on Nuclear Pollution, et al. v. NRC (D.C. Cir. No. 82-1581)
In July 1982, the petitioners challenged the NRC's final rule which eliminated financial qualification reviews for public utility licensees (see 47 *Fed. Reg.* 13750 (March 31, 1982)). In February 1984, the D.C. Circuit issued an order holding that the rule was not adequately supported by its accompanying statement of basis and purpose, and remanded it to the agency (727 F.2d 1127). The court questioned the internal consistency of the Commission's explanation for dispensing with the financial qualifications review for electric utilities. The court found that the Commission's reasoning, if supported by the facts, would apply generally to all license applicants and would not support a rule which singled out utilities for special treatment. In April 1984, the Commission responded to the court's remand by promulgating a proposed rule reinstating financial qualification review at the construction permit stage, but eliminating review at the operating license stage for regulated utilities on the ground that such utilities will recover all reasonable costs of safe operation through the ratemaking process. In June 1984, the Commission issued a Statement of Policy which declared the remanded rule valid pending further action to comply with the court's mandate. In July 1984, NECNP filed a Petition for a Writ to Enforce the court's mandate, contending that the court's mandate vacated the March 31, 1984, rule, and, therefore, the Commission's June 1984 Statement of Policy was in violation of the mandate. On August 16, 1984 the Commission approved its proposed final qualifications rule. On September 20, 1984, the D.C. Circuit granted the Commission's unopposed motion to dismiss as moot the petition for a writ of enforcement.

San Luis Obispo Mothers for Peace v. NRC (D.C. Cir. No. 84-1410)
In August 1984, plaintiffs filed a petition for review by the D.C. Circuit of the Commission's decision allowing the Licensing Board decision authorizing a full power license for Diablo Canyon to become effective. Plaintiffs also moved for a preliminary injunction to stay operation of the reactor. The D.C. Circuit granted the stay motion on August 17, 1984, and established an expedited briefing schedule. All briefs have been filed and oral argument has been held. The court lifted the stay in a brief order following oral argument.

Union of Concerned Scientists v. NRC (D.C. Cir. No. 82-2053)
In September 1982, the Union of Concerned Scientists (UCS) challenged NRC's July 1982 amendments to the emergency planning rules permitting (1) issuance of initial licensing decisions without the results of preparedness exercises and (2) staff authorization of low power operating licenses without any review of off-site emergency preparedness (47 *Fed. Reg.* 30232 (July 13, 1982)). The Attorney-General of Massachusetts then intervened in the lawsuit. In October, UCS filed a petition for rulemaking in which it asked, in effect, that the NRC reconsider the exercise portion of the rule. Subsequent discussions confirmed that the exercise rule was the focus of the UCS lawsuit (see 47 *Fed. Reg.* 51889 (November 18, 1982)). The parties agreed to hold this case in abeyance until March 1983 to allow the NRC time to act on the UCS petition. In December 1983, the court granted the motion. The NRC denied the UCS petition on April 12, 1983 (48 *Fed. Reg.* 16691). In May 1984, the D.C. Circuit Court of Appeals vacated NRC's July 1982 amendments to the emergency planning rules. The court ruled that the Atomic Energy Act does not permit the Commission to exclude the results of emergency preparedness exercises from operating license hearings. The D.C. Circuit subsequently denied the NRC petition for rehearing and suggestion for rehearing *en banc*. On October 31, 1984, the intervenor utilities filed before the Supreme Court a petition for a writ of certiorari. The D.C. Circuit's mandate remained stayed pending the Supreme Court's disposition of the case.

Resolved Cases

Carstens v. NRC, No. 83-1879 (D.C. Cir., September 7, 1984)
A unanimous panel of the U.S. Court of Appeals for the District of Columbia Circuit affirmed the Commission's decision on the adequacy of the seismic design of Units 2 and 3 of the San Onofre (Cal.) Generating Station. In reaching its decision, the court emphasized the Commission's broad discretion to determine what technical specifications are necessary to protect the public health and safety and the deferential role of the court in reviewing

complex regulatory decisions. In particular, the court rejected petitioner's argument that uncertainty in the science of seismology coupled with the regulatory requirement that seismic inquiries be conducted in a conservative manner required the NRC to accept the worst prediction of earthquake potential proffered to it.

General Public Utilities Corp, et al. v. U.S. (E.D. Pa. No. 81-4950); 3rd Cir. No. 83-1017

In December 1981, the owners and operators of the Three Mile Island Unit 2 (Pa.) nuclear facility sued the United States, alleging damages in excess of $4 billion resulting from the accident at the facility. Plaintiffs theories of liability are that the United States, in its role as a regulator, violated statutory, regulatory or other self-imposed requirements and failed to warn the licensee of defects in the equipment, analyses, procedures and training, or, alternatively, failed to direct the licensee to correct certain deficiencies. In November 1982, the District Court denied the Government's motion to dismiss this case on both the discretionary function and the misrepresentation exemptions to the Tort Claims Act. However, recognizing that these issues were close and important, the District Court certified an immediate appeal to the Third Circuit. Subsequent to briefing and argument, the Third Circuit deferred its decision until it could consider the impact of the U.S. Supreme Court's decision in *United States v. Varig Airlines,* concerning the discretionary function and misrepresentation exceptions to government liability under the Federal Tort Claims Act. In September 1984, the Third Circuit reversed the District Court and ordered the case dismissed.

Philadelphia Newspapers, Inc. v. NRC (D.D.C. No. 83-1330) *on appeal* (D.C. Cir. No. 83-1698)

In May 1983, Philadelphia Newspapers, Inc. (Philadelphia Inquirer), sued the NRC claiming that the Commission could not close a proposed Commission meeting on TMI under Exemption 10 of the Sunshine Act. In the alternative, the Philadelphia Inquirer argued that the public interest required that the meeting be open to the public. In June 1983, the District Court heard argument on Philadelphia Newspapers' request for a preliminary injunction and cross-motions for summary judgment. On June 3, the court granted NRC's motion for summary judgment, holding that the proposed meeting fell within Exemption 10 and that the NRC did not act arbitrarily or capriciously in deciding that the public interest did not require opening of the meeting. On June 23, 1983, plaintiff appealed. On appeal to the D.C. Circuit, the Court of Appeals held that the Commission could close a TMI-1 restart meeting devoted to the on-the-record restart proceeding conducted by the Licensing Board but was required to open meetings devoted to whether the Licensing Board's decisions should be made immediately effective during the pendency of administrative appeals (727 F.2d 1195). In May 1983, the court denied the Commission's motion asking the court to modify or clarify its decision.

Rockland County & New York Public Interest Research Groupv. NRC (2nd Cir. No. 83-4225)

In August 1983, Rockland County and the New York Public Interest Research Group (NYPIRG) filed a petition in the D.C. Circuit to review the Commission's June 10, 1983 order declining to initiate enforcement action against the Consolidated Edison Company for emergency preparedness deficiencies at the Indian Point facility. Petitioners sought, among other things, to overturn three Commission decisions and to obtain a court order suspending reactor operations until the emergency preparedness deficiencies are cured.

The NRC moved to transfer the case to the Second Circuit. In December 1983, the D.C. Circuit granted the motion to transfer. Pursuant to agreement of the parties for voluntary withdrawal of the petition, the Second Circuit on May 29, 1984 dismissed the petition for review.

Union of Concerned Scientists v. NRC (D.C. Cir. No. 82-2000) (Environmental Qualifications)

In August 1982, the Union of Concerned Scientists sought review by the D.C. Circuit Court of Appeals of Commission action which suspended the June 30, 1982 deadline for documentation and completion of environmental qualification of safety-related equipment as required by a Commission order of May 27, 1982. Petitioner contended that this suspension violated the Atomic Energy Act and that it was promulgated without notice and opportunity for comment in violation of the Administrative Procedure Act (APA). In June 1983, the court vacated the rule, holding among other reasons for its decision, that hearing and notice requirements of the Atomic Energy Act (AEA) preclude use of the APA "good cause" exception in NRC rulemaking, at least when the rulemaking explicitly amends reactor licenses (711 F.2d 370). The Commission moved the court to vacate this part of the opinion. After further briefing on the issue of the availability of the "good cause" exception in NRC rulemakings, the court on December 5, 1983, denied the motion and issued its mandate.

Management and Communication

CHAPTER 13

Progress on Consolidation

The 2,500 NRC employees working in the Washington, D.C. area are located in 10 separate buildings in the District of Columbia and Maryland. The NRC and the Government Services Agency (GSA) are developing options to consolidate NRC offices in Bethesda, Md., or in the District of Columbia. The GSA has obtained expressions of interest from several real estate developers in providing the necessary space at a single venue. Options are to include a single or multi-building tenancy, to be delivered in phases within 18 to 36 months of the lease award, which is scheduled for summer of 1985.

STRENGTH AND STRUCTURE

Personnel Management

In fiscal year 1984, the NRC used 3,441 staff years in carrying out its mission. This number includes part-time and temporary workers and consultants, as well as full-time permanent staff. Total expenditure of staff years was within 1 percent of the OMB target of 3,416 staff years.

Commission and Director Changes

Commissioner Victor Gilinsky's term ended on June 30, 1984, and on July 5, 1984, Lando W. Zech was appointed to the Commission, bringing it back to its full strength of five members.

In September 1984, Sharon R. Connelly was appointed Director, Office of Inspector and Auditor, succeeding James J. Cummings.

In October 1984, Robert D. Martin was appointed Regional Administrator of Region IV, Dallas, Texas, succeeding John T. Collins.

Recruitment

In fiscal year 1984, NRC hired 374 people and lost 251 through an attrition rate of 7.7 percent per year. The agency's recruitment program included visits to 30 college campuses and participation in approximately 10 job fairs during the year. About 14 percent, i.e., 50, of the new hires for the year were entry-level professionals.

Training and Development

During 1984, NRC gave added emphasis to its Intern and Upward Mobility Programs. In September, the NRC began recruiting for its third class of candidates for the Senior Executive Service Candidate Development Program. The NRC also participated in the Office of Personnel Management's Women's Executive Leadership Program.

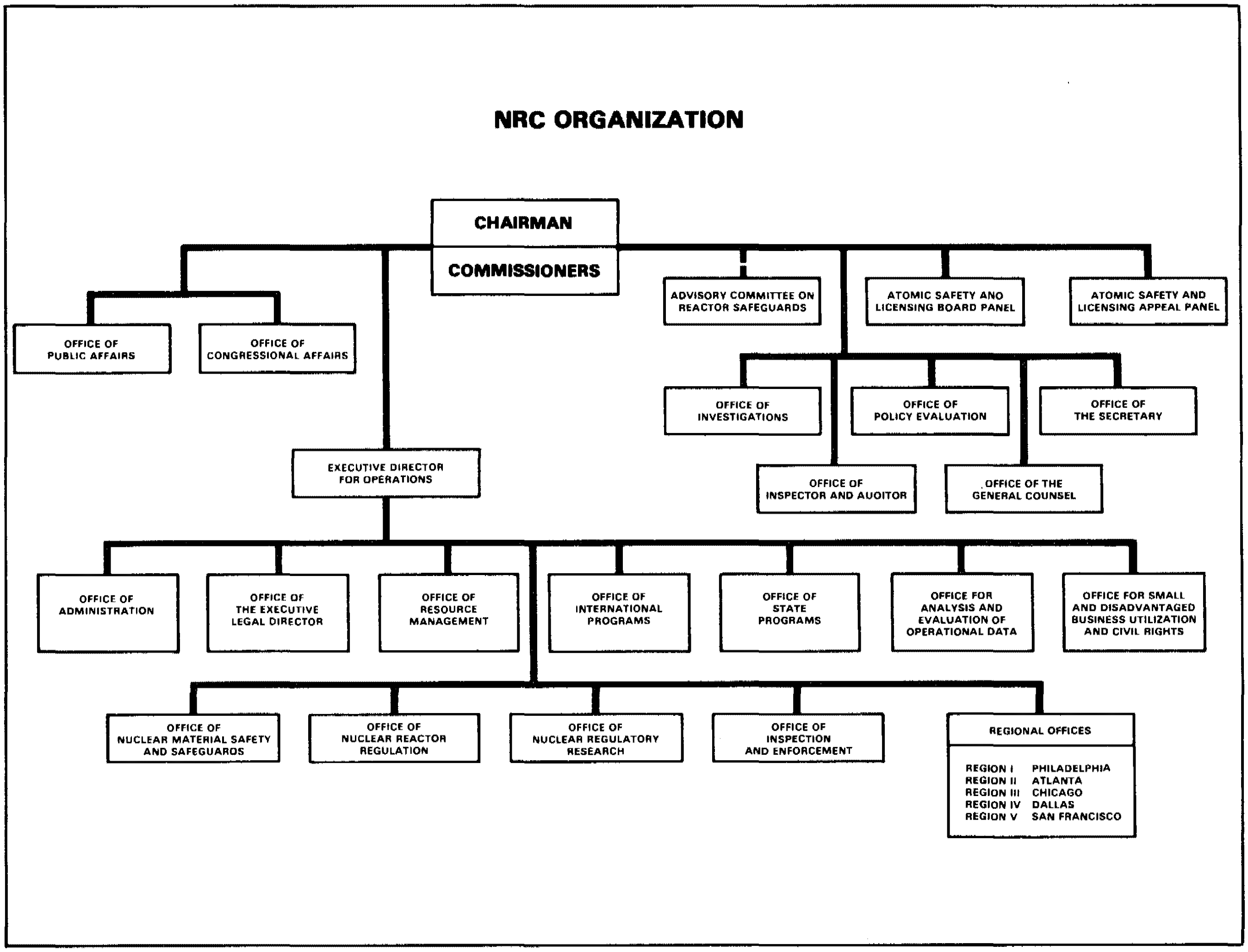
The NRC provides over 60 different technical courses in reactor technology and methodology to agency inspectors and other technical employees. Nineteen additional courses are available to improve executive and management skills, and a separate segment of 5 courses is provided to enhance secretarial and clerical performance. NRC employees also participate in a wide range of private sector and government-wide education and development activities.

ORGANIZATIONAL CHANGES

Within the Office of Nuclear Reactor Regulation (NRR), the Clinch River Breeder Reactor Program organization was abolished because funding for the Clinch River Breeder Reactor was discontinued by the Congress. Some essential work being done under the program was transferred to a new Advanced Reactors Group in NRR's Division of Safety Technology.

The Office of Nuclear Regulatory Research was reorganized to accommodate shifting mission priorities with reduced staff resources. The previous five-division structure was realigned into four divisions. The new divisions are titled Engineering Technology, Accident Evaluation, Risk Analysis and Operations, and Radiation Programs and Earth Sciences.

The Vendor Inspection Program was moved to the Office of Inspection and Enforcement from the Agency's Region IV office to provide Headquarters direction and management to this nationwide program, enhance implementation of Commission-level policy and guidance on vendor-related issues, and improve interaction between vendor inspection programs and other Headquarters programs.



Decentralization of NRC Activities

Late in 1981, the Commission concluded that there would be advantages to bringing regulatory functions as close as practicable to the people and facilities affected by them. Consequently, the Commission developed policy goals calling for expansion of the NRC regional office operations. The NRC organizational structure was changed in October 1981 to bring the regional offices under the direct control of the Executive Director for Operations (EDO), and the new post of Deputy Executive Director for Regional Operations and Generic Requirements was created to assist the EDO in managing regional operations.

Throughout 1982 and 1983, the scope of regional activity was carefully expanded. By the end of 1983, the Commission's policy goals had been achieved. In March 1984, the Commission issued a final policy statement on regionalization, and all of the regulatory functions planned for the Regions had been transferred to them (see Table 1).

EMPLOYEE-MANAGEMENT RELATIONS

Incentive Awards

NRC managers recognized high quality work performed by staff members during 1984 with 174 special achievement awards, 309 high quality performance increases, 56 certificates of appreciation, 38 SES bonuses, 4 distinguished service awards, 27 meritorious service awards, and 3 equal employment opportunity awards.

Two NRC executives received Presidential Rank Awards. William J. Dircks, Executive Director for Operations, received the Distinguished Executive Rank Award. James P. O'Reilly, Administrator of Region II, Atlanta, Ga., received the rank of Meritorious Executive.

Labor Relations

In July 1984 the Collective Bargaining Agreement between the NRC and the National Treasury Employees

Table 1. NRC Headquarters Functions Transferred to Regional Offices

(Headquarters Office in parentheses)

1. Operating Reactor licensing technical review (NRR).
2. Licensing authority for Fort St. Vrain (NRR—to Region IV).
3. Administer reactor operator license examinations (NRR).
4. Uranium mill tailings (NMSS—to Region IV).
5. Authority to issue materials license (NMSS).
6. Review safeguards license amendments which do not decrease effectiveness for reactors and SNM facilities (NMSS).
7. Conduct transportation route surveys and review contingency plans for spent fuel and Category 1 SNM shipments (NMSS—to Region III).
8. Perform closeout surveys and terminations of uranium fuel fabrication licenses (NMSS).
9. Maintain oversight of 10 CFR 70 licenses for advanced fuel (Pu) plants that have initiated decontamination and decommissioning activities (NMSS).
10. Issue proposed civil penalties (IE).
11. Issue orders and make 10 CFR 2.206 decisions consistent with the transfer of licensing authority from IE, NRR, and NMSS.
12. Review License Amendments of Emergency Plans for Operating Reactors.
13. Observe and appraise the annual emergency preparedness exercises for operating reactors (IE).
14. Provide legal assistance to Regional Administrators of functions to review severity level III violations, proposed civil penalties and orders, 2.206 decisions, material licenses and mill tailings licenses (ELD).
15. Provide State Agreement Officer (SP).
16. Continue State liaison functions.
17. Perform budget formulation/execution and management information reporting activities.
18. Perform various administrative support services.

Union (NTEU) came up for comprehensive renegotiation. NTEU had indicated its desire to renegotiate major portions of the three-year old agreement. The NRC and NTEU have agreed to begin negotiations in the fall of 1984.

Mid-term negotiations on the current Agreement were completed in September 1984. The NRC and NTEU reached agreement on the provision of a compressed work schedule for bargaining unit employees. Eligible bargaining unit employees may begin a fixed 5-4/9 compressed work schedule in November 1984. By opting to work a combination of eight 9-hour days and one 8-hour day per pay period, employees would have one extra day off per pay period. All other mid-term issues were dropped, to be brought up again as part of the comprehensive renegotiations.

Approximately 127 grievances, 51 mid-contract negotiations, and 11 unfair labor practice charges were handled during Fiscal Year 1984.

INSPECTION AND AUDIT

The NRC's Office of Inspector and Auditor (OIA) continued to pursue agency goals concerning the efficiency and integrity of NRC operations, and issued 20 audit reports—and 12 follow-up audit reports—toward improving various NRC programs and activities. OIA also completed 61 investigative actions, of which 31 were Reports of Investigation, and referred seven matters to the Department of Justice for review and possible action. In addition, annual inspections of the Office of Investigations' Headquarters and five regional field offices were performed.

Highlights of some of the audit reports issued during 1984 follow.

Reactor Licensing

In October 1984, OIA issued a report on the results of an audit of the review process for issuing an operating license for a nuclear power plant. The report disclosed three areas in which OIA believed the licensing process needed improvement: the experience of Project Managers; the scheduling of operating license reviews; and the utilization of the technical review staff in view of a declining licensing workload. OIA made recommendations to correct the problems identified.

Internal Controls

On September 8, 1983, the President signed the Federal Managers' Financial Integrity Act of 1982 which requires the head of each executive agency to prepare annual statements to the President and the Congress stating whether the agency's system of internal accounting and administrative control do or do not comply with standards established by the Comptroller General. OIA conducted a limited review of the NRC's evaluation of the agency's internal accounting and administrative controls to determine whether it was carried out in a reasonable and prudent manner and consistent with the guidelines and standards. OIA's report to the Commission in December 1983 contained the results of its review.

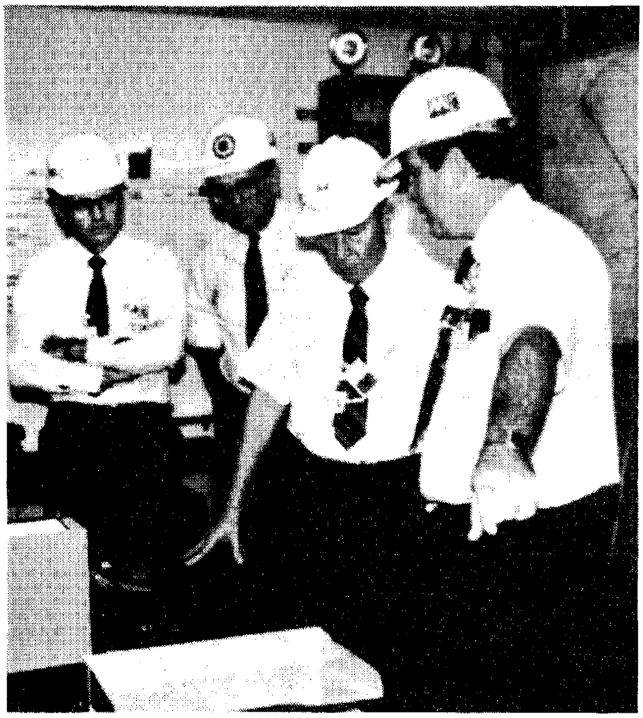
The report notes that during OIA's review nothing came to light that would indicate that the NRC did not comply with established guidelines and standards. The report also contains recommendations to further strengthen the evaluation process and make NRC managers more aware of their responsibilities.

Committee to Review Generic Requirements

In March 1984, OIA issued a report to the Commission which examined the implementation of the Committee to Review Generic Requirements (CRGR) charter and the effectiveness of the CRGR in controlling the number and nature of requirements imposed by NRC on licensees. The report identified four areas in which OIA questioned whether the charter was being fully implemented and one area in which OIA believed the staff was exceeding the provisions of the charter. The report also concluded that because some generic requirements were still being imposed on licensees without CRGR's review, the CRGR was not totally effective in controlling the imposition of generic requirements on licensees.

IE's Quality Assurance Inspection Program For Reactors Under Construction

This August 1984 audit report addressed NRC's inspection program for reactors under construction with particular emphasis on the regional offices' implementation of selected quality assurance inspection modules. OIA found that some regional offices were experiencing difficulty in maintaining their planned inspection schedules and that some of the contributing causes were: (1) giving a higher priority to inspection resources for operating plants because of public health and safety; (2) diverting inspection resources to perform reactive inspections, investigate allegations, and follow-up on other activities, causing construction inspections to be missed or not completed; and (3) failure to fully implement program requirements for inspecting utility QA program management. OIA concluded that if NRC is to rely on a utility's QA program and have confidence that utility management is dedicated to its implementation, then NRC must test the effectiveness of the utility's QA program through consistent routine inspections.



Before the Commission issued an operating license to the Mississippi Power and Light Company for its Grand Gulf Unit 2 nuclear power station, Chairman Nunzio Palladino and NRC staff members visited the plant. Chairman Palladino is shown being briefed by an official of the licensee.

Concerns Expressed by Sandia National Laboratory

In January 1984, representatives of Sandia National Laboratory told the Commission that NRC staff offices apply "Pressure For Research Not to Impact Licensing Issue Resolution," and "Pressure For Research Not to Impact Previous Licensing Decisions." As a result, the Commission directed OIA to conduct an audit of these statements. OIA's August 23, 1984 report concluded that the problems experienced by Sandia related to the unique nature of Sandia's Equipment Qualification research effort. OIA did not identify any programmatic or policy changes needed to solve the problems but believed NRR management needed to be more aware of Sandia's unique problems. The report noted that communication with the national laboratories and Sandia in particular had improved since Sandia expressed its concerns.

Program Direction, Management, and Utilization of Research

This August 1984 audit report included an overview of RES' functions and responsibilities as well as a broad look at research direction, management, and utilization. OIA concluded that RES' coordination efforts for research projects were adequate; the planned research pertained to licensing and other regulatory needs; planned research appeared to be geared toward operating plants rather than the licensing of new plants; a substantial portion of RES funds were expended at DOE national laboratories but legitimate reasons substantiated their use; and technical direction was provided to the labs by RES. However, OIA did identify four areas in need of improvement.

Technical Assistance

In a December 1983 audit report OIA concluded that NRR's overall management practices associated with the use of technical assistance needed improvement. Specific improvements were needed in planning technical performance during the life of projects and in closing out contracts and task orders. OIA also felt that all of the 1981 NRR task force recommendations in the area of technical assistance should be implemented by NRR. The report contained 30 recommendations to improved NRR's use of technical assistance.

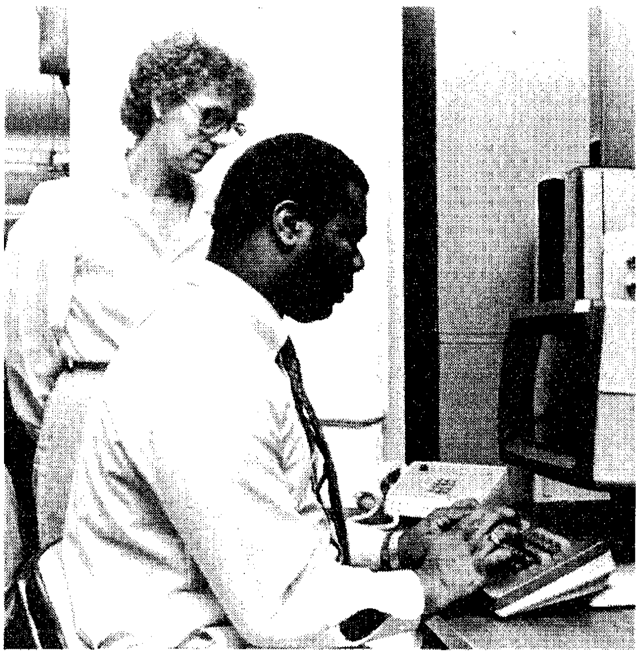
IE's Program Assessment Function

OIA's August 1984 report concluded that IE management had made a "good faith" attempt to implement a program assessment effort. Although assessments were being performed, OIA believed that adequate resources may not be available to conduct comprehensive assessments of IE's two largest programs, the reactor construction and operations inspection programs. OIA also believed that the lack of recent field inspection experience by the people performing assessments, may have caused a lack of credibility in the eyes of the regional people being assessed. The report contained recommendations which, if implemented, would enhance the IE assessment function.

DOCUMENT CONTROL SYSTEM

The NRC Document Control System (DCS) data base has grown to over 1 million data records which reference over 12 million pages of information. Over 80 percent of this information is available to the general public; the remainder is restricted to the NRC in order to ensure nuclear plant security and protect proprietary information.

To enhance DCS capabilities the NRC has established a tracking system to monitor on-going resource expenditures, develop a resource utilization audit trail and accurately and efficiently project resource requirements and priorities for future NRC document processing needs. An in-house system review identified over 50 recommendations for improving system operations and effectiveness.



John Davenport, of the NRC's Public Document Room (PDR), negotiates a computer search of the PDR's machine-readable bibliographic data base, as PDR Records Manager Jean Rathje looks on.

The first action, scheduled for completion in 1985, will reduce annual operating costs by more than $300,000.

In fiscal year 1984 the agency also began a major effort to upgrade current DCS operating software and hardware and to develop on-line subject search and report capabilities.

FUNDING AND BUDGET MATTERS

Contracting and Reimbursable Work

The NRC programs are supported by substantial amounts of contractual effort for confirmatory research and technical assistance. Reimbursable arrangements with the Department of Energy (DOE) and other Federal agencies provide much of this support. Other assistance comes from contracts with commercial sources and through grants for research related programs. (N.B. Specific research programs are described in Chapter 11.)

Contracts with commercial firms for technical assistance, research work, and general purchases totaled approximately $50 million in fiscal year 1984. Contracts under the Small Business Innovation Research Program totaled $1.1 million and grants with educational and non-profit institutions totaled $1.9 million. These projects are administered through the Division of Contracts, Office of Administration.

NRC LICENSE FEES

In fiscal year 1984, the Commission collected $16.7 million in fees for the processing of applications, permits, licenses and approvals and for routine health and safety inspections. These fees are sent to the Department of the Treasury for deposit as miscellaneous receipts. Table 2 shows a breakdown of these collections. The total collected since fees were first imposed (October 1968 through September 1984) is $177.4 million. Of this amount, $6.5 million has been refunded to licensees because of a 1974 Supreme Court decision negating annual license fees.

New Fee Schedule

The Commission adopted a revised schedule of fees which became effective June 20, 1984. The revised schedule is designed to recover more completely NRC costs incurred for providing services to identifiable recipients, including both materials and facility applicants and licensees. Unlike the previous fee schedule which specified fixed fees for each fee category, the revised schedule, in some instances, eliminated the ceilings or upper limits on fees charged for the review of facility and major fuel cycle applications. Fees for some of these applications are now based on the actual cost of staff-hour and contractual services expended on the reviews. Likewise, inspection fees for certain facility and major fuel cycle licensees and radioactive waste burial and storage facilities are based on the actual costs incurred in conducting the inspections.

The revised rule established fees for the first time for non-routine or reactive inspections and for Part 55 requalification and replacement operator examinations. The fees charged for non-routine inspections are based on the actual cost to conduct the inspections; fees to administer the operator examinations are based on the actual costs incurred, up to the maximum fee specified in the schedule. Other fees in the revised schedule were adjusted to take into account the NRC's increased licensing and inspection costs.

Table 3 provides information relating to the costs of issuance of the operating licenses and the fees paid for them.

PUBLIC COMMUNICATION

Public Information

Media Workshops. A series of one-day educational seminars were conducted for the news media by the five

Table 2. FY 1984 License Fee Collections

Fees	*Materials*	*Facilities*	*Total*
Applications	$ 165,998		$ 165,998
Construction Permits*		884,275	884,275
Operating Licenses**		2,648,129	2,648,129
Amendments	301,132	1,965,450	2,266,582
Renewals	442,543		442,543
Inspection Fees	1,210,100	8,390,408	9,600,508
Special Projects	946	525,295	526,241
Approvals	109,706	50,000	159,706
Decommissioning		150	150
TOTALS	$2,230,425	$14,463,707	$16,694,132

*Collected for the withdrawn application for Black Fox Units 1 and 2.

**Four operating licenses were issued which were subject to the full cost requirement of the March 23, 1978 rule up to the maximum fee. One operating license (Catawba 1) was issued under the revised rule where the fee to be assessed will be based on the full licensing and inspection costs, up to a maximum of $3.1 million.

Regional Offices of the Nuclear Regulatory Commission for the fourth consecutive year. Reporters and editors from national wire services, broadcast networks, news magazines and daily newspapers were briefed on the fundamentals of nuclear power and the risks of exposure to radiation. The seminars were held in Portland, Ore., May 2; Seattle, Wash., May 3; Minneapolis, Minn., June 5; Charlotte, N.C., June 20; Houston, Tex., October 25; and Baltimore, Md., November 20.

Public Affairs. The NRC's Office of Public Affairs maintained daily contact with the news media and the public by arranging interviews and press briefings, issuing public announcements and responding to thousands of telephone calls. Public announcements of Commission rulemaking, public hearings, proposed fines against licensees and other agency activities were distributed to the news media, the scientific community, universities and the general public.

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 material. Located at 1717 H Street, N.W., in Washington, D.C., the PDR is a specialized documentatin center that houses significant documents on nuclear regulation and makes them available to the public. Visitors to 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 radioactive wastes (their use, transport and disposal). The PDR features extensive accession listings and an on-line computer data base. The PDR staff was selected among the finalists for the U.S. Congressional Award for Exemplary Service to the Public (1982).

The PDR contains about 1.3 million documents, and the collection is enlarged by an average of 312 new items every day. During an average month, the PDR serves 1,250 users. The staff retrieves an average of 6,728 files per month containing multiple documents or microfiche for researchers on-site and provides about 2,200 documents in response to letters and telephone requests. The public purchased 3.4 million pages of documents and about 33,000 microfiche cards in fiscal year 1984.

Persons wishing to use or obtain additional information regarding the holdings, file organization, reference, reproduction services and procedures of the PDR may call (202) 634-3274 or write to the U.S. Nuclear Regulatory Commission, Public Document Room, 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

Table 3. Cost of OL Issuances in FY 1984

Operating Licenses	*Issue Date*	*Licensing Cost*	*Inspection Cost*	*Total Cost*	*Fee Paid*
LaSalle 2	12/16/83	$ 563,860	$ 418,139	$ 981,999	$ 302,800
WNP-2	12/20/83				1,024,000
Susquehanna 2	03/23/84				302,800
Callaway 1	06/11/84				1,024,500
Catawba 1	07/18/84				2,750,467

(N.B. Catawba 1 fees represent costs through 6/23/84; billed but not paid.)

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).

LOCAL PUBLIC DOCUMENT ROOMS

Through its local public document room (LPDR) program, the NRC makes document collections available to the public near the sites of proposed and operating nuclear power plants. These collections contain information about the licensing, construction, operation, inspection, and regulation of nearby nuclear facilities. They include documents dealing with health and safety, safeguards, environmental, and antitrust considerations. LPDR collections usually are located in university or public libraries that have copying facilities and are open to the public during the evening and on weekends. Currently, there are more than 110 LPDRs in operation. (See Appendix 3 for a list of LPDR locations.)

To inform the public about the existence and availability of documents at the local level, NRC publishes a quarterly newsletter and conducts evening workshops at individual LPDR libraries. The workshops provide instruction to the public in identifying, locating, and retrieving information. A toll-free telephone number (1-800-638-8081) is available to library staffs and individuals who need rapid, convenient answers to questions about such topics as collection content, search strategies, use of reference tools and indices, and locating and retrieving information at LPDR sites.

COMMUNICATIONS AND DATA PROCESSING

The NRC is installing a secure voice network that will link all NRC Regional Offices and Headquarters and other Federal agencies that might be involved in nuclear safeguards incidents. The network will protect all classified and sensitive unclassified information transmitted by telephone. Similarly, secure telecommunications techniques have been established with selected licensees.

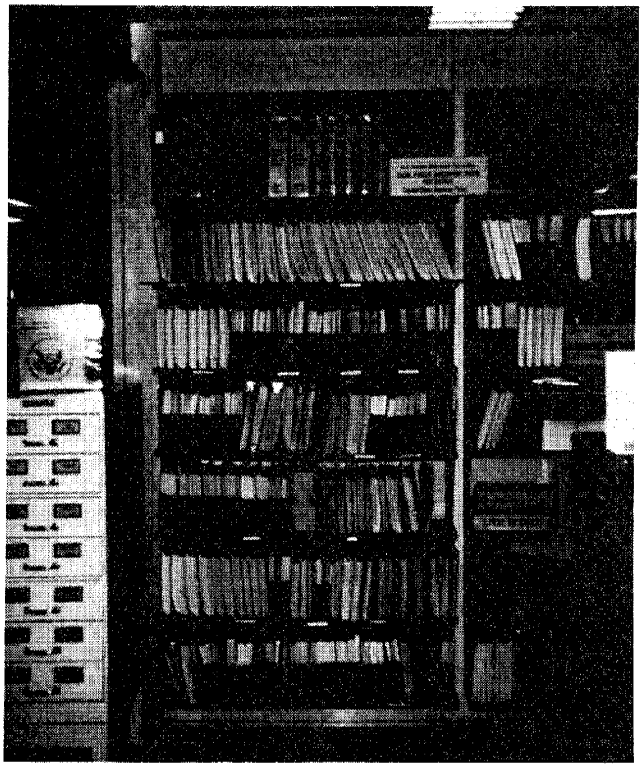
During the past year, the NRC has improved the protection of sensitive unclassified data processed on NRC automatic data processing (ADP) facilities. In addition to issuing policies and guidelines in accordance with Office of Management and Budget guidance, NRC has conducted ADP security surveys of agency and contractor facilities and undertaken risk analyses and other studies to assess their strengths and weaknesses.

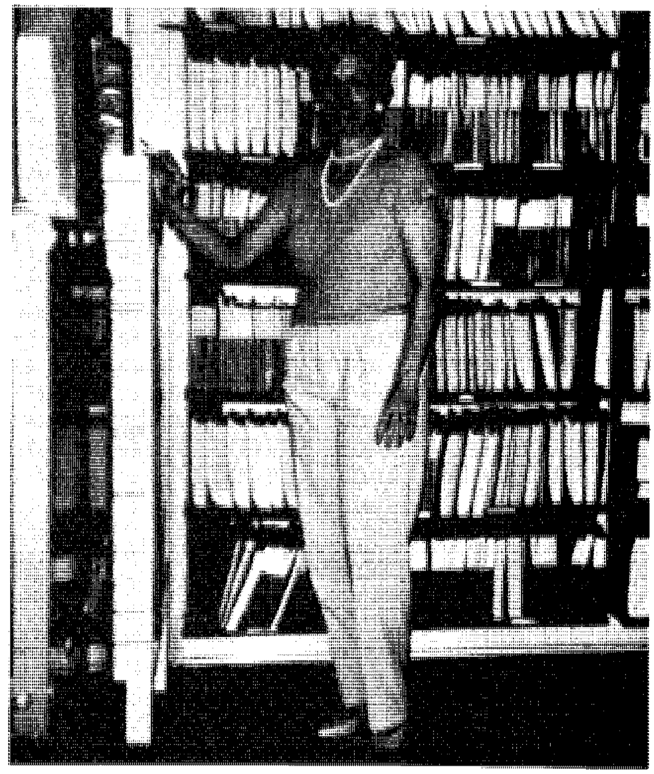
Office of Resource Management ADP Activity

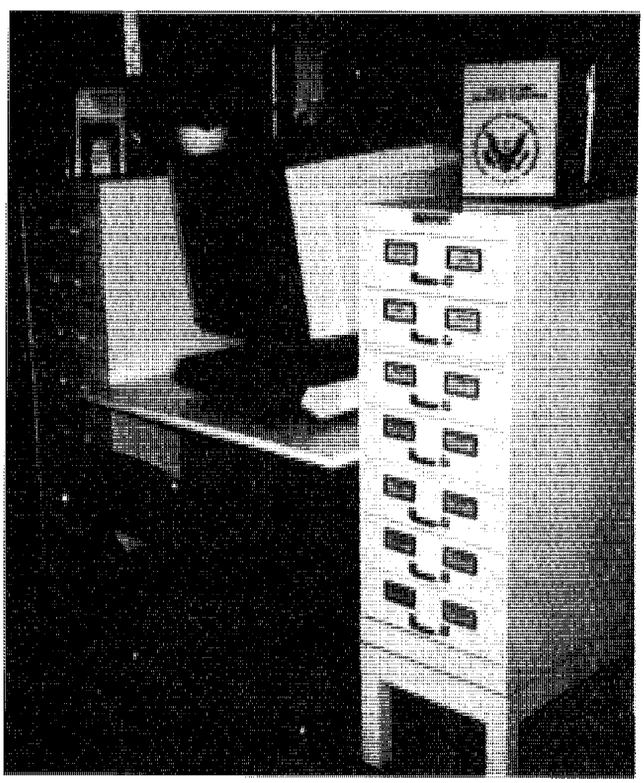
In the area of automatic data processing (ADP), the Office of Resource Management embarked on a comprehensive program in 1984 to improve the agency's data- and word-processing capability. Over the next few years, the primary ADP objective is to set up a user-oriented information network which will, by means of the newest technologies, integrate data and word processing and provide for the transfer of information to and from every organizational unit of the agency.

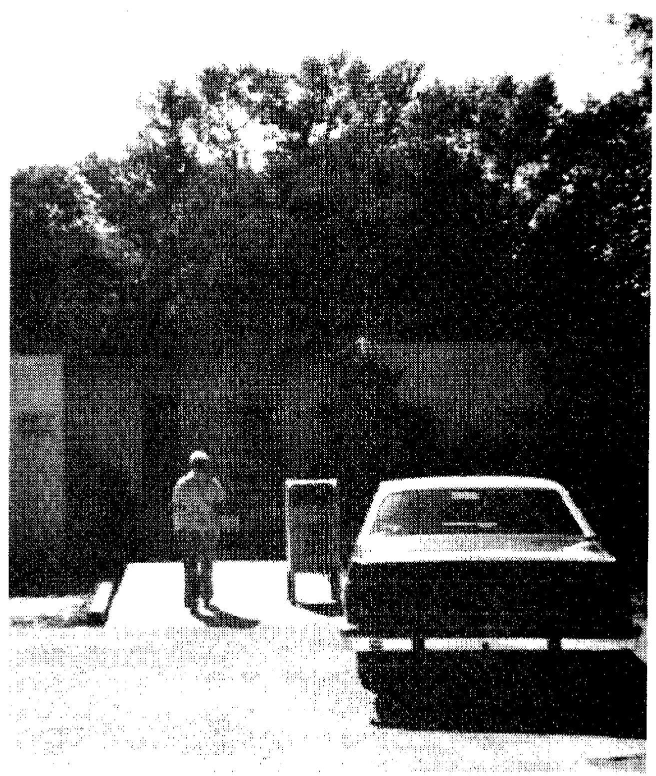
Achievement of this objective will mean that the entire NRC staff will be provided with access to central processors and systems. Information will be extracted from "shared" or "corporate" data bases, copied to microcomputers for local processing, and then electronically transmitted to the central systems for updating. Standard compatible software and processing techniques will provide flexibility for *ad hoc* reporting and "plain English" queries, thus increasing staff productivity and leaving more time for analysis, while reducing the need for programming intervention by the ADP staff. Reduction in the need for intervention will free the ADP staff to perform the more important task of systems analyses. This is an extremely significant development in times of budget and staff reductions, when new systems analysis might otherwise have to be carried out at some expense to the smooth and efficient operation of existing systems.

In order to assist the NRC ADP user-community and move forward to a shared, "corporate data" environment,









NRC's local Public Document Rooms (LPDRs) are, as a rule, set up in public libraries at communities near nuclear power or other facilities. This LPDR, at Crystal River, Fla., is the repository for all documents related to the nearby Crystal River nuclear power plant. Shown clockwise from upper left are: the hard-copy collection of plant documents, with (on left) the microfiche storage cabinet; Ms. Julie De Busk, Librarian, who handles the filing on nuclear plant matters; a front view of the Crystal River Public Library; and NRC-furnished LPDR equipment, including two microfiche storage cabinets, and a microfiche reader/printer.

the following major actions—along with countless minor ones—were taken in 1984:

Installation of Microcomputers. Several Headquarters and Regional Offices, along with all Resident Inspectors received system compatible microcomputers by the end of the year. The number of microcomputers installed in NRC tripled compared to 1983. At the end of 1982, NRC had only two microcomputers compared to over 230 at the end of 1984. When counted with other terminals, long-range plans call for a 2:1 staff-to-workstation ratio by 1987.

Initiation of the NRC Corporate Data Network (CDN). The NRC began the analysis of NRC's corporate data needs and the acquisition of a large data base management system package. These are the first two steps in the long-range improvement project for NRC, known as the NRC Corporate Data Network (CDN).

Establishment of an ADP Planning Staff. Prior to 1984, there was little top-down, agency-wide, comprehensive ADP planning in the NRC. In January, an ADP Planning Staff was established by reassigning valuable, senior-level ADP staff from operational functions. Since that time, the ADP Planning Staff has been planning NRC's long-range ADP efforts with guidance from NRC's ADP Steering Group composed of top-level managers from program and staff offices.

Creation of the Information Technology Services Section. An Information Technology Services (ITS) section was established within NRC to enable the staff to maximize the use of ADP equipment and software for both scientific and management information system applications. The ITS is divided into two major functional areas. Through the ITS Support Center, NRC staff members are provided with walk-in and telephone consulting. In addition, the Support Center provides computer and video-based tutorials, one-on-one demonstrations and equipment and software for trial use. Through the ITS Training Laboratory, NRC staff members are provided with ADP courses, hands-on microcomputer training and training for use with the NRC-accessible mainframe computers and the NRC-owned minicomputers.

Expansion of the NRC Office Automation Network. By the end of the year, a network of word processing/electronic document distribution systems was operating in the NRC which linked all major offices and geographical locations of the agency for the first time. Not only could clerical staff at any site prepare and electronically distribute documents, thereby realizing time savings, but professional staff using microcomputers could also take advantage of the system's large storage capacity, its electronic mail facility, high speed laser printers and its communications capability with agency computers through emulation features or file transfers. As the CDN is developed, data and word processing capabilities will be integrated through multi-purpose workstations.

Initiation of the NRC Software Improvement Program (SIP). The initial use of a "fourth-generation" database managment system (DBMS) package was made to improve access to existing NRC systems at the National Institutes of Health timeshared facility. The DBMS is intended to serve as a bridge to the CDN, emphasizing end-user computing while the CDN is beginning. With this package, users are allowed to submit on-line, ad hoc queries utilizing remote terminals and printers without the intervention or assistance of the ADP technical staff. Training in the use of this software is provided by the ITS. Systems which benefitted from use of the software in 1984 included the new NMSS Licensing Management System

Representatives of the Illinois Department of Nuclear Safety (IDNS) met with NRC Region III (Chicago) officials in 1984 to discuss a Memorandum of Understanding on low-level waste inspections. Shown left to right are Dennis Patovani (IDNS), John Cooper (IDNS), Roland Lickus (NRC), and Don Etchison, Director of IDNS.

The NRC established an Information Technology Services (ITS) section within the Office of Resource Management during 1984 to enable more and better-trained NRC people to use Automatic Data Processing (ADP) equipment. The main features of the new section are two educational and support facilities in Bethesda, Md., accessible to all NRC employees. The first facility, opened in October 1984, is the ITS Support Center, providing walk-in and telephone consultation to users of ADP resources. The second, opened soon afterwards, is the ITS Training Laboratory, a teaching facility offering hands-on training experience. Ms. Francine Goldberg, NRC's ITS Section Chief, is shown here (center foreground) at the inauguration of the Training Laboratory, briefing Commission and staff officials on the purposes and functions of the lab.

(LMS), the NRR Operator Licensing Tracking System (OLTS), the NRR Allegations Management System (AMS), and the multi-office Resource Information Tracking System (RITS). The DBMS not only improves existing applications, but is also used to develop new systems.

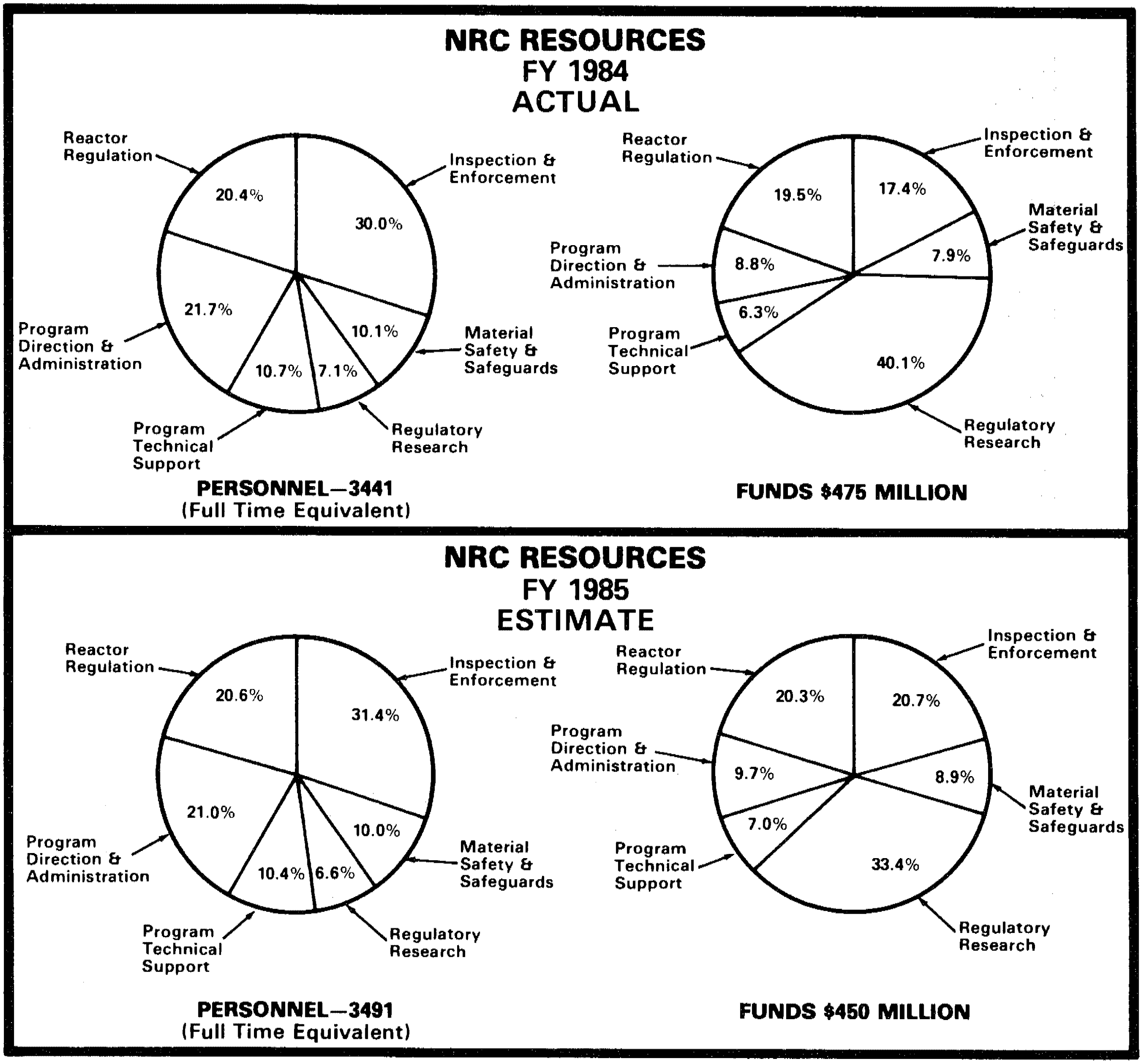
Support for the new NRC Operations Center. The NRC Operations Center will serve as the hub of NRC activity during emergency preparedness tests at reactor sites and for accidents involving nuclear materials and will provide NRC managers in Headquarters and the Regions with up-to-date information during the course of any event. The Division of Automated Information Services has the responsibility for providing ADP support for the Center. This includes the operation of two super-mini-computers and the development of the computer software for the radiation dose assessment system, an information management system, and the duty officer support system. These systems were operational by the end of 1984. As construction of the new facility in the Maryland National Bank Building in Bethesda, Md., was taking place during the year, the computers were maintained by contractors. In November, the two minicomputers were moved to their new location and RM/D staff assumed complete control over their operation. Final completion and initial operation of the new Center is expected early in 1985.

Expansion of Regional Office Support. Consistent with the expanding role of the Regional Offices, ADP support for the Regions was increased significantly. Recognizing the Regions' greater interaction with headquarters offices, numerous systems were installed beyond the more traditional inspection and enforcement systems. These systems included the Licensing Management System, the Operator Licensing Tracking System, the Allegations Management System and the Resource Information Tracking System. In addition, ADP support contracts were negotiated and implemented for each Region.

Development of the new NMSS Licensing Management System (LMS). An entirely new Licensing Management System (LMS) was developed for the Office of Nuclear Material Safety and Safeguards (NMSS) and the regional offices using a DBMS package and the NRC Office Automation Network. It will facilitate the issuance of materials licenses and provide user-friendly access to a database containing information pertaining to all materials licenses. In addition, it provides fee information, demographic data and inspection and enforcement data. The system will be operational in February 1985.

COMMISSION HISTORY PROGRAM

In November 1984, the University of California Press published a history of the first phases of nuclear power regulation in the United States entitled, *Controlling the Atom, The Beginnings of Nuclear Regulation, 1946-1962*. The 500-page book was co-authored by the two NRC professional historians, George T. Mazuzan and J. Samuel Walker. It is a comprehensive study of the early history of nuclear power regulation and it inaugurates a planned multi-volume series that will document nuclear regulatory history up to the present. *Controlling the Atom* reconstructs the context in which the Atomic Energy Commission (AEC), the predecessor of the Nuclear Regulatory Commission, established its regulatory programs. It weighs the relationship between the AEC's regulatory policies and its other major functions: developing and testing nuclear weapons and encouraging the expanded use of civilian atomic energy. The role of the many organizations and groups outside the AEC which have had an effect on the content and process of nuclear regulation is also assessed and their impact analyzed. This first volume provides a full and thoroughly documented account of



debates over such critical issues as the development of standards for protection against radiation hazards, the licensing of nuclear power reactors, the siting of nuclear plants, the framing of the Price-Anderson legislation, the disposal of radioactive wastes. The discussions range from matters as broad as the sustained public controversy over the effects of radioactive fallout from bomb testing to matters as specific as the structure of the AEC's internal regulatory organization. The documentation in the history comes from an array of government records and private manuscript collections. The book is designed as a resource for the general reader as well as an authoritative reference for agency staff

OFFICE OF SMALL AND DISADVANTAGED BUSINESS UTILIZATION/CIVIL RIGHTS

Small and Disadvantaged Business Utilization Program

In cooperation with the Division of Contracts, the following procurement preference and dollar thresholds were achieved in fiscal year 1984:

- $53,430,000 for total prime contracts greater than $10,000.
- $22,210,000 of this total for prime contract awards greater than $10,000 to small business.
- $9,505,000 for Section 8(a) awards.
- $943,000 for prime contracts to small and disadvantaged business other than 8(a).
- $1,114,000 for prime contracts to women-owned business concerns.
- $1,304,000 for subcontracts awarded to small business.
- $47,000 for subcontracts awarded to small and disadvantaged business.

During the year, 75 interviews were conducted with firms wanting to do business with the NRC, and 37 follow-up meetings were arranged with NRC technical personnel.

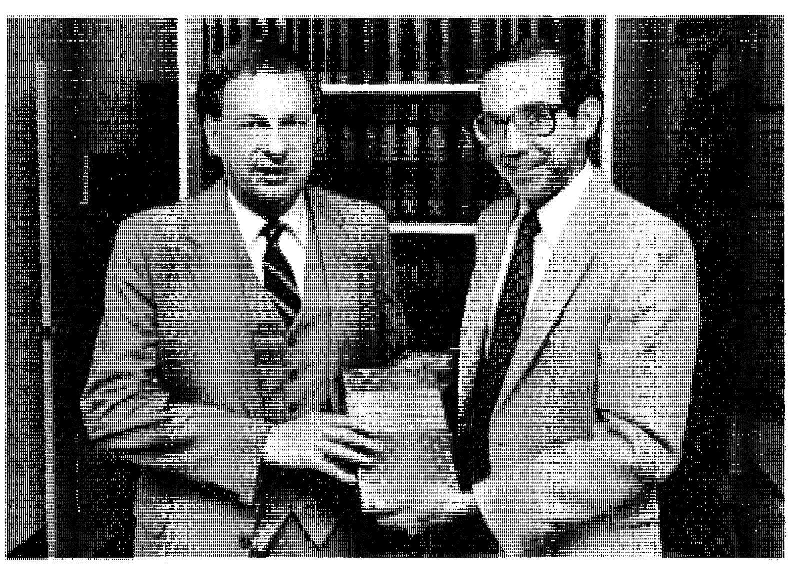
Civil Rights Program

The NRC Affirmative Action Plan was approved by the Equal Employment Opportunity Commission, with hiring goals established for each Office and Region of the agency.

On April 18,1984, the Office of Small and Disadvantaged Business Utilization/Civil Rights (OSBUD/CR) briefed the Commission on EEO and affirmative action programs. As a result of this briefing, staff was requested to prepare a Consolidated EEO Program Plan. The plan reviews NRC's current EEO programs and outlines a comprehensive series of initiatives in this area for fiscal year 1985.

OSBDU/CR has begun to hold quarterly meetings with the 18 headquarters Equal Employment Opportunity (EEO) counselors; the EEO counselors can often be effective in addressing concerns of employees as they arise and in preventing formal discrimination complaints by recognizing and dealing with potential problems early on.

In fiscal year 1984, the U.S. Department of Agriculture Graduate School, in conjuction with OSDBU/CR, developed and administered a comprehensive two-day training program for managers and supervisors. The program was designed to help participants develop, among other things, a more sophisticated awareness of historical



George T. Mazuzan, NRC Historian, and J. Samuel Walker, Associate Historian, display their recently published history of the early phases of nuclear power regulation, entitled *Controlling the Atom.*

and socio-economic factors which have caused—and could continue to cause—discrimination in the workplace, and to show participants how, in job-related ways, they could personally take action to alter discriminatory behavior patterns. A total of 424 NRC managers and supervisors participated in the program.

Federal Women's Program

During fiscal year 1985, a primary effort of the Federal Women's Program was to continue to monitor personnel policies and procedures related to the hiring, training and advancement of women and to make recommendations as appropriate. To raise the level of awareness among employees regarding women's issues, training in the prevention and elimination of sexual harassment was completed for all employees. The Federal Women's Program Advisory Committee also sponsored a special lunch-time program for all employees commemorating National Women's History Week.

Regional Federal Women's Program Coordinators were brought to headquarters to receive annual training, guidance and assistance to help them better advise and assist their Regional Administrators with respect to women's specific employment concerns and to the agency's EEO objectives for women.

Conferences were held with the agency's top management to address the concerns of NRC women in Headquarters and in the Regions. Attention to these concerns and other program initiatives were incorporated into the Federal Women's Program plan of action for fiscal year 1985 and subsequently into the EEO Consolidated Program Plan for fiscal year 1985.

FY 1983/1984 NRC Financial Statements

Balance Sheet (in thousands)

Assets	September 30, 1984	September 30, 1983
Cash:		
Appropriated Funds in U.S. Treasury	$ 169,677	$ 165,961
Other—Notes 1 & 3	10,527	11,560
	180,204	177,521
Accounts Receivable:		
Federal Agencies	132	127
Miscellaneous Receipts—Note 2	3,623	1,920
Other	50	50
	3,805	2,097
Plant:		
Completed Plant and Equipment	24,429	20,621
Less—Accumulated Depreciation	8,026	5,710
	16,403	14,911
Advances and Prepayments:		
Federal Agencies	–0–	–0–
Other	5,492	4,286
	5,492	4,286
Total Assets	$ 205,904	$ 198,815

Liabilities and NRC Equity	September 30, 1984	September 30, 1983
Liabilities:		
Funds Held for Others—Notes 1 & 3	$ 10,527	$ 11,560
Accounts Payable and Accrued Expenses:		
Federal Agencies	41,683	39,297
Other	24,004	20,416
Accrued Annual Leave of NRC Employees	12,285	11,271
Deferred Revenue—Note 3	–0–	–0–
Total Liabilities	$ 88,499	$ 82,544
NRC Equity: Balance at October 1	116,271	121,799
Additions:		
Funds Appropriated—Net	465,800	465,274
Non-Reimbursable Transfers from Other Gov't Agencies	–0–	277
	582,071	587,350
Deductions:		
Net Cost of Operations	446,249	451,301
Funds Returned to U.S. Treasury—Note 2	18,417	19,778
	464,666	471,079
Total NRC Equity	117,405	116,271
Total Liabilities and NRC Equity	$ 205,904	$ 198,815

Note 1. As of September 30, 1984, includes $3,886,510.05 of funds received under cooperative research agreements involving NRC, DOE, Euratom, France, Federal Republic of Germany, Japan, Austria, the Netherlands, Belgium, and the United Kingdom.

Also included is $5,972,245.90 of funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processing the applications has been finalized and, accordingly, are not available for NRC use. See Note 3.

Note 2. These funds are not available for NRC use.

Note 3. On March 24, 1978, 10 CFR 1 was revised. Contained therein by category of license are maximum fee amounts to be paid by applicants at the time a facility or material license is issued. Also, after the review of the license application is complete, the expenditures for professional manpower and appropriate support services are to be determined and the resultant fee assessed. In no event will the fee exceed the maximum fee for that license category, which generally has been paid. This could involve the refunding of a significant portion of the initial amount paid. Therefore, the revenue is recorded in a Deferred revenue account at the time of billing and is removed from this account and recorded in Funds Held for Others when the bill is paid. The balance in the Deferred revenue account consists of deferred revenue on billings issued but not collected. See Note 1.

Note 4. Represents current year cost of plant and equipment acquisitions for use at DOE facilities.

FY 1983/1984 NRC Statement of Operations
(in thousands)

	Fiscal Year, 1984 (October 1, 1983, thru September 30, 1984)	*Fiscal Year, 1983 (October 1, 1982, thru September 30, 1983*
Personnel Compensation	$ 143,643	$ 136,038
Personnel Benefits	16,196	14,719
Program Support	250,608	267,253
Administrative Support	41,516	38,324
Travel of Persons	10,475	8,847
Equipment (Technical)—Note 4	359	3,922
Construction—Note 4	–0–	–0–
Taxes and Indemnities	72	11
Refunds to Licensees	–0–	–0–
Representational Funds	2	1
Reimbursable Work	90	75
Increase in Annual Leave Accrual	1,013	1,216
Depreciation Expense	2,287	1,840
Equipment Write-Offs and Adjustments	200	27
Total Cost of Operations	$ 466,461	$ 472,273
Less Revenues:		
Reimbursable Work for Other Federal Agencies	92	73
Fees (Deposited in U.S. Treasury as Miscellaneous Receipts—Note 2):		
Material Licenses	2,274	1,482
Facility Licenses	14,750	16,567
Other	3,096	2,850
Total Revenue	20,212	20,972
Net Cost of Operations Before Prior Year Adjustments	446,249	451,301
Prior Year Adjustment	–0–	–0–
Net Cost of Operations	$ 446,249	$ 451,301

U.S. Government Investment in the Nuclear Regulatory Commission
(in thousands)
(From January 19, 1975 through September 30, 1984)

Appropriation Expenditures:	
Fiscal Year 1975 (January 19, 1975 through June 30, 1975)	$ 52,792
Fiscal Year 1976 (July 1, 1975 through September 30, 1976)	226,248
Fiscal Year 1977 (October 1, 1976 through September 30, 1977)	230,559
Fiscal Year 1978 (October 1, 1977 through September 30, 1978)	270,877
Fiscal Year 1979 (October 1, 1978 through September 30, 1979)	309,493
Fiscal Year 1980 (October 1, 1979 through September 30, 1980)	377,889
Fiscal Year 1981 (October 1, 1980 through September 30, 1981)	416,867
Fiscal Year 1982 (October 1, 1981 through September 30, 1982)	441,902
Fiscal Year 1983 (October 1, 1982 through September 30, 1983)	514,613
Fiscal Year 1984 (October 1, 1983 through September 30, 1984)	462,084
	3,303,324
Unexpended Balance of Appropriated Funds in U.S. Treasury September 30, 1984	169,677
Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975	429
Funds Appropriated—Net	3,473,430
Less:	
Funds Returned to U.S. Treasury—Note 2	140,712
Assets and Liabilities Transferred from Other Federal Agencies Without Reimbursement	1,673
Net Cost of Operations from January 19, 1975 through September 30, 1984	3,213,640
Total Deductions	3,356,025
NRC Equity at September 30, 1984 as Shown on Balance Sheet	$ 117,405

Appendix 1

NRC Organization

(As of December 31, 1984)

COMMISSIONERS

Nunzio J. Palladino, Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Lando W. Zech, Jr.

The Commission Staff

General Counsel, Herzel H.E. Plaine
Office of Policy Evaluation, John E. Zerbe, Director
Office of Public Affairs, Joseph J. Fouchard, Director
Office of Congressional Affairs, Carlton C. Kammerer, Director
Office of Inspector and Auditor, Sharon R. Connelly, Director
Secretary of the Commission, Samuel J. Chilk
Office of Investigations, Ben B. Hayes, Director

Advisory Committee and Panels

Advisory Committee on Reactor Safeguards, Jesse C. Ebersole, 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, William J. Dircks
Deputy Executive Director for Operations, Jack W. Roe
Deputy Executive Director for Regional Operations
and Generic Requirements, Victor Stello, Jr.
Assistant for Operations, Thomas A. Rehm

Program Offices

Office of Nuclear Reactor Regulation, Harold R. Denton, Director
Office of Nuclear Material Safety and Safeguards, John G. Davis, Director
Office of Nuclear Regulatory Research, Robert B. Minogue, Director
Office of Inspection and Enforcement, Richard C. DeYoung, Director

Staff Offices

Office of Administration, Patricia G. Norry, Director
Executive Legal Director, Guy H. Cunningham
Office of Resource Management/Controller, Learned W. Barry
Office of International Programs, James R. Shea, Director
Office of State Programs, G. Wayne Kerr, Director
Office for Analysis and Evaluation of Operational
Data, Clemens J. Heltemes, Jr., Director
Office of Small and Disadvantaged Business Utilization/
Civil Rights, William B. Kerr, Director

Regional Offices

Region I Philadelphia, Pa., Thomas E. Murley, Regional Administrator
Region II Atlanta, Ga., James P. O'Reilly, Jr., Regional Administrator
Region III Chicago, Ill., James G. Keppler, 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 licenses nuclear power, test and research reactors under 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 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 a 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 Inspection and Enforcement develops and oversees programs of inspection of nuclear facilities and materials licensees in order to determine whether those facilities are constructed and operations are conducted in compliance with license provisions and Commission regulations; in order to identify conditions that may adversely affect the protection of the public health and safety, of nuclear materials and facilities, or of the environment; and in order to provide a basis for recommending issuance or denial of licenses. IE develops and oversees a program of investigation of accidents, incidents and allegations of improper actions that involve nuclear material and facilities; enforces NRC regulations and license provisions; and manages and directs all NRC actions related to emergency preparedness, including evaluations of State and local emergency plans performed by the Federal Emergency Management Agency (FEMA). It also performs audits of its programs as carried out by NRC Regional 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; maintains a forecast of matters for future Commission consideration; processes and controls Commission correspondence; maintains the Commission's official records; controls the handling and service of documents issued and received 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 serves the Commission in a variety of legal capacities. The Office assists the Commission in the review of Appeal Board decisions, of petitions seeking direct Commission relief, and of rulemaking proceedings; the Office drafts the legal documents necessary to carry out the Commission's decisions. The General Counsel provides a legal analysis of proposed legislation affecting the Commission's functions and assists in drafting legislation and preparing testimony. The General Counsel also represents the Commission in court proceedings, frequently in conjunction with the Department of Justice.

The Office of Policy Evaluation plans and manages activities involved in performance of an independent review of judgments and positions developed by the NRC staff which require policy determinations by the Commission. The Office also conducts analyses and projects which are either self-generated or requested by the Commission.

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 Public Affairs plans and administers NRC's program to inform the public of Commission policies, programs and activities and keeps NRC management informed of public affairs activities of interest to the Commission. OPA reports directly to the Chairman.

The Office of Congressional Affairs provides advice and assistance to the Commission and senior staff on congressional matters, coordinates NRC's congressional relations activities, and maintains liaison for the Commission with congressional committees and members of Congress. OCA reports directly to the Chairman.

SUPPORT STAFF

The Office of Administration directs the Agency's programs for organization and personnel management; security and classification; technical information and document control; facilities and materials license fees; contracting and procurement; rules, proceedings and document services, including administration of Freedom of Information Act and Privacy Act requests; management development and training; telecommunications; transportation services; management of space; and other administrative services.

The Office of Resource Management develops and maintains NRC's financial and manpower management programs, including policies, procedures and standards of accounting, budgeting, cost analysis, resource planning and analysis, and automated data processing systems development and support. The Office provides management information for other offices and issues special reports from the NRC to Congress, other government agencies and the public. The Office assists NRC offices in statistical matters and in the budget process, keeping the EDO and Commission informed on programs and issues of significance. RM also maintains liaison with the Office of Management and Budget, the Congress, other agencies of government, and the private sector.

The Office of the Executive Legal Director provides legal advice and services to the Executive Director for Operations and staff, including representation in administrative proceedings involving the licensing of nuclear facilities and materials, and the enforcement of license conditions and regulations; counseling with respect to safeguards matters, contracts, security, patents, administration, research, personnel, and the development of regulations to implement applicable Federal statutes.

The Office of International Programs plans and implements programs of international nuclear safety cooperation, creating and maintaining relationships with foreign regulatory agencies and international organizations; coordinates NRC export-import and international safeguards policies; issues export and import licenses; and coordinates responses by NRC to other agencies related to export-import actions and issues.

The Office of State Programs directs programs related to regulatory relationships with State governments and organizations and interstate bodies, manages the NRC State Agreements program, administers the indemnification program and performs financial qualifications reviews of applicants and licensees. The Office also verifies that applicants are not in violation of the antitrust laws.

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 oversees action taken in response to the feedback and assesses the overall effectiveness of the agency-wide operational safety data program, serving as a focal point for interaction with the ACRS and industry groups involved in operational safety data analysis and evaluation.

The Office of Small and Disadvantaged Business Utilization/ Civil Rights develops and implements the NRC's program in accordance with the Small Business Act, as amended, insuring that appropriate consideration is given to labor surplus area firms and women-owned businesses. The Office develops and recommends NRC policy providing for equal employment opportunity and develops, monitors and evaluates the affirmative action program to assure compliance with the policy. The Office also serves as contact with local and national public and private organizations with related interests.

OTHER OFFICES

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, 1984, the members were:

MR. JESSE C. EBERSOLE, Retired Head Nuclear Engineer, Division of Engineering Design, Tennessee Valley Authority, Knoxville, Tenn.

MR. DAVID A. WARD, Vice Chairman, Research Manager, Reactor Safety Research, E. I. du Pont de Nemours & Company, Savannah River Laboratory, Aiken, S.C.

DR. ROBERT C. AXTMANN, Professor of Chemical Engineering, Princeton University, Princeton, N.J.

DR. MAX W. CARBON, Professor and Chairman of Nuclear Engineering Department, University of Wisconsin, Madison, Wisc.

DR. WILLIAM KERR, Professor of Nuclear Engineering and Director of the Office of Energy Research, University of Michigan, Ann Arbor, Mich.

DR. HAROLD W. LEWIS, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.

DR. CARSON MARK, Retired Division Leader, Los Alamos Scientific Laboratory, Los Alamos, N.M.

MR. CARLYLE MICHELSON, Retired Principal Nuclear Engineer, Tennessee Valley Authority and Retired Director, 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.

DR. DAVID OKRENT, Chairman, School of Engineering and Applied Science, University of California, Los Angeles, Cal.

MR. GLENN A. REED, Retired Plant Manager, Pt. Beach Nuclear Power Plant, Wisconsin Electric Power Company, Two Rivers, Wisc.

DR. FORREST J. REMICK, Acting Vice President for Research and Graduate Studies and Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.

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.

MR. CHARLES J. WYLIE, Retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.

Atomic Safety and Licensing Board Panel

PANEL MEMBERS:

CHIEF ADMINISTRATIVE JUDGE B. PAUL COTTER, JR., ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Executive) Robert M. Lazo, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Technical) Frederick J. Shon, ASLBP 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 LAWRENCE BRENNER, 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, Retired Physicist, Union Carbide Corporation, Oak Ridge, Tenn.

JUDGE JAMES H. CARPENTER, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE HUGH K. CLARK, Retired Attorney, E.I. duPont deNemours & Company, Kennedyville, Md.

JUDGE RICHARD F. COLE, ASLBP Environmental Scientist U.S. Nuclear Regulatory Commission, Bethesda, Md.

JUDGE FREDERICK P. COWAN, Retired Physicist, Brookhaven National Laboratory, Boca Raton, Fla.

JUDGE MICHAEL A. DUGGAN, Economist, University of Texas, Austin, Tex.

JUDGE GEORGE A. FERGUSON, Physicist, Howard University, Washington, D.C.

JUDGE HARRY FOREMAN, Medical Doctor, 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 ANDREW C. GOODHOPE, Retired Administrative Law Judge, Federal Trade Commission, Wheaton, Md.

JUDGE HERBERT GROSSMAN, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, 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, Lawrence Livermore Laboratory, Livermore, Cal.
JUDGE ERNEST E. HILL, Nuclear Engineer, Lawrence Livermore Laboratory, Livermore, Cal.
JUDGE FRANK F. HOOPER, Marine Biologist, 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, Retired Physicist, Oak Ridge Laboratories, Oak Ridge, Tenn.
JUDGE JAMES L. KELLEY, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
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 JAMES A LAURENSON, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
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, Oklahoma State University, Stillwater, Okla.
JUDGE GARY L. MILHOLLIN, Attorney, University of Wisconsin, Madison, Wis.
JUDGE MARSHALL E. MILLER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
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 PAUL W. PURDOM, Retired Environmental Engineer, Decatur, Ga.
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 QUENTIN J. STOBER, Biologist, University of Washington, Seattle, Wash.
JUDGE SEYMOUR WENNER, Retired Administrative Law Judge, Postal Rate Commission, Chevy Chase, Md.
JUDGE SHELDON J. WOLFE, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

PROFESSIONAL STAFF:

STEVEN F. CROCKETT, Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DONNA D. DUER, Legal Intern, U.S. Nuclear Regulatory Commission, Bethesda, Md.
CHARLES J. FITTI, Executive Secretary, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELEANOR L. FRUCCI, Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELLEN C. GINSBERG, Legal Intern, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELVA W. LEINS, Assistant Executive Secretary, 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 which would otherwise be performed by the 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 panel 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. The Commission retains review authority over decisions and actions of appeal boards. The appeal board panel, on October 1, 1984, 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.
DR. REGINALD L. GOTCHY, 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.

PART-TIME MEMBERS:

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.
LYNN M. CLANCY, Law Clerk, 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, 1984, 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. FRANK H. DE LAND, Chief, Nuclear Medicine Department, Veterans' Administration Hospital, Lexington, Ky.
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. B. LEONARD HOLMAN, Chief, Clinical Nuclear Medicine, Department of Radiology, Brigham and Women's Hospital Boston, Mass.
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.
DR. JOSEPH B. WORKMAN, Associate Professor of Radiology, Duke University Medical Center, Durham, N.C.

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 deci- sional process regarding cleanup plans for Three Mile Island, Unit 2. The Panel consists of the following members representing agencies of the Common- wealth 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, Chairman, Mayor of Lancaster, Pa.
JOSEPH J. DINUNNO, Private Consultant, Annapolis, Md.
THOMAS B. COCHRAN, Senior Staff Scientist, National Resources Defense Council, Washington, D.C.
THOMAS GERUSKY, Director of the Pennsylvania Bureau of Radiation Protection, Department of Environmental Resources, Harrisburg, 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.
ROBERT G. REID, Mayor of Middletown, Pa.
GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.
JOEL ROTH, resident of Elizabethville, Pa.
THOMAS SMITHGALL, resident of Lancaster, Pa.
ANN TRUNK, resident of Middletown, Pa.
NIEL WALD, Professor of Radiation Health, Department of Radiology, University of Pittsburgh, Pittsburgh, Pa.

Appendix 3

Local Public Document Rooms

Copies of most documents originating in the NRC or submitted to it for review are placed in the Commission's Public Document Room (PDR) 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

- Ms. Anne E. Anderson
  Reference Librarian
  Houston-Love Memorial Library
  212 W. Burdeshaw Street
  P.O. Box 1369
  Dothan, Ala. 36302
  Joseph M. Farley Nuclear Plant

- Mrs. Peggy McCutchen
  Director
  Scottsboro Public Library
  1002 S. Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

ARIZONA

- Ms. Harriet Meckfessal
  Documents Librarian
  Sciences
  Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde Nuclear Generating Station

ARKANSAS

- Ms. Marifran Bustion
  Serials Librarian
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One

CALIFORNIA

- Ms. Judy Klapproth
  Director
  Eureka-Humboldt County Library
  636 F Street
  Eureka, Cal. 95501
  Humboldt Bay Power Plant

- Mrs. Fontayne Holmes
  Senior Librarian
  West Los Angeles Regional Library
  11360 Santa Monica Boulevard
  Los Angeles, Cal. 90025
  UCLA Training Reactor

- Miss Diana Gin
  Librarian
  Government Documents Collection
  Sacramento Public Library
  828 I Street
  Sacramento, Cal. 95814
  Rancho Seco Nuclear Generating Station

- Ms. Ann Douthett
  Reference Librarian
  San Clemente Public Library
  242 Del Mar
  San Clemente, Cal. 92672
  San Onofre Nuclear Generating Station

- Mr. Chi Su Kim
  Head
  Government Documents and Maps Department
  Robert E. Kennedy Library
  California Polytechnic State University
  San Luis Obispo, Cal. 93407
  Diablo Canyon Nuclear Power Plant

- Mr. Gregory Cook
  Public Affairs Officer
  U.S. Nuclear Regulatory Commission, Region V
  1450 Maria Lane
  Suite 300
  Walnut Creek, Cal. 94596
  Vallecitos Boiling Water Reactor

COLORADO

- Miss Shirley Soenksen
  Reference Librarian
  Greeley Public Library
  City Complex Building
  919 7th Street
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Generating Station

CONNECTICUT

- Ms. Vickie Johnson
  Reference Librarian
  Russell Library
  123 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Plant

- Mr. Vincent Juliano
  Director
  Waterford Public Library
  49 Rope Ferry Road
  Waterford, Conn. 06385
  Millstone Nuclear Power Station

FLORIDA

- Ms. Cheryl P. Gardieff
  Supervising Librarian
  Crystal River Public Library
  668 N. W. First Avenue
  Crystal River, Fla. 32629
  Crystal River Nuclear Plant

- Mrs. Ramona Scott
  Circulation Librarian
  Charles S. Miley Learning Resources Center
  Indian River Community College
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Plant

- Ms. Renee Pierce
  Branch Librarian
  Miami-Dade Public Library
  Homestead Branch
  700 N. Holmstead Blvd.
  Holmstead, Fla. 33030
  Turkey Point Plant

- Ms. Esther Gonzales
  Documents Librarian
  Urban and Regional Documents Collection
  Environmental and Urban Affairs Library
  Florida International University
  Tamiami Trail and 107th Avenue
  Miami, Fla. 33199
  Turkey Point Plant

GEORGIA

- Mrs. Wynell Bush
  Librarian
  Appling County Public Library
  301 City Hall Drive
  Baxley, Ga. 31563
  Edwin I. Hatch Nuclear Plant

- Mrs. Juanita Smith
  Assistant Documents Librarian
  Burke County Library
  412 4th Street
  Waynesboro, Ga. 30830
  Alvin W. Vogtle Jr. Nuclear Plant

ILLINOIS

- Mrs. Penny A. O'Rourke
  Librarian
  Byron Public Library District
  218 W. Third Street
  P.O. Box 434
  Byron, Ill. 61010
  Byron Station

- Ms. Cheryle Rae Nyberg
  Assistant Law Librarian
  University of Illinois Law Library
  504 E. Pennsylvania Avenue
  Champaign, Ill. 61820
  Clinton Power Station

- Mrs. Betsy Taubert
  Library Clerk
  Vespasian Warner Public Library
  120 W. 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. Marie Hoscheid
  Head
  Reference Department
  Moline Public Library
  504 17th Street
  Moline, Ill. 61265
  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

- Mrs. Pam Wilson
  Morris Public Library
  604 Liberty Street
  Morris, Ill. 60451
  GE-Morris Facility

- Ms. Evelyn Moyle
  Documents Librarian
  Jacobs Memorial Library
  Illinois Valley Community College
  Rural Route 1
  Oglesby, Ill. 61348
  LaSalle County Station

- Mr. Richard A. Gray
  Librarian
  Business Science and Technology Department
  Rockford Public Library
  215 North Wyman Street
  Rockford, Ill. 61101
  Byron Station

- Mrs. Karen Stott
  Librarian
  Savanna Township Public Library
  326 Third Street
  Savanna, Ill. 61074
  Carroll County Station

- Ms. Kay Sauer
  West Chicago Public Library
  332 E. Washington Street
  West Chicago, Ill. 60185
  West Chicago site

- Mrs. Sharon Ruda
  Director
  Government Documents Collection
  Wilmington Public Library
  201 S. Kankakee Street
  Wilmington, Ill. 60481
  Braidwood Station

- Ms. Susan G. Clark
  Adult Services Librarian
  The Memorial Library Center
  Zion-Benton Public Library District
  2400 Gabriel Avenue
  Zion, Ill. 60099
  Zion Nuclear Plant

INDIANA

- Mrs. Charlene M. Peters
  Adult Services Librarian
  Madison-Jefferson County Public Library
  420 W. Main Street
  Madison, Ind. 47250
  Marble Hill Nuclear Generating Station

IOWA

- Ms. Janice Horak
  Reference Librarian
  Cedar Rapids Public Library
  428 Third Avenue, S.E.
  Cedar Rapids, Ia. 52401
  Duane Arnold Energy Center

KANSAS

- Ms. V. Sue Hatfield
  Head
  Government Documents Division
  William Allan White Library
  Emporia State University
  1200 Commercial Street
  Emporia, Kans. 66801
  Wolf Creek Generating Station

- Ms. Jan Brown
  Librarian
  NRC Documents Collection
  Washburn University School of Law
  Topeka, Kans. 66621
  Wolf Creek Generating Station

KENTUCKY

- Mr. Hunter Seitz
  Head
  Government Documents Division
  Louisville Free Public Library
  4th and York Streets
  Louisville, Ky. 40203
  Marble Hill Nuclear Generating Station

- Ms. Beverly B. Schneider
  Library Director
  Campbell County Public Library
  4th and Monmouth Streets
  Newport, Ky. 41071
  William H. Zimmer Nuclear Power 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

- Mrs. Barbara Shelton
  Librarian
  Wiscasset Public Library
  High Street
  P.O. Box 367
  Wiscasset, Me. 04578
  Maine Yankee Atomic Power Plant

MARYLAND

- Ms. Marcia G. Hammett
  Reference Librarian
  Calvert County Public Library
  Fourth Street
  P.O. Box 300
  Prince Frederick, Md. 20678
  Calvert Cliffs Nuclear Power Plant

MASSACHUSETTS

- Mrs. Margaret E. Howland
  Director of Learning Resources
  Library/Learning Resource Center
  Greenfield Community College
  1 College Drive
  Greenfield, Mass. 01301
  Yankee Rowe Nuclear Power Station

- Ms. Grace E. Karbott
  Library Associate
  Plymouth Public Library
  11 North Street
  Plymouth, Mass. 02360
  Pilgrim Nuclear Power Station

MICHIGAN

- Mr. Gig Stewart
  Library Director
  North Central Michigan College
  1515 Howard Street
  Petoskey, Mich. 49770
  Big Rock Point Nuclear Plant

- Mrs. Lelane Hardie
  Documents Librarian
  Kalamazoo Public Library
  315 South Rose Street
  Kalamazoo, Mich. 49007
  Palisades Nuclear Plant

- Ms. Sandra Krchmar
  Reference Librarian
  Grace A. Dow Memorial Library
  1710 W. St. Andrews Road
  Midland, Mich. 48640
  Midland Plant

- Ms. Janice Murphy
  Reference/Government Documents Coordinator
  Ellis Reference and Information Center
  Monroe County Library System
  3700 S. Custer Road
  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. Thomas Smisek
  Librarian
  Technology and Science Department
  Minneapolis Public Library
  300 Nicollet Mall
  Minneapolis, Minn. 55401
  Monticello Nuclear Generating Plant
  Prairie Island 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 Plant

MISSOURI

- Mrs. Evelyn Hillard
  Public Services Librarian
  Callaway County Public Library
  709 Market Street
  Fulton, Mo. 65251
  Callaway Plant

- Mr. B. J. Johnston
  Government Publications Librarian
  John M. Olin Library
  Washington University
  Skinker and Lindell Boulevards
  St. Louis, Mo. 63130
  Callaway Plant

NEBRASKA

- Mrs. Loy Mowery
  Director
  Auburn Public Library
  1118 15th Street
  P.O. Box 324
  Auburn, Neb. 68305
  Cooper Nuclear Station

- Mrs. Theresa Jehlik
  Librarian
  Business, Science and Technology Department
  W. Dale Clark Library
  215 S. 15th Street
  Omaha, Neb. 68102
  Fort Calhoun Station

NEW HAMPSHIRE

- Ms. Pamela Gjettum
  Director
  Exeter Public Library
  Front Street
  Exeter, N.H. 03833
  Seabrook Nuclear Station

NEW JERSEY

- Miss Joanne L. Owens
  Junior Library Clerk
  Pennsville Public Library
  190 S. Broadway
  Pennsville, N.J. 08070
  Hope Creek Nuclear Generating 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

- Mr. Stanley Zubowski
  Buffalo and Erie County Public Library
  Lafayette Square
  Buffalo, N.Y. 14203
  West Valley Demonstration Project

- Dr. Peter B. Allison
  Head
  Social Science/Documents Center
  Elmer Holmes Bobst Library
  New York University
  70 Washington Square, South
  New York, N.Y. 10012
  Indian Point Station

- Mr. Sol Becker
  Chief Librarian
  Public Health Library
  New York City Department of Health
  125 Worth Street
  New York, N.Y. 10013
  Columbia University Research Reactor

- Mrs. Adele Mangano
  Library Technician
  Reference and Documents Department
  Penfield Library
  State University of New York
  Oswego, N.Y. 13126
  Nine Mile Point Nuclear Station
  James A. Fitzpatrick Nuclear Power Plant

- Ms. Cynthia A. Dana
  Senior Library Clerk
  Business and Social Science Division
  Rochester Public Library
  115 South Avenue
  Rochester, N.Y. 14610
  Robert E. Ginna Nuclear Power Plant

- Ms. Cathy McGowan
  Reference Clerk
  Shoreham-Wading River Public Library
  Route 25A
  Shoreham, N.Y. 11786
  Shoreham Nuclear Power Station

- Ms. Annette Gernatt
  Town of Concord Public Library
  23 N. Buffalo Street
  Springfield, N.Y. 14141
  West Valley Demonstration Project

- 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. Linda Hickman
  Documents Librarian
  Reference Department
  Olivia Rainey Library
  Wake County Public Libraries
  104 Fayetteville Street
  Raleigh, N.C. 27601
  Shearon Harris Nuclear Power Plant

- Ms. Emma Myles
  Library Assistant
  Brunswick County Library
  109 W. Moore Street
  Southport, N.C. 28461
  Brunswick Steam Electric Plant

OHIO

- Ms. Vera E. Ehas
  Public Services Librarian
  Clermont County Public Library
  180 South Third Street
  Batavia, Ohio 45103
  William H. Zimmer Nuclear Power Station

- Mrs. Shirley Morgan
  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

- Mrs. O.J. Grosclaude
  Sallisaw City Library
  111 North Elm
  Sallisaw, Okla. 74995
  Kerr-McGee Corporation

OREGON

- Mr. Jim Takita
  Head
  Social Science and Science Department
  Government Documents Collection
  Library Association of Portland
  801 S. W. 10th Avenue
  Portland, Ore. 97205
  Trojan Nuclear Plant

PENNSYLVANIA

- Ms. Nancy Luezinger
  Reference Librarian
  B.F. Jones Memorial Library
  663 Franklin Avenue
  Aliquippa, Pa. 15001
  Beaver Valley Power Station
  Shippingport Project

- Ms. Datz
Apollo Memorial Library
219 N. Pennsylvania
Apollo, Pa. 15613
Babcock & Wilcox Parks Township
Fuel Facility

- Mr. Phil Hearne
Librarian
Dauphin Library System
101 Walnut Street
Harrisburg, Pa. 17101
Three Mile Island Nuclear Station

- Mr. Lawrence H. Peterson
Reference Librarian
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
Fulton Nuclear Station

- Mr. William A. Felker
Reference and Information Manager
Government Publications
Department
Free Library of Philadelphia
19th and Vine Streets
Philadelphia, Pa. 19103
Three Mile Island Nuclear Station
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 Experimental Reactor
Project

- Ms. Diane H. Smith
Head
Government Documents
Pattee Library
Room C 207
Pennsylvania State University
University Park, Pa. 16802
Three Mile Island Nuclear Station
Susquehanna Steam Electric
Station
Beaver Valley Power Station

- Ms. Lisa Stegmueller
Reference Librarian
Reference Department
Osterhout Free Library
71 S. Franklin Street
Wilkes-Barre, Pa. 18701
Susquehanna Steam Electric
Station

- Mr. David Van de Streek
Librarian
Library
Pennsylvania State University
York Campus
1031 Edgecomb Avenue
York, Pa. 17403
Three Mile Island Nuclear Station

RHODE ISLAND

- Ms. Ann Crawford
Cross Mill Public Library
Old Post Road
Charlestown, R.I. 02813
Wood River Junction

- Mr. Tom Reynolds
Univ. of Rhode Island
University Library, Govt.
Publications Section
Kingston, R.I. 02881
Wood River Junction

SOUTH CAROLINA

- Miss Ava G. Black
Head Librarian
Barnwell County Public Library
Hagood Avenue
Barnwell, S.C. 29812
Barnwell Reprocessing Plant
Low-level Waste
Burial Site

- Ms. Maureen Harris
Head
Public Documents Collection
Robert M. Cooper Library
Clemson University
Clemson, S.C. 29631
Oconee Nuclear Plant

- Ms. Mary Toll
Reference Librarian
Technical Services Department
South Carolina State Library
1500 Senate Street
Columbia, S.C. 29201
Catawba Nuclear Station

- Mrs. Jane Mason
Librarian
Nuclear Information Depository
Hartsville Memorial Library
220 N. Fifth Street
Hartsville, S.C. 29550
H. B. Robinson Plant

- 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
Library Clerk Typist
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
Department
Chattanooga-Hamilton County
Library
1001 Broad Street
Chattanooga, Tenn. 37402
Sequoyah Nuclear Plant
Watts Bar Nuclear Plant

- Ms. Carol A. Goris
Reference Assistant
Lawson McGhee Public Library
500 W. Church Avenue
Knoxville, Tenn. 37902
Clinch River Breeder Reactor
Project

- Mrs. Carol P. Cooper
Library Assistant
Depository Collection
Reference Department
Oak Ridge Public Library
Civic Center
Oak Ridge, Tenn. 37830
Clinch River Breeder Reactor
Project

TEXAS

- Miss Willie K. Farmer
  Documents Assistant
  U.S. Documents Collection
  Documents Department
  University of Texas
  701 South Cooper
  P. O. Box 19497
  Arlington, Tex. 76019
  Comanche Peak Steam, Electric Station

- Ms. Nancy Byrd
  Assistant Curator
  Austin-Travis County Collection
  Austin History Center
  Austin Public Library
  810 Guadalupe Street
  P. O. Box 2287 (78768)
  Austin, Tex. 78701
  South Texas Project

- Ms. Patsy G. Norton
  Director
  Wharton County Junior College
  J M Hodges Learning Center
  911 Boling Highway
  Wharton, Tex. 77488
  South Texas Project

- Ms. Peggy Oldham
  Assistant 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 Department
  San Antonio Public Library
  203 S. St. Mary's Street
  San Antonio, Tex. 78205
  South Texas Project

VERMONT

- Mrs. Junia A. Bryant
  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 Department
  Alderman Library
  University of Virginia
  Charlottesville, Va. 22901
  North Anna Power Station

- Ms. Mary Ann Manrique
  Accounting Clerk
  Louisa County Courthouse
  P. O. Box 160
  Louisa, Va. 23093
  North Anna Power Station

- Mr. Alan Zoellner
  Documents Librarian
  Documents Department
  Swem Library
  College of William and Mary
  Williamsburg, Va. 23185
  Surry Power Station

WASHINGTON

- Mrs. Lois McCleary
  Library Assistant
  W. H. Abel Memorial Library
  125 Main Street, South
  Montesano, Wash. 98563
  Washington Nuclear Projects 3 & 5

- Ms. Joan Hamilton
  Technical Processing Librarian
  Richland Public Library
  Swift and Northgate Streets
  Richland, Wash. 99352
  Washington Nuclear Projects 1, 2, and 4
  Basalt Waste Isolation Project
  Richland Low-level Waste Burial Site

WISCONSIN

- Mrs Kathy Pletcher
  Head
  Government Documents Secton
  Library Learning Center
  University of Wisconsin
  2420 Nicolet Drive
  Green Bay, Wis. 54301
  Kewaunee Nuclear Power Station

- Miss Dolores Hendersin
  Reference Librarian
  LaCrosse Public Library
  800 Main Street
  LaCrosse, Wis. 54601
  LaCrosse Nuclear Generating Station

- 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 1984

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 1984, are described briefly below.

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Operator Licensing Function; Completion of Regionalization by Assignmentof the Function to Regions IV and V—Part 55

On October 4, 1983 (48 FR 45223), the NRC published amendments to its regulations, effective immediately, concerning the further implementation of NRC's regional licensing program. This amendment delegated the authority and responsibility for issuing licenses for operators and senior operators of licensed nuclear reactors located in Regions IV and V to the Regional Administrators of Regions IV and V.

Transportation of Radioactive Material—Part 71

On October 5, 1983 (48 FR 45381), the NRC published a document that made corrections to and clarifications of a final rule published August 5, 1983, that revised regulations for the transportation of radioactive material to make them compatible with those of the International Atomic Energy Agency. This document corrects a number of typographical errors and announces that two petitions for rulemaking are granted.

Temporary Operating Licenses—Parts 2 and 50

On October 13, 1983 (48 FR 46489), the NRC published amendments to its regulations, effective November 14, 1983, that provide for the issuance of temporary operating licenses for nuclear power reactors. Public Law 97-415, enacted January 4, 1983, amended section 192 of the Atomic Energy Act of 1954 to authorize the issuance of a temporary operating license under certain prescribed circumstances. These amendments are designed to conform NRC regulations and procedures to the new temporary operating licensing authority.

Codes and Standards for Nuclear Power Plants—Part 50

On November 4, 1983 (48 FR 50878), the NRC amended its regulations to incorporate by reference the Summer 1982 Addenda of the ASME Boiler Pressure Vessel Code. These amendments, effective December 7, 1983, will permit the use of improved methods for construction.

Deletion of Exception Filing Requirement for Appeal From Initial Decision; Consolidation of Responsive Briefs—Part 2

On November 17, 1983 (48 FR 52282), the NRC published amendments to its regulations relating to appeals from an initial adjudicatory decision. The amendments, effective December 19, 1983, require parties to file a notice of appeal rather than exceptions to an initial decision. In addition, the amendments require parties to file a single responsive brief regardless of the number of appellant briefs filed. The amendment reduces procedural requirements for appealing an initial decision.

NRC Export Licensing Authority; Interpretation—Part 110

On January 24, 1984 (49 FR 2881), the NRC published an interpretative rule, effective immediately, that specified those components especially designed or prepared for use in a gas centrifuge uranium enrichment plant which are subject to the Commission's export licensing authority.

General Statement of Policy and Procedure For Enforcement Action—Part 2

On March 8, 1984 (49 FR 8583), the NRC published amendments to its regulations, effective immediately, that made minor revisions to its enforcement policy based on agency experience in implementing the policy. This policy statement, codified as Appendix C to 10 CFR Part 2, is intended to inform licensees and the public of the bases for taking various enforcement actions.

Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions and Related Conforming Amendments—Parts 2, 30, 40, 50, 51, 61, 70, 72, and 110

On March 12, 1984 (49 FR 9352), the NRC published a document that revised 10 CFR Part 51 to implement section 102(2) of the National Environmental Policy Act of 1969 (NEPA) in a manner consistent with NRC's domestic licensing and related regulatory authority. This amendment, effective June 7, 1984, reflects Commission policy to voluntarily take into account the regulations of the Council on Environmental Quality that implement the procedural provisions of NEPA.

Codes and Standards for Nuclear Power Plants—Part 50

On March 15, 1984 (49 FR 9711), the NRC amended its regulations which incorporate by reference national codes and standards for the construction of nuclear power plant components. These amendments, effective May 14, 1984, increase specific references to the ASME Boiler and Pressure Vessel Code to include subsections that provide rules for the construction of certain safety systems. These amendments also remove obsolete rules no longer applicable.

Financial Protection Requirements and Other Agreements; Facility Form Policy—Part 140

On March 26, 1984 (49 FR 11146), the NRC published a document, effective April 23, 1984, that adds statements to its regulations regarding the text of the Facility Form Policy, including any codified amendatory endorsement or change to the policy. The statement indicates that this text is an example of a contract that has been "accepted" as evidence of financial protection, but that variations on the text would be considered by the Commission. This document also contains two amendatory endorsements that modify certain definitions in the Facility Form Policy, publishes the standard secondary master policy form for codification, and makes other minor conforming amendments.

Office of Investigations—Part 1

On April 20, 1984 (49 FR 16760), the NRC published an amendment to its regulations, effective immediately, that added a description of the Office of Investigations and its functions to the NRC's Statement of Organization and General Information. The Office of Investigations was established to conduct NRC investigations of licensees, permittees, applicants, contractors, and vendors.

Information Collection Requirements: Display of OMB Control Numbers—Chapter I

On May 9, 1984 (49 FR 19623), the NRC published amendments to its regulations, effective immediately, to indicate the Office of Management and Budget control numbers under which the information collection requirements imposed by NRC regulations have been approved by OMB. This action is necessary to comply with OMB regulations that implement the Paperwork Reduction Act.

Regional Licensing Program: Further Implementation—Parts 30, 40, and 70

On May 9, 1984 (49 FR 19630), the NRC published amendments to its regulations, effective April 2, 1984, concerning the domestic licensing of byproduct, source, and special nuclear material. These amendments broadened the scope of NRC's decentralized licensing program by including additional types of materials licensing actions in its delegation of licensing authority to the regions.

Revision of License Fee Schedule—Part 170

On May 21, 1984 (49 FR 21293), the NRC published a document, effective June 20, 1984, amending the schedule of fees for inspections and for the review of applications and requests for permits, licenses, approvals, amendments, renewals, and special projects. The revised schedule of fees is based on the costs of providing services and will enable NRC to more completely recover the costs it incurs in providing services to identifiable recipients.

Abolition of the Position of Appeal Panel Vice Chairman—Part 2

On June 12, 1984 (49 FR 24110), the NRC published amendments to its regulations, effective immediately, abolishing the position of permanent Vice Chairman of the Atomic Safety and Licensing Appeal Panel. The amendments also authorize the most senior available full-time Appeal Panel member to perform certain functions previously performed by the Vice Chairman.

Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions and Related Conforming Amendments—Parts 2, 30, 40, 50, 51, 61, 70, 72, and 110

On June 14, 1984 (49 FR 24512), the NRC published a document confirming the effective date of June 7, 1984 for the final rule published March 12, 1984 that revised 10 CFR Part 51 to implement section 102(2) of the National Environmental Policy Act of 1969 in a manner consistent with NRC's domestic licensing and related regulatory authority. This document also includes the section listing the collection of information requirements approved by OMB on June 6, 1984.

Change in Mailing Address for Submittal of Personnel Monitoring Reports—Part 20

On June 14, 1984 (49 FR 24513), the NRC published an amendment to its regulations, effective immediately, informing NRC licensees of a change in the mailing address to be used for the submittal of personnel monitoring reports.

Tritium and Source Material Reports—Parts 30, 40, and 150

On June 15, 1984 (49 FR 24705), the NRC published an amendment to its regulations, effective July 16, 1984, applicable to NRC and Agreement State licensees who transfer or receive other than U.S. origin source material or who import or export source material of any origin. The amendments lower the reportable quantity of certain source material transfers from 1,000 kilograms to 1 kilogram in order to satisfy existing international commitments. The amendments also remove the requirement that NRC and Agreement State licensees report tritium inventories.

Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants—Part 50

On June 26, 1984 (49 FR 26036), the NRC published amendments to its regulations, effective July 26, 1984, to require improvements in the design and operation of light-water-cooled nuclear power plants to reduce the likelihood of failure of the reactor protection system to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of an anticipated transients without scram (ATWS) event. The amendments require the installation of certain equipment in nuclear power plants and encourage the development of a reliability insurance program for the reactor trip system on a voluntary basis.

Emergency Planning and Preparedness—Part 50

On July 6, 1984 (49 FR 27733), the NRC published an amendment to its regulations that reduced the frequency of participation by state and local governmental authorities in emergency preparedness exercises at nuclear power reactor sites. This amendment, effective August 6, 1984, reflects the experience gained in observing and evaluating over 150 emergency preparedness exercises since 1980.

Consolidation of Materials License Applications on Form NRC 313—Parts 30, 33, 34, 35, and 40

On July 9, 1984 (49 FR 27923), the NRC published amendments to its regulations concerning the domestic licensing of source and byproduct material that consolidated five application forms into one simplified form for materials license applications. The amendments, effective immediately, simplify the regional review process and provide an improved format for the automatic data entry of the information submitted as part of the license application.

Charges for the Production of Records—Part 9

On July 31, 1984 (49 FR 30457), the NRC published an amendment to its regulations, effective immediately, that revised the charges for copying records publicly available at the NRC Public Document Room in Washington, DC. This action reflects the change in copying charges resulting from the Commission's award of a new contract for this service.

Revised Access Authorization Fees for Licensee Personnel—Part 25

On August 13, 1984 (49 FR 32171), the NRC published an amendment to its regulations, effective immediately, to revise the access authorization fees charged to licensee personnel who require access to National Security Information and/or Restricted Data. The revised fees reflect the current access authorization investigation cost charged to the NRC by the Office of Personnel Management plus part of NRC's overhead associated with the processing of access authorization requests.

Waste Confidence Decision—Parts 50 and 51

On August 31, 1984 (49 FR 34658), the NRC published its final decision in the "Waste Confidence Rulemaking." This proceeding was intended to assess generically the degree of assurance now available that radioactive waste can be safely disposed of, to determine when disposal of off-site storage will be available, and to determine whether radioactive wastes can be safely stored on-site past the expiration of existing facility licenses until off-site disposal or storage is available.

Requirements For Licensee Actions Regarding the Disposition of Spent Fuel Upon Expiration of Reactor Operating Licenses—Parts 50 and 51

On August 31, 1984 (49 FR 34688), the NRC published an amendment to its regulations, effective November 29, 1984. Because the Commission has reasonable assurance that no significant environmental impacts will result from the storage of spent fuel for at least 30 years beyond the expiration of nuclear reactor operating licenses, discussion of any environmental impact of spent fuel storage for the period following expiration of the license or amendment applied for is not required in connection with the issuance or amendment of an operating license for a nuclear reactor or in connection with the issuance or amendment of an initial license for an independent spent fuel storage installation. The amendment also requires nuclear power reactor licensees to submit plans for managing spent fuel at their site until it is transferred to DOE for disposal. The plans must be submitted to the NRC for review and approval no later than 5 years before the expiration of the operating license.

Glass Enamel and Glass Enamel Frit Containing Small Amounts of Uranium—Part 40

On September 11, 1984 (49 FR 35611), the NRC published an amendment to its regulations, effective immediately, that deletes an exemption from licening requirements applicable to the possession and use of glass enamel and glass enamel frit containing small amounts of source material. The rule is intended to prevent any unnecessary exposure to radiation that may be received by users or consumers of products containing source material by prohibiting the future domestic manufacture or importation of these materials or products.

Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants; Elimination of Review—Parts 2 and 50

On September 12, 1984 (49 FR 35747), the NRC published an amendment to its regulations, effective October 12, 1984, that eliminates financial qualification review and findings for electric utilities that are applying for operating licenses for utilization facilities if the utility is a regulated public utility or is authorized to set its own rates. This amendment also reinstates a requirement for financial qualifications review and findings for electric utilities that are applying for construction permits.

PROPOSED REGULATIONS AND AMENDMENTS

Financial Protection Requirements and Indemnity Agreements; Facility Form Policy—Part 140

On October 19, 1983 (48 FR 48474), the NRC published a notice of proposed rulemaking that would add statements to its regulations regarding the text of the Facility Form Policy, including any codified amendatory endorsement or change to the policy. The proposed statement would indicate that this text is an example of a contract that has been "accepted" as evidence of financial protection but that other variations on the text would be considered by the Commission. The proposed amendments would remove the misimpression that the Commission requires the exact language presented in the text of the Facility Form Policy. (Issued as a final rule on March 26, 1984 (49 FR 11146))

Tritium and Source Material Reports—Parts 30, 40, and 150

On November 29, 1983 (48 FR 53714), the NRC published a notice of proposed rulemaking that would amend reporting requirements applicable to NRC and Agreement State licensees who transfer source material. The proposed rule would lower the reportable quantity of source material transfers from 1,000 kilograms to 1 kilogram in order to satisfy existing international commitments. The proposed rule would also remove the requirement that NRC and Agreement State licensees report tritium inventories. (Issued as a final rule on June 15, 1984 (49 FR 24705))

Hybrid Hearing Procedures For Expansions Of Onsite Spent Fuel Storage Capacity At Civilian Nuclear Power Reactors—Parts 2 and 72

On December 5, 1983 (48 FR 54499), the NRC published a notice of proposed rulemaking that contained two options for implementing the hybrid hearing process set out in the Nuclear Waste Policy Act. This hybrid hearing process is to be used in certain contested proceedings on a license application or amendment to expand the spent nuclear fuel storage capacity at the site of a civilian nuclear power reactor. Thy hybrid hearing process would employ less formal procedures in the initial stages of the hearing process and would designate only genuine and substantial issues for resolution in an adjudicatory hearing.

Improved Personnel Dosimetry Processing—Part 20

On January 10, 1984 (49 FR 1205), the NRC published a notice of proposed rulemaking that would require NRC licensees to utilize the specified services of dosimetry processors who have been accredited by the National Voluntary Laboratory Accreditation Program of the National Bureau of Standards.

Material Control and Accounting Requirements for Facilities Possessing Formula Quantities of Strategic Special Nuclear Material—Part 70

On February 2, 1984 (49 FR 4091), the NRC published a notice of proposed rulemaking that would significantly strengthen material control and accounting capabilities at all fuel cycle facilities authorized to possess and use formula quantities of strategic special nuclear material, including reprocessing plants but not waste disposal operations or nuclear reactors. The proposed rule would require more timely detection of material losses and provide for more rapid and conclusive resolution of discrepancies.

Pressurized Thermal Shock Events—Part 50

On February 7, 1984 (49 FR 4498), the NRC published a notice of proposed rulemaking that would amend its regulations for light water nuclear power plants. The proposed rule would—(1) establish a screening criterion related to the fracture resistance of pressurized water reactor vessels during pressurized thermal shock events; (2) require analyses and a schedule for implementation of flux reduction programs that are reasonably practicable to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plant operation beyond the screening criterion will be considered.

Disposal of High-Level Radioactive Waste in the Unsaturated Zone—Part 60

On February 16, 1984 (49 FR 5934), the NRC published a notice of proposed rulemaking that would amend regulations pertaining to the disposal of high-level radioactive waste in geologic repositories. The proposed rule would make the technical criteria for geologic disposal in the saturated zone equally applicable to disposal within the unsaturated zone.

Export and Import on Nuclear Equipment and Material—Part 110

On March 1, 1984 (49 FR 8445), the NRC published a notice of proposed rulemaking that would expand the authority of the public to export nonsensitive nuclear equipment and material under a general license. The proposed rule would incorporate the U.S. Government policy of facilitating nuclear cooperation with countries sharing U.S. nonproliferation goals in the new general license, but would not affect the existing rigorous controls over the export of proliferation sensitive nuclear commodities.

Environmental Qualification of Electric Equipment—Part 50

On March 7, 1984 (49 FR 8445), the NRC published a notice of proposed rulemaking seeking public comment on an amendment that would delete from power plant operating licenses a June 30, 1982 deadline for the environmental qualification of electric equipment imposed by a previous commission order. This rulemaking proceeding responds to a decision by the United States Court of Appeals for the District of Columbia vacating and remanding the final rule published June 30, 1982 that suspended the June 30, 1982 deadline.

Exceptions to Notice and Comment Rulemaking Procedures—Part 2

On April 2, 1984 (49 FR 13043), the NRC published a notice of proposed rulemaking that would amend the Commission's rules of practice to clarify the Commission's use of the exceptions to notice and comment rulemaking contained in the Administrative Procedure Act (5 U.S.C. 553 (b)). This clarification is necessary to leave no doubt that the Commission does assert its discretion to make exceptions to the general requirements for notice and opportunity for comment in informal rulemaking to the extent possible.

Financial Qualifications of Electric Utilities in Operating License Reviews and Hearings for Nuclear Power Plants; Elimination of Review—Parts 2 and 50

On April 2, 1984 (49 FR 13044), the NRC published a notice of proposed rulemaking that would eliminate financial qualifications review and findings for electric utilities that are applying for operating licenses for utilization facilities if the utility is a regulated public utility or is authorized to set its own rates. (Issued as a final rule on September 12, 1984 (49 FR 35747))

Rules of Practice and Rules for Licensing Production and Utilization Facilities; Request for Public Comment on Regulatory Reform Proposals—Parts 2 and 50

On April 12, 1984 (49 FR 14698), the NRC published a document requesting public comment on suggestions for procedural changes in the nuclear power plant licensing process. This document presents suggestions from the Regulatory Reform Task Force suggesting improvements to three principal parts of the hearing process: screening, conduct of hearings, and decision making. If the Commission decides that, based on comment received, a particular proposal warrants adoption through rulemaking, the Commission will then issue a formal notice of proposed rulemaking to implement the suggestion.

Glass Enamel and Glass Enamel Frit Containing Small Amounts of Uranium—Part 40

On April 30, 1984 (49 FR 18308), the NRC published a notice of proposed rulemaking that would delete an exemption from licensing requirements applicable to the possession and use of glass enamel and glass enamel frit containing small amounts of source material. The proposed rule is intended to prevent any unnecessary radiation exposure that may be received by users or consumers of material or products containing source material by prohibiting the future domestic manufacture or importation of these materials or products. (Issued as a final rule on September 11, 1984 (49 FR 35611))

Modification of Protection Requirements for Spent Fuel Shipments—Part 73

On June 8, 1984 (49 FR 23867), the NRC published a notice of proposed rulemaking that would amend its regulations for the physical protection of irradiated reactor fuel in transit. The proposed amendment, which reflects a safeguards rather than a safety issue, would take into account new research indicating that the consequences of a successful sabotage of an irradiated fuel shipment in a heavily populated area would be small compared to the consequence estimates that prompted the issuance of the current rule. The proposed rule would relieve the licensee of non-essential requirements for certain spent fuel shipments while providing continued protection against sabotage.

Charges for the Production of Records—Part 9

On June 21, 1984 (49 FR 25482), the NRC published a notice of proposed rulemaking that would revise the charges for copying records publicly available at the NRC Public Document Room. The proposed rule would reflect the change in copying charges resulting from the Commission's award of a new contract for the copying of records. (Issued as a final rule on July 31, 1984 (49 FR 30457))

Limiting the Use of Highly Enriched Uranium in Domestic Research and Test Reactors—Part 50

On July 6, 1984 (49 FR 27769), the NRC published a notice of proposed rulemaking that would limit the use of highly enriched uranium (HEU) fuel in domestic research and test reactors (non-power nuclear reactors). The proposed amendment would require that new non-power reactors use low enriched uranium (LEU) fuel and existing non-power reactors replace HEU fuel with LEU fuel when available.

Production or Disclosure in Response to Subpoenas or Demands of Courts or Other Agencies—Part 9

On July 10, 1984 (49 FR 28072), the NRC published a notice of proposed rulemaking that would prescribe procedures with respect to the production of documents or disclosure of information in response to subpoenas or demands of courts or other judicial or quasi-judicial authorities in state and Federal proceedings.

Access Authorization Program—10 CFR Parts 50 and 73

On August 1, 1984 (49 FR 30726), the NRC published a notice of proposed rulemaking that would amend its regulations to require an access authorization program for individuals seeking unescorted access to protected areas and vital islands at nuclear power plants. The proposed rule, which would affect all nuclear power plant licensees, would consist of three major industry run components: background investigation, psychological assessment, and continual behavioral observation programs.

Physical Protection of Nuclear Power Plants—Part 73

On August 1, 1984 (49 FR 30735), the NRC published a notice of proposed rulemaking that would amend its nuclear power plant safeguards regulations. The proposed regulation would clarify and refine requirements for the designation and protection of vital locations containing safety-related equipment. The proposed requirements are intended to provide a more safety-conscious safeguards system while maintaining current levels of protection.

Searches of Individuals at Power Reactor Facilities—Part 73

On August 1, 1984 (49 FR 30738), the NRC published a notice of proposed rulemaking that would amend its regulations pertaining to entry searches at power reactor facilities. The proposed regulation would require equipment searches of all individuals seeking access to protected areas, except on-duty police officers, and pat-down searches when detection equipment fails or cause to suspect exists. The proposed amendment would increase assurance that power reactors are adequately protected against sabotage by an insider.

Training and Qualifications of Civilian Nuclear Power Plant Personnel and Operators' Licenses—Part 55

On August 8, 1984 (49 FR 31700), the NRC published a notice of proposed rulemaking that would amend its regulations governing the training and qualifications of civilian nuclear power plant personnel. The proposal would conform the literal language of the regulations to the long-standing agency practice of treating the satisfactory completion on an NRC-approved program for training reactor operators as the equivalent of actual operating experience at a reactor.

Enforcement of Nondiscrimination on the Basis of Handicap in Federally Conducted Programs—Part 4

On August 28, 1984 (49 FR 34132), the NRC published a notice of proposed rulemaking that would provide for the enforcement of section 504 of the Rehabilitation Act of 1973, as amended, which prohibits discrimination on the basis of handicap, as it applies to programs or activities conducted by the NRC.

ADVANCE NOTICES OF PROPOSED RULEMAKING

On October 7, 1983 (48 FR 45787), the NRC published a correction to an advance notice of proposed rulemaking published on September 28, 1983, concerning the establishment of requirements for the long-term management of its process for the imposition of new regulatory requirements for power reactors. This document corrected an error in the presentation of the separate views of Commissioner Roberts.

Role of NRC Staff in Adjudicatory Licensing Hearings—Part 2

On November 2, 1983 (48 FR 50550), the NRC published an advance notice of proposed rulemaking to request comment on whether and to what extent changes should be made in the NRC staff's present role as a full party in adjudicatory hearings in initial licensing proceedings for nuclear power reactors.

Appendix 5

Regulatory Guides—Fiscal Year 1984

NRC regulatory guides describe methods for 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 with the U.S. Government Printing Office to become a consigned sales agent for certain NRC publications including regulatory guides, except for draft guides issued for public comment which receive free distribution. Active guides are sold on a subscription or individual copy basis. NRC licenses receive, at no cost, pertinent draft and active regulatory guides as they are issued.

The following guides were issued or revised during the period October 1, 1983, to September 30, 1984.

Division 1—Power Reactor Guides

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 22)

1.85 Materials Code Case Acceptability—ASME Section III, Division 1 (Revision 22)

1.89 Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Revision 1)

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 3)

Division 2—Research and Test Reactor Guides

None

Division 3—Fuels and Materials Facilities Guides

3.54 Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation

Division 4—Environmental and Siting Guides

None

Division 5—Materials and Plant Protection Guides

5.9 Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material (Revision 2)

5.11 Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste (Revision 1)

5.21 Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry (Revision 1)

5.23 In Situ Assay of Plutonium Residual Holdup (Revision 1)

5.34 Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection (Revision 1)

5.37 In Situ Assay of Enriched Uranium Residual Holdup (Revision 1)

5.38 Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry (Revision 1)

5.53 Qualification, Calibration, and Error Estimation Methods for Non-destructive Assay (Revision 1)

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

10.2 Guidance to Academic Institutions Applying for Specific Byproduct Material Licenses of Limited Scope (Errata to Revision 1)

10.4 Guide for the Preparation of Applications for Licenses To Process Source Material (Errata to Revision 1)

10.5 Applications for Type A Licenses of Broad Scope (Errata to Revision 1)

10.6 Guide for the Preparation of Applications for Use of Sealed Sources and Devices for Performing Industrial Radiography (Errata to Revision 1)

10.7 Guide for the Preparation of Applications for Licenses for Laboratory and Industrial Use of Small Quantities of Byproduct Material (Errata to Revision 1)

10.9 Guide for the Preparation of Applications for Licenses for the Use of Gamma Irradiators (Errata to Revision 1)

DRAFT GUIDES

Division 3

CE 227-4 Standard Format and Content for the Health and Safety Sections of License Renewal Applications for Uranium Hexafluoride Production Plants

Division 5

SG 301-4 Standard Format and Content Guide for Access Authorization Plans for Nuclear Power Plants

Division 8

OP 032-5 Test and Calibration of Radiation Protection Instrumentation

OP 212-4 Radiation Protection Training for Personnel Employed in Medical Facilities

Division 10

TM 608-4 Guide for the Preparation of Applications for Licenses in Medical Teletherapy Programs (Errata)

Appendix 6

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1984)

The following listing includes nuclear power reactor electrical generating units which were in operation or under construction or under construction in the United States as of December 31, 1984, representing a total capacity of approximately 116,000 MWe. 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, CP—has construction permit. The dates for operation are either actual or as scheduled by the utilities as of December 31, 1984.

This listing includes nine fewer units than a year ago, reflecting cancellations of plans for seven future facilities and permanent shutdown of two units.

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 Unit 1	1,235	PWR	CP 1974	Tennessee Valley Authority	1988
Scottsboro	Bellefonte Nuclear Plant Unit 2	1,235	PWR	CP 1974	Tennessee Valley Authority	1990
ARIZONA						
Wintersburg	Palo Verde Nuclear Generating Station Unit 1	1,304	PWR	OL 1984	Arizona Public Service Co.	1985
Wintersburg	Palo Verde Nuclear Generating Station Unit 2	1,304	PWR	CP 1976	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 3	1,304	PWR	CP 1976	Arizona Public Service Co.	1987
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	CP 1970	Pacific Gas & Electric Co.	1985
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	CP 1974	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	CP 1974	Georgia Power Co.	1987
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 2	1,100	PWR	CP 1974	Georgia Power Co.	1988
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
Byron	Byron Station Unit 1	1,120	PWR	OL 1984	Commonwealth Edison Co.	1985
Byron	Byron Station Unit 2	1,120	PWR	CP 1975	Commonwealth Edison Co.	1986
Braidwood	Braidwood Unit 1	1,120	PWR	CP 1975	Commonwealth Edison Co.	1986
Braidwood	Braidwood Unit 2	1,120	PWR	CP 1975	Commonwealth Edison Co.	1987
Clinton	Clinton Nuclear Power Plant Unit 1	950	BWR	CP 1976	Illinois Power Co.	1986
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	CP 1977	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	CP 1977	Gulf States Utilities Co.	1985

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
MAINE						
Wiscasset	Maine Yankee Atomic Power	810	PWR	OL 1972	Maine Yankee Atomic Power Co.	1972
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
Lagonna Beach	Enrico Fermi Atomic Power Plant Unit 2	1,093	BWR	CP 1972	Detroit Power Co.	1984
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
Midland	Midland Nuclear Power Plant Unit 1	492	PWR	CP 1972	Consumers Power Co.	1985
Midland	Midland Nuclear Power Plant Unit 2	818	PWR	CP 1972	Consumers Power Co.	1985
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	CP 1976	Public Service of N.H.	1985
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	CP 1974	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	CP 1974	Niagara Mohawk Power Co.	1986
Ontario	R. E. Ginna Nuclear Power Plant Unit 1	470	PWR	OL 1969	Rochester Gas & Elec. Co.	1970
Brookhaven	Shoreham Nuclear Power Station1	820	BWR	OL 1984	Long Island Lighting Co.	1985
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	CP 1978	Carolina Power & Light Co.	1986

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	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	1985
Perry	Perry Nuclear Power Plant Unit 2	1,205	BWR	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	1992
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	Peach Bottom Atomic Power Station Unit 3	1,035	BWR	OL 1974	Philadelphia Elec. Co.	1974
Pottstown	Limerick Generating Station Unit 1	1,065	BWR	OL 1984	Philadelphia Elec. Co.	1985
Pottstown	Limerick Generating Station Unit 2	1,065	BWR	CP 1974	Philadelphia Elec. Co.	Indef
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	CP 1974	Duquesne Light Co. Ohio Edison Co.	1986
Goldsboro	Three Mile Island Nuclear Station, Unit 1	776	PWR	OL 1974	GPU Nuclear Corp.	1974
Goldsboro	Three Mile Island Nuclear[2] Station, Unit 2	906	PWR	OL 1978	GPU Nuclear Corp.	1978
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

[1]Has license to load fuel, but restricted to 0.001 percent of power.
[2]Shut down indefinitely (not included in summary)

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
SOUTH CAROLINA (continued)						
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	CP 1975	Duke Power Co.	1984
Lake Wylie	Catawba Nuclear Station Unit 2	1,145	PWR	CP 1975	Duke Power Co.	1987
TENNESSEE						
Daisy	Sequoyah Nuclear Power Plant Unit 1	1,128	PWR	OL 1980	Tennessee Valley Authority	1981
Daisy	Sequoyah Nuclear Power Plant Unit 2	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	1984
Spring City	Watts Bar Nuclear Plant Unit 2	1,165	PWR	CP 1973	Tennessee Valley Authority	1986
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station Unit 1	1,150	PWR	CP 1974	Texas Utilities	1985
Glen Rose	Comanche Peak Steam Electric Station Unit 2	1,150	PWR	CP 1974	Texas Utilities	1986
Bay City	South Texas Nuclear Project Unit 1	1,250	PWR	CP 1975	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)	1,266	PWR	CP 1975	Wash. Public Power Supply System	Indef
Richland	WPPSS No. 2 (Hanford)	1,103	BWR	OL 1983	Wash. Public Power Supply System	1984
Satsop	WPPSS No. 3	1,242	PWR	CP 1978	Wash. Public Power Supply System	Indef
WISCONSIN						
LaCrosse	LaCrosse (Genoa) Nuclear Generating Station	48	BWR	OL 1967	Dairyland Power Coop.	1969
Two Creeks	Point Beach Nuclear Plant Unit 1	495	PWR	OL 1970	Wisconsin Michigan Power Co.	1970
Two Creeks	Point Beach Nuclear Plant Unit 2	495	PWR	OL 1971	Wisconsin Michigan Power Co.	1972
Kewanee	Kewanee Nuclear Power Plant	515	PWR	OL 1973	Wisconsin Public Svc. Corp.	1974

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