U. S. Nuclear Regulatory Commission

Annual Report 1978

February 14, 1979

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

Dear Mr. President:

We have the honor to submit herewith the fourth Annual Report of the United States Nuclear Regulatory Commission for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974. This report covers the major activities of the NRC from October 1, 1977, through September 30, 1978, and briefly describes some additional actions through December 31, 1978.

Respectfully,

Joseph M. Hendrie
Chairman

U. S. Nuclear Regulatory Commission

Annual Report 1978

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Statutory Reporting Requirements Addressed

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 Chapters 1 and 2. Specific goals concerning fuel cycle are discussed in Chapter 3; safeguards, Chapter 4; wastes, Chapter 5; inspection and enforcement, Chapter 6; abnormal occurrences, Chapter 7; emergency response planning, Chapter 8; nuclear nonproliferation, Chapter 9; standards, Chapter 10; and research and risk assessment, 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, 7, 10 and 11; materials facilities, devices and transportation packages, Chapters 3, 7, 10 and 11; waste facilities, Chapters 5 and 10.)

"(2) investigating abnormal occurrences and defects in nuclear power plants and other licensed facilities ..." (See Chapters 2 and 7.)

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle ..." (See Chapters 4, 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 4, 6 and 10.)

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

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities ..." (See Chapters 1, 2, 3, 7 and 10.)

Section 205, as amended in 1977, 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 210, added in 1977, 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 9.)

Atomic Energy Act of 1954, as Amended

Section 170 i, directs the Commission to report annually on indemnity operations implementing the Price-Anderson Act which provides a system to pay public liability claims in the event of a nuclear incident. (This report, which has been submitted separately in the past, appears in Chapter 2 under "Indemnity and Insurance." A report on Advisory Committee on Reactor Safeguards activities, which has been submitted annually with the indemnity operations report, also is included in Chapter 2.)

Overview and Summary

This is the fourth Annual Report of the U.S. Nuclear Regulatory Commission, to be submitted to the President for transmittal to the Congress, under Section 307(c) of the Energy Reorganization Act of 1974.

This report sets forth major NRC activities in fiscal year 1978—licensing decisions, regulatory studies and directives, policy matters—under headings which correspond with the various facets of the agency's statutory responsibility. This introductory chapter provides a brief overview and summary of those activities. Throughout this chapter, certain significant actions or events taking place after the close of the report period (September 30, 1978) are cited; these will be treated in more detail, as warranted, in next year's report.

As the NRC completed its fourth year, a number of issues, as controversial as they are critical, continued to be of concern to the Commission in carrying out the nuclear regulatory mission. NRC efforts to improve nuclear licensing and regulation without compromise to completeness reflected the continuing national preoccupation with the need for reliable and safe domestic energy sources. Other evidences of the broadening ramifications of the agency's mandate, at home and internationally, can be found in virtually every part of the report. Amid the shifts in priorities and commitment of resources, the basic mission remains unchanged: to regulate civilian nuclear activities so that the public health and safety, national security and environmental quality are protected and the antitrust laws obeyed.

Chapter 1 discusses salient actions and events of the fiscal year in the general areas of safety, research, the nuclear fuel cycle, the licensing process, and new statutory tasks, and includes brief updating through December 1978.

SAFETY

The Commission's primary concern for safety in civilian nuclear activities involves two major considerations: the risks posed by serious nuclear accidents, on the one hand, and by exposure to routine releases of low levels of radioactivitiy on the other. (The risks associated with proliferation are of a different sort, and are discussed separately.) The Commission's safety goal, implemented with guidance from national radiation protection standards, is to see that its licensees and applicants for licenses take the actions considered necessary to assure that there are no undue risks to the public and workers from both normal activities and potential accidents.

The NRC has increased its studies of the potential health effects from exposure to low-level radiation. During the year, NRC funded research on the effects of specific radioisotopes, analyzed current research in radiobiology and epidemiology, drew up preliminary plans to study the feasibility of a large-scale epidemiology investigation on low-level radiation effects, and conducted a public meeting to review and critique recent studies in this field. The NRC also assisted the Department of Health, Education, and Welfare in its Presidential assignment to develop a program responding to concern about the effects of radiation exposure on workers in nuclear-related projects.

At the end of 1978, the Commission was working with the Environmental Protection Agency to develop preliminary plans for a broad program of epidemiological research on health effects of low-level ionizing radiation as directed by the NRC Authorization Act for Fiscal Year 1979, which was signed into law on November 6. NRC and EPA concluded a memorandum of understanding on their respective roles in December, and will report progress to Congress in April and September 1979.

In related actions, the NRC initiated a two-year study of the environmental impact of consumer products containing radioactive material, and issued proposed policy and rule changes designed to improve regulation of the uses of radioisotopes in medicine. The consumer products assessment is concentrating initially on the health and policy aspects of the increasing use of ionization-type smoke detectors.

Licensees' Experience

On the basis of NRC inspections and personnel exposure information, licensees continued to achieve a generally good overall radiation safety record during 1978.

During the fiscal year, NRC reported to Congress on a quarterly basis nine abnormal occurrences in licensed operations, compared with 19 in the previous year. In addition, there were four events reported by Agreement States which met the criteria for abnormal occurrences. These occurrences—events considered to be significant from the standpoint of safety but which do not always imply a direct, imminent threat to people—are summarized in Chapter 7.

Seven of the nine abnormal occurrences reported during the year concerned power reactors, and involved such problems as design deficiencies, unqualified electrical equipment, degradation of components, and deficiencies in procedures. Some of these events revealed technical problems generic to a number of reactors. The NRC took appropriate actions to assure correction of deficiencies, involving, in some cases, the shutdown or extended outages of plants.

As in previous years, there was no nuclear accident causing detectable injury to members of the public at any licensed power reactor in the United States. By year-end, licensed nuclear power plants had accumulated more than 400 reactor-years of operation without experiencing such an accident.

During the year, the staff continued a systematic evaluation of 11 nuclear power plants licensed before 1972 to determine to what extent they meet current licensing requirements for new plants. The program will determine whether changes will be necessary in the interests of safety and what the implications of the findings are for operating plants. Technical reviews of the environmental qualification of equipment were initiated in December 1977 and an interim assessment of staff findings was published (NUREG-0458) in May 1978 as part of a response to a petition on this subject from the Union of Concerned Scientists. Work has also begun on other topics requiring long review times such as seismic design and effects of postulated pipe breaks.

In January 1978, as required by an amendment to the Energy Reorganization Act, the Commission transmitted to the Congress a plan for the specification and analysis of "unresolved safety issues" relating to nuclear reactors. Progress reports on the resolution of these issues are required in the Annual Report. The NRC has identified 17 "unresolved safety issues," and progress and schedules toward their resolution are discussed in Chapter 2 of this Annual Report.

Occupational Exposures

During the year the NRC compiled 1977 personnel monitoring data collected from 457 licensees in the four categories having the greatest potential for significant personnel radiation exposures: power reactors, industrial radiographers, fuel fabricators and processors, and certain processors and distributors of radioisotopes. These are the only categories currently required to report personnel monitoring data to the NRC. The information showed that 98,212 individuals were monitored, with a collective dose of 38,944 man-rems and an average individual dose of 0.40 rem. This represented a slight increase over the 1976 average dose of 0.36 rem but continued to be well below the annual dose permitted by NRC regulations.

In order to obtain a comprehensive picture of the exposure experience of workers in licensed nuclear operations, the Commission adopted an amendment, effective in December 1978, requiring annual statistical summary reports from all NRC licensees. The expanded requirement will last two years, covering 1978 and 1979, after which the NRC will consider whether or not to extend or modify the rule change.

The NRC expended substantial effort during 1978 toward upgrading safety in radiography operations. Several seminars were conducted for radiographers at the regional offices, and the Commission published for comment possible equipment design requirements and several proposed minor changes in regulations.

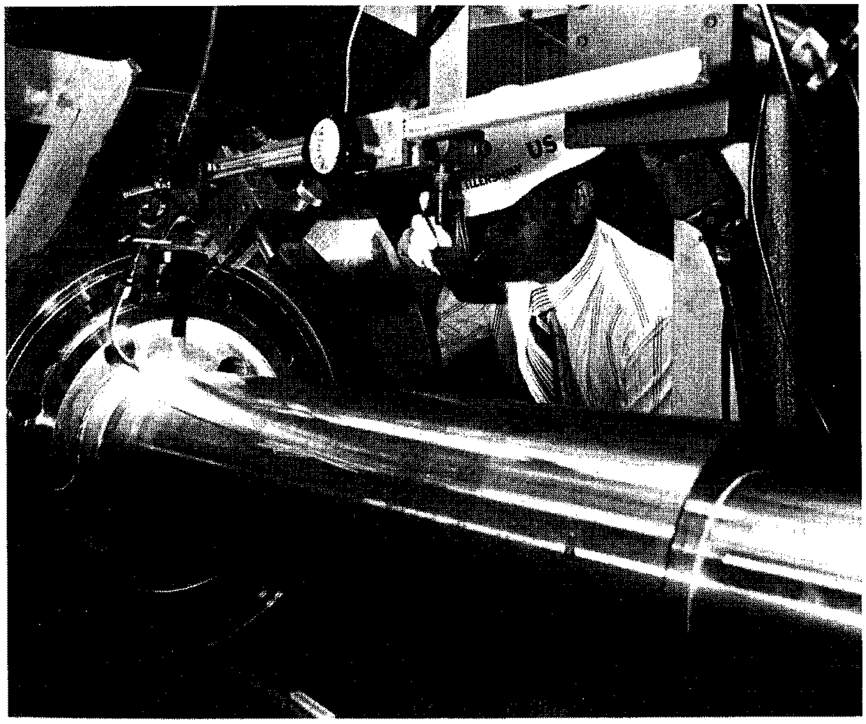
The Commission is considering other rule changes to strengthen and make more inspectable and enforceable its requirements that workers' exposures be kept not only within regulatory limits, but also as low as is reasonably achievable within those limits. In addition, the Commission is committed to holding a public hearing in 1979 on the adequacy of present occupational standards for radiation protection.

Inspection and Enforcement

Inspections of all types—approaching 6,600 in fiscal year 1978—are being conducted at a rate almost doubling that achieved when the NRC was created in 1975. An important development during the year was the assignment of resident inspectors to the sites of 20 nuclear power stations where 45 reactors are either in operation or are in advanced stages of construction, and to three major nuclear fuel facility sites. Such deployment will continue over the next several years in a program to improve inspection effectiveness.

In October, the NRC staff submitted to the Commission the results of a program begun in

The independent Risk Assessment Review Group of scientists appointed to review the Reactor Safety Study (WASH-1400) and comments made on it reported results of its year-long study to the Commission on September 7, 1978. Dr. Harold Lewis of the University of California at Santa Barbara, chairman of the group, is shown at left center, addressing the Commission, seated on the opposite side of the table, from left: Commissioner Ahearne, Commissioner Kennedy, Chairman Hendrie, Commissioner Gilinsky, and Commissioner Bradford. (The group's membership and its conclusions and recommendations regarding the Reactor Safety Study and development and use of risk assessment methodology will be found in Chapter 11.)



NRC's Licensee Contractor and Vendor Inspection Program, coordinated from the Region IV (Dallas) Office, ensures that organizations supplying services, equipment, components or systems to licensees/applicants carry out quality assurance programs that meet exacting NRC guidelines. This photo shows inspector Lawrence E. Ellershaw examining gas-metal arc welding being performed on a pump rotor during his inspection of a pump fabrication plant in Vernon, California.

1976 to develop methods for evaluating the regulatory performance of major licensees. Technical reports describing the two-year effort were released to the public and drew widespread attention from licensees, industry and citizen groups, and local news media. Using inspection and enforcement data for the year 1976, the staff explored three distinct evaluation methods: (1) statistical analysis of noncompliance information, (2) trend analysis of "licensee event" data, and (3) the subjective opinions of NRC inspectors. The Commission supports the staff's concept of developing and applying a comprehensive evaluation approach that will combine the best features of each of the three methods developed to date, beginning with 1978 data. If successful, the new two-year trial program will improve the quality of regulation by providing a systematic way of identifying key factors that influence licensee regulatory performance and, at the same time, assist the NRC in allocating inspection resources more efficiently and effectively.

The more severe sanctions imposed in 1978 citations of licensees for failure to comply with NRC requirements included 14 civil monetary penalties and 10 orders to cease and desist operations, or for modifications, suspension, or revocation of licenses. The Commission plans to resubmit proposed legislation not acted on by the 95th Congress which would sharply increase the amount of a fine that could be levied as a measure to provide greater incentive for licensee compliance.

Transportation

In August, the NRC certified to the Congress, in conformity with Public Law 94-79, that it had developed and tested a safe plutonium container which would not rupture under crash and blast testing equivalent to the crash and explosion of a high-flying aircraft. This culminated a three-year effort involving extensive design and testing and reviews by the Advisory Committee on Reactor Safeguards and the National Academy of Sciences' Assembly of Engineering.

A final NRC environmental statement assessing impacts of radioactive materials transport by all modes was released in December 1977. It concluded that shipments are being conducted under the present regulatory system in an adequately safe manner. A draft environmental statement on the transportation of radioactive material in urban areas will be issued in early 1979. A preliminary report of a joint NRC-DOT study of the adequacy of existing requirements for shipping low-level radioactive material was completed in July. It concluded that the low hazard associated with uranium concentrate (yellow-cake) would not justify more accident-resistant packaging.

Litigation continued over a New York City ordinance which virtually bans the transport of significant amounts of radioactive material within the city. DOT has announced a rulemaking proceeding for routing restrictions of highway movements, and the NRC is considering joint participation in the proceeding.

The 1977 Annual Report noted that shipments of highly enriched uranium through Chicago's O'Hare Airport would cease pending a joint study by NRC and the Office of the Mayor of Chicago. In February 1978, the Mayor decided against proceeding with the short-term study at that time. The Mayor subsequently indicated that the use of a military air base or the military side of a civilian-controlled airport would go far to reassure the general public that every possible precaution is being taken relative to the transportation of these materials. The NRC referred this question to the Executive Branch. Meanwhile, there has been a *de facto* suspension of air shipments of highly enriched uranium through O'Hare Airport since December 1977.

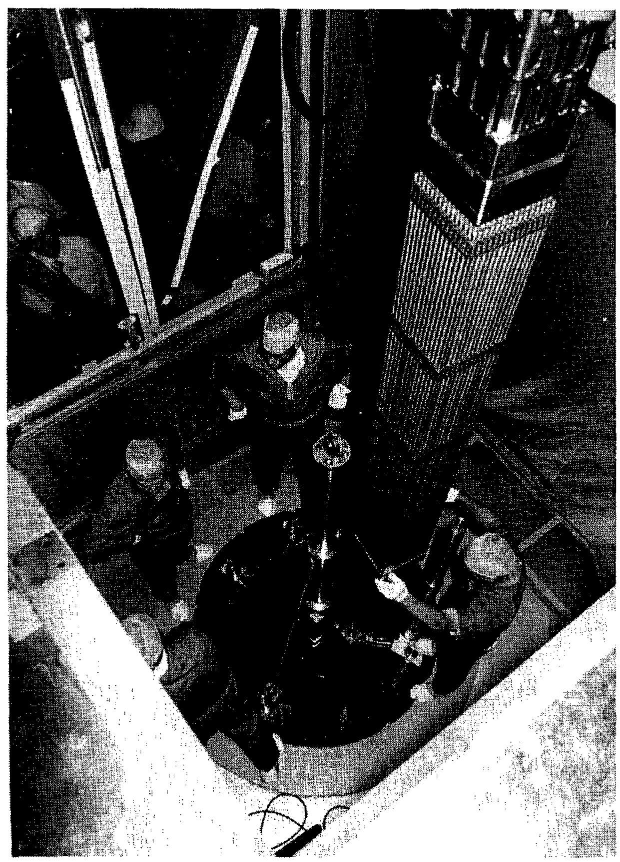
RESEARCH

NRC's confirmatory research program continued to expand and to produce useful results during 1978, particularly with regard to light water reactor safety. Reactor research adds to the understanding of the margins of safety which NRC licensing requirements are intended to provide.

The loss-of-fluid test facility (LOFT)—the Commission's largest test apparatus located at the Department of Energy's Idaho National Engineering Laboratory—was brought to its full design power of 50 megawatts (thermal) on October 8 after completion of a series of non-nuclear tests. The LOFT project investigates the integral thermal-hydraulic and nuclear fuel behavior aspects of loss-of-coolant accidents to permit the validation of analytical models developed for reactor safety analysis and the evaluation of emergency core cooling systems. The first nuclear experiment was successfully conducted in December and others will continue into the 1980's, dealing with a variety of pipe-break sizes and locations.

Other research activities during the year included initiation of a program to evaluate safety margins in seismic design methodology for reactors, operation of the modified Annular Core Research Reactor at its upgraded design power, completion of the first loss-of-coolant accident blowdown test in the Power Burst Facility, and development of production versions of major systems, component and containment computer codes. The research program also contributed to development of safe plutonium air-shipment containers, as discussed above.

In April the NRC provided to the Congress, as directed in the Fiscal Year 1978 Authorization

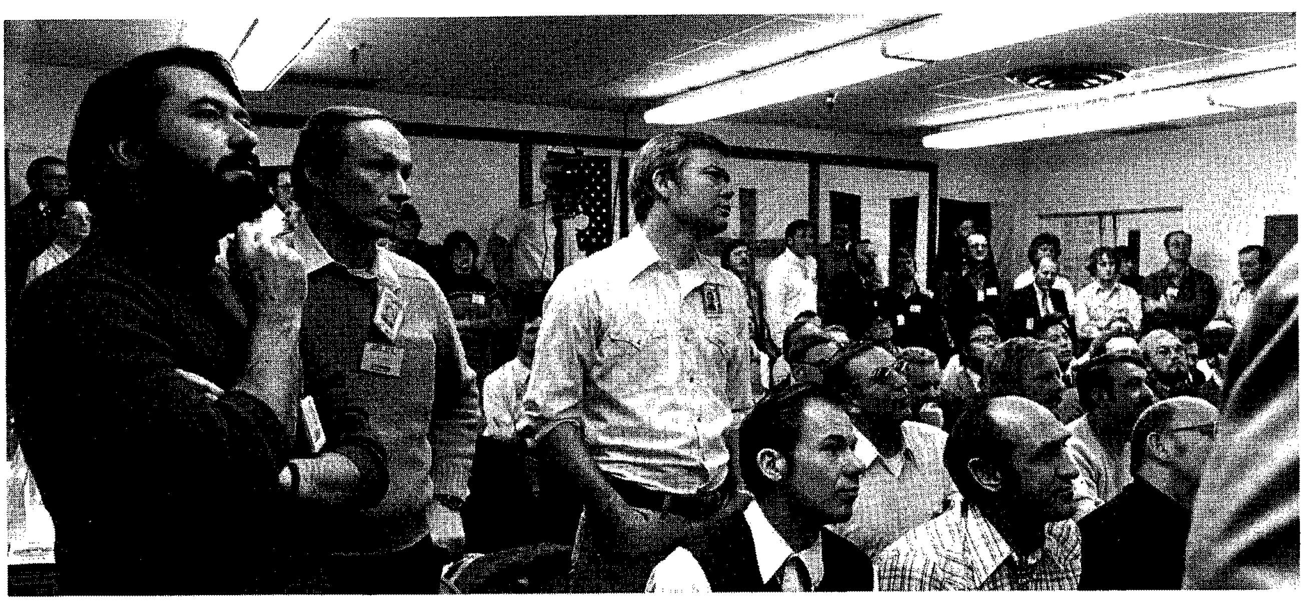


Work crews load nuclear fuel modules into the LOFT reator in preparation for bringing the reactor critical for the first time. Criticality occurred on February 5, 1978. (Note: New nuclear fuel can be handled directly, whereas after operation in the reactor, remote handling is required.) The first fuel loaded was a corner module, containing a neutron source to help start the reactor. The second module was the center module shown in the reactor, held in place by temporary supports. The assembly shown going into the reactor is one of the four which contain the control rods. Fuel loading was done inside a clean area to prevent foreign material from entering the reactor.

Scenes before and during the successful conduct of the first of a series of nuclear tests in NRC's LOFT (Loss-of-Fluid Test) reactor on the night of December 9, 1978 at the Department of Energy's Idaho National Fngineering Laboratory (INEL).

Above, in the LOFT control room two hours before the test, computer specialists discuss code predictions with NRC, DOE and LOFT contractor project personnel. The 50 thermal megawatt test reactor, largest in NRC's program of confirmatory research and the only complete experimental system of its kind in the world performing loss-of-coolant experiments, is crammed with hundreds of instruments to gather data on temperatures, pressures and coolant flow rates throughout a brief but extremely complex event: a deliberate loss of coolant, simulating a reactor pipe break, nuclear fuel heatup, and fuel cooldown by emergency cooling systems.

Below, seconds before "blowdown" in the test, NRC, DOE, contractor and foreign scientists watch intently as critical items of information are recorded on a display panel in the visitors' room adjacent to the control room in a concrete control complex. (Thomas E. Murley, director of NRC's Division of Reactor Safety Research, is shown standing at left.) After "countdown," in the 90-foot-tall steel containment building nearby, two large blowdown valves opened in about 18/1000 of a second and steam and water was rapidly discharged into a suppression tank. After automatic shutdown of the reactor, on loss of the coolant, the temperature of the nuclear fuel cladding rose from 650 degrees Fahrenheit to about 950 degrees, then leveled off and lowered as the emergency systems successfully forced cooling water back into the reactor.



Act (P.L. 95-209), a plan for developing new or improved safety systems for nuclear power plants. A status report, also required annually by the Act, is contained in Chapter 11.

In September, a seven-member group of scientists appointed by the Commission to review the Reactor Safety Study (also known as the Rasmussen Report), delivered its final report. The charter of the group, headed by Professor Harold W. Lewis of the University of California, had four basic elements: (1) to clarify the achievements and limitations of the Reactor Safety Study (WASH-1400); (2) to assess peer comments on it, and the response to those comments; (3) to study the present state of risk assessment methodology; and (4) to recommend to the Commission how (and whether) such methodology can be used in the regulatory and licensing process. In general, the report of the Lewis group agrees with much of the criticism that has been expressed of the Reactor Safety Study, particularly of the Executive Summary of the study, while endorsing the basic fault tree/event tree methodology that was employed in the study. (The Lewis group's summary of its findings and recommendations appears in Chapter 11 under "Risk Assessment Research.") At the end of the year, the Commission was reviewing the report and its recommendations.

THE NUCLEAR FUEL CYCLE

Salient developments affecting the NRC's responsibilities in nuclear fuel cycle regulation included: (1) enactment of the Nuclear Nonproliferation Act of 1978, exerting strong impact on export licensing considerations; (2) a Congressional mandate to study possible extension of NRC regulatory authority to existing and future Federal radioactive waste storage and disposal activities; and (3) enactment of uranium mill tailings control legislation which gives the NRC direct regulatory authority over tailings and provides for remedial actions at inactive mill sites.

Other significant fuel cycle activities requiring NRC participation were the study by an Interagency Review Group on Nuclear Waste Management, which is to be reported to the President in early 1979; and ongoing national and international evaluations of nuclear fuel cycle systems to explore means of minimizing nuclear proliferation risks.

Waste Management

The importance of resolving the issue of safe storage and disposal of nuclear wastes—particularly high-level radioactive waste—was emphasized by the President's action in establishing an Interagency Review Group on Nuclear Waste Management (IRG) to develop a strategy to deal with handling existing and future waste from military and civilian activities. The NRC, participating as a non-voting member, provided technical assistance and staff comments on successive drafts of the IRG report. Recommendations based on the report and public comments thereon are expected to be sent to the President in final form early in 1979.

On the basis of a draft of the IRG report released in October for public comment, the Commission feels that it objectively identifies key issues and establishes a philosophic basis for a disciplined approach to solving waste management problems. The Commission will study the recommendations in the final report to determine their potential impact on the direction and scope of the agency's waste management programs.

In November, the Commission issued for public comment a proposed policy statement on procedures for reviewing a possible license application from DOE for a high-level nuclear waste repository. The proposed process would involve (1) pre-application consultation by DOE and NRC staff on site suitability matters; (2) formal safety and environmental review of the application by NRC, and an opportunity for a public hearing before a decision on construction; (3) a second NRC formal review and an opportunity for a hearing before the Commission could authorize receipt of waste for storage; and (4) an NRC staff review after the repository had been filled to capacity prior to a Commission decision on the closing of the facility and amendment of the license.

The NRC plans to issue by early 1980 proposed regulations on high-level waste classification, form, packaging, and repository siting and design.

As directed by the Fiscal Year 1979 Authorization Act (P.L. 95-601), the NRC is exploring with DOE possible extension of NRC regulation to existing and future Federal waste storage and disposal activities, and also ways to improve State participation in siting, licensing and developing waste facilities. Reports on both these investigations are expected to be submitted to the Congress by March 1, 1979.

Recent developments have raised the question of whether adequate regionally distributed commercial capacity for low-level radioactive wastes will be available at currently operating facilities. Two of the six licensed commercial burial grounds (West Valley, N.Y., and Maxey Flats, Ky.) are closed and a third at Sheffield, Ill., has reached its licensed capacity. A limit has been placed by South Carolina on the acceptable volume at Barnwell, S.C. Thus, a large fraction of the low-level waste generated in the Eastern and Midwestern United States must soon be transported to the operating sites at Beatty, Nev., and Hanford, Wash. The NRC believes that the industry can work out cooperative arrangements for use of shielded containers, transport vehicles, and interim storage for the immediate future. However, the NRC has requested DOE to develop a contingency plan for use of its disposal sites for commercial wastes if needed.

During the year, the NRC conducted a study, scheduled for completion early in 1979, of methods other than shallow land burial for disposal of low-level wastes, including: (1) engineered structures, (2) ocean disposal, (3) mined cavities, and (4) burial at greater depths (approximately 30 feet) than the four to six feet of cover in present practice. Public comments have been requested on development of a regulatory program for alternative disposal methods.

At year-end, the Commission elevated its waste management organization to divisional status, with some 50 staff members and $10 million allocated to these activities during fiscal year 1979.

Spent Fuel

Termination in 1977 of proceedings on reprocessing and recycle of plutonium in light water reactors accentuated the need for interim storage of the growing accumulation of spent fuel discharged from nuclear power plants. The problem was addressed in a draft environmental impact statement issued by the NRC staff in March 1978. The statement indicated that commercial spent fuel generated through the year 2000 could be accommodated in a safe and environmentally sound manner either by modifying storage pools at reactor sites or by providing independent storage facilities. The final statement, to be completed in early 1979, will take into account the extensive public comments received.

Meanwhile, NRC has taken a number of actions to authorize pool expansions and to prepare for licensing of offsite storage. As of September 30, 1978, 36 of 50 applications to expand pools had been approved. Development of regulatory guidance on temporary spent fuel storage has received high priority, and a proposed rule on independent installations was published for public comment in October.

The NRC staff also has provided guidance to DOE and to the Tennessee Valley Authority, at their request, regarding potential license applications for interim spent fuel storage installations.

Mill Tailings Control

The Uranium Mill Tailings Radiation Control Act of 1978 gives the NRC direct licensing authority over mill tailings by amending the definition of byproduct material in the Atomic Energy Act of 1954. It also provides for NRC participation in a DOE-implemented remedial action program to control tailings piles at inactive milling sites which have long been recognized as needing corrective action. (Provisions of the Act are described later in this chapter.) During the year, the NRC had already set performance objectives for the uranium milling industry as part of an extensive program to upgrade tailings management. The staff is preparing a generic environmental impact assessment which will present alternative solutions and provide for public participation in regulatory decisions.

In the interim, NRC is requiring a stabilization and control program at all uranium mills as part of the license review for new mills or applications for license renewals. Licenses are being conditioned to require tailings stabilization and financial security arrangements to ensure this. NRC also is working with its Agreement States to ensure compatibility of regulatory requirements in this area.

Exports and International Safeguards

The quest for means of minimizing nuclear proliferation risks in the operation of nuclear fuel cycle systems continued to dominate international regulatory concerns during 1978.

The NRC participated throughout the year in support activities associated with the International Nuclear Fuel Cycle Evaluation (INFCE), being conducted by more than 50 countries and international organizations, and with DOE's Non-proliferation Alternative Systems Assessment Program (NASAP) which is providing technical input to INFCE. NRC has furnished technical expertise to U.S. support groups in the INFCE project, and has been reviewing and commenting on the health and safety, environmental, safeguards and licensing aspects of reactor and fuel cycle concepts being studied in NASAP.

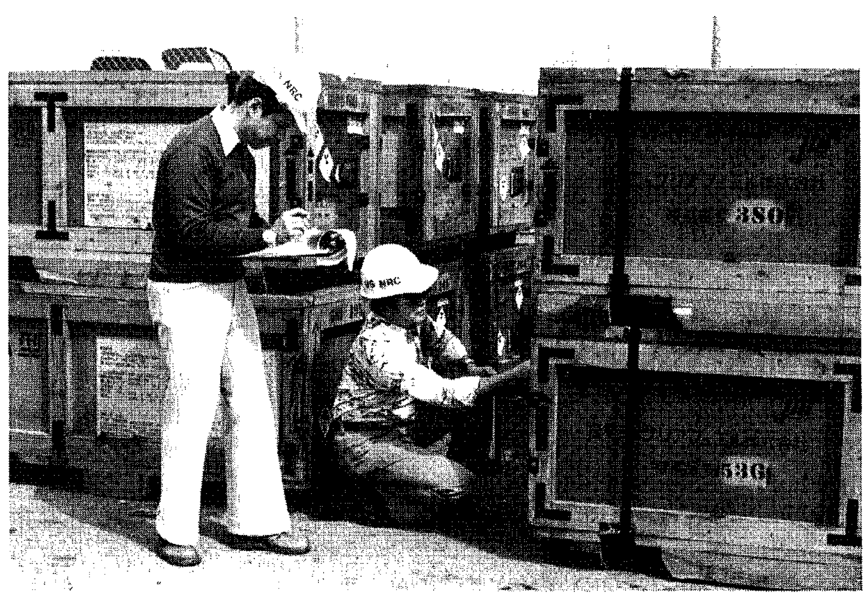
Under provisions of the 1979 NRC Authorization Act (described later in this chapter), NRC will broaden its activities in monitoring and reporting to Congress on these fuel cycle studies.

With the enactment of the Nuclear Nonproliferation Act of 1978, United States policy took a new direction which significantly affected the NRC's activities and decision-making in export/import licensing matters. Before the Act became law, the NRC had already begun actions to consolidate and codify export/import licensing regulations, and these rules (10 CFR Part 110), which became effective in May, serve to implement provisions of that legislation.

Shortly after the nonproliferation legislation was enacted, the four NRC commissioners then in office divided in a tie vote on the question of whether or not India met all the criteria in the new Act in their consideration of a license application to export nuclear fuel for the Tarapur reactors. As a result, the Commission did not certify that the statutory criteria were met. This resulted in referral of the case to the President who authorized the export on April 27. After a 60-day Congressional review period expired without a resolution disapproving the proposed export, the material was shipped to India in July.

The Commission's views concerning the experience in discharging its new responsibilities under the Act, which are required to be reported annually, are presented in Chapter 9.

International safeguards continued to be a major focus of Commission interest in 1978. In February, the Commission informed the cognizant committees of Congress of its views regarding safeguards deficiencies in various countries as identified in the Special Safeguards Implementation Report of the International Atomic Energy Agency (IAEA). The Commission further noted that the staff has indicated its inability on the basis of available information to provide independent assessments of the adequacy of IAEA safeguards. In correspondence with this situation, the Commission has supported the development of an interagency U.S. Government action plan to strengthen IAEA safeguards. Meanwhile, in accordance with pertinent statutes, the NRC continues to assess the international safeguards aspects of proposed exports on the basis of available information.



NRC auditors Yutaka Kobori and Arnold Wieder from the Region V Office in Walnut Creek, Cal., inventory shipment of reactor fuel bound for Japan as part of NRC's safeguards inspection program.

Domestic Safeguards

A number of safeguards efforts were initiated or continued during the fiscal year. Among these was the implementation of the requirement that power reactor licensees' safeguards meet a specified hypothetical threat involving sabotage of a nuclear facility (see Chapter 4). In addition, a comprehensive evaluation program to assess safeguards was continued at licensed fuel cycle facilities. Inspections were performed to assure compliance with safeguards regulations and license conditions. Safeguards provisions for non-power reactor operations are established on an individual basis at a level commensurate with the safeguards risk posed by the facility. Most non-power reactors involve only a small risk of either theft or sabotage; however, NRC is currently reevaluating such risks with a view toward improving physical protection measures at these facilities. The intransit domestic shipping of strategic special nuclear material (SSNM) also came under scrutiny during the year: from January to October of 1978, all eight shipments made were inspected while in progress, with no items of non-compliance detected.

Several guides and reports were issued by NRC during fiscal year 1978 in support of the safeguards regulations of Part 73, furnishing assistance to licensees in designing intrusion alarm systems, as part of the physical security required for power reactors or activities involving SSNM; in the training of security personnel at the facility site and for transportation purposes; for assessing the potential benefits of automation in tracking SNM in storage or accounting for SNM through sampling methods; and the like (see Chapter 10).

The Safeguards Contingency Planning Program produced detailed contingency plans to be carried out at both the individual facility or licensed operation and at the national level in the event of a theft of SSNM or sabotage of a nuclear facility.

(A detailed report on domestic safeguards for fiscal year 1978 is being sent separately to the Congress as required by Public Law 95-601, amending Section 209 of the Energy Reorganization Act of 1974.)

THE LICENSING PROCESS

Improving the process for licensing nuclear power plants, which has been a continuing goal of the Commission, became a focal point of public, Congressional and Executive Branch concern during 1978.

Some basic problems were underlined in the Seabrook case which has been before public agencies and in and out of court, and still awaits final resolution. In the years following initial authorization for construction of the Seabrook Nuclear Power Plant in New Hampshire, the Commission twice ordered suspension of construction; the licensing and appeal boards expended thousands of hours on the case; and NRC and Environmental Protection Agency decisions have several times been taken to court. The Commission found its decisions complicated by decisions made by other agencies.

The Seabrook experience offered some valuable lessons for the NRC. First, the staff and the hearing boards need to do a good job in developing an adequate evidentiary record, both for efficiency in the process and to avoid repeated court challenges. Second, Seabrook pointed up the value of an early site review in order to resolve at an early stage issues concerning basic land use and the environment. Third, the Seabrook case helped identify some ambiguities in the NRC alternative sites review that need correcting.

As a direct outgrowth of the Seabrook experience, the Commission also has directed that a comprehensive study be conducted of the "immediate effectiveness" rule which will focus particularly on the implications of permitting construction of nuclear power plants to proceed while challenges to construction permits are under adjudication.

The Commission has also ordered a study of whether the Commission should involve itself to a greater degree in the licensing process by taking direct appeals, at least of some issues, from the decisions of the licensing boards. At present, an NRC Atomic Safety and Licensing Appeal Board reviews decisions of the Atomic Safety and Licensing Boards. The Commission may review decisions of an appeal board in cases of exceptional legal or policy importance, either on its own motion or by accepting a petition for

A new Memorandum of Understanding establishing an overall management policy for the Department of Energy and the NRC with regard to interagency relationships in the conduct of research programs and related activities was signed on February 24, 1978, at Commission headquarters. Left to right: DOE Under Secretary Dale D. Myers, NRC Chairman Joseph M. Hendrie, and NRC Commissioner Richard T. Kennedy.

review. The Commission has not taken a position on the matter pending completion of the study.

A number of other administrative improvements in licensing procedures have been initiated or are continuing on the basis of Commission-approved recommendations coming out of an extensive staff study (see Chapter 2).

During the year, the Commissioners presented their individual views in Congressional testimony concerning the Administration's proposed nuclear siting and licensing legislation which failed of enactment during the second session of the 95th Congress. The proposal featured, among other things, statutory recognition and extension of some of the Commission's policies and initiatives in the areas of early site review and standardization of nuclear power plants. At year-end, the Commission was preparing its views concerning possible legislative initiatives for communication to the cognizant committees during the new session.

NEW TASKS MANDATED

New NRC responsibilities, special tasks and reporting requirements were mandated in three Acts which became law during the final session of the 95th Congress: the Nuclear Nonproliferation Act of 1978 (discussed earlier in this chapter and in Chapter 9), the NRC Authorization Act for Fiscal Year 1979 (Public Law 95-601, signed November 6), and the Uranium Mill Tailings Radiation Control Act of 1978 (Public Law 95-604, signed November 8).

Authorization Act Requirements

The fiscal year 1979 NRC Authorization Act contains many provisions affecting the NRC's activities and authority, and requiring new reports to Congress, both through new mandates and through amendments to the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974.

The principal new requirements in the Act are as follows:

Low-Level Radiation Health Effects Study. NRC and EPA are directed to conduct, in consultation with the Secretary of Health, Education, and Welfare and other Federal and State agencies, preliminary planning and design studies for epidemiological research on the health effects of low-level ionizing radiation. By April 1, 1979, NRC and EPA are to report to Congress on agency capabilities and research needs and by September 30, 1979, on options for Federal research in this area.

Safeguards Reports. The Commission is to submit a special report to Congress before February 1, 1979 on the status of domestic safeguards matters during fiscal year 1978, with a report on each succeeding year to be included in the NRC Annual Report to Congress.

Fuel Cycle Evaluation. The NRC must monitor, and assist as requested, the International Nuclear Fuel Cycle Evaluation and the studies being carried out by DOE of the various fuel cycle systems, and report to Congress on the status of these studies semiannually through calendar year 1980 and yearly thereafter through 1982.

Employee Protection. Employers, including NRC licensees, license applicants, and their contractors and subcontractors, may not discharge or otherwise discriminate against employees for assisting the NRC enforcement process. Any employee who believes he has been discriminated against for any such assistance may file charges with the Secretary of Labor, who is authorized to investigate and rule on the merits of the complaint and to enforce a finding of a violation by all appropriate means.

Waste Disposal. Several provisions relate to studies and reports concerning radioactive waste storage or disposal. NRC is directed to:

- Investigate, in cooperation with DOE, possible extension of NRC's regulatory authority to existing and future Federal radioactive waste storage and disposal activities. The Commission is directed to report the results to Congress by March 1, 1979, including a listing and inventory of all radioactive waste storage and disposal activities now being conducted or planned by Federal agencies.
- Notify the Governor and legislature of any State when the Commission has knowledge of a proposed site in such State for radioactive waste storage or disposal.
- Explore improving the opportunities for State participation in the siting, licensing, and developing of waste storage and disposal facilities. The report of results and any necessary legislative proposals (including a possible grant program) is to be submitted to Congress by March 1, 1979.

Conflicts of Interest. NRC must carry out rulemaking to establish regulations ensuring that persons under contract to the agency conduct their activities free from any real or perceived interest conflicts.

Contractor Use. NRC must report to Congress annually, beginning January 1, 1979, on its use of contractors, consultants, and the national laboratories.

Licensing Boards. The Commission is directed to review the selection and training process for members of the Atomic Safety and Licensing Boards, report the findings to Congress, and revise the process as appropriate.

At the end of 1978, in compliance with the Act, NRC-EPA efforts were underway to develop preliminary plans for an epidemiological research program on health effects of low-level radiation; a special report to Congress on the status of domestic safeguards during 1978 was nearing completion; work was in progress concerning possible extension of NRC regulatory authority to existing and future Federal radioactive waste disposal activities and on possible improvements of opportunities for State participation in the waste disposal area; the first annual report to Congress on NRC use of contractors, consultants and the national laboratories was near issuance; and the Commission was reviewing the selection and training process for licensing board members.

Uranium Mill Tailings Control Act

The Uranium Mill Tailings Radiation Control Act of 1978 extends NRC's regulatory authority to include uranium mill tailings and provides for a program of remedial action at inactive mill sites which also places new responsibilities on the agency.

As this Annual Report went to press, the NRC was taking appropriate steps to comply with provisions of the Act.

Licensing and Regulation. Title II of the Act gives NRC direct licensing authority over mill tailings by amending the definition of licensable "byproduct material" in Section 11e. of the Atomic Energy Act of 1954, as amended, to include "(2) the tailing or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content."

The Commission is required to implement specific licensing conditions and determinations expressly set forth in the statute. Similarly, Agreement States which regulate mill tailings by arrangement with the Commission must follow a more narrowly circumscribed pattern with less discretion permitted in the choice of regulatory procedures. Portions of the new regulatory regime for mill tailings will not come into full force until three years after the date of enactment of this legislation. Among the features of the Act are the following:

- Three years after enactment, any byproduct or source material license issued or renewed for an activity that produces tailings must be conditioned to assure (1) compliance with NRC decontamination, decommissioning and reclamation standards; and (2) transfer of ownership of the tailings to either the State where the activity occurred or to the United States. NRC is authorized to require the custodial agency to monitor and maintain the tailings to assure safety and compliance with NRC and EPA requirements.
- NRC is authorized to require adequate financial arrangements by a licensee to assure decontamination, decommissioning, and reclamation of sites, structures and equipment used in conjunction with mill tailings and mining wastes.
- Continued State regulation of mill tailings will have to be brought within the scope of the NRC's Agreement State program, provided for in Section 274 of the Atomic Energy Act. Section 274 is amended extensively, including provision for termination of "all or part of" an agreement upon a finding either that public health and safety so requires or that the State has not complied with one or more of the requirements in Section 274.
- EPA must promulgate "standards of general application" for protection of public health and safety and the environment from radiological and nonradiological hazards from mill tailings, to be implemented and enforced by the NRC or Agreement States, as appropriate.
- The Act authorizes appropriations up to $500,000 to NRC for fiscal year 1980 for grants to Agreement States to assist in developing their programs to implement provisions of the amended Section 274.

Remedial Action at Inactive Sites. Title I of the Act establishes a State-Federal cooperative program, generally administered by the Secretary of Energy, for cleaning up uranium tailings piles at inactive mill sites subject to NRC consultation and concurrence. NRC is given a consultative role in the designation of the "processing sites" needing remedial action, and in reports to the Congress by DOE and EPA. NRC concurrence is required regarding:

- Terms and conditions of cooperative agreements for remedial action.
- Determination that removal of the materials from a processing site is appropriate.
- Sale or transfer to the United States of lands or interests (designated processing site) acquired by a State or permanent use of such land by the State for park, recreational or other public purposes.
- DOE's determination that remedial action has been completed, for purposes of transferring title to the materials and lands on which they are disposed to the United States.
- Selection and performance of remedial action in accordance with EPA general standards.
- Recovery of additional minerals from residual radioactive materials.

NRC licenses will be required for (1) custody by DOE or other Federal agency of land and residual radioactive materials after completion of remedial action; (2) sale or lease by the Government of subsurface mineral rights on lands where such materials are disposed; and (3) removal from Indian lands and retention and maintenance elsewhere by DOE of residual radioactive materials.

The Commission is directed to encourage public participation in its activities concerning the remedial action program and, in cooperation with DOE, to document and make public information obtained in the program.

2

Reactor Regulation

The primary goal of the NRC in licensing and regulating nuclear reactors in the United States is to assure the health and safety of the public and the protection of the environment. The reactor licensing process is centered in the NRC's Office of Nuclear Reactor Regulation (NRR), where each proposed nuclear power plant is reviewed by a staff which is drawn from a broad spectrum of professional disciplines; the staff is organized into four divisions, plus an antitrust and indemnification group. (See Appendix 1 for a description of NRR organization.)

This chapter discusses major aspects of the reactor licensing actions and develops the relationship between licensing actions and the primary objective: the safe operation of nuclear power plants. The chapter covers specific licensing actions during fiscal year 1978; steps to ensure safe design under the "defense-in-depth" concept; highlights of special technical reviews; action to improve the licensing process through standardization, early site review, and other means; environmental protection; antitrust reviews; indemnity and insurance matters; and other subjects related to safety in reactor operations. (Safeguards against sabotage of reactors are discussed in Chapter 4.)

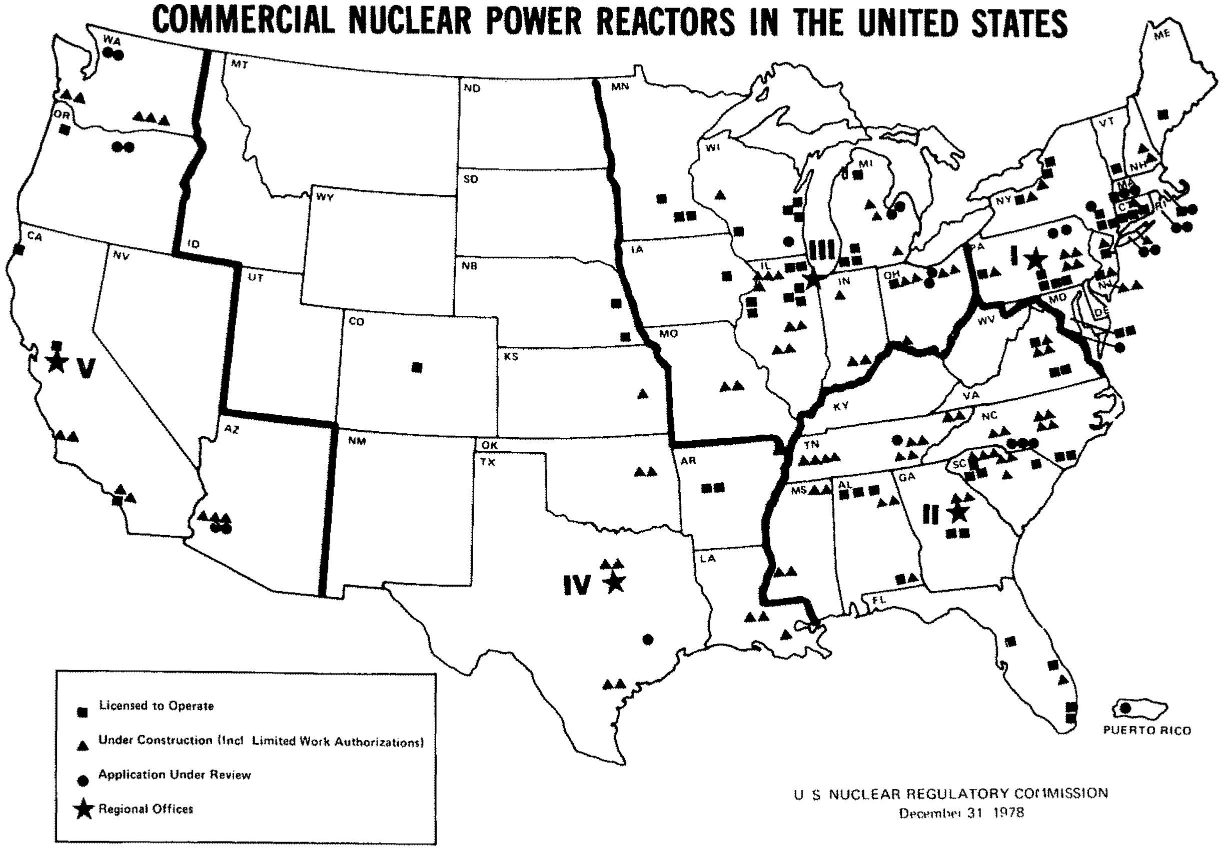
Status of Nuclear Power Generation

As of September 30, 1978, there were 212 nuclear power units either in operation, being built or being planned, representing a total capacity of 209,000 net megawatts electric (MWe). Of these 212 units, 195 had entered the NRC licensing process, as follows:

- 70 licensed to operate, with a total capacity of 51,000 MWe.
- 88 with construction permits representing 96,000 MWe capacity.
- 37 under review for construction permits, representing 44,000 MWe capacity. (Initial construction work was proceeding on four of these under limited work authorizations.)

Of the remaining 17 units—those which had not entered the

NRC REGIONAL OFFICES
AND
COMMERCIAL NUCLEAR POWER REACTORS IN THE UNITED STATES



NRC licensing process—nine had been ordered and eight publicly announced.

Licensing Reactor Operators

The safety of a nuclear facility depends not only on its design but on the qualifications of the people who operate it. To assure that the people in charge of each nuclear power plant are capable of directing and performing the activities necessary to reactor operation, the NRC requires each individual who handles the controls of the reactor to be licensed. The requirements for issuance of operators' licenses are set forth in 10 CFR Part 55. Two types of licenses are issued by the NRC: one for "operators" and one for "senior operators." During fiscal year 1978, the NRC issued 238 new operator licenses, 212 renewals, and 33 amendments, bringing the number of operator licenses in effect on September 30, 1978 to 1,052. During the same period 243 new licenses, 499 renewals and 82 amendments were issued for senior operators, bringing the total to 1,438 in effect.

ACTION ON TECHNICAL PROBLEMS

NRC actions on technical problems related to nuclear power plant safety can take a number of different forms. They can be (1) specific licensing actions to resolve a problem experienced or identified at an operating reactor, (2) long term research programs, (3) standards development efforts, (4) part of licensing (construction permit or operating license) reviews, or (5) generic reviews of issues that involve several nuclear power plants.

Items of the first type above that are determined to involve a major reduction in the degree of protection of the public health and safety are reported to Congress quarterly as Abnormal Occurrences (see Chapter 7). Discussions of several additional items involving licensing actions at operating reactors are discussed below, under "Other Technical Issues."

NRC research programs are discussed in Chapter 11 and the development of regulatory standards is discussed in Chapter 10.

Table 1. Nuclear Power Plant Licensing Actions—Fiscal Year 1978

LIMITED WORK AUTHORIZATIONS

Applicant	Facility	Date Issued	Location
1. Tennessee Valley Authority	Phipps Bend 1 & 2	10-18-77	Phipps Bend, Tenn.
2. Tennessee Valley Authority	Yellow Creek 1 & 2	2-9-78	Yellow Creek, Miss.
3. Public Service Co. of Oklahoma	Black Fox 1 & 2	7-26-78	Inola, Okla.

CONSTRUCTION PERMITS

Applicant	Facility	Date Issued	Location
1. Northern States Power Co.	Tyrone 1	12-27-77	Durand, Wis.
2. Duke Power Co.	Cherokee 1, 2 & 3	12-30-77	Cherokee County, S.C.
3. Tennessee Valley Authority	Phipps Bend 1 & 2	1-16-78	Phipps Bend, Tenn.
4. Carolina Power & Light Co.	Harris 1, 2, 3 & 4	1-27-78	Bonsal, N.C.
5. Washington Public Power Supply System	WPPSS 4	2-21-78	Richland, Wash.
6. Public Service Co. of Indiana	Marble Hill 1 & 2	4-4-78	Madison, Ind.
7. Washington Public Power Supply System	WPPSS 3 & 5	4-11-78	Satsop, Wash.

OPERATING LICENSES

Applicant	Facility	Date Issued	Location
1. Virginia Electric & Power Co.	North Anna 1	11-26-77	Mineral, Va.
2. Indiana & Michigan Electric Co.	Cook 2	12-23-77	Bridgman, Mich.
3. Metropolitan Edison	Three Mile Island 2	2-8-78	Goldsboro, Pa.
4. Georgia Power Co.	Hatch 2	6-13-78	Baxley, Ga.
5. Arkansas Power & Light Co.	Arkansas 2	7-18-78	Russelville, Ark.

THE LICENSING PROCESS

Obtaining an NRC construction permit — or a limited work authorization, pending 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 license. The process is set in motion with the filing and acceptance of the application, generally comprising ten 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 as 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-PDR 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 September 1975 and updated periodically. This plan states the acceptance criteria used in evaluating the various systems, components and structures important to safety and in assessing the proposed site, and it describes the procedures 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 summarizing the results of their review regarding the anticipated effects of the proposed facility on the 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 site, on safety aspects of the licensing decision.

In appropriate cases, 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 public 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 a review of the applicant's Environmental Report (ER) for acceptability. Assuming the ER is 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 time period 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 conducted on environmental and site suitability aspects of 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 concurrently 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 the plant is scheduled to be complete, the applicant files an application for an operating license. A process similar to that for the construction permit is followed. The application is filed, 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 require "backfitting" of a licensed plant, that is, the addition, elimination or modification of structures, systems or components of the plant.

Unresolved Safety Issues Plan

In 1977, the NRC's Office of Nuclear Reactor Regulation (NRR) instituted a program to define, categorize and manage generic technical activities on a systematic basis. The initial effort under this program resulted in the identification of 133 generic tasks. These tasks cover a variety of topics. Some are related to safety, some to environmental matters, and some to improving the regulatory process.

Subsequent to the inception of the NRR program, the Congress acted, in late 1977, to amend the Energy Reorganization Act of 1974 to include, among other things, a new Section 210, as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"Section 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report to the Commission thereafter."

In response to this reporting requirement, the NRC provided a report to the Congress (NUREG-0410) in January 1978 describing the generic issues program of the Office of Nuclear Reactor Regulation that had been implemented earlier in 1977. The NRR program described in NUREG-0410 provides for the identification of generic issues, the assignment of priorities, the development of detailed Task Action Plans to resolve the issues, projections of dollar and manpower costs, continuing high level management oversight of task progress, and public dissemination of information related to the tasks as they progress. The NRR program is, however, of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. As noted above, the program also includes other generic tasks of importance to the NRC's mission, such as those for the resolution of environmental issues; for the development of improvements in the reactor licensing process; for consideration of less conservative design criteria or operating limitations, in areas where overly conservative requirements may be unnecessarily restrictive or costly; for the maintenance and development of the NRC staff's capabilities to perform independent audit calculations; and for the actual performance of independent audit calculations.

This Annual Report section is limited to describing the progress on that portion of the NRR program required to be reported to the Congress by Section 210.

The following definition of an "Unresolved Safety Issue" was developed for use in identifying the generic issues in the broader NRR staff program that should be reported to Congress, pursuant to Section 210: "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected."

All of the generic issues reported to the Congress last year in NUREG-0410, as well as any other issues identified since that time, were considered as candidates for "Unresolved Safety Issues." A systematic review of these issues was undertaken by the NRC staff. As an aid to this review, an evaluation was made of the subject areas involved according to their relative importance from the standpoint of public risk. This risk-based characterization was used together with a substantial body of additional information (e.g., heavy weight was given to issues arising from events reported to the Congress as "Abnormal Occurrences") to determine which issues met the definition of an "Unresolved Safety Issue." The review resulted in the identification of seventeen "Unresolved Safety Issues." The Subcommittee on Generic Items of the Commission's Advisory Committee on Reactor Safeguards (ACRS) has been briefed on the identified issues. The NRC staff will continue to coordinate with the ACRS on these issues and future issues considered for reporting as "Unresolved Safety Issues." (The selection process and the rationale for decisions regarding particular issues are described in a separate report, NUREG-0510, "Identification of 'Unresolved Safety Issues' Relating to Nuclear Power Plants—A Report to Congress.")

Although the term "Unresolved Safety Issue" has been in use for some time, and the Congress used the term to identify those issues about which it wished to be kept informed, it has been frequently misunderstood. An immediate question is: if a generic safety issue (i.e., a safety issue relating to more than one plant) is "unresolved," then how can NRC grant a license to operate a specific nuclear power plant for which that issue is relevant? The answer is that before the license is granted the NRC staff must determine that licensing and operation of the specific plant can continue pending a generic resolution of the issue. The bases for such a determination include one or more of the following: (1) the issue does not apply to or has been resolved for the plant under consideration; (2) interim measures assuring adequate safety of operation are being required at affected plants pending final resolution of the issue; (3) resolution of the issue can reasonably be expected before the plant under consideration begins operation; or (4) the likelihood of occurrence and/or the consequences of an accident scenario, for which the issue under study is an important consideration, is small.

The NRC staff's conclusions in this regard are subjected to the scrutiny of the licensing process in individual cases. Specifically, the NRC staff's conclusions on individual applications are reviewed by the Advisory Committee on Reactor Safeguards and are specifically addressed in the public hearing process (see previous section in this chapter describing the licensing process).

The seventeen generic issues listed in Table 2 were determined to be "Unresolved Safety Issues." These issues are addressed by twenty-two generic tasks in the NRR Program for the Resolution of Generic Issues. The task numbers of the applicable generic tasks are provided in parentheses following the title of each issue in Table 2. Three of the twenty-two generic tasks addressing these seventeen issues have been completed. Generic Task A-6 was completed and documented in a report, NUREG-0408, "Mark I Containment Short Term Program Safety Evaluation Report," in December 1977; Generic Task A-26 was completed and documented in NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," in September 1978; and Generic Task A-31 was completed and documented in Regulatory Guide 1.139, "Guidance for Residual Heat Removal," in May 1978.

A discussion of each of the "Unresolved Safety Issues" follows.

Table 2: Unresolved Safety Issues and Related Task Numbers

1. Water Hammer — (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System — (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity — (A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments — (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram — (A-9)
6. BWR Nozzle Cracking — (A-10)
7. Reactor Vessel Materials Toughness — (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports — (A-12)
9. System Interactions in Nuclear Power Plants — (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment — (A-24)
11. Reactor Vessel Pressure Transient Protection — (A-26)
12. Residual Heat Removal Requirements — (A-31)
13. Control of Heavy Loads Near Spent Fuel — (A-36)
14. Seismic Design Criteria — (A-40)
15. Pipe Cracks in Boiling Water Reactors — (A-42)
16. Containment Emergency Sump Reliability — (A-43)
17. Station Blackout — (A-44)

Water Hammer

Water hammer events are intense pressure pulses in fluid systems, such as commonly experienced when rapidly closing a water faucet, and they often occur in nuclear power plant fluid systems. Since 1971, about 100 incidents involving water hammer in nuclear power reactors have been reported. These incidents have involved many types of fluid systems, including steam generator feed-rings, feedwater and steam supply piping, residual heat removal systems, emergency core cooling systems, containment spray systems, and service water systems. Water hammer has been attributed to various causes, such as the rapid condensation of steam pockets, steam-driven slugs of water, pump start-up with partially empty lines, and rapid valve motions. Most of the damage has been relatively minor, though there have been several cases of failure or partial failure of system piping.

While no water hammer incident has resulted in the release of radioactivity outside of a plant, the concern is that water hammer could result in the failure of a pipe in the reactor coolant system or disable a system required to cool the plant after a reactor shutdown.

The means to prevent one particular type of water hammer caused by the rapid condensation of steam in the steam generator feed-rings of some pressurized water reactors are being instituted. In addition, applicants with new steam generator designs are being required to demonstrate through test or analysis that water hammer will not occur in these designs. Plants with steam generators—of the top feeding type that are subject to water hammer—are being required to modify the feed-rings and/or test the systems to assure water hammer will not occur. And other actions to correct the specific causes of water hammer identified to-date are being required.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-1. The potential for water hammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that water hammer is given appropriate consideration in all areas of licensing reviews. A technical report providing the results of a staff review of water hammer events in nuclear power plants is scheduled for publication in February 1979. Issuance of this report completes a major subtask of Generic Task A-1. The remaining subtasks are expected to be completed in 1980.

Asymmetric Blowdown Loads On the Reactor Coolant System

In the very unlikely event of a rupture of the primary coolant piping in light water reactors, large non-uniformly distributed loads would be imposed upon the reactor vessel, reactor vessel internals, and other components in the reactor coolant system. The potential for such asymmetric loads, which result from the rapid depressurization of the reactor coolant system, was only recently identified and was not considered in the original design of some facilities. The forces associated with a postulated break in the reactor coolant piping near the reactor vessel, for example, could affect the integrity of the reactor vessel supports and reactor pressure vessel internals. A significant failure of the reactor vessel support system, besides impacting the reactor internals, has a potential for (1) damaging systems designed to cool the core following the postulated piping break, (2) affecting the capability of the control rods to function properly, (3) damaging other reactor coolant system components, and (4) causing other ruptures in the initially unbroken reactor coolant system piping loops and attached systems.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-2. The issue was originally identified in May 1975 by the Virginia Electric and Power Company in relation to its North Anna Units 1 and 2 nuclear power plants. A survey of all operating pressurized water reactors (PWRs) was conducted in October 1975 which showed that asymmetric blowdown loads had not been considered in the design of the reactor vessel supports for any operating PWR facility. In June 1976, the NRC staff requested all operating PWR licensees to assess the adequacy of the reactor vessel supports at their facilities with respect to these newly identified loads.

Most licensees with plants using Westinghouse nuclear steam supply systems initially proposed an augmented in-service inspection program

(ISI) of the reactor vessel safe-end-to-end pipe welds in lieu of providing the detailed analysis requested by the NRC staff. Licensees with Combustion Engineering nuclear steam supply systems submitted a probability study in support of a conclusion that the probability of a break at the location in the piping necessary to produce the postulated load was so low that no further analysis was necessary. Licensees with Babcock and Wilcox nuclear steam supply systems took an approach similar to Combustion Engineering licensees.

The NRC staff's review of these proposed alternatives to detailed plant-specific analyses has been completed with the conclusion that proposed alternatives to the requested analysis should not be accepted. Accordingly, the NRC staff sent letters on January 25, 1978, to all PWR licensees and applicants stating that an analysis must be undertaken to assess the design adequacy of the reactor vessel supports and other structures to withstand the loads when asymmetric loss-of-coolant accident forces are taken into account. As part of Task A-2, the NRC staff will review and approve analytical models and computer codes developed by reactor vendors to calculate asymmetric blowdown loadings, prior to their use by licensees and applicants in plant-specific analyses. In addition, the staff will develop explicit guidelines and acceptance criteria for the asymmetric load analyses and will conduct a pipe break probability study.

Plant modifications to assure that the postulated loads are accommodated have been implemented late in the construction stage of several plants and have been proposed and are under staff review for some operating plants. For plants still under operating license review, the NRC staff requires that plant-specific analyses be completed and any necessary plant modifications completed prior to issuance of an operating license. The generic efforts for pressurized water reactors under Task A-2 are currently scheduled for completion in early 1979.

The NRC staff has been investigating this phenomenon as it applies to boiling water reactors and has determined that asymmetric loads are also significant and therefore need to be evaluated for these lower pressure systems. The staff is currently developing plans for expanding Task A-2 to resolve this issue for boiling water reactors.

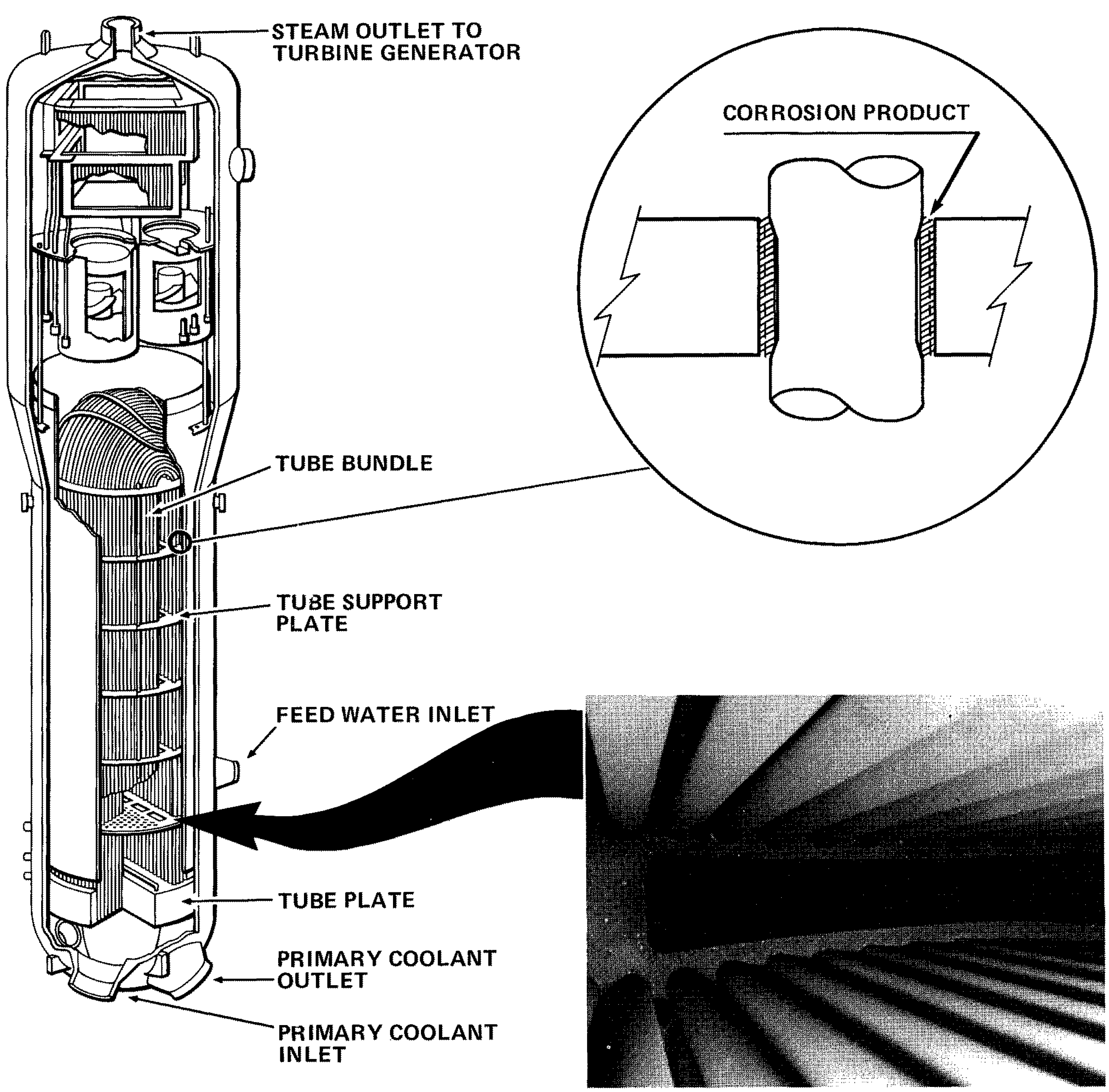
PWR Steam Generator Tube Integrity

The heat produced in the reactor at a nuclear power plant is used to convert water into steam which will drive the turbine-generators. In plants employing pressurized water reactors, the primary coolant water which extracts heat by circulating through the reactor core is kept under pressure sufficient to prevent boiling. This high-pressure water passes through tubes around which a secondary coolant (also water) is circulating, under somewhat lower pressure. The water in the secondary system is allowed to boil and produce steam to drive the turbine-generators. The assembly in which the transfer takes place is the steam generator. The tubes within it are an integral part of the primary coolant boundary, keeping the radioactive primary coolant in a closed system and isolated from the environment. The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions. In addition, the requirements for increased steam generator tube inspections and repairs have resulted in significant increases in occupational exposures to workers.

A detailed discussion of the specific problems associated with steam generator tube integrity that were occurring at operating reactors was provided in the 1977 NRC Annual Report, page 95. The information below is provided to supplement and update that information.

Corrosion resulting in steam generator tube wall thinning has been observed in several Westinghouse and Combustion Engineering (CE) plants for a number of years. Major changes in their secondary water treatment process essentially eliminated this form of degradation. Another major corrosion-related phenomenon has also been observed in a number of plants in recent years, resulting from a build-up of support plate corrosion products in the annulus between the tubes and the support plates. This build-up eventually causes a diametral reduction of tubes, called "denting," and deformation of the tube support plates. This phenomenon has led to other problems, including stress corrosion cracking, leaks at the tube/support plate intersections, and U-bend section cracking of tubes which were highly stressed because of support plate deformation.

PWR STEAM GENERATOR



The buildup of corrosion deposits between the steam generator tubes and the tube support plates, in addition to constricting the tubes, exerts stresses on the tube support plates. The stresses cause hourglassing of the normally rectangular internal bypass flow holes located between the innermost tube rows.

The significant developments in Westinghouse and Combustion Engineering steam generators, since June 1977, were the following:

- Continued tube denting at Indian Point Unit 2, San Onofre Unit 1, Surry Units 1 and 2, Turkey Point Units 3 and 4, and lesser amounts of denting at a number of other Westinghouse designed reactors. Steam generator replacement is planned for early 1979 or 1980 at Surry Units 1 and 2. Replacement or retubing is also being considered for Turkey Point Units 3 and 4. In the interim, the units are operating under restrictions imposed by the NRC.

- Discovery of support plate cracking (related to denting) at Indian Point Unit 2 and San Onofre Unit 1.
- Removal of several tubes and a section of support plate at Indian Point Unit 2 to investigate the potential for steam generator cleaning revealed continued active corrosion of the support plate.
- Continuation of tube denting at Millstone Unit 2 and Maine Yankee and discovery of denting in St. Lucie 1. Millstone Unit 2, Maine Yankee, and Arkansas Nuclear One Unit 2 have removed lugs and portions of the solid rim in the uppermost support plates to reduce the susceptibility of the plates to denting-related cracks (CE designs).
- Palisades Nuclear Power Station is sleeving degraded tubes instead of plugging them. This process restores the structural integrity of the tubes while keeping them in service (CE design).

Another form of steam generator tube degradation in Babcock and Wilcox (B&W) steam generators was found in the Oconee Nuclear Plant where the first tube leak occurred in July 1976. To-date, 14 tube leaks, all at the Oconee units, have occurred in B&W steam generators. The majority of these leaking tubes were located adjacent to the open inspection lane. Laboratory examination of removed defective tubes indicated that the tube failures were caused by the propagation of circumferential fatigue cracks by flow-induced vibration.

The significant developments in B&W steam generators, since May 1977, were the following:

- Continued tube leaks at the Oconee units.
- Initiation of a demonstration tube sleeving program by Duke Power Company at the Oconee units. The tube sleeves will not serve as part of the primary coolant boundary but will be installed to change the vibrational characteristics of the tubes and decrease the dynamic stresses and the susceptibility of the tubes to fatigue cracking.

*NRC approval or concurrence prior to a return to power following a steam generator inspection for tube leak is only required for those units whose steam generators are judged by the staff (taking performance history into account) to be so severely degraded that they require close, continuous monitoring.

Following inspections by licensees of their steam generators and the completion of any necessary repair programs, the NRC approves or concurs in the restart of each of the severely affected facilities.* To-date, the units severely affected by the tube denting have completed inspection and repair programs and received NRC approval for operation for limited time periods. Safe operation is assured by the imposition of strict conditions on licensed operation, requiring the plugging of affected tubes and restricting allowable leak rates during operation.

As the NRC staff continues to closely monitor, evaluate, and approve the acceptability of continued operation of plants experiencing steam generator tube problems, it has undertaken a number of generic reviews and studies as part of three generic tasks in the NRC Program for the Resolution of Generic Issues; specifically, Generic Tasks A-3, A-4, and A-5 each directed at the particular problems of Westinghouse, Combustion Engineering, and Babcock and Wilcox plants, respectively.

Under these tasks generic studies will be conducted to (1) evaluate inservice inspection results from operating reactors, (2) evaluate the consequences of tube failures under postulated accident conditions, (3) evaluate tube structural integrity, (4) establish tube plugging criteria based on new information, (5) define the requirements for monitoring secondary coolant chemistry, (6) evalute inservice inspection methods, and (7) review design improvements proposed for new plants. These studies will be used to revise current NRC staff requirements and guidance regarding these subjects. In addition, under Task A-3, the NRC staff will review and evaluate the first proposed steam generator replacement operation to establish acceptance criteria and guidance on a generic basis for use in the review of subsequent replacement operations. These generic tasks are currently scheduled to be completed in early 1980.

BWR Mark I and Mark II Pressure Suppression Containments

In the course of performing large scale testing of an advanced design pressure-suppression containment (Mark III), and during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads were identified which had not explicitly been included in the original Mark I or Mark II containment design basis. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to various modes of safety relief valve (SRV) operation generally associated with plant transient operating conditions. Since these new hydrodynamic loads had not been explicitly considered in the original design of the Mark I and Mark II containments, the NRC staff determined that a detailed reevaluation of these containment system designs was required.

As a result of the need for this reevaluation the affected utilities formed ad hoc Mark I and Mark II Owners' Groups and each has engaged the General Electric Company as its program manager. Both Owners' Groups developed two-phase programs consisting of a short-term program and a long-term program for resolution of the pool dynamic concerns for their respective containment designs. The Owners' Groups' programs include a number of comprehensive experimental and analytical programs to establish generic pool dynamic loads, load combinations and design criteria.

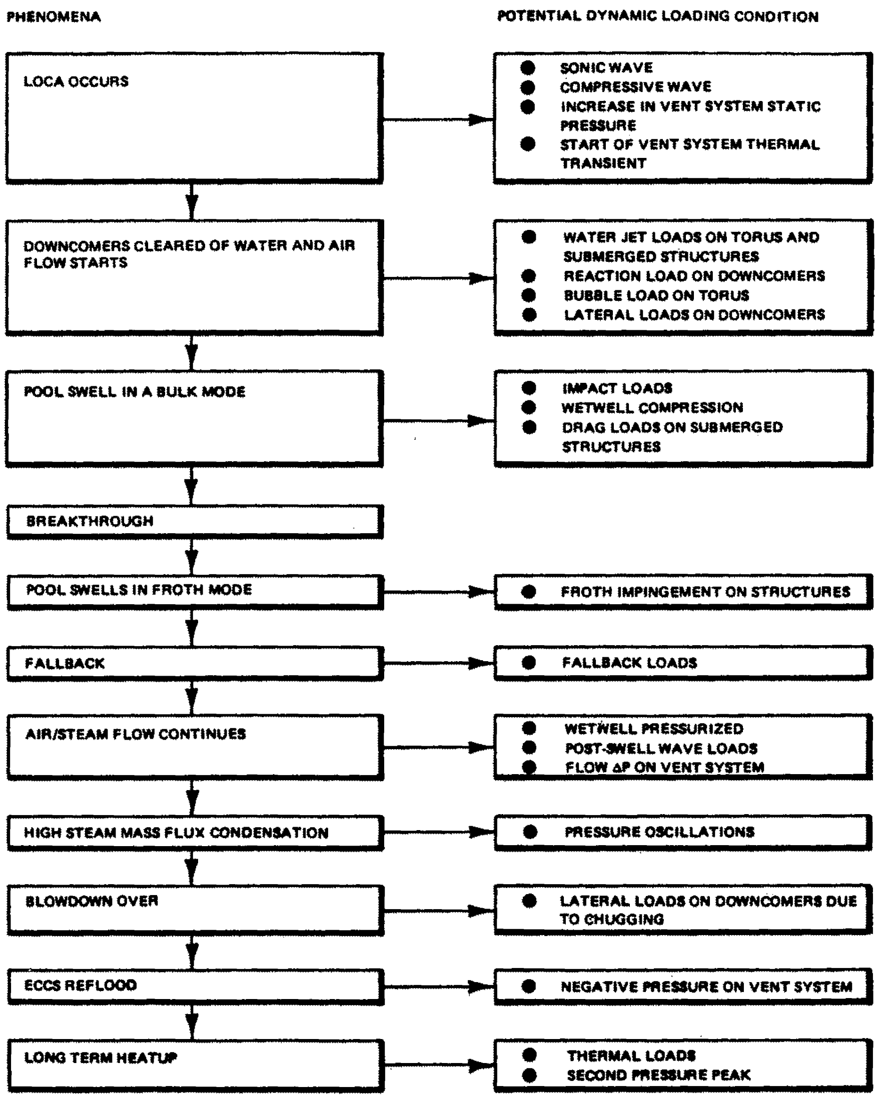
The NRC staff has identified and initiated a number of generic tasks to review and evaluate the results of the Mark I and Mark II Owner's Group short-term and long-term programs to develop technical positions for use in licensing actions on individual plants utilizing the Mark I and Mark II containment designs. These generic tasks are included in the NRC Program for Resolution of Generic Issues (described in NUREG-0410 as noted above). Specifically, they are Task A-6, Mark I Short-Term Program; Task A-7, Mark I Long-Term Program; Task A-8, Mark II Containment Program; Task A-39, Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments.

The objectives of the Mark I Short-Term Program were: (1) to examine the containment system of each BWR facility with a Mark I containment design to verify that it would maintain its integrity and functional capability when subjected to the most probable hydrodynamic loads induced by a postulated design basis loss-of-coolant accident; and (2) to verify that licensed Mark I BWR facilities may continue to operate safely, without undue risk to the health and safety of the public, while a methodical, comprehensive Long-Term Program is conducted. The NRC determined that, for the Short-Term Program, "maintenance of containment integrity and function" would be adequately assured if a safety factor to failure of at least two were demonstrated to exist for the weakest structural or mechanical component in the Mark I containment system (i.e., if the calculated stresses in all components of the affected containment structure were shown to be less than one-half the stress which would cause the component to lose its structural integrity). The NRC concluded that the objectives of the Short-Term Program had been satisfied and documented the basis for this conclusion in the "Mark I Containment Short-Term Program Safety Evaluation Report," NUREG-0408, dated December 1977. (Thus Task A-6 was completed in December 1977.)

The objectives of the Mark I Long-Term Program are: (1) to establish design basis loads that are appropriate for the anticipated life of each Mark I BWR facility, and (2) to restore the original intended design safety margins for each Mark I containment system. The Mark I Long-Term Program consists of a series of major tasks and subtasks which are designed to provide a detailed basis for hydrodynamic load definition and the methodology and acceptance criteria for the structural assessments. The generic aspects of the Mark I Long-Term Program will be described in a Plant Unique Analysis Applications Guide, scheduled to be completed in February 1979, and in the Load Definition Report, a portion of which was completed in December of 1978. The remainder of the Load Definition Report is scheduled to be completed in March 1979. Subsequently, each utility with a Mark I plant will perform a plant-unique analysis using approved load definition and structural analysis techniques to demonstrate conformance with the Mark I Long-Term Program structural acceptance criteria. These analyses are currently scheduled for completion in October 1979.

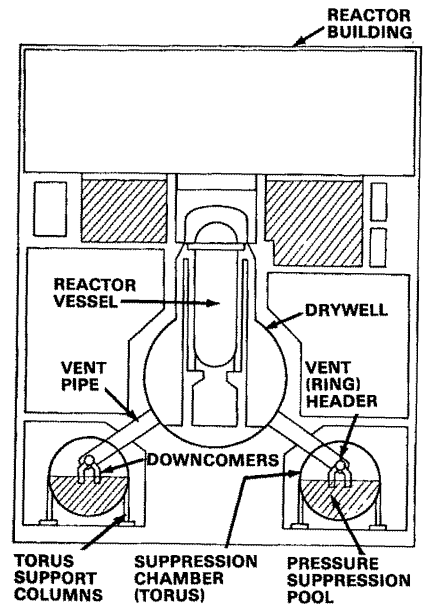
The scheduled completion date for the Mark I Long-Term Program (Task A-7), including the issuance of license amendments and the implementation of any plant modifications necessary to satisfy the Mark I Long-Term Program structural acceptance criteria, is December 1980. In recognition of this schedule, a number of facilities are adopting their own schedules to

Sequence of Events and Potential Loading Conditions Following a Postulated LOCA



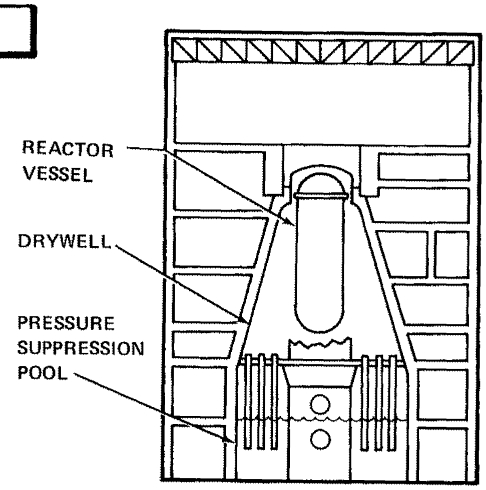
The design objective of the Mark I containment system is to condense the steam released during a postulated loss-of-coolant accident (LOCA) event, to limit the release of the fission products associated with the accident to the reactor building, and to serve as a source of water for the emergency core cooling systems. (From "Mark I Containment Short-Term Program Safety Evaluation Report.")

Mark I Containment System



implement anticipated plant modifications and minimize the potential for extended plant outages or unscheduled outages.

The objective of the NRC staff's efforts under Generic Task A-8 related to the Mark II Short-Term Program (STP) was to review and evalute the pool dynamic loads associated with a postulated large loss-of-coolant accident proposed by the Mark II Owner's Group to determine their acceptability for use in plant unique analyses. The Mark II Short-Term Program was completed in October 1978 and documented in NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance



MARK II

Criteria." With regard to the Mark II Long-Term Program (LTP), the NRC staff will evaluate the results of the Mark II confirmatory experimental and analytical programs to assess the margin for selected loads. The Mark II Long-Term Program is currently scheduled for completion in October 1980.

Under Generic Task A-39, the NRC staff will review and evalute the results of the Mark I and Mark II Owners' Group's experimental and analytical programs to establish and justify the safety relief valve-related pool dynamic loads for BWR Mark I and Mark II containment designs. The results of Generic Task A-39 will be an integral part of the final acceptability of the Mark I and Mark II pressure suppression containment designs. This generic task is currently scheduled for completion in December 1979. An interim assessment of multiple-consecutive SRV discharges was performed for the operating Mark I facilities to support deferral of the resolution of this issue until the completion of the Mark I Long-Term Program. This review was completed in December 1978 and deferral was found to be acceptable. A safety evaluation describing the NRC staff's interim assessment will be issued in early 1979.

Anticipated Transients Without Scram

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated-transient-without-scram," or ATWS, would have occurred.

This issue has been discussed throughout the nuclear industry for a number of years. Historically, the regulatory staff has excluded very low probability events from the design basis. At issue in the ATWS discussions is whether or not the probability of an ATWS event is sufficiently low to warrant the continuance of the current staff practice with regard to ATWS, i.e., continued exclusion from the design basis for nuclear power plants because of its low probability.

Because of the perceived potential for serious consequences resulting from ATWS events, a number of studies have been undertaken to assess the probabilities and consequences of such events. These studies have been performed by vendors, utility groups, and by the AEC and NRC regulatory staff. The ATWS issue was incorporated in the NRC Program for Resolution of Generic Issues (described in NUREG-0410, as noted above) as Generic Task A-9.

In September 1973, the then-AEC staff published WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water Cooled Power Reactors," which set forth staff "acceptance criteria" to protect against ATWS events. During the two-year period following publication of the staff report, each of the four reactor manufacturers submitted analyses and supporting information on ATWS which was reviewed by the NRC staff and addressed in four status reports published in December 1975. The staff reports evaluated the information for conformance to the WASH-1270 criteria and noted where design changes and additional analyses were required.

The vendors and owners have questioned whether the NRC staff's requirements are necessary and justified. The industry contends that the probability of an ATWS event is significantly less than estimated by the NRC staff and so low as to make ATWS events minor safety concerns in light water reactor operations.

Because of the continuing controversy over the NRC staff position since its publication in WASH-1270, a staff review and evaluation of all the information available on the subject of ATWS, and in particular, the material developed subsequent to the publication of the staff status reports referred to above, was undertaken in the latter part of 1977 and early 1978. A report, NUREG-0460, was published in April 1978 providing the results of this review and evaluation.

It was concluded in NUREG-0460 that considering the expected frequency of transients, the reliability of current reactor scram systems necessary to meet the safety objectives has not been demonstrated and may well have not been attained. NUREG-0460 recommended that means of mitigating the consequences of ATWS events be provided in plant designs.

The recommendations presented in NUREG-0460 have been criticized by industry and some members of the NRC staff as unnecessarily conservative and therefore too costly. The staff is now evaluating alternative means of reducing the probability or consequences of ATWS events, other than that recommended in NUREG-0460. The effectiveness, cost and other factors, such as the effect on the licensing process of these alternatives, is being evaluated. Based on this evaluation, the staff will recommend to the Commission the alternatives which provide the best balance between safety and cost for new designs, plants under construction and operating plants. The staff expects to provide its recommendations to the Commission in early 1979.

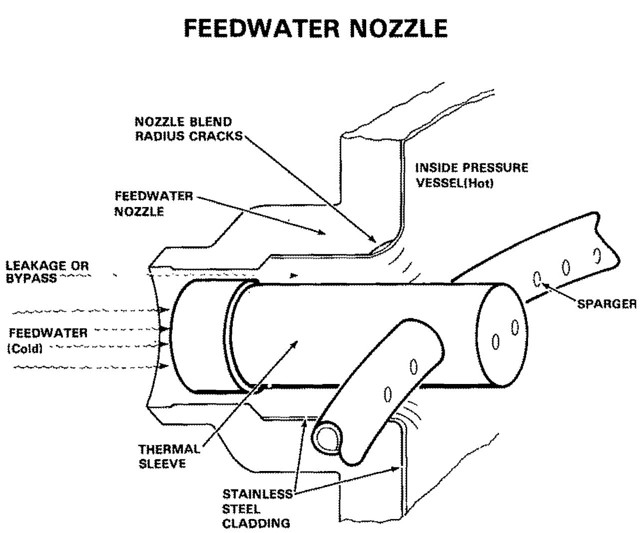
BWR Nozzle Cracking

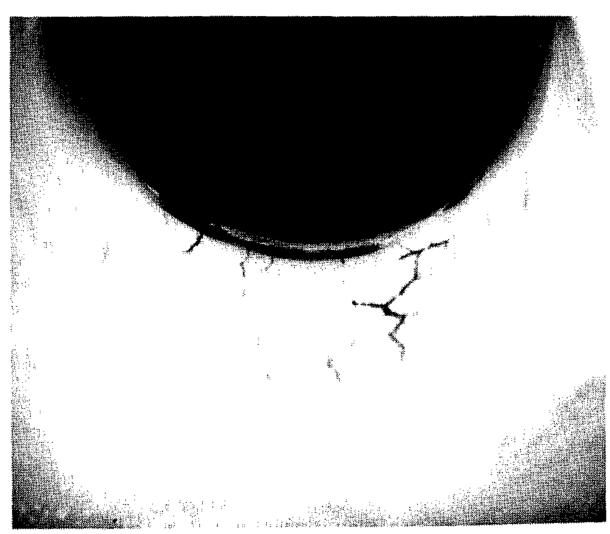
Over the last several years, inspections at 21 of the 23 boiling water reactor (BWR) plants licensed for operation in the U.S. have disclosed some degree of cracking in the feedwater nozzles of the reactor vessel at all but three facilities. Two facilities have not yet accumulated significant operating time and have not yet been inspected, although all BWR plants will eventually be inspected for this problem.

The feedwater nozzles, part of the "pressure vessel," are an integral part of the primary pressure boundary of the reactor coolant system and the second barrier (after the fuel cladding) to the release of radioactive fission products. All of the repaired BWR feedwater nozzles met the ASME pressure vessel code limits, however, and no immediate action was necessary. Because only relatively small amounts of base metal have been removed by repair operations, there has been no significant reduction in safety margins. Several plants have removed the stainless steel nozzle cladding as a means of eliminating crack initiation, since the clad thickness was not necessary to meet code reinforcement requirements. Nevertheless, the cracking is potentially serious because:

- Excessive crack growth could lead to impairment of pressure vessel safety margins requiring more complicated repair work than simple grinding.
- The design safety margin could be reduced by excessive removal of base metal.
- The exposure to radiation of the personnel performing inspection and repair tasks can be considerable.
- The repair of these kinds of cracks can result in considerable shutdown time at the plant affected.

The reactor vendor (the General Electric Company) and the NRC have concluded from their respective studies that the cracking is caused by fluctuations or "cycling" of the temperature on the inside surface of the nozzles; that the





Cracks in the nozzle blend area of a reactor pressure vessel feedwater nozzle are illustrated above. The area affected is shown in the drawing at left, and actual cracks are shown in the photograph at right (taken from inside the pressure vessel looking out through the nozzle). The inside diameter of the nozzle is approximately 10 inches.

stainless steel cladding exhibited less resistance to crack initiation than the underlying low-alloy steel; and that, after initiation in the stainless steel cladding, cracks can be propagated by operational startup and shutdown cycles or other operationally-induced transients. The vendor has performed extensive analysis and testing to confirm the suspected cause of the cracking and to uncover possible long-term solutions — a newly designed sleeve, removal of the stainless steel cladding, reduction of the temperature differential at the nozzle, or some combination of these. The licensees involved have increased the number and extent of inspections of feedwater nozzles, with careful repair and reinspection where cracks were found. The vendor advised these licensees to closely monitor startup and shutdown procedures in an effort to substantially reduce the time during which cold feedwater is being injected into the hot pressure vessel.

In a closely related area, the NRC was informed in March 1977 by the General Electric Company that a crack had been found in the nozzle of the "control rod drive (CRD) return line" in a reactor vessel in a foreign country. The CRD return line nozzles are the openings in BWR pressure vessels through which the high pressure water in excess of that needed to operate and cool the CRDs is returned to the pressure vessel. Later in March, the Philadelphia Electric Company reported that similar cracking had been found in the CRD return line nozzle at its Peach Bottom Atomic Power Station, Unit 3. The cracks resembled those found in the feedwater nozzles and seemed to be the result of the same kind of cyclic thermal stresses that were causing feedwater nozzle cracks. Both the foreign reactor and the Peach Bottom Unit 3 reactor are representative of a small number of BWRs which do not have a thermal sleeve in the CRD return line nozzle.

The licensee removed the cracks in the Peach Bottom CRD nozzle by grinding out the cracked area, the maximum crack depth being 7/8-inch, and returned the unit to operation with the CRD return line "valved out" and with the flow and pressure in the CRD hydraulic system modified.

Inspection of other CRD return line nozzles which incorporated thermal sleeves indicated that these sleeves may not be effective in preventing this cracking phenomenon. The Georgia Power Company found a crack in the CRD return line nozzle at its Hatch Plant, Unit 1, which did have a thermal sleeve. (The crack was removed, the nozzle capped, and the return line rerouted to the reactor water cleanup system.)

The NRC staff efforts related to the resolution of these two similar issues regarding nozzle cracking in boiling water reactors were consolidated into a single staff effort, Generic Task A-10, in 1977. Under Generic Task A-10, the staff issued interim guidance to operating plants in a report entitled, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," in July 1977. The staff is often requiring in-service inspection using liquid penetrant examinations at operating reactors in accordance with the frequency, procedures and acceptance criteria described in the above report.

Additional efforts under Generic Task A-10 include following and reviewing advancements in (1) the development and testing of effective feedwater nozzle thermal sleeves and spargers, (2) life-cycle testing of certain CRD system valves, (3) the development of various feedwater system and CRD system modifications, and (4) the development of viable ultrasonic system techniques by the nuclear industry to allow reliable and consistent early determination of cracking from positions exterior to the reactor vessel.

Generic Task A-10 is scheduled for completion in late 1979.

Reactor Vessel Materials Toughness

Resistance to brittle fracture, a rapidly propagating catastrophic failure mode for a component containing flaws, is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessel, three considerations are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to that combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material is accounted for in developing and revising these Technical Specification limitations over the life of the plant.

For the service times and operating conditions typical of current operating plants, reactor vessel fracture toughness provides adequate margins of safety against vessel failure. Further, for most plants the vessel material properties are such that adequate fracture toughness can be maintained over the life of the plants. However, results from a reactor vessel surveillance program indicate that up to 20 older operating pressurized water reactors were fabricated with materials that will have marginal toughness after comparatively short periods of operation.

The objective of Task A-11 is to evaluate material degradation mechanisms resulting from neutron irradiation and determine appropriate licensing criteria and corrective action for low toughness reactor vessel materials in these currently licensed plants. Task A-11 is currently scheduled for completion in July 1979. This completion date is well in advance of the date needed to assure that adequate fracture toughness is maintained in these older plants.

Fracture Toughness and Potential For Lamellar Tearing of PWR Steam Generator and Reactor Coolant Pump Supports

During the course of licensing review for a specific Pressurized Water Reactor (PWR) a number of questions were raised as to (1) the adequacy of the fracture toughness properties of the material used to fabricate the reactor coolant pump supports and steam generator supports, and (2) the potential for failure due to lamellar tearing of these same supports. The safety concern is that, although these supports are designed for worst-case accident conditions, poor fracture toughness or lamellar tearing could cause the supports to fail during such accidents. Support failure could conceivably impair the effectiveness of systems designed to mitigate the consequences of the accident. (An example of a postulated event sequence of potential concern would be a large pipe break in the reactor coolant system which severely loads the supports, followed by a support failure of sufficient magnitude that a major component such as a steam generator is severely displaced resulting in failure of the emergency core cooling system piping which is needed to provide cooling water to the core.)

Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for the supports of the PWR in question. To address the fracture toughness question (lamellar tearing is discussed separately below), tests not originally specified and not in the relevant ASTM specifications were made on those heats of steel for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of the PWR in question, the applicant agreed to a license condition which stated that he would raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F—prior to pressurizing the reactor coolant system above 1,000 psig—thereby assuring adequate toughness in the event of an accident. Auxiliary electrical heat will be used to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Because similar materials and designs have been used in other plants and therefore similar problems may exist, review of this issue was included in the NRC Program for Resolution of Generic Issues as Generic Task A-12.

A consultant was engaged to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in the later stages of operating license review. The staff thereafter completed a review of the materials utilized in the supports of 34 potentially affected PWRs. Based on the consultant's preliminary evaluation, it was determined that there are approximately 15-20 plants whose supports are of questionable toughness. We expect that these plants may be required to utilize in-service inspection or auxiliary heating if adequate toughness properties cannot be demonstrated.

Upon completion of the generic study, the generic phase of the fracture toughness program will be documented and the results implemented on a plant-specific basis. Lessons derived from the generic solution will be incorporated into the Standard Review Plan for use in future license reviews.

The staff has concluded that continued operation (and licensing) of PWRs is justified pending completion of this task and implementation of the task results because support failure is not expected to occur except under the unlikely combination of:

(1) The occurrence of an initiating event (e.g., a large pipe break) which has been determined to be of low probability (normal operating stresses on piping are very low)

(2) The existence of non-redundant and critical support structural member(s) with low fracture toughness (many supports contain redundant members).

(3) The existence of support structural members at operating temperatures low enough that the fracture toughness of the support material is reduced to a level at which brittle failure could occur if a large flaw existed.

(4) The existence of a flaw of such size that the stresses imparted during the initiating event could cause the flaw to rapidly propagate, resulting in brittle failure of the member(s).

The second potential concern, lamellar tearing*, may also be a problem in those support structures which are similar in design to those of the aforementioned PWR. However, continued operation of PWRs during the continuing generic review of this concern was judged acceptable, based on a review of approximately 400 relevant technical documents which revealed only one instance of known failure from lamellar tearing. This failure occurred in often-stressed truck brakes. In addition, the factors considered above for the fracture toughness concern—such as low stresses during normal operation and the low probability of an initiating event—apply equally to this concern.

The generic fracture toughness program is expected to be completed in August 1979. The lamellar tearing evaluation is a longer term effort and is expected to be completed in 1981.

*Lamellar tearing is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The tearing always lies within the parent plate, often outside the transformed (visible) heat-affected zone (HAZ) and is generally parallel to the weld fusion boundary. Lamellar tearing occurs at certain critical joints usually within large welded structures involving a high degree of stiffness and restraint. Restraint may be defined as a restriction of the movement of the various joint components that would normally occur as a result of expansion and contraction of weld metal and adjacent regions during welding ("Lamellar Tearing in Welded Steel Fabrication," The Welding Institute).

Systems Interactions In Nuclear Power Plants

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multi-disciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialties—such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated. Simply stated, the left hand may not know or understand what the right hand is doing in all cases where it is necessary for the hands to be coordinated.

The NRC staff believes that its current review procedures and safety criteria provide reasonable assurance that an acceptable level of redundancy and independence is provided for systems that are required for safety. Nonetheless, in mid-1977, this task (Task A-17) was initiated to confirm that present procedures adequately take into account the potential for undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organizational units and assign secondary responsibility to other units where there is a functional or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 will provide an independent investigation of safety functions—and systems required to perform these functions—in order to assess the adequacy of current review procedures. This investigation will be conducted by Sandia Laboratories under contract assistance to the NRC staff.

The contract éffort, Phase I of the task, began in May 1978 and is expected to be completed in September 1979. The Phase I investigation is structured to identify areas where interactions are possible between and among systems and have the potential of negating or seriously degrading the performance of safety functions. The investigation will then identify where NRC review procedures may not have properly accounted for these interactions. Finally, in a follow-on Phase II of the task, specific corrective measures will be taken in areas where the investigation shows a need.

As noted above, the NRC staff believes that its review procedures and acceptance criteria currently provide reasonable assurance that an acceptable level of system redundancy and independence is provided in plant designs and this task is expected to confirm this belief. Nonetheless, because adverse systems interactions are potentially of large significance to plant safety, this issue has been identified as an "Unresolved Safety Issue." If no significant system interactions are identified in the Phase I investigation described above, as is expected, this issue will not be treated in subsequent reports as an "Unresolved Safety Issue."

Environmental Qualification of Safety-Related Electrical Equipment

In addition to the conservative design, construction and operating practices and quality assurance measures required for nuclear power plants, safety systems are installed at nuclear plants to mitigate the consequences of postulated accidents. Certain of these postulated accidents could create severe environmental conditions inside the containment. The most serious of these accidents would be a high energy pipe break in the reactor coolant system piping or in a main steam line. In either case, the release of hot pressurized water and steam to the containment would create a high temperature environment (250 to 400°F) at high humidity (including steam) and pressure (as high as c. 50 psig). For some applications, chemicals are added for fission product removal to the containment sprays that are used to reduce the pressure in the containment. Additionally, some electrical equipment is predicted to be submerged following a large pipe break. Thus, the safety equipment is exposed to such environmental conditions and needs to remain operable during this period, as well as for the long-term post-accident period.

In order to assure that electrical equipment in safety systems will perform its function under accident conditions, the NRC requires that such equipment—principally equipment associated with the emergency core cooling system and containment isolation and cleanup systems—be "environmentally qualified." Specific electrical equipment of concern during postulated accident conditions includes: (1) the instrumentation needed to initiate the safety systems and provide diagnostic information to the plant operators (e.g., electrical penetrations into containment, any electrical connectors to cabling which transmits signals, and the instruments themselves), (2) control power to motor operators for certain valves (e.g., ECCS and containment isolation valves located inside containment), and (3) fan cooler motors for those plants that utilize fan coolers for containment heat removal.

The current NRC safety review process for nuclear power plants applies certain criteria for confirming the capability of electrical equipment important to safety to function in the environment that might result from various accident conditions. Although such criteria have been applied to varying degrees since the early days of commercial nuclear power, they have come to be defined in clearer detail over the years.

The process of clarifying the criteria has given rise to certain questions regarding: (1) the degree to which electrical equipment used in older plant designs (those now operating) is capable of withstanding the environmental conditions (pressure, temperature, humidity, steam, chemicals, vibration, and radiation) of various accident conditions under which it must be able to function (i.e., the "qualification of equipment" in these older plants), and (2) the adequacy of test or analyses conducted for electrical equipment in newer plants to "qualify" such equipment as capable of withstanding the conditions of the environment created by various accidents during which the equipment must function (i.e., the "adequacy" of qualification tests).

With regard to older plants, the following actions have taken place in recent months.

As a result of a Sandia testing program being conducted for the NRC Office of Nuclear Regulatory Research, a generic safety concern with the adequacy of environmental qualification of certain electrical equipment was identified. This issue was highlighted by a November 4, 1977 petition from the Union of Concerned Scientists which requested immediate action by the NRC regarding operating power reactors and licensing actions for other proposed plants. (See "Abnormal Occurrences—1978," in Chapter 7 for extended discussion of specific actions following the Sandia tests.) Subsequent NRC staff investigations in response to this issue led, as of June 30, 1978, to seven plant shutdowns for corrective action and extended outages for two other plants to make modifications. These actions were taken for the most part as a result of a lack of conclusive information regarding the qualification of certain safety equipment.

Having identified the problems associated with qualification of electrical equipment, the NRC conveyed that information to the licensees of all operating reactor facilities through an Inspection and Enforcement Circular which was issued on May 31, 1978. The purpose of this Circular was to ensure that the knowledge gained by the NRC staff would be appropriately factored into future actions by licensees. The NRC staff also initiated an augmented inspection effort, to become part of the normal inspection activities, which will concentrate on the inspection of installed safety-related electrical equipment and on an audit of the records for environmental qualification.

In addition, a review of the environmental qualification of safety-related electrical equipment has been initiated for 11 operating reactor facilities in the Systematic Evaluation Program (SEP).

With regard to the second question above—the adequacy of qualification tests for newer plants—the NRC staff has worked with the industry to develop standards for equipment qualification and documentation which will assure the high level of equipment reliability required for nuclear applications. This effort has culminated in the development of IEEE Standard 323, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." This standard and its ancillary standards have provided the focal point for the development of environmental qualification requirements in recent years.

IEEE Standard 323 was first issued as a trial use standard (IEEE Std. 323-1971) in 1971 and later, after substantial revision, as a final standard (IEEE Std. 323-1974) in 1974. Both versions of the standard set forth basic requirements for environmental qualification of electrical equipment but do not provide details for implementation of these requirements. Specific qualification techniques have been reviewed and approved by the NRC staff on a case-by-case basis as a part of individual licensing actions. These licensing actions include initial construction permit and operating license application reviews and requalification actions for operating reactors, where documentation of the initial qualification was not available.

The evolutionary nature of the process of developing environmental qualification requirements and the case-by-case implementation of them has resulted in a diversity of methods in use and different levels of documentation of the extent to which equipment is qualified.

Several aspects of equipment qualification are being pursued at this time by the NRC staff and the nuclear industry on a generic basis, in order to achieve a more uniform implementation of requirements established in IEEE Standard 323-1974. One such activity is the development of interim NRC staff positions regarding how the requirements of IEEE Standard 323-1974 can be met. This activity is a part of Generic Task A-24, "Environmental Qualification of Safety-Related Electrical Equipment," in the NRC Program for the Resolution of Generic Issues and is scheduled for completion in 1979.

Further efforts under Generic Task A-24 involve the review of the environmental qualification programs of reactor vendors and architect/engineers as a basis for qualifying safety-related electrical equipment, pursuant to the requirements of IEEE-Standard 323-1974. Performing these reviews on a generic basis rather than on case-by-case licensing reviews will save time and resources for the NRC staff and the industry. This follow-on portion of the generic task will be scheduled following completion of the development of the interim NRC staff positions referred to above.

Reactor Vessel Pressure Transient

Over the past several years, incidents known as "pressure transients" have taken place at various PWR facilities. A pressure transient oc curs when the pressure-temperature limits included in the technical specifications for the facility have been exceeded. As of the close of the report period, there had been a total of 33 such events. Half of them occurred before the plant achieved initial criticality (i.e., before initial operation of the reactor); the majority occurred during startup or shutdown operations. In all of these incidents fracture mechanics and fatigue calculations indicated that the reactor vessels were not damaged and continued operation of the vessels was acceptable. Nevertheless, the staff concluded that appropriate regulatory actions were necessary (1) to reduce the frequency of pressure transient events, and (2) to provide equipment which would restrict future transients to acceptable pressures. This action was necessary because reactor vessel safety margins would be reduced over the lifetime of the vessel by neutron irradiation, which reduces material toughness.

The NRC staff's review of this safety issue was incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-26. Task A-26 was completed in September 1978 with the issuance of the final report, NUREG-0224, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors."

Upgraded procedural controls were implemented at operating PWR facilities which significantly reduced the occurrence of pressure transient events. The few events which have occurred were not significant and were of the type that will be precluded by equipment changes.

Most of the equipment changes carried out at operating PWR facilities involve the addition of a second lower set point on existing power operated relief valves, the addition of new spring-loaded relief valves, or modifications to allow use of existing spring-loaded relief valves. A few newly licensed facilities must complete similar design changes by their first refueling shutdown. The extended equipment implementation schedule for new facilities was based upon the reduced frequency of occurrence of pressure transient events, a result of improved procedural controls and the large safety margins for new pressure vessels.

Residual Heat Removal Shutdown Requirements

The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident (LOCA) has been typically interpreted as achieving a "hot-standby" condition (i.e., the reactor is shutdown, but system temperature and pressure are still at or near normal operating values). Considerable emphasis has been placed on the hot-standby condition of a power plant in the event of an accident or abnormal occurrence. A similar emphasis has been placed on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby condition values.

Even though it may generally be considered safe to maintain a reactor in a hot-standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. For this reason the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for both PWRs and BWRs. It is essential that a power plant be able to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under any accident conditions.

This issue was designated as Task A-31, "RHR Shutdown Requirements," in 1977, and included in the NUREG-0410 Report to Congress. In accordance with the Task Action Plan for this task, the staff's views on requirements for residual heat removal systems were translated into proposed changes to Standard Review Plan Section 5.4.7. These proposals were considered by the Regulatory Requirements Review Committee (RRRC) during its 71st meeting on January 31, 1978.

The RRRC recommended approval of the proposed changes and further recommended that (1) the changes be applied on a case-by-case basis to all operating reactors and all other plants (custom or standard) for which the issuance of the operating license is expected before January 1, 1979, and (2) the changes be backfitted to all plants (custom or standard) for which construction permit or preliminary design approval applications were docketed before January 1, 1978, and for which the operating license issuance is expected after January 1, 1979. These recommendations were approved by the Director of NRR and are being implemented. Accordingly, Task A-31 has been completed.

Subsequently, the staff positions on design requirements for residual heat removal systems were incorporated into Regulatory Guide 1.139, "Guidance for Residual Heat Removal", which was issued for public comment in May 1978. Comments were received during the latter part of 1978 and it is expected that this Regulatory Guide can be issued in its final form in late 1979 or early 1980.

Control of Loads Near Spent Fuel

Overhead cranes are used to lift heavy objects, sometimes in the vicinity of spent fuel, in both PWRs and BWRs. If a heavy object, such as a spent fuel shipping cask or shielding block, were to fall or tip onto spent fuel in the storage pool or in the reactor core during refueling and damage the fuel, there could be a release of radioactivity to the environment and a potential for radiation over-exposures to in-plant personnel. If the dropped object is large, and is assumed to drop on fuel containing a large amount of fission products with minimal decay time, calculated offsite doses could exceed the siting guideline values in 10 CFR Part 100.

The NRC staff's review of this safety issue has been incorporated in the NRC Program for Resolution of Generic Issues as Generic Task A-36. The objective of the task is to develop a revision to the Standard Review Plan (SRP) based on a reevaluation of current NRC requirements and procedures currently utilized at operating plants. If necessary, the revision will provide criteria to further reduce the potential for heavy loads causing unacceptable damage to spent fuel in a storage pool or in the reactor core during refueling. The revised SRP will provide the basis for implementing additional requirements and procedures in existing plants where warranted and can be used in future reviews of new plants.

It is the NRC staff's view that continued operation during review of this generic issue presents no undue risk to the health and safety of the public. Operating facilities use a variety of design and administrative measures to minimize the potential for dropping a heavy object over the reactor core or over the spent fuel pool. These design and administrative measures have been effective since no heavy load handling accidents resulting in damaged fuel have occurred in over 300 reactor years of U.S. operating experience. For facilities that have requested increases in spent fuel pool storage capacity, the NRC has prohibited the movement of loads over fuel assemblies in the spent fuel pool that weigh more than the equivalent weight of one fuel assembly. And for those plants where the review of the cask drop or the crane handling system is not complete, movement of shielded casks over or near spent fuel has been prohibited.

Concurrent with the NRC review, licensees have examined their current procedures for the movement of heavy loads over spent fuel to assure that the potential for a handling accident that could result in damage of spent fuel is minimized while the generic evaluation proceeds. Most of the licensees' submittals of their reviews have been received and were under review at the end of 1978.

Generic Task A-36 is expected to be completed in early 1979. The Task will result in the development of generic criteria, but implementation of these criteria will be dependent on plant design characteristics and the specific procedures in effect at each particular plant.

Seismic Design Criteria

NRC regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants is provided in the NRC regulations and in Regulatory Guides. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, re-reviews of the seismic design of various plants are being undertaken (principally as part of the Commission's Systematic Evaluation Program) to assure that these plants do not present an undue risk to the public.

The NRC staff is conducting Generic Task A-40, as part of the NRC Program for Resolution of Generic Issues. Task A-40 is, in effect, a compendium of short-term efforts to support the reevaluation of the seismic design of operating reactors. The objective of the task is, in part, to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, and to quantify the overall conservatism of the design sequence. In this manner the program will aid the NRC staff in performing its reviews of the seismic design of operating reactors.

Generic Task A-40 is separated into ten separate subtasks. The subtasks are described in the Task Action Plan for Task A-4, which is included in NUREG-0371. Most of the subtasks are scheduled for completion in September 1979. However, three of the subtasks—related to developing state-of-the-art methodology in order to better define earthquake ground motion near earthquake sources—are longer term efforts. These three subtasks are scheduled for completion in 1981.

Pipe Cracks At Boiling Water Reactors

Pipe cracking has occurred in the heat-affected zones of welds in primary system piping in boiling water reactors (BWRs) since the mid-1960's. These cracks have occurred mainly in Type 304 stainless steel, which is the type used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized," either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were very early (late 1960's) found to be susceptible to IGSCC. Because of this, the Atomic Energy Commission took the position in 1969 that furnace sensitized safe ends should not be used on new applications. Most of the furnace-sensitized safe ends in older plants have been removed or clad with a protective material, and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop-bypass lines and in 10-inch diameter core spray lines. More recently cracks were discovered in recirculation riser piping (12-inch to 14-inch) in foreign plants. Cracking is most often detected during Inservice Inspection using ultrasonic testing techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

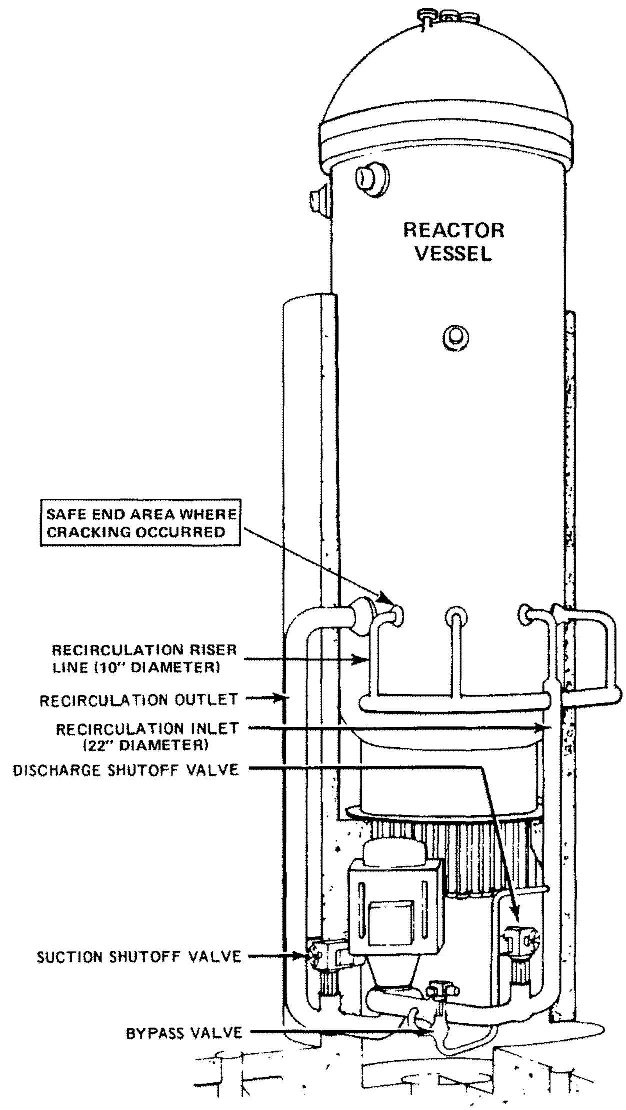
In response to these occurrences of BWR primary system cracking, the NRC has taken a number of measures. These actions included:

- Issuance of Regulatory Guide 1.44 on "Control of the Use of Sensitized Stainless Steel."
- Issuance of Regulatory Guide 1.45 on "Reactor Coolant Boundary Leak Detection Systems."
- Closely following the incidence of cracking in BWRs, including foreign experience.
- Encouraging replacement of furnace-sensitized safe ends.
- Requiring augmented in-service inspection (additional more frequent ultrasonic examination) of "service sensitive" lines, i.e., those that have experienced cracking.
- Requiring upgrading of leak detection systems.

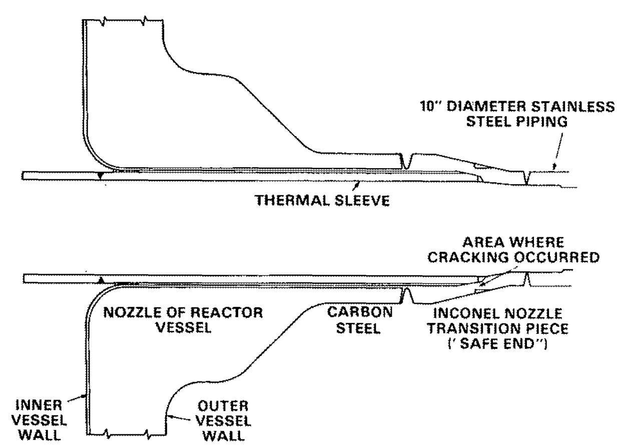
Pipe cracking and furnace sensitized safe end cracking has been recently reported in larger (24-inch diameter) lines in a BWR (designed by the General Electric Company) in Germany with over 10 years of service. Because the safe ends on that facility had been furnace-sensitized during fabrication, IGSCC was suspected. One of the safe ends was removed for destructive examination. During laboratory examination of the removed safe end, and also a small section of attached pipe, cracks were discovered at various locations in the safe end and in the weld heat affected zone of the pipe. The cracks in the pipe weld area were very shallow, with the maximum depth less than about 5 mm (about 1/8-inch). Cracking in the furnace-sensitized safe end was somewhat deeper. The German experience was the first known occurrence of IGSCC in pipes as large as 24-inch in diameter.

In June 1978, a through-wall crack was discovered in an Inconel recirculation riser safe end (10-inch diameter) at the Duane Arnold facility (see discussion under "Abnormal Occurrences—1978," in Chapter 7). The crack has been attributed to IGSCC although the material in this instance is different from the Type 304 stainless steel that has been historically found to crack. Subsequent ultrasonic examination discovered indications in some of the other seven safe ends. Following their removal, cracking was discovered in all eight safe ends. The cracking appeared to have originated in a tight crevice between the inside wall of the safe end and an internal thermal sleeve. Such crevices are known to enhance IGSCC. Differences in materials, geometry, stress levels and crevices appear to make the problem at Duane Arnold unique to a particular type of recirculation riser safe end (Type I). As a result of this event, ultrasonic examination of the other Type I safe ends in U.S. BWRs (i.e., at Brunswick Units 1 and 2) was conducted. No significant indications were found in Unit 2, and one indication was identified at Unit 1. Although this indication is relatively minor and is not "reportable" pursuant to the NRC regulations, evaluation of it is continuing. The ultrasonic indication which was found was to be reevaluated at another plant shutdown scheduled for later in 1978, after the close of the report period.

In discussions with General Electric (the reactor vendor) regarding recent pipe cracking experience, the company was asked by the NRC to



These diagrams show area where cracking occurred in the fitting of the recirculation pipe of the Duane Arnold reactor vessel. Above is the reactor vessel, with the safe end area indicated. Below is a close-in view of the area.



provide an in-depth report on the significance of recent events regarding current inspection, repair, and replacement programs. They were also asked to address any new safety concerns related to the occurrence of cracking in large main recirculation piping. Based on information presented by the vendor and extensive staff evaluation, it was concluded that the recent occurrences did not constitute a basis for immediate concern about plant safety, nor require any new immediate actions by licensees.

The staff briefed the Commission on pipe cracking in BWRs on August 31, 1978, and on September 14, 1978, re-established an NRC Pipe Crack Study Group. The Study Group will specifically address the following issues:

- The significance of the cracks discovered in large diameter pipes relative to the conclusions and recommendations set forth in the referenced report and in its implementation document NUREG-0313.
- Resolution of concerns raised over the ability to use ultrasonic techniques to detect cracks in austenitic stainless steel.
- The significance of the cracks found in large diameter sensitized safe ends, and any recommendations regarding the current NRC program for dealing with this matter.
- The potential for stress corrosion cracking in PWRs.
- The significance of the safe end cracking at Duane Arnold relative to similar material and design aspects at other facilities.

The Study Group is scheduled to complete its evaluation and report in January 1979. In addition to the Study Group effort, the NRC has underway several generic technical review efforts which are aimed at improving piping inspection techniques and requirements. These generic efforts and any follow-on efforts resulting from the Study Group's evaluation will be incorporated into a new generic task, Task A-42, "Pipe Cracks at Boiling Water Reactors."

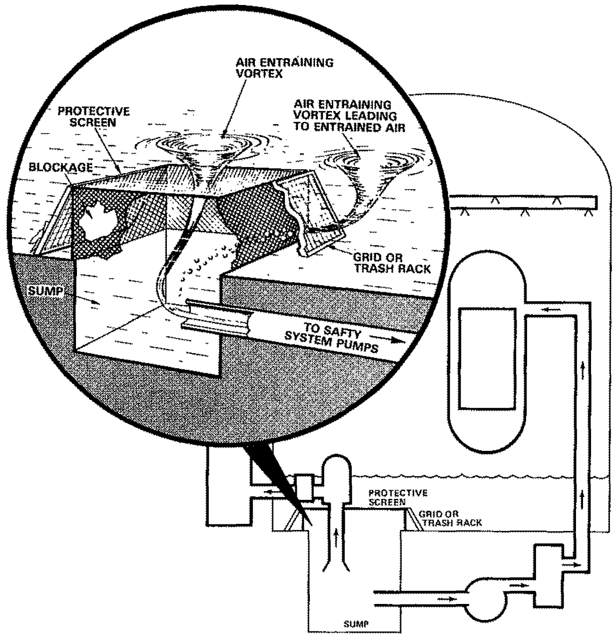
Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the emergency sump at the low point in the containment. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the containment. Loss of the ability to draw water from the emergency sump could disable the emergency core cooling and containment spray systems. The consequences of the resulting inability to cool the reactor core or the containment atmosphere could be melting of the core and/or breaking of the containment.

One postulated means of losing the ability to draw water from the emergency sump would be blockage by debris. A principal source of such debris could be the thermal insulation on the reactor coolant system piping. In the event of a piping break, the subsequent violent release of the high pressure water in the reactor coolant system could rip off the insulation in the area of the break. This debris could then be swept into the sump, potentially causing damage.

Currently, regulatory positions regarding sump design are presented in Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," which addresses the question of debris (insulation). The regulatory guide recommends that, in addition to providing redundant separated sumps, two protective screens be installed. A low approach velocity in the vicinity of the sump is required to allow insulation to settle out before reaching the sump screening; it is also required that the sump remain functional assuming that one-half of the screen surface area is blocked. The NRC staff believes that sump designs in accordance with this regulatory guide acceptably resolve this issue. Nonetheless, the NRC staff is continuing to study the behavior of insulation under pipe break conditions to gain a better understanding of how it might behave.

A second postulated means of losing the ability to draw water from the emergency sump would be abnormal conditions in the sump or at the pump inlet—such phenomena as air entrainment, vortices, or excessive pressure drops. These conditions could result in pump cavitation, reduced flow and possible damage to the pumps.



Currently, regulatory positions regarding sump testing are contained in Regulatory Guide 1.79, "Pre-Operational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," which addresses the testing of the recirculation function. Both in-plant and scale model tests have been performed to demonstrate that circulation through the sump can be reliably accomplished. The NRC staff believes that sumps tested in accordance with this regulatory guide acceptably resolve this issue. As supplemental guidance, the staff, through a contractor, is studying whether further guidance for the design and review of emergency sumps to assure adequate hydraulic design can be developed.

The NRC staff initially planned to study the issue of containment emergency sump blockage from insulation as part of Generic Task C-3, "Insulation Usage Within Containment." In addition, initial plans were to study the vortex formation issue as part of Generic Task B-18, "Vortex Suppression Requirements for Containments." However, containment emergency sump operability is fundamental to the successful operation of both the emergency core cooling system (needed to cool the core) and the containment spray system (needed to assure containment integrity), following a loss-of-coolant accident. For this reason, these portions of Tasks C-3 and B-18 have been combined and elevated to the highest priority (category A) as Generic Task A-43, under the more general title of "Containment Emergency Sump Reliability." Because this action has only recently been taken, a Task Action Plan and schedule for this task have not yet been developed.

Station Blackout

Electrical power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current (a. c.) power connection, a standby emergency diesel generator a. c. power supply, and direct current (d. c.) sources.

The issue of station blackout was originally included as Generic Task B-57 in the NRC Program for Resolution of Generic Issues. The task involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all a. c. power, i.e., a loss of offsite a. c. sources and both onsite emergency diesel generator sources. Loss of all a. c. for an extended period of time in pressurized water reactors, accompanied by loss of the auxiliary feedwater pumps (usually one of two redundant pumps is a steam turbine driven pump that is not dependent on a. c. power for actuation or operation), could result in an inability to cool the reactor core, with potentially serious consequences. If the auxiliary feedwater pumps are dependent on a. c. power to function, then a loss of all a. c. power for an extended period could of itself result in an inability to cool the reactor core. Although this is a low probability event sequence, it could be a significant contributor to risk.

Current NRC safety requirements require as a minimum that diverse power drives be provided for the redundant auxiliary feedwater pumps. As noted above, this is normally accomplished by utilizing an a. c. powered electric motor driven pump and a redundant steam turbine driven pump. One concern is the design adequacy of plants licensed prior to adoption of the current requirements.

An initial survey of operating plants has been completed which indicates that all operating pressurized water reactors have either steam turbine driven or diesel driven auxiliary feedwater pumps (neither of which is dependent on a.c. power). This assures at least that some capability exists for accommodating an extended loss of all a. c. power. Further review of older plants in this regard will be conducted as part of the NRC's Systematic Evaluation Program (see earlier discussion in this chapter). Further study will include determining if any requirements beyond providing diverse power drives for the auxiliary feedwater pumps are needed—such as specific time requirements for the period during which the plant must be capable of accommodating a station blackout.

This safety issue was previously included in the NRC Program for the Resolution of Generic Issues as Generic Task B-57, but has recently been elevated to category A as Generic Task A-44. Because this action has only recently been taken, a Task Action Plan and schedule for this task have not yet been developed. A Task Action Plan will be developed by March 1, 1979.

OTHER TECHNICAL ISSUES

Burnable Poison Rod Assembly Failures

On February 17, 1978, while the Crystal River Unit 3 (Citrus County, Florida) was operating during its first cycle, a Loose Parts Monitoring System alarm occurred from the "B" steam generator. To minimize potential damage from any loose part in the primary coolant system, the licensee reduced reactor power and shut off one of the two reactor coolant pumps in the affected loop.

On March 3, 1978, the reactor was shut down for inspection of the "B" steam generator. On March 6, 1978, several parts of a Burnable Poison Rod Assembly were found in the steam generator. Burnable poison rods are similar in size and shape to fuel rods, but the burnable poison rods contain neutron absorbing materials that reduce the excess reactivity of a fresh core in such a way that the absorbing capacity (reactivity worth) diminishes with burnup. In the Babcock & Wilcox design, burnable poison rods are mounted in detachable fixtures (BPRAs) that are normally removed from the fuel assemblies at the end of the first cycle of operation.

Inspection revealed damage to the steam generator tubes and to welds on the tube sheet (a lattice to which the tubes are attached). There were indications of a small primary-system-to-secondary-system coolant leak. BPRA parts were found in the "B" steam generator, core support assembly, various fuel assemblies, and in the plenum and bottom of the reactor vessel. The reactor was completely defueled to facilitate further inspection.

The failure of the BPRA was attributed to a vibration-induced wear of the mechanism that couples the BPRA to the fuel assemblies. The BPRAs for Crystal River Unit 3 were all removed from the core and were not replaced, since they were no longer needed to help control the reactivity of the core associated with the early stages of core life.

Similar BPRAs were also in use at Davis-Besse Nuclear Power Station, Unit 1, and at Three Mile Island Nuclear Station, Unit 2. The BPRAs were removed from Davis-Besse Unit No. 1 and will not be replaced. However, the BPRAs were still needed to help control core reactivity at Three Mile Island Unit 2, since the plant is operating in the early part of its first fuel cycle; therefore, retaining collars were added to the BPRAs to keep them in place.

During an inspection at Davis-Besse Unit 1, some wear was also discovered on various orifice rod assemblies (flow regulating devices) which had coupling mechanisms similar to the BPRAs. These were also removed at Davis-Besse Unit 1, as well as at Crystal River Unit 3, Three Mile Island Unit 2, and Oconee Nuclear Station, Unit 3. Further evaluations by the reactor vendor, Babcock and Wilcox, however, indicate that orifice rod assembly wear may not be significant and the orifice rod assemblies will not have to be removed until planned reactor shutdowns for refueling.

The actions being taken and proposed by the affected licensees and the reactor vendor are being closely monitored by the NRC. The corrective actions taken will be inspected and reviewed by the NRC.

The event at Crystal River Unit 3, and subsequent actions at the other affected licensees, did not result in a major reduction in the degree of protection of the public health and safety. The consequences of the event at Crystal River Unit 3 were minor steam generator damage and loss of electrical generating capacity for several months, while repairs are made. Some loss of electrical generating capacity was also associated with some of the other affected licensees.

Design Errors in Control Building

On April 13, 1978, Portland General Electric Company (PGE), operator of the Trojan Nuclear Plant, orally informed the NRC of potential design errors related to the "shear walls" of the control building at the facility. PGE investigated the matter and reported that design errors did, in fact, exist and that the control building walls did not conform to the design criteria set forth in the Final Safety Analysis Report for the facility.

A detailed NRC staff review of PGE's investigation and analysis of the design revealed the following errors:

(1) The steel reinforcement in the reinforced concrete core of the walls was permitted to be generally discontinuous and, therefore, the concrete core could not be relied upon to resist shear (in case of an earthquake) to the extent assumed in the approved design.

(2) The shear capacity of the reinforced concrete and grouted masonry block was computed incorrectly resulting in a lower level of conservatism than intended.

(3) The steel reinforcement needed to resist shear beyond the capacity of the concrete and grouted masonry block was computed incorrectly, resulting in a lower level of conservatism than intended.

As a result of these identified design errors the NRC concluded that the control building did not comply with the requirements of the Trojan license in that the shear walls do not have the intended margin to resist Trojan's Operating Basis Earthquake (OBE) nor the Safe Shutdown Earthquake (SSE).

As a result of the identification of the nonconformances, a detailed reevaluation of the control building in its existing configuration was performed by PGE to assess the present capability of the structure to withstand the Operating Basis Earthquake and the Safe Shutdown Earthquake. The NRC staff determined that there had been a reduction in conservatism and design margins, with respect to the control building seismic capability, below the level intended and desired for the 33 years remaining in the expected plant life. Because this reduction in margin was significant, the NRC staff concluded that the appropriate margins should be restored by modifications to the control building. PGE indicated its intent to make such modifications.

The NRC staff also determined that, based on data supplied by PGE, there was adequate assurance of safety until control building modifications could be implemented, since the Trojan Plant had the capability to withstand an SSE of the magnitude established for that facility and could be brought to a safe shutdown condition. In addition, the NRC staff determined that the facility could be operated in the interim without endangering the health and safety of the public, provided that no modifications to the control building were made that would in any way reduce the strength of the existing shear walls. Also, since the NRC staff had concluded on the best available information that the OBE capability for the control building had been reduced to 0.11g (0.15g was established for the facility), actions that would otherwise be required for a 0.15g earthquake would have to be taken in the event that a 0.11g peak ground acceleration earthquake were to occur at the plant site.

Having made these determinations, the Acting Director of the Office of Nuclear Reactor Regulation on May 26, 1978, issued an Order dealing with this matter. The Order, which offered an opportunity for hearing, was to be effective June 26, 1978, or on a date specified in an Order made following a hearing, if one were held in connection with the Order.

The May 26, 1978 Order called for:

- Design modifications to restore the seismic design margins originally intended to the control building with the control building brought into substantial compliance by June 1, 1979.
- An implementation schedule, to be reviewed and approved by the NRC, by July 1, 1978.
- Detailed design information by September 1, 1978, for NRC staff review and approval, together with supporting analyses and application for license amendments

as necessary to implement these modifications.

- Conditional license waiver of the areas of non-conformance noted above until the control building has been brought into substantial compliance in these areas. The conditions called out were that no modifications affecting the strength of the control building shear walls were to be made without NRC approval and the facility should be brought to cold shutdown in the event that an earthquake reaching 0.11g ground acceleration should occur at the site and that subsequent restart would require prior NRC approval. The Order noted that since the facility—shut down at the time—did not conform with existing license requirements, it could not be operated without violating the license.

Numerous requests for a hearing were received, and a hearing was ordered to begin September 6. However, on August 22, PGE advised the NRC of new information resulting from a new finite-element analysis which differed in several respects from information previously provided. Accordingly, the hearing on interim operation was postponed, and subsequently held October 23 to November 3, and December 11 to 14, at which time the new information was considered. Final resoluton of the matter, including the question of interim operation prior to completion of modifications, was to be decided by the Atomic Safety and Licensing Board. The Board's Initial Decision on the question of interim operation was scheduled to be issued about December 22, 1978.

Further hearings on the nature and timeliness of modifications to the control building were to take place later.

Control Rod Guide Tube Integrity

In December 1977, extensive wear and some holes were observed in the upper section of numerous control rod guide tubes at Northeast Nuclear Energy Company's Millstone Unit 2 facility. Subsequent inspections at other facilities with reactors designed by Combustion Engineering (CE) disclosed similar indications of guide-tube wear.

The guide tubes serve in a dual capacity as the principal structural members of the fuel assemblies and as guide channels for the movement of the control rods. The structural integrity of the guide tubes is required to assure that the control rods can be inserted to shut down the reactor, when that is required by activation of a reactor safety system.

The licensees for Millstone Unit 2, St. Lucie Unit 1, Calvert Cliffs Unit 1, Arkansas Unit 2, and Maine Yankee modified the fuel assemblies during refueling outages by installing stainless steel sleeves in both worn and unworn guide tubes. The sleeves stiffen and strengthen the worn tubes, minimize further wear, and assure safety in activities involving the assemblies with worn guide tubes. In new fuel assemblies, sleeving prevents guide tube damage in areas affected by control rod positions. The sleeving modification serves as an interim solution which mitigates the effects of guide tube wear but does not completely eliminate the cause of the wear. Because of major design differences there, no evidence of abnormal guide tube wear was found at the Ft. Calhoun facility.

Calvert Cliffs Unit 2 was the last reactor to be shut down for refueling since the identification of this generic wear problem and was in its refueling outage at the close of the report period. That plant had been operating with the control rods inserted three inches further into the reactor core than originally intended. The repositioning of the control rods was intended to reduce the local severity of the guide tube wear and improve the assurance of control rod scram capability. Also, a more frequent exercising of the control rods was required. All of the affected reactors were operated in this manner prior to modification of the guide tubes; the justification for this requirement was supported by data obtained at St. Lucie Unit 1 and Maine Yankee during refueling.

Investigations by CE are continuing through out-of-reactor flow visualization tests, in an effort to understand the mechanism of flow-induced control-rod vibration, which causes the wear. These test results indicate that the amplitude of control-rod vibration is proportionate to the magnitude of coolant flow through the guide tube. Various prototype fuel assemblies, designed for either decreased guide-tube flow or flow diversion, have been successfully tested in the out-of-reactor test facility.

Demonstration fuel assemblies that utilize the new design concept are being used at Maine Yankee and Arkansas Unit 2 in an attempt to test the design under actual operating conditions. Sixteen fuel assemblies with reduced guide tube flow were to be loaded at Calvert Cliffs Unit 2 during the refueling outage to gain further knowledge. It is anticipated that these tests will provide the data necessary to enable CE to find a permanent cure for the problem.

The NRC reviewed and approved actions taken by affected licensees to assure safe continued operation of their facilities. The NRC issued Safety Evaluations approving the sleeving modifications, as a part of the core reload evaluation. The NRC staff is closely monitoring the results of both the out-of-reactor and in-reactor prototype testing of the newly designed fuel assemblies. The NRC will review any new designs that are proposed as a result of the ongoing tests.

(See Chapter 7, "Abnormal Occurrences — 1978.")

BWR Offgas Explosions

Operating experience with BWRs has resulted in several explosions or rapid burning of hydrogen gas in an auxiliary system to the reactor called the offgas system. Hydrogen gas is generated in the reactor by the radiolytic decomposition of water. When the hydrogen is transferred to the offgas system, it may become potentially explosive if the mixture of hydrogen and oxygen is within certain limits. There have been 29 incidents of BWR offgas hydrogen explosions reported to the NRC. (The actual number of incidents that have occurred could exceed 29 since not all such incidents are required to be reported to the NRC.) The majority have occurred within the offgas system (OGS) which is designed to withstand internal hydrogen explosions. As a result, these explosions did not cause personnel injury, significant radioactivity release or equipment damage. Five other reported offgas explosions have occurred external to the OGS following leakage of the hydrogen-rich offgas mixture from the OGS. These explosions have caused injuries to personnel and significant local physical damage to systems not required for reactor shutdown.

The NRC reviewed the incidents of offgas explosions, their probable causes and consequences, and preventive measures taken to meet them. A technical report (NUREG-0442) on operating experience with BWR offgas systems was issued following the review. No serious design flaws in the engineering of the offgas system have been identified that require immediate remedial action. However, because offgas explosions external to the OGS have caused personnel injuries and property damage, and have necessitated reactor power reductions or shutdowns, interrupting electric power production, preventive measures to minimize the probability of offgas explosions were outlined in the NRC technical report.

An NRC Office of Inspection and Enforcement Bulletin was issued to all BWR licensees requesting their review of the OGS to identify measures that would lessen the likelihood of an offgas leakage, accumulation and potential explosion. The bulletin also requested the review of operation and maintenance procedures to assure proper operation in accordance with all design parameters and to identify measures to prevent inadvertent actions which might cause an ignition of the offgas mixture in the offgas piping. Responses to the Bulletin from all BWR licensees have been received and independent inspections by the Office of Inspection and Enforcement have been made to review the licensees' systems in light of their responses to the Bulletin. These results have been forwarded to the Office of Nuclear Reactor Regulation for consideration in assessing the possibility of additional requirements. Final staff evaluation of the items addressed in the Bulletin is expected by mid-1979.

In addition to the above NRC review and issuance of the technical report and Bulletin, the NRC in 1977 had contracted for a review of operating experience with offgas systems to determine if common factors exist and could be corrected to reduce the possibility of hydrogen explosions.

Fire Protection

Following the fire at the Brown's Ferry Plant in March 1975, the NRC initiated a review of the fire protection programs for all operating plants and for plants not yet operational. Improved guidelines have been developed and are being implemented. At the close of calendar year 1978, the fire protection program reviews had been completed for 27 of the 70 licensed power plants and modifications to improve plant capabilities are being implemented. The reviews of the remaining operating plants will be completed by July 1979, and modifications to most plants will be made by late calendar year 1980.

On November 4, 1977 the Union of Concerned Scientists (USC) filed a Petition for Emergency and Remedial Action. Part of this petition dealt with fire protection concerns at plants under construction and at operating plants, stating that "fire can destroy redundant electrical cables previously thought to be protected by current flame retardancy and cable separation standards."

After consideration of public and staff comments concerning the petition, the Commission issued an order on April 13, 1978 denying the UCS petition and directing certain accelerated staff actions regarding its ongoing fire protection testing program. The basis for Commission denial of the petition is that plants under construction or in operation are in compliance with General Design Criterion 3-Fire Protection, and that fire protection test results do not demonstrate a violation of this criterion.

On May 2, 1978, the UCS submitted a petition which requests that the Commission reconsider its April 13, 1978 decision on the earlier petition filed on November 4, 1977. The Commission has this petition under consideration, and is reviewing public and staff comments which have been developed as a result of the reconsideration. (See Chapters 7 and 13.)

Occupational Radiation Exposures

In the period since 1969, when collective occupational radiation exposure records were first required from reactor licensees, the yearly average man-rems per megawatt year of total power produced has remained relatively constant, below a value of 2.0. Nevertheless, the NRC has been actively concerned with the fact that collective occupational exposure per reactor at commercial light-water nuclear power plants has increased from a 1969 yearly average of 178 man-rems per reactor to a 1977 average of 500 man-rems per reactor. Among the causes for this increase is the increase in radiation fields around reactor plant components, primarily because of the buildup of activated corrosion products, and a need to perform more maintenance and safety-related inspections as plants get older. The NRC reviews the construction permit and operating license applications, and the accompanying Safety Analysis Report (SAR), including a review of the facility's radiation protection program. This review is to assure that the facility is designed to protect the health and safety of the work force against the radiation and radioactivity contained within the facility, resulting from the reactor operation. This latter review includes a determination that the radiation protection program will assure that occupational radiation exposure will be as low as is reasonably achievable. Radiation exposures currently being experienced result after approved and appropriate radiation protection practices are implemented. Additional actions have been taken during the year with respect to the buildup of radioactivity and preparation for maintenance work.

A 1976 petition of the Natural Resources Defense Council called for reduction of the radiation exposure limit. In response, NRC staff proposed further actions to control risks associated with occupational radiation exposure in licensed activities. Staff proposals are under consideration by the Commission and a public hearing on the subject is planned.

Welding Material Deficiency

The NRC was informed on August 4, 1978, of the possibility that weld wire used in some of the reactor vessel welds in 12 vessels manufactured by the Babcock & Wilcox Company (B&W) may have differed from the kind of wire specified for that use. A chemical analysis of one sample of archive material by B&W disclosed that the nickel content in the material was 0.1 percent and the silicon content was 1.0 percent. The *minimum* specified percentage for nickel content was 0.6 percent, and the *maximum* specified percentage for silicon content was 0.5 percent.

The NRC staff undertook a study of the possible effects on reactor vessel integrity of the use (or possible use) of the improper or atypical weld material. Licensees for facilities with atypical material in the "belt-line" region of the vessel have introduced, as needed, new and more conservative pressure-temperature operating limits during bolt up, heat up and cool down to maintain reactor vessel safety margins.

While this particular problem has been identified as one possibly affecting 12 B&W reactor vessels, it is not possible without positive evidence to conclude that similar atypical weld material was not also supplied to other vessel manufacturers and used by them in making reactor vessels. Thus all other power reactor facilities with an operating license or construction permit have also been asked to provide certain information regarding their reactor vessels in order to ascertain whether or not atypical weld material was used in the construction of the vessels.

ADVANCED NUCLEAR POWER PLANTS

On April 7, 1977, President Carter issued a statement on Nuclear Power Policy which restated the role that nuclear energy was to have in the total energy prospects of the country. The President's policy would also defer indefinitely the commercial reprocessing and recycling of plutonium produced in nuclear power reactors, restructure the U.S. breeder reactor program to give high priority to alternative designs, and defer the time when breeder reactors are to be commercialized.

During the report period, the NRC participated in the review and assessment of a variety of reactor types and fuel cycles being considered by the Department of Energy (DOE) as part of the Nonproliferation Alternative Systems Assessment Program (NASAP) and also performed reviews and provided comments on the studies and assessments being performed under the International Nuclear Fuel Cycle Evaluation (INFCE) program. In its reviews and comments the staff focused on the potential licensability of these reactor types and associated fuel cycles, with respect to safety and safeguards concerns and environmental acceptability. (See "Nuclear Fuel Cycle Evaluations" in Chapter 9.)

Clinch River Breeder Reactor

The status of the staff review of the Clinch River Breeder Reactor remained inactive throughout the year and will remain so pending enactment of legislation clarifying the status of this facility.

Fast Flux Test Facility

The Fast Flux Test Facility (FFTF) is a major LMFBR test facility which, with a power of 400 megawatts (thermal), will provide an intense field of fast neutrons for irradiating fuels and materials in connection with advanced reactor research and development. The facility, which is located about 10 miles north of Richland, Wash., is owned by the Department of Energy (DOE) and is not subject to licensing by the NRC. An NRC staff safety review was performed, however, under terms of an interagency agreement with DOE. The staff completed the major part of its review effort and, in August 1978, issued its Safety Evaluation Report (NUREG-0358). Sodium filling of one secondary sodium loop took place in July 1978. Fuel loading is expected in May 1979. Full power operation is not expected until early 1980.

The Advisory Committee on Reactor Safeguards (ACRS) was extensively involved in the review of FFTF with meetings addressing the review held in July, August, September and November 1978. The ACRS concluded that—if due regard is given NRC staff recommendations concerning weld inspections, mitigation of possible consequences of certain low probability accidents, and other matters—the startup and operation of the FFTF is acceptable.

Gas-Cooled Reactors

As a consequence of the withdrawal of the General Atomic Company from the commercial nuclear power market in late 1975, regulatory activities related to gas-cooled reactors have been confined primarily to the Fort St. Vrain reactor, now undergoing power ascension testing. A limited licensing review related to a standardized, large, high-temperature gas-cooled reactor and to a gas-cooled fast breeder reactor has also been undertaken.

Fort St. Vrain. Fort St. Vrain, a 330-MWe high-temperature gas-cooled reactor (HTGR), was designed by the General Atomic Company and is being operated by the Public Service Company of Colorado near Platteville, Colorado. Details of the operation may be found on p. 14 of the 1977 NRC Annual Report.

On October 31, 1977, cyclic temperatures were noted at the Fort St. Vrain reactor at 58 percent power, during the initial rise in power above the previously authorized 40 percent level. Subsequent fluctuations have been observed under a variety of core conditions and at power levels between 40 percent and the present limit of 70 percent. The fluctuations were observed in outlet helium temperatures, external thermal neutron flux, steam temperatures and PCRV movement. Temperature fluctuations have usually remained within design and Technical Specification limits, and the average core thermal power and average helium temperatures remain relatively constant during the fluctuations.

Based on tests performed in 1978, conditions have been established which permit operation of the reactor in a steady-state mode for routine power production below 70 percent of rated power without fluctuations. A public meeting was held in Denver, Colo., on November 3 and 4, 1978, to discuss the fluctuations and possible remedies for them.

Large High-Temperature Gas-Cooled Reactor. A preliminary Safety Evaluation Report for General Atomic's design of a large, standardized HTGR was prepared by NRC staff and discussed with the ACRS Subcommittee on HTGRs in July 1977. This report updated the staff's safety evaluation of the Summit and Fulton HTGRs which had been made prior to the cancellation of these projects. The preliminary SER emphasized the status of the graphite structural design, the seismic design and the thermal and fluid mechanical design.

In early 1978 a group of utilities (now sixteen in number) formed an organization, Gas Cooled Reactor Associates (GCRA), for the purpose of developing a commercially viable HTGR. GCRA manages the DOE funds supporting the project and is responsible for carrying out initial phases of the licensing review. Commercial operation of the first of a series of 900 MW(e) steam cycle HTGRs is foreseen for 1990. Current plans include submittal of a safety analysis report in April 1980 in support of a standardized plant which would form the basis for the issuance of a construction permit in mid-1983. GCRA has requested that the NRC staff undertake at present a pre-application review of selected technical topics pertinent to the HTGR concept. The stated purpose of this review would be to aid development of HTGR licensing criteria and provide for an orderly and effective review of the standard plant application when it is submitted.

As of the end of 1978, the future of the large HTGR program remained uncertain, according to information received from DOE. A final decision will be made early in 1979 whether to redirect the program toward development of an HTGR gas turbine cycle (vs. the steam cycle pursued to date) or to terminate the program during fiscal year 1980.

Gas-Cooled Fast Breeder Reactor. In late 1976, an organization of utilities, Helium Breeder Associates, was formed to work with both General Atomic and DOE (then ERDA) toward the development and demonstration of the Gas-Cooled Fast Breeder Reactor (GCFR). The GCFR demonstration unit would produce 330 MWe. General Atomic is currently studying a revised reactor design that would permit emergency core cooling by means of natural convection. In mid-1977, the staff met with the ACRS Subcommittee on the GCFR, representatives of Helium Breeder Associates (HBA), General Atomic and the Southwestern Public Service Company to review the planned program. A member of HBA, Southwestern Public Service Company had formerly planned to operate the GCFR demonstration plant on a site near Amarillo, Texas, but withdrew this plan in mid-1978.

Floating Nuclear Power Plants

Floating nuclear power plants (FNPs) are electrical generating stations of standardized design which would be constructed at a shipyard facility, using assembly line techniques, and ultimately could be sited at offshore ocean sites or in estuaries and rivers. They are planned to be of conventional reactor system design (using pressurized water reactors) mounted on floating platforms similar to the hull of a barge. Offshore Power Systems (OPS), a subsidiary of Westinghouse Electric Corporation, filed an application with the NRC in 1973 for a license to manufacture eight identical floating nuclear power plants at a site in Jacksonville, Fla.

An NRC staff Safety Evaluation Report (NUREG-75/100) was issued in September 1975; Supplement No. 1 (NUREG-0054) was issued in March 1976 and Supplement No. 2 in October 1976. The staff's Final Environmental Statement (FES) issued in October 1975 (Part I), relates to the construction and nonnuclear testing of the floating plants at the manufacturing site in Jacksonville, Florida. That FES concluded that there is nothing inherent in the operation of the manufacturing facility that would warrant denial of the manufacturing license and recommended its issuance subject to several license conditions. The Final Environmental Statement issued in September 1976 (Part II), relates to the siting and operation of the eight floating plants.

At the request of the Council on Environmental Quality, the NRC prepared an Addendum to the Final Environmental Statement (Part II) which elaborated upon the discussion material and analyses presented in Part II relative to the estuarine and riverine siting of FNPs. The Draft and Final Addendums were issued in March and June 1978 respectively. The staff concluded in the FES, Part II and in the Addendum to Part II that there was reasonable assurance that eight FNPs could be sited with acceptable environmental impact at offshore sites along the Atlantic Ocean and Gulf of Mexico and at carefully selected shoreline locations, including estuarine waters. The Environmental Protection Agency, however, believes that it will be extremely difficult to find environmentally acceptable sites in any of the estuarine or barrier island areas along the East and Gulf Coasts.

A revised Draft Environmental Statement (Part III) was issued in May 1978 which compared the total risk to the public—for both floating and land-based nuclear power plants—from accidental releases of radioactivity to the environment for a spectrum of accidents, including "Class-9" or core-melt events. Part III also presented an overall cost-benefit analysis for all elements of the environmental statement. In the Draft Environmental Statement, Part III, the manufacturing license was recommended for issuance subject to conditions related to mitigating the effects of accidental radioactive releases to the environment resulting from Class 9 events. These conditions include the use of a material beneath the reactor vessel to delay the melting of the core through the barge and, for estuary siting, the use of a closed breakwater. A principal reference used in the preparation of Part III was the Liquid Pathway Generic Study report (NUREG-0040) which is discussed below. The Final Environmental Statement (Part III) was issued in December 1978 and confirmed the earlier staff conclusions contained in the Draft Environmental Statement, Part III. Public hearings on safety and environmental issues were started in March 1975 and continued during 1976, 1977 and 1978. During the 1978 hearings, the applicant requested that the Atomic Safety and Licensing Appeal Board certify the question of whether Class 9 accidents are a proper subject for consideration in the staff's environmental statement. On September 29, 1978, the question was certified to the Commission. On December 8, 1978, the Commissioners agreed to consider the question. Briefs from all parties to the proceeding were filed on December 29, 1978. The major issue being contested is consideration of "Class 9" accidents.

The first application for a permit to construct and operate an offshore floating nuclear power station was filed in 1973 by the Public Service Electric and Gas Company (PSE&G) of New Jersey. The proposed Atlantic Generating Station (AGS), consisting of two floating units (1150 MWe each), would be located approximately three miles off the coast of New Jersey, some 11 miles northeast of Atlantic City. The staff's Draft Environmental Statement (NUREG-0058) issued in October 1976 recommended the issuance of a CP to the applicant. The Safety Evaluation Report (NUREG-0293) was issued in July 1977. In early 1978, the Public Service Electric and Gas Company and Offshore Power Systems agreed upon a three-year delay on the delivery of the floating nuclear power plants for the Atlantic Generating Station. As a result of this delay, as well as to allow for possible consideration of alternative sites for the AGS by the utility, the NRC suspended the safety and environmental reviews of this application until further notice. In December 1978, PSE&G cancelled its contract with OPS, citing among its reasons the lower than anticipated growth rate in its generating area.

Liquid Pathway Generic Study. In connection with its licensing actions on proposed floating nuclear power plants, the staff completed a report on the impacts of accidental radioactive releases to the hydrosphere from floating and land-based nuclear power plants. The report, entitled "Liquid Pathway Generic Study," NUREG-0440, was released in February 1978. As a result of this study, the staff found that the risks and impacts via the liquid pathway from postulated accidents at FNPs at representative sites are expected to be substantially the same as those expected for land-based plants (LBPs), with one exception. That exception is the increase in risk associated with releases to water bodies in the event of a core-melt accident. If such an unlikely event were to occur, the core might melt through the bottom of the barge and introduce radioactive material into the water. The study also concludes that the risks associated with releases to the liquid pathway at an FNP are less than those at an LBP for a spectrum of design basis accidents and are greater than those at an LBP for a core-melt accident.

In November 1978, the Commission submitted a statement to the Congress—pursuant to Section 236 of the Legislative Reorganization Act of 1970—on the actions that have been, are being and will be taken with regard to recommendations made by the Comptroller General of the U.S. in a report entitled, "Before Licensing Floating Nuclear Plants, Many Answers Are Needed."

PROTECTING THE ENVIRONMENT

Health Effects of Coal and Nuclear Fuel Cycle

As noted in the 1977 Annual Report, the NRC is actively developing comprehensive estimates of the potential health effects of the coal and nuclear fuel cycles. The efforts are continuing and have been updated to include estimates of the potential long-term health effects (up to 1,000 years) associated with releases of Radon-222 from mining and milling of uranium, and Carbon-14 releases from electric power generation and fuel reprocessing. (The draft NUREG-0332, "Health Effects Attributable to Coal and Nuclear Fuel Cycles," is being revised to reflect the most recent health effects data from the National Academy of Sciences, and to respond to comments received on the draft.)

A rulemaking petition filed by the New England Coalition on Nuclear Pollution in 1975 challenged the value for radon-222 in 10 CFR 51 as greatly underestimating releases of radon from fuel cycle activities. The petitioner also noted that the value did not include estimates of long-term radon releases and health effects associated with them. As a result of this petition and an ASLBP member's memorandum, the Commission amended the rule to remove the value for radon from Table S-3 and to permit litigation of the issue in individual licensing proceedings. The staff developed new release estimates for periods up to 10,000 years, and estimates of health impacts for up to 1,000 years, and presented these estimates in several licensing hearings during the report period.

Interim staff estimates of the impact of radon-222 and carbon-14 for environmental periods ranging from 100 years to 1,000 years into the future are under consideration in licensing hearings in which the issue has been raised. (See "Environmental Survey of the Uranium Fuel Cycle," in Chapter 3.)

To improve estimates of other environmental impacts associated with the nuclear and coal fuel cycle alternatives, model development for health effects is continuing at Argonne National Laboratory, and a more detailed study of the potential environmental impacts of the coal fuel cycle is being considered for fiscal year 1979.

Assessment of Radiological Consequences of Radionuclide Releases

Before issuing a license, the NRC assesses the probable radiological impact to the public of both the normal operation of nuclear power plants and of adverse but improbable events, of varying likelihood. Such assessments are necessary to assure the health and safety of the public and the protection of the environment. From the results of continuing research, as well as from regular monitoring of both the radioactive effluents and radioactivity in the environment, these assessments are regularly upgraded to insure accuracy and reliability.

Control of Effluents

Standard Technical Specifications. The staff has completed the development of radiological effluent Standard Technical Specifications which implement the requirements of Appendix I to 10 CFR Part 50. The specifications have been reviewed and approved by the Regulatory Requirements Review Committee, and have been published as NUREG-0472 and 0473, applicable to pressurized water reactors and boiling water reactors, respectively. Copies of the applicable specifications have been forwarded to all licensees with operating licenses. Licensees have been requested to submit specifications for their plants using the Standard Technical Specifications on a schedule consistent with submittal dates provided to them. Operating license applicants have also been provided with the applicable Standard Technical Specifications and requested to submit proposed specifications at least six months prior to their scheduled operating license.

To assist in the preparation of the radiological effluent technical specifications, the staff has prepared a guidance manual entitled "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, a Guidance Manual for Users Of Standard Technical Specifications," NUREG-0013. This manual provides methods that are acceptable to the staff for determining parameters used in the specifications.

Change in "ALARA" Rule. On September 5, 1975, the Commission amended Appendix I to 10 CFR Part 50 to allow applicants whose applications for construction permits were docketed between January 2, 1971, and June 4, 1976, the option of dispensing with the cost-benefit analysis required by Appendix I, provided that the proposed or installed radwaste systems and equipment satisfy the site design objectives for nuclear power reactors proposed by the staff in the rulemaking proceeding on Appendix I (Docket No. RM 50-2).

The amended version of Appendix I did not explicitly extend the option (to use the criteria of the Annex) to applicants and licensees whose applications were docketed prior to January 2, 1971 (referred to as "pre-71 plants"). However, a review by the NRC staff of the radwaste systems of the pre-71 plants indicated that these plants had already proposed or installed radwaste systems and equipment designed to satisfy numerical design objectives set forth in either RM 50-2 or in an earlier document which contained similar but more restrictive criteria. Since the radwaste systems of these pre-71 plants contain equipment designed to meet the criteria of the Annex to Appendix I, the staff performed a generic cost-benefit analysis for the pre-71 plants to determine if these plants satisfy the cost-benefit criteria of Section II.D. When this analysis (contained in NUREG-0389) showed that certain of the pre-71 plants satisfy these criteria, the option of using the Annex was extended to these pre-71 plants on a generic basis.

Therefore, if the detailed analysis of the individual radwaste systems of these plants shows that, in addition to meeting the criteria of the Annex, these systems are capable of meeting the design objectives of Sections A, B and C of Appendix I to 10 CFR 50, then the staff would conclude that these plants satisfy the criterion that radioactive materials released in their effluents to unrestricted areas are as low as is reasonably achievable.

In-Plant Measurements Program. In promulgating Appendix I to 10 CFR Part 50, the NRC indicated its desire to use the best available data for improving the calculational models used by the NRC staff to determine conformance with the regulation. To obtain additional data for use in improving its calculational models, NRC contracted with the Idaho National Engineering Laboratory to perform in-plant measurements on pressurized water reactors (PWRs). The measurements will provide a database for radioisotope inventory in plant systems, radioactive waste management system performance, and source terms for both liquid and gaseous systems. Measurements have been completed or are near completion at three plants (Zion, Fort Calhoun, and Turkey Point). Measurements at Maine Yankee are scheduled to begin in 1979.

Site-Related Problems

Potential for Faulting. (The background to licensing problems associated with this facility can be found in the 1977 NRC Annual Report, pp. 26-27.)

In August 1977, the NRC staff informed the licensee for Humboldt Bay, the Pacific Gas and Electric Company, that it could not reasonably conclude from the latter's most recent report that surface faulting would not occur at the plant site. The staff also stated its intention to recommend denial by the licensing board of the application for amendment permitting restart of the unit. The licensee was given the NRC's evaluations and those of the U.S. Geological Survey concerning the potential for surface faulting at the site, and additional information was requested by the licensee regarding these evaluations.

A meeting was held on December 14, 1977 to provide the licensee and its consultants the opportunity to discuss the evaluations with the staff, the U.S. Geological Survey personnel, and representatives of the California Division of Mines and Geology. The written response of the NRC staff to the licensee's request for additional information was also discussed.

On March 3, 1978, representatives of the licensee met with the NRC staff to inform them of a proposed program for further geological investigation near the site of Humboldt Bay Unit 3. Based on the results of these studies, expected by late 1979, the licensee will convey its intentions to the NRC regarding its proposed amendment application.

On May 16, 1978, the Atomic Safety and Licensing Board ruled that the request for a hearing on the proposed amendment submitted in July 1977 by citizens of the Humboldt Bay area would be granted and that a hearing will be held at a time to be determined. It is expected that, at the conclusion of the licensee's geological investigation, a hearing schedule will be established.

Reevaluation of Seismic Capability. In 1971, the existence of a geologic fault about 3.5 miles offshore from the site of the Diablo Canyon Nuclear Plant was discovered. The plant was under construction at the time of the discovery and when, in 1973, application was made for an operating license for the facility, an extensive investigation of the fault (the Hosgri Fault) was undertaken. That investigation led to the conclusion by the NRC and the U.S. Geological Survey that the maximum earthquake ground motion at the proposed site "may be more severe than that for which the plant had been originally designed." The applicant for an operating license—the Pacific Gas and Electric Company—was thus advised in April 1976 that the plant's seismic capabilities should be reanalyzed "to determine what modifications would be necessary to withstand the more severe ground motion."

The Advisory Committee on Reactor Safeguards finished its review of the matter in July 1978 and public hearings before the Atomic Safety and Licensing Board were scheduled to begin in December 1978. A decision on the operating license application is expected in the spring of 1979.

Possible Faulting Near Reactor. While a request for a license renewal for the General Electric Test Reactor (GETR) at the Vallecitos Nuclear Center in California was under review by the NRC, the U.S. Geological Survey published a map showing a fault, the Verona Fault, within 200 feet of the GETR. Field investigations ensued which revealed evidence of possible faulting at the newly identified location of the Verona Fault, and the potential for surface faulting at the GETR site became a licensing concern. The licensee had not identified surface displacement as a design basis and it had not been considered by the AEC when it authorized operation of the GETR in 1959. In the absence of assurance that the GETR could withstand surface displacements induced by earthquake, the NRC staff ordered suspension of operation of the GETR on October 24, 1977, and directed the licensee to show cause why the suspension should not be continued.

On November 11, 1977, the General Electric Company made a written response to the show-cause order which averred that the Verona Fault did not exist, that the geological features thought to be a low angle thrust fault were actually landslide features. The NRC staff evaluated this response and concluded that it did not give sufficient technical information to support a decision that the safety issues raised in the show-cause order had been resolved.

The licensee attempted to formulate the seismic implications of the postulated Verona Fault—in lieu of doing extensive site investigation—and to demonstrate that the GETR is capable of withstanding the consequences of surface displacement associated with the fault. Based on its own review, the NRC staff suggested to the licensee that the value of surface displacement which could be supported by available information would be in excess of the licensee's projection. As a result of the NRC posture, the licensee proposed additional GETR site investigations to resolve the issues regarding the existence of the Verona Fault and its characteristics. These were in progress at the close of the report period.

Mass Mortality of Commercial Lobsters. A mass mortality of commercially held lobsters occurred during October 1977 in Seabrook Harbor, N.H., at a location 400–600 feet south of where a barge dock associated with Seabrook Station was being constructed. NRC staff investigated the incident because it was alleged that construction activities were responsible for the mortality. The investigation included a site inspection, discussions with affected lobstermen, the permittee, State and Federal agencies, and other knowledgeable individuals, and a review of pertinent literature was carried out.

Lobster mortalities may have occurred in small numbers during September, but were greatest during mid-to-late October and apparently continued at reduced levels through early November. It was estimated that about 2400–3400 lobsters, weighing a total of about 3000–3400 pounds were lost.

Several lobsters that had died during late October and early November were obtained through the cooperative efforts of lobstermen, State, and Federal agencies. An independent postmortem pathological examination to determine the cause(s) of death of those specimens was performed for NRC by the U.S. Environmental Protection Agency Research Laboratory, Narragansett, Rhode Island. The pathology study concluded that the immediate cause of death of the lobsters examined was due to gaffkemia or "red tail disease," a virulent bacterial disease of lobsters.

The incidence of gaffkemia mortality was evidently aggravated by coinciding environmental factors. Preceding the incident, an unusual combination of environmental conditions existed in Seabrook Harbor, including heavy rainfall, low and fluctuating levels of salinity, extreme tidal flushing, a mild warming trend, a phytoplankton bloom of unusual magnitude, and the potential for increased turbidity and siltation and for reduced dissolved oxygen. It appears unlikely that construction activities alone could have accounted for excessive turbidity, although an increase in local turbidity levels was possible from those activities.

To minimize the potential for adverse effects from increased turbidity or siltation on commercially held lobsters in Seabrook Harbor during future construction activities at the barge dock site, several precautionary measures were recommended by NRC staff, and incorporated by the Army Corps of Engineers into its permit amendment, which was then implemented by the permittee.

Socioeconomic Assessments. The construction and operation of a nuclear power plant may have considerable impact on the social and economic life of communities near the plant site. The degree of stress and disruption a community will experience is partially dependent on the ability of the community to anticipate and plan for that impact. For that reason, several efforts have been continued or initiated by NRC to help forecast socioeconomic impacts of nuclear power plant sitings more accurately. A study to estimate the likelihood that people would avoid beaches in the vicinity of future floating nuclear power plants was published as NUREG-0394 (E. J. Baker et al., "Impact of Offshore Nuclear Generating Stations on Recreational Behavior at Adjacent Coastal Sites," December 1977) and used as the basis of staff testimony on this issue in ASLB hearings on the floating nuclear power plants. A study, "Visual Change Within a Region Due to Alternative Closed Cycle Cooling Systems and Associated Socioeconomic Impacts" was completed and findings are now being incorporated in environmental impact statements. Considerable progress was made in forecasting the number of construction workers coming into an area, their family characteristics, and probable residential location.

The NRC staff has developed and begun implementation of procedures to promote early cooperation in socioeconomic impact analysis among NRC staff, State and local officials and utilities. These procedures will provide better and more timely information for those local officials who must develop plans to mitigate potentially severe impacts, and utilities will be encouraged to participate more fully in that process.

INTERAGENCY COORDINATION

Coordination on Environmental Matters

The environmental review of NRC licensing actions entails extensive coordination with other Federal and State agencies. Under the National Pollutant Discharge Elimination System permit program, the Environmental Protection Agency (EPA) or delegated State agencies are reviewing impacts to water quality and aquatic biota. In accordance with the Second NRC/EPA Memorandum of Understanding, NRC and EPA have coordinated reviews to avoid duplication of effort and dual regulation. (See 1976 NRC Annual Report, page 70.) NRC provided technical input into EPA hearings on cooling water requirements at the Indian Point and Brunswick facilities. (See Chapter 8 for discussion of NRC-EPA coordination on emergency response planning.)

The Council on Environmental Quality is also involved in a major effort to coordinate the Federal Government's activities in the area of hazardous substances, and at the direction of the President established the Toxic Substances Strategy Committee (TSSC), with representation from 18 Federal agencies. NRC requested membership in this group and, since joining, has contributed significant staff effort both on the TSSC itself and on seven of the eight major Task Groups. The tasks include the development of strategies in the areas of research roles and responsibilities; assessment of research activities in the context of regulatory and policy needs, trade secrets and confidentiality problems; mechanisms for addressing information needs and their impacts; analysis of historical lessons as background for strategy development; policies relating to common approaches for risk assessment; and recommendations for handling of crisis materials. A report to the President on these efforts is planned for early 1979.

In accordance with provisions of the Endangered Species Act, NRC has consulted with the U.S. Fish and Wildlife Service concerning the Pink Mucket Pearly Mussel at the Hartsville and Watts Bar sites (Tennessee), and the Yuma Clapper Rail at the Sundesert site (California), and has consulted with NOAA's Department of Marine Fisheries concerning the Short Nosed Sturgeon at the Montague site (Massachusetts). Such consultation facilitates a determination of whether an NRC licensing action might further imperil an endangered species of wildlife.

NRC has also participated in several interagency task forces focusing on environmental management issues. The Council on Environmental Quality is leading a task force on developing an environmental data base and standardizing monitoring programs. The Heritage Conservation and Recreation Service employed the task force approach to developing a national recreational policy, which included the NRC. The U.S. Department of the Interior, with NRC sharing in the financial support, is developing a transmission line operational manual. This manual is scheduled for publication in April 1979. NRC has reviewed several drafts of State Coastal Zone Management Plans in anticipation of coordinating a review of licensing actions for projects in coastal zones. NRC has provided EPA with data related to pending revisions of the latter's effluent limitation guidelines for the steam electric industry.

A Federal Interagency Task Force on Emergency Instrumentation for Nuclear Incidents at Fixed Facilities is developing guidance on the establishment of emergency radiation detection and measurements systems, in order to provide data directly to State and local governments to complement any measurement systems they may have. In parallel with this effort, an NRC/EPA Task Force was formed for the purpose of providing a clear definition of the types of radiological incidents for which States and local governments should plan and develop preparedness programs. As a result, two reports are being developed: (1) Interim Guidance on Offsite Radiation Measurements Systems, and (2) Planning Bases for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (NUREG-0396).

Such cooperative efforts have made a greater range of expertise available to NRC for its environmental reviews while reducing duplicative efforts.

COOPERATION WITH STATES

NRC and the States cooperate extensively in the environmental review process. There remains however some duplication of effort, particularly in assessing the need for power and in evaluating water quality impacts. And in States which have NEPA-type laws requiring an independent assessment of environmental impact, duplicative environmental reviews may be conducted.

Joint Hearings

The decision as to whether to hold joint hearings with States is made on a case-by-case basis, depending upon the compatibility of NRC and State environmental review schedules and other factors. A successful example of a joint NRC/State hearing was the hearing on the proposed Douglas Point facility in Charles County, Maryland. This hearing was held in July and August of 1976 and involved close coordination between the State and NRC which resulted in the avoidance of much duplicative effort.

In the case of the proposed Greene County facility, the joint hearings with New York State began in early January 1977 and had not been concluded at the close of the report period. Difficulties have been encountered because of differences in NRC and New York's statutory requirements and procedures. The New York State siting law has been recently revised and it is expected that many of the problems experienced in the Greene County proceeding can be avoided in future proceedings in New York State. Substantial efforts are currently underway to develop a detailed agreement and hearing protocol for a forthcoming facility proposed to be located in New York State.

To date, the proposed Douglas Point and Greene County facilities are the only nuclear power plants for which joint NRC/State hearings have been held or are being conducted. The matter of joint hearings is discussed in more detail on pages 31 and 32 of the 1977 NRC Annual Report.

Cooperative Agreements

The NRC, starting in March 1977, increased its efforts to cooperate closely with those States to which EPA has granted authority to issue National Pollutant Discharge Elimination System Permits (402 permits), required for every nuclear power plant licensed by NRC. The initial purpose was to enter into agreements for cooperation that embody principles similar to those set forth in the Second NRC-EPA Memorandum of Understanding (discussed in the 1976 NRC Annual Report, p. 70) under the Federal Water Pollution Control Act (FWPCA). Interest on the part of some States has resulted in broadening the scope of cooperative agreements to cover many areas subject to the jurisdiction of the State or NRC or both.

To date the following agreements have been consummated:

(1) **Virginia** — Effective October 26, 1977. This Agreement is very similar to the second NRC-EPA Memorandum of Understanding under the FWPCA except that it applies primarily to water-related matters and only to nuclear power plants.

(2) **New York** — Effective March 30, 1978. This Agreement (a Memorandum of Understanding) is broader than the Virginia Agreement. It provides for cooperation in the entire environmental review process for nuclear power plants where the State and NRC have overlapping responsibilities under Federal and State law. The intent of the Memorandum is to assure that delays in the siting of nuclear power plants and duplication of effort will be minimized and that effective use will be made of resources of the State agencies and NRC, particularly in the areas of professional expertise. It provides for exploring means whereby the staffs of the State agencies would prepare for NRC, under mutually acceptable guidelines and criteria, analyses in areas of concurrent jurisdiction—such as need for baseload facility, water quality, air quality, terrestrial and aquatic ecology, and land-use aesthetics.

It is anticipated that analyses prepared for NRC by the State will be the subject of separate ancillary agreements. Two ancillary agreements, one in the area of "need for baseload facility" and the other on "water-related matters" currently are being negotiated.

(3) **South Carolina** — Effective April 21, 1978. The South Carolina Agreement is

very similar to the Virginia Agreement except that it applies to all fuel cycle facilities (other than those transferred to the State under the Atomic Energy Act of 1954, as amended, Section 274b).

(4) **Washington** — Effective September 6, 1978. The Agreement with Washington (entitled a Memorandum of Agreement) sets forth the following main principle of cooperation:

The State and NRC agree to explore together the development of detailed subagreements in areas of mutual concern, including, but not necessarily limited to, environmental reviews (or portions thereof) of nuclear facilities subject to licensing by NRC or certification by the State Energy Facility Site Evaluation Council (EFSEC); siting requirements; conduct and format of hearings; confirmatory radiological environmental monitoring around operating nuclear facilities; decommissioning of nuclear facilities; emergency preparedness planning; response to radiological incidents; and radioactive material transportation monitoring.

ANTITRUST ACTIVITIES

As required by law since December 1970, the NRC (then AEC) has conducted prelicensing antitrust reviews of all applications to construct nuclear power plants and certain other nuclear facilities for commercial use. These reviews assure that the issuance of a particular license will neither create nor maintain a situation inconsistent with the antitrust laws. The NRC holds a hearing whenever one is recommended by the Attorney General and must also consider whether antitrust issues raised by the NRC staff or intervenors should be the subject of a hearing. Remedies to antitrust problems usually take the form of conditions attached to licenses; such license conditions may result either from hearings or from non-hearing negotiated settlements.

Antitrust hearings are held separately from those on environment, health, and radiological safety matters. So that antitrust reviews do not delay NRC licensing decisions, applicants are required to submit specified antitrust information to the NRC at least nine months, but not earlier than 36 months, before other parts of the construction permit applications are filed for acceptance review. NRC also performs antitrust reviews prior to issuing operating licenses to determine whether significant changes in applicants' activities have occurred since the construction permit antitrust reviews.

Since the inception of NRC's antitrust program 91 initial construction permit antitrust reviews have been or are being performed. As a result of reviews by the Department of Justice, 17 were "recommended for hearing"; 24 were recommended for "no hearing" because applicants agreed to antitrust license conditions; 49 were recommended for "no hearing," without need for conditions; and one is pending. In addition to these initial reviews, NRC has reviewed and sought advice from the Department of Justice in 27 cases in which additional applicants are seeking part ownership participation in nuclear plants for which applications had been reviewed previously.

The NRC has also sought the Attorney General's advice for two applications for operating licenses where the Commission determined that significant changes in the applicants' activities had occurred. The Attorney General has recommended hearings in both cases. The NRC staff has also conducted operating license reviews of seven applications in which it found no significant changes to have occurred.

In its antitrust program, NRC has reviewed over 170 private, public, and cooperative utilities, which accounted for 84 percent of total kilowatt hour sales in the United States in 1977. (The NRC has reviewed 72 of the top 100 utilities, ranked by kwh sales, in the United States.)

Significant developments have occurred during fiscal year 1978 in several antitrust proceedings. These developments include:

The Atomic Safety and Licensing Appeal Board, on December 30, 1977, reversed the antitrust decision of an Atomic Safety and Licensing Board with regard to Consumers Power Company's application to construct and operate its Midland Nuclear Power Plant (Michigan). The Appeal Board determined that issuance of an unconditioned license to Consumers Power would tend to maintain a situation inconsistent with the antitrust laws. The Appeal Board remanded the case to the Licensing Board to consider an appropriate remedy. The matter is now pending before the Licensing Board.

As a result of a review of a complaint by the City of Cleveland, the NRC sent, on June 28, 1978, a Notice of Violation to the Cleveland Electric Illuminating Company regarding non-compliance with antitrust license conditions that were imposed on the Davis-Besse and Perry construction permits. Responses to the Notice from all parties involved with the complaint are currently under review.

The Florida Municipal Utilities Association and several Florida cities filed late intervention petitions in connection with the St. Lucie, Unit 2 proceeding. An Atomic Safety and Licensing Board granted intervention to the cities. The decision of the Licensing Board was affirmed by the Appeal Board and subsequently by the Commission on June 28, 1978. Pre-hearing Discovery is now underway in the St. Lucie 2 proceeding.

In response to a request, the Commission, in connection with an Operating License application for the South Texas Facility determined that for the purpose of antitrust review "significant changes" have occurred since the prior review of this application by the Attorney General and requested the Attorney General's advice as to whether an antitrust hearing was required. The Attorney General in a letter dated February 21, 1978 advised the Commission that he recommended that an antitrust hearing be held in connection with this application. An Atomic Safety and Licensing Board has been constituted and has ruled with respect to several petitions for leave to intervene. The Atomic Safety and Licensing Board has adopted a general statement of issues and has ordered the initiation of discovery. In a related matter, the Commission on June 21, 1978 determined that "significant changes" have occurred since the construction permit antitrust review of the application for the Comanche Peak Nuclear Power Plant, Units 1 and 2 (Texas). The Commission directed the staff to seek additional advice from the Attorney General with respect to the antitrust aspects of this application. On July 31, 1978 the Attorney General recommended an antitrust hearing.

Discovery has been progressing in the antitrust proceeding for Pacific Gas and Electric Company's application for its Stanislaus Nuclear Power Plant (Calif.). Several sets of interrogatories have been propounded by the parties, and document production has commenced.

INDEMNITY AND INSURANCE

NRC's regulations implementing the Price-Anderson Act provide a three-layered system to pay public liability claims in the remote event of a nuclear incident causing personal injury or property damage. The first layer of this system 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 $140 million.

The second layer provides a mechanism—payment of a retrospective premium—whereby the utility industry would share liability for any damages exceeding $140 million that result from a nuclear incident. In the event of a nuclear incident causing damages exceeding $140 million, each licensee of a commercial reactor rated at 100 electrical megawatts or more would be assessed a prorated share of damages of up to the statutory maximum of $5 million per reactor per incident.

The third layer—Government indemnity—equals the difference between the $560 million limit of liability and the sum of the first and second layers. Currently, the third layer is $85 million. Government indemnity for reactors will be phased out when the sum of the first and second layers provides liability coverage of $560 million. Under the current level of primary financial protection required by the Commission, this will occur when 84 commercial reactors have been licensed. After that point, the limit of liability for a single nuclear incident would increase without limit in increments of $5 million for each new commercial reactor licensed.

Constitutionality of the Price-Anderson Act. On June 26, 1978, the U.S. Supreme Court unanimously upheld the constitutionality of the Price-Anderson Act's limitation on liability for nuclear incidents. This decision reversed a decision by the U.S. District Court for the Western District of North Carolina.

The opinion of the Court, written by Chief Justice Burger, stated that the record "fully supports the need for the imposition of a statutory limit on liability to encourage private industry participation." Thus, the Court concluded that the Price-Anderson Act "bears a rational relationship to Congress' concern for stimulating the involvement of private enterprise in the production of electric energy through the use of atomic power." Further, the court held "the congressional decision to fix a $560 million ceiling, at this stage in the private development and production of electric energy by nuclear power, to be within permissible limits and not violative of due process." (See discussion in Chapter 13, under "Judicial Review.")

Indemnification of Storage of Spent Fuel at Distant Reactor Locations. In November 1977, after public notice, the Commission issued amendments to the operating licenses of Carolina Power and Light Company's Brunswick Steam Electric Plant, Units 1 and 2 (N.C.), and H. B. Robinson Steam Electric Plant, Unit 2 (S.C.), to authorize Carolina Power and Light to store irradiated fuel from the Robinson reactor in either of the spent fuel storage pools at the Brunswick facility. After public notice the Commission also amended the Brunswick indemnity agreement to redefine the term "radioactive material" in the agreement to provide indemnity coverage for storage at Brunswick of the spent fuel generated by the Robinson facility. Any future requests by licensees for similar amendments will be handled by the Commission on a case-by-case basis.

Indemnity Operations. As of September 30, 1978, 137 indemnity agreements with NRC licensees were in effect. Indemnity fees assessed by the NRC from October 1, 1977, through September 30, 1978, totalled $1,992,535. Total fees collected since the inception of the program are almost $20 million. Future collection of indemnity fees will decrease as the indemnity program is phased out for commercial reactor licensees. No payments have been made under the NRC's indemnity agreements with licensees during the 21 years of the program's existence.

Insurance Premium Refund. The two private nuclear energy liability insurance pools — American Nuclear Insurers (also known as the Nuclear Energy Liability-Property Insurance Association) and the Mutual Atomic Energy Liability Underwriters — paid to policy holders the twelfth 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 policy holders over the preceding 10-year period. Refunds paid in 1978 totalled $2,178,638, which is approximately 71 percent of all premiums paid on the nuclear liability insurance policies issued in 1968. The refunds represent 99 percent of the premiums placed in reserve in 1968.

IMPROVING THE LICENSING PROCESS

Improving Effectiveness and Efficiency

In 1977, the Commission directed a staff study of recently completed licensing actions for the purpose of identifying ways to improve the effectiveness and efficiency of the nuclear power plant licensing process. The study and its findings were discussed in detail in the last Annual Report and were published in NUREG-0292 in June 1977. Nine of the Study Group recommendations were approved by the Commission and have been implemented. The recommendations and their status at the end of the report period are as follows:

(1) *Improve the Quality of Applications by Improving Guidance and Strengthening Acceptance Criteria.* This calls for updating the Standard Review Plan and the Standard Format Guide for SARs and making them effective as soon as possible. A system also is to be developed for periodic and timely updating of the Standard Format Guide. Considerable progress was made during the report period. Those changes designated as short-term revisions have been completed and issued. The entire effort is scheduled for completion by late fiscal year 1979.

(2) *Improve the Quality of Applications by Eliminating Unnecessary Information.* This task seeks to identify information which is no longer necessary in applications and to consider the efforts and benefits of eliminating such information. A Task Force was formed and has completed its study. The Task Force concluded that there is little information now

contained in Safety Analysis Reports and Environmental Reports that is not necessary for staff review. Accordingly, the Task Force recommended that no further effort be expended on this recommendation. The Commission has directed that the Task Force study and recommendations be published for public comment.

(3) The next three recommendations are interrelated and are being developed for application on a trial basis for selected applications. They are:

(a) *Increase Pretendering Coordination with Applicants.* This involves expanded management and working level coordination and is designed to provide specific guidance and direction to applicants during the preparation of the application and should result in a more acceptable application being filed.

(b) *Expand and Restructure the Acceptance Review.* The review for completeness will be increased in scope and depth, evaluating acceptability in terms of detail, quality, and clarity. An application will be considered acceptable for docketing and detailed technical review by the staff if the staff can complete its review of that application, as docketed, without any significant additional major information or clarification from the applicant.

(c) *Modify the Current Review Process by Developing an Early Safety Evaluation Report Based on the Application as Docketed.* This relies on successful implementation of parts (a) and (b) and provides for an intensive and detailed safety evaluation without the usual question-answer cycles. The staff's positions and conclusions will be given in the Safety Evaluation Report which will be issued about six months after docketing.

Detailed plans and procedures have been prepared to implement these recommendations. A review of the Palo Verde Units 4 and 5 application is being made with partial use of these procedures and will serve as the first test case. NRC plans to apply these recommendations fully to at least three of the four CP applications expected to be submitted in fiscal years 1979 and 1980. Some assessment of the effectiveness of these procedures should be possible by fiscal year 1980.

(4) *Increase Public Participation During Staff Review.* The plan is to hold a number of staff/applicant meetings at strategic points in the review cycle in the vicinity of the proposed site so that the public will have a chance to observe the interaction of the two groups in the review process and to have questions answered. Another aspect of this plan is to consolidate and integrate the present staff-public interactions into a coordinated and structured plan with well-defined goals and responsibilities.

(5) *Improve the Hearing Process.* This involved a study to determine means for increasing the efficiency and effectiveness of the hearing process by adhering more strictly to the requirements of 10 CFR Part 2. A Commission paper has been prepared with suggested areas for improvement.

(6) *Modify the LWA Rules.* This addresses the need for improved guidance to applicants, staff, and the hearing boards as to the type of activities which are and are not permitted under an LWA. A paper together with a draft rule change will be prepared for Commission consideration by late 1978.

(7) *Increase Use of Rulemaking.* This recommendation considers the desirability of increased use of rulemaking as a mechanism for improving the efficiency and effectiveness of the licensing process. A Steering Committee was established with senior representatives from three NRC offices. The Steering Committee developed criteria for identifying suitable issues for rulemaking and prepared a staff paper identifying and evaluating proposed issues for rulemaking. The Commission approved its publication for public comment.

Board Notification

In May 1978, the Commission approved an agency-wide policy on notifying Licensing Boards, Appeal Panel, and the Commission of new information which is uncovered or developed by NRC staff and is considered by staff to be relevant to one or more licensing proceedings. Each office was required to develop detailed procedures for carrying out the approved policy.

The procedures became effective in July 1978 and a panel was formed to provide training to all NRC professional staff members on board notification policy and procedures. This training has been completed.

The Commission also stated that, after a period of one year, the agency-wide policy and procedures will be reviewed and modified, as necessary. (See discussion of events leading to adoption of notification policy on pp. 187-189 of the 1977 NRC Annual Report. See also Chapter 12 of this report.)

Progress in Standardization

During 1978 a number of significant steps were taken affecting standardization of nuclear power plants. The NRC regards standardization of plant designs—complemented by the early review of sites proposed for nuclear plants—as one of the most important means for improving the efficiency and effectiveness of the regulatory process.

Four procedural options are available (see 1976 NRC Annual Report, page 36, for details) to applicants for standardization of nuclear power plants: "Reference Systems" (approved design used repeatedly by reference), "Duplicate Plants" (approved design for several identical plants), "License to Manufacture" (approved design for manufacture of identical units at the central location), and "Replicate Plants" (reuse of recently approved custom design).

Since the standardization policy was adopted by the Atomic Energy Commission in 1973, the following has been accomplished:

(1) Twenty-four applications for preliminary design approvals under the reference system concept have been received. Twelve preliminary design approvals for reference system designs have been issued as of the end of the fiscal year. Eleven construction permit applications (for a total of twenty-seven units) referencing five of the reference system designs have been received. Construction permits for 16 of the units have been issued.

(2) One application for a manufacturing license for eight floating nuclear plants has been received and is currently under review.

(3) Eight applications for construction permits, for a total of fifteen units, have been received under the duplicate plant concept. Construction permits for 12 of the units have been issued.

(4) Five applications for construction permits, for a total of 10 units, have been received under the replicate plant concept. Construction permits for two of the units have been issued.

In a policy statement issued on June 29, 1977, the Commission reaffirmed its support for standardization and requested public comments on proposed program changes and suggestions to enhance the use of standardization. The comments received from the public were considered by the staff in its continuing study of standardization. On the basis of its study, the staff concluded that certain changes to the Commission's standardization program should be made and that these changes could be implemented within existing regulations. In addition, the staff concluded that the revised standardization program will continue to allow applicants to utilize a wide variety of design options in ways that can avoid the development of significant adverse antitrust consequences. The report, "Review of the Commission Program for Standardization of Nuclear Power Plants and Recommendations to Improve Standardization Concepts," NUREG-0427, issued June 7, 1978, provides a summary of the information used in the staff's study, presents the public comments received in response to the Commission's June 29, 1977 policy statement, and the staff's assessment of this information together with its conclusions and recommendations.

Following review of the staff's recommendations the Commission, in August 1978, issued a policy statement, "Statement on Standardization of Nuclear Power Plants," which expanded on the standardization concept for nuclear plants and described specific policy changes being made to improve the usefulness of the Commission's standardization program. All of the changes can be implemented within existing regulations. These changes (1) define the effective time periods for design approvals under each of the four standardization concepts, (2) provide for forward-referencing of an approved final plant design, (3) define the criteria for qualification reviews under the duplicate plant and replicate plant concepts, (4) establish the requirements for updating a plant design under the manufacturing license concept, (5) provide for the extension of current Preliminary Design Approvals, and (6) introduce the concept of a Standard Design Approval as a means of achieving a one-step licensing review process.

In order to provide an organizational focus on standardization, in May 1978, the Standardization Branch was created under an Assistant Director for Standardization and Advanced Reactors in the Division of Project Management, Office of Nuclear Reactor Regulation. This Branch will be responsible for the development of NRC policy in the area of standardization, as well as the project management function for applications for approvals of standard plant designs.

Table 3 lists the applications for preliminary design approvals of reference system designs, and for construction permits for plants utilizing one or more of the available standardization options. Since the standardization policy was enunciated in 1973, more than one-half of the construction permit applications have utilized one or more of the standardization options and the fraction has increased to about two-thirds during the last three years.

Early Site Reviews

During the review of applications for nuclear power reactor construction permits, site-related issues often become "critical-path" items. In order to remove such items from the critical path and take better advantage of the standard plant concept, the NRC established procedures for Early Site Reviews (ESR). Applications utilizing the ESR process include Blue Hills (Texas), North Coast (Puerto Rico), Douglas Point (Maryland), and Fort Calhoun Unit 2 (Nebraska). (See 1978 NRC Annual Report, pages 36 and 37.)

Environmental Standard Review Plans

Environmental standard review plans are being prepared to guide and direct the staff's environmental review of nuclear power plant applications. The plans are intended to give guidance to both applicants and staff as to the information and criteria that are considered essential to the environmental review process. Ninety-three draft plans have been published (NUREG-0158, Parts 1, 2 and 3). The plans will specify NRC internal procedures and positions, document the content and bases for each environmental review, and frame the extent of the review to assure that only essential items are considered. Upon their completion, the review plans will be used as the basis for a revision of Regulatory Guide 4.2 so that the NRC data requirement is more explicitly stated.

All plans were issued for review and comment by the end of 1977. Comments on the plans were received through the first half of 1978, and it is expected that final plans will be issued in 1979.

Systematic Evaluation of Operating Reactors

The Systematic Evaluation Program (SEP) staff is responsible for the review of 11 older licensed operating power reactors, applying current licensing criteria, and for documenting the results—including the need for any necessary plant changes. The major objectives of the SEP are:

(1) The program will assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

(2) The program will establish documentation which shows how well each operating plant reviewed meets current criteria on significant safety issues, and should provide a rationale for acceptable departure from these criteria.

(3) The program will provide the capability to make integrated and balanced decisions with respect to any required backfitting.

(4) The program is structured for early identification and resolution of any significant deficiencies.

Table 3. Standardization Applications

(as of September 30, 1978)

PROJECT	APPLICANT	DOCKET DATE	COMMENTS
Reference Systems			
Nuclear Island			
GESAR-238(NI)	General Electric	7/30/73	Nuclear Island, PDA-1 (Preliminary Design Approval) issued 12/22/75
Turbine Island			
C F BRAUN SSAR	C. F. Braun	12/21/74	Turbine Island matched to GESSAR-238(NI). PDA-5 issued 5/07/76
Nuclear Steam Supply System (NSS)			
BSAR-205	Babcock & Wilcox	3/01/76	PDA-12 issued 5/31/78
BSAR-241	Babcock & Wilcox	5/14/74	(Withdrawn)
CESSAR	Combustion Engineering	12/19/73	PDA-2 issued 12/31/75
GASSAR	General Atomic	2/05/75	Review suspended at request of applicant.
GESSAR-238	General Electric	10/16/75	PDA-10 issued 3/10/77
GESSAR-251	General Electric	2/14/75	PDA-9 issued 3/31/77
RESAR-3S	Westinghouse	7/31/75	PDA-7 issued 12/30/76
RESAR-41	Westinghouse	3/11/74	PDA-3 issued 12/31/75
RESAR-414	Westinghouse	12/30/76	
Balance of Plant (BOP)			
BOPSSAR/ BSAR-205	Fluor Pioneer	10/31/77	BOP matched to BSAR-205
BOPSSAR/ RESAR-41	Fluor Pioneer	1/27/76	PDA-11 issued 8/17/77. BOP matched to RESAR-41
ESSAR/BSAR-205	Ebasco	5/19/78	BOP matched to BSAR-205
ESSAR/CESSAR	Ebasco	2/02/78	BOP matched to CESSAR
ESSAR/RESAR-414	Ebasco	11/23/77	BOP matched to RESAR-414
GAISSAR/ BSAR-205	Gilbert Commonwealth		BOP matched to BSAR-205
GAISSAR/ CESSAR	Gilbert Commonwealth		BOP matched to CESSAR

PROJECT	APPLICANT	DOCKET DATE	COMMENTS
GAISSAR RESAR-414	Gilbert Commonwealth		BOP matched to RESAR-414
GIBBSAR	Gibbs & Hill	5/10/77	BOP matched to RESAR-41
SWESSAR/ BSAR-205	Stone & Webster	12/22/75	BOP matched to BSAR-205
SWESSAR/ CESSAR	Stone & Webster	10/21/74	BOP matched to CESSAR PDA-6 issued 8/16/76
SWESSAR/ RESAR-3S	Stone & Webster	10/02/75	BOP matched to RESAR-3S BPDA-8 issued 3/31/77
SWESSAR/ RESAR-41	Stone & Webster	6/28/74	BOP matched to RESAR-41 PDA-4 issued 5/05/76

Utility Applications Using Reference Systems

Cherokee 1, 2 & 3	Duke Power	5/24/74	References CESSAR. CP issued 12/30/77
Perkins 1, 2 & 3	Duke Power	5/24/74	References CESSAR
South Texas 1 & 2	Houston Light and Power Co.	7/05/74	References RESAR-41 CP's issued 12/22/75
WPPSS 3 & 5	Washington Public Power Supply System	8/02/74	References CESSAR. CP's issued 4/11/78
Palo Verde 1, 2 & 3	Arizona Public Service		References CESSAR. CP's issued 05/25/76
Hartsville 1, 2, 3 & 4	Tennessee Valley Authority		References GESSAR-238(NI) CP's issued 05/09/77
Palo Verde 4 & 5	Arizona Public Service	03/31/78	References CESSAR
Black Fox 1 & 2	Public Service of Oklahoma	12/23/75	References GESSAR-238 (NSSS)
Phipps Bend 1 & 2	Tennessee Valley Authority	11/07/75	References GESSAR-38 CP's issued 1/16/78 (NI)
Erie 1 & 2	Ohio Edison Co.	3/01/77	References BSAR-205
Yellow Creek 1 & 2	Tennessee Valley Authority	7/16/76	References CESSAR

Duplicate Plants

Bryon 1 & 2 Braidwood 1 & 2	Commonwealth Edison	9/20/73	Two units at each of two sites. CP's issued 12/31/75
Cherokee 1, 2 & 3 Perkins 1, 2 & 3	Duke Power	5/24/74	Three units at each of two sites. Also references CESSAR. Cherokee CP's issued 12/30/77
SNUPPS			Five units at four sites.
Wolf Creek	Kansas Gas & Electric Co., Kansas City Power & Light	5/17/74	CP issued 5/17/77
Callaway 1 & 2	Union Electric	6/21/74	CP's issued 4/14/76
Tyrone 1	Northern States Power	6/21/74	CP's issued 12/27/77
Sterling	Rochester Gas & Electric	6/21/74	CP issued 9/01/77

PROJECT	APPLICANT	DOCKET DATE	COMMENTS
WNP Koshkonong 1 & 2	Wisconsin Electric Power Madison Gas & Electric Wisconsin Power & Light Wisconsin Public Service	8/09/74	Initially submitted under duplicate plant option with intent for as many as six total units at three sites. Utility's change in plans led led to removal from standardization program by staff. Review discontinued because of site problems.
License to Manufacture			
Floating Nuclear Plant (FNP) 1-8	Offshore Power Systems	7/05/73	Entire plant design
Utility Applications Using License to Manufacture			
Atlantic 1 & 2	Public Service Electric & Gas	3/01/74	References Floating Nuclear Plant
Replication			
Jamesport 1 & 2	Long Island Lighting	9/06/74	Replicates Millstone 3
Marble Hill 1 & 2	Public Service of Indiana	9/17/75	Replicates Byron
New England 1 & 2	New England Power & Light	9/09/76	Replicates Seabrook
Palo Verde 4 & 5	Arizona Public Service	3/31/78	Replicates Palo Verde 1, 2, and 3
Haven 1 & 2	Wisconsin Electric Power	5/26/78	Replicates Koshkonong 1 & 2

(5) The program will efficiently use available resources and minimize requirements for additional resources by NRC or industry.

The planned systematic evaluation will assess the adequacy of 11 of the older operating power reactors with respect to safety and provide clear written documentation of the basis for the assessment. The technical evaluation will be based on the evaluation of some 130 selected safety topics in the context of how they affect a plant's ability to withstand certain Design Basis Events. These technical evaluations will also provide the basis for action on licensee requests to convert 7 of the 11 licensees from Provisional Operating Licenses to Full Term Operating Licenses.

For future reactors, NRC staff has instituted procedures which will eliminate the need for such a program. Specifically, the operating license review will document deviations from current licensing requirements and the basis, if such exists, for acceptance. In addition to this, each new licensing requirement which is identified by the Regulatory Requirements Review Committee as applicable to operating facilities will be assessed for each facility and the conclusions documented, thus keeping the evaluations current in the future. Coupled with the Systematic Evaluation Program, the new procedures will assure that every operating plant will have a record of the results of staff review for all safety concerns and that the record will be continuously updated as new issues are identified by the staff.

Phase I of the SEP, the development of a list of topics to be used in performing the systematic evaluations, has been completed. As a result, a comprehensive lists of topics and definitions of staff safety objectives, together with a review procedure that considers the effect of these topics on Design Basis Events, were developed. Phase II of the SEP, the actual evaluation of the eleven older facilities was approved by the Commission in November 1977 and is scheduled for completion by January 31, 1981.

Quality Assurance

The application of disciplined engineering practices and thorough management and programmatic controls to the design, fabrication, construction, and operation of nuclear power plants is essential to the protection of public health and safety and of the environment. Quality Assurance (QA) provides this necessary discipline and control. Through a QA program that meets NRC requirements, all organizations performing work that is important to safety are required to conduct work in a preplanned and documented manner; to independently verify the adequacy of completed work; to provide records that will confirm the acceptability of work and manufactured items; and to assure that all individuals are properly trained and qualified to carry out their responsibilities.

Each NRC licensee is held responsible for assuring that his nuclear power plants are built and operated safely and in conformance with the NRC regulations. In addition, the NRC has several specific QA responsibilities. First, it has a responsibility for developing the criteria and guides for judging the acceptability of nuclear power plant QA programs. Second, it has a responsibility for reviewing the QA programs of each licensee and its principal contractors to assure that sufficient management and program control exist. Finally, NRC inspects selected activities to determine that the QA programs are being implemented effectively.

Where QA programs are found deficient, the NRC requires appropriate upgrading. In those cases where the QA program is not being properly implemented, the NRC uses enforcement authority as necessary to achieve proper implementation. If a generic QA problem develops, improvements in QA programs are made industry wide.

Through the NRC topical report program, the industry has widely adopted standardized QA programs which can be used on new projects without a new review. As of the end of the fiscal year, a total of 32 topical reports on quality assurance from manufacturers of nuclear steam supply systems, architect-engineering firms, constructors, and utilities have been found acceptable by the NRC, and other reports are under review.

NRC is engaged in activities, also under the topical report program, that are intended to minimize or eliminate the need for redundant audits of suppliers without reducing the confidence that work is proceeding satisfactorily in accordance with regulations. NRC has reviewed and found acceptable a topical report from the Coordinating Agency for Supplier Evaluation that should reduce the need for pre-award audits for potential suppliers. NRC is also in the process of reviewing a topical report describing the ASME certification and inspection program which, if found acceptable, could be endorsed as a "third party" audit program. Successful achievement of this objective should further reduce the need for pre-award audits and for yearly programmatic audits by purchasers.

An independent assessment of the adequacy of NRC's regulatory practices in the area of QA was contracted to Sandia Laboratories and completed in August 1977. The results of the study generally endorsed current practices and suggested additional measures and potential improvements for NRC consideration. Some of the recommendations have been implemented, some are being implemented, and others are the subject of further study.

Areas where recommendations have been implemented are:

(1) The establishment of a revised documented agreement between the NRC Office of Nuclear Reactor Regulation and the NRC Office of Inspection and Enforcement to provide a management system for identifying, scheduling for completion, and reporting the status of those problems requiring action by both offices.

(2) Providing improved documented communication to those vendors inspected under the Licensee Contractor & Vendor Inspection Program to assure that they are aware of the continuing responsibility and authority of the licensee (purchaser) with respect to vendor quality assurance.

(3) Clarifying the responsibilities within the NRC Office of Inspection and Enforcement with respect to the inspection of independent architect engineering firms and those utilities who perform their own in-house architect-engineering activities.

Value Impact Analysis

During the report period, the staff prepared a technical analysis for its proposed requirements concerning Anticipated-Transients-Without Scram (ATWS). The staff determined that these incidents—described above, under "Unresolved Safety Issues Plan"—had the potential for becoming core-melt accidents with significant offsite releases, and proposed requirements to reduce the probability of such accidents to 10^{-6} per reactor-year. Employing methods developed during the Reactor Safety Study (RSS) to measure radiological risk, the staff also performed a value impact analysis which measured the benefits (averted risks and other associated impacts) that would be attained by the proposed ATWS requirements and weighed that value against the dollar costs and other impacts entailed in meeting the new requirements. This analysis, although clearly subject to large uncertainties in calculation, appeared to support the need for the requirements. However, because of uncertainties in the RSS it was not the primary basis for decision-making regarding the staff's initial recommendations regarding ATWS requirements.

Value-impact is defined as an evaluation of all significant adverse impacts of a particular action as measured against all significant beneficial values of that action, synonymous with NRC's NEPA cost-benefit analysis. The staff views value-impact analyses as being an aid to the decision-maker. Such analyses provide a format for formal analysis and display of results regarding all of the many significant costs and benefits that must be considered in making a complex decision. The information should be quantified where possible; however, not all significant information is amenable to quantification, and even that which is has some associated uncertainty. Judgment as to the extent of uncertainty and the significance of unquantifiable information must be a part of the analysis as well as of the subsequent decision. In the case of the ATWS value-impact analysis there are important considerations with which rather large uncertainties are associated. Unless the values clearly and substantially outweigh the costs (or vice-versa) after appraisal of these uncertainties, the value-impact analysis cannot be expected to dictate the decision of itself. What it can do is provide the decision-maker with additional valuable perspectives on which judgment must be exercised.

The staff will be considering all comments received on this subject, as well as the recommendations of the Lewis Committee on the use of the RSS, before making any final recommendations concerning any ATWS requirements.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards is a panel of independent advisors established by law to review and report to the Nuclear Regulatory Commission (NRC) on safety studies and on construction permit and operating license applications for nuclear power reactors and other major nuclear facilities.The Committee also provides advice to the Commission on a wide range of safety-related matters such as the adequacy of proposed reactor safety standards, reactor safety research, specific technical issues of a topical nature, and the safety of operating reactors. In addition, upon request by the Department of Energy (DOE), the Committee reviews and provides reports with regard to the possible hazards of DOE nuclear activities and facilities. The Committee may also on its own initiative conduct reviews of specific safety-related items.

Recently added to the Committee's functions (Public Law 95-209) is the requirement for Committee review of the NRC's Reactor Safety Research Program and an annual report to the Congress concerning the adequacy of the program. The first report by the Committee was provided to the Congress in December 1977 ("Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program," NUREG-0392).

During fiscal year 1978, the Committee provided reports on its review of Construction Permit applications for seven licensed nuclear power stations which included a total of 12 individual nuclear power plants. The Committee also reviewed and reported on operating license applications for five nuclear power stations consisting of a total of seven individual nuclear power plants.

The continued effort toward standardization of nuclear power plant design was reflected by the Committee's review and approval of the Stone and Webster Engineering Corporation's application for a Preliminary Design Approval of a standardized nuclear balance-of-plant design (SWESSAR-P1) that would interface with a single unit Babcock & Wilcox pressurized water nuclear steam supply system.

The Committee also completed a review and reported favorably on an application for Preliminary Design Approval for the Westinghouse Electric Corporation's standardized nuclear steam supply system—RESAR 414, a 3800 MWt nuclear power system.

The Committee reviewed and approved requests for power level increases for the Maine Yankee Nuclear Power Station and Unit 3 of the Indian Point Nuclear Generating Station.

The Committee performed a review requested by DOE on the Naval S8G prototype propulsion system and its shipboard application.

Special reports were provided to the NRC by the Committee during the report period on the following matters:

- Resolution of generic items related to the Shearon Harris Nuclear Plant.
- Modification of Recirculation and Quench Spray Systems at the North Anna Power Station Unit I.
- Regional Tectonics of the Pacific Northwest.
- Status of Generic Items Relating to Light Water Reactors.
- Liquid Pathways Generic Study.
- Containers for Air Shipment of Plutonium.
- Decommissioning of Nuclear Facilities.
- Proposed Research on Systems to Improve Safety of LWR's.
- Westinghouse Critical Heat Flux Correlation and Thermal Design Procedure.
- Evaluation of Alternative Sites to Those with High Population Densities.

During fiscal year 1978, the Committee met with the NRC staff on numerous occasions to hear and discuss reports of operating experiences and proposed changes at nuclear facilities.

During the fiscal year, the Committee prepared reports to Congress and Congressional Oversight Committees as follows:

- First annual report to Congress on Reactor Safety Research in the U.S. (required by Public Law 95-209). This first annual report focused on the NRC Safety Research Program with particular attention directed to Systems Engineering, Analysis Development, Fuel Behavior, Metallurgy and Materials, Site Safety, Advanced Reactor Safety, Fuel Cycle and Environmental Safeguards, and Risk Assessment.
- Report to Hon. Morris K. Udall, Chairman, Committee on Interior and Insular Affairs, House of Representatives, on the advisability of establishing an independent quasi-judicial board for nuclear and related accidents.

The Committee also responded to inquiries from the President, North Anna Environmental Coalition, regarding pressure vessel structure and pump performance at the North Anna Nuclear Power Station (Va.).

In providing advice to the NRC on proposed Regulatory Guides and Standards, the Committee reviewed and approved a total of 21 proposed guides or revisions to guides including those on:

- Site investigations for nuclear power plants.
- Material for concrete containment.
- Electrical penetrations for light-water reactors.
- Service limits and loading combinations for component supports.
- Seismic design classification.
- Test programs for water cooled reactors.
- Combustible gas control systems in light water reactors.
- Tornado design classification.

Several proposed amendments to NRC criteria were also reviewed including those on General Design Criterion 50 (Appendix A, 10 CFR Part

50), Containment Design Basis, and standards for combustible gas control systems in light water reactors.

In addition to the items noted specifically above the Committee devoted considerable attention to the following areas of interest:

- Treatment, storage, and disposal of high and low level radioactive wastes.
- Security and physical protection provisions at nuclear facilities.
- Provisions for mitigating the consequences of anticipated transients without scram in light water reactors.
- Reduction of radioactive exposure to nuclear plant personnel.
- Review and evaluation of operating experience at nuclear facilities and its application to improve facility designs and procedures.
- Application of probablistic methods of analysis and related data to the evaluation of reactor safety issues, particularly the seismic design of nuclear plants.
- Review of NRC siting policies and practices.

In performing the reviews and preparing the reports referenced above, the Committee met in full session 12 times. In addition, 92 Subcommittee and Working Group Meetings were held and eight site-facility visits were made. All the full Committee meetings were largely open to the public and 91 of 92 Subcommittees and Working Group Meetings were either fully or partly open. Comments were received from members of the public with respect to several matters evaluated by the Committee.

Members of the Committee also participated in visits and meetings with representatives of the Japanese, French, British and German regulatory and research agencies.

Materials Regulation

The NRC regulates all steps involved in supplying fuel to nuclear reactors except for uranium mining and the enrichment of uranium in Government-owned plants. Thus, in the reactor "fuel cycle," the NRC licenses and maintains surveillance over the construction and operation of facilities for uranium milling, uranium hexafluoride conversion, fuel processing and fabrication, "spent" fuel storage, and spent fuel reprocessing. The NRC also regulates the uses of reactor-produced radioisotopes (byproduct materials) in medicine and industry, the transportation of nuclear materials, and the ultimate disposal of radioactive wastes (discussed in Chapter 5). In all of these areas, the NRC requires that licensees conform to standards established to protect public health and safety, national security and the environment.

Among highlights in radioactive materials regulation during fiscal year 1978, the NRC:

- Completed five uranium mill licensing actions.
- Initiated a program of technical assistance to Agreement States on environmental analyses for uranium mills.
- Completed more than 8,300 materials licensing actions.
- Set performance objectives for the uranium milling industry as part of an intensive program to resolve the problems of mill tailings management.
- Terminated Commission proceedings on the issue of reprocessing spent light water reactor fuel and recycling the recovered plutonium in fresh mixed oxide fuel. The action also ended NRC proceedings on pending or further major plutonium recycle related license applications.
- Certified to the Congress that a safe container had been developed for air transport of plutonium.
- Issued a final environmental statement indicating that radioactive material transportation generally is being conducted under current regulations in an adequately safe manner.

The Nuclear Fuel Cycle

URANIUM MILLING AND PROCESSING

Mined uranium ore is physically and chemically treated in uranium mills to recover a uranium concentrate. In this process, large quantities of waste material, termed "mill tailings," are produced. Because most of the radioactivity originally contained in the ore is retained in the tailings, this material can cause environmental problems unless adequate control measures are taken. Although the concentration of radioactive material in the mill tailings is relatively low, the material presents a waste management problem because of the large quantities produced and the long half-lives of a number of the contained radionuclides.

There are currently 21 uranium mills in operation, all located in western States. Of these, 10 are licensed by NRC and the remaining 11 by Agreement States (see Chapter 8). Currently and previously operating mill sites already contain approximately 140 million tons of accumulated tailings, and a number of new mills are under construction or are in the planning stage. It is estimated that, by the year 2000, as many as 90 uranium mills may be in operation and as much as 750 million tons of tailings may have been generated.

Mill Licensing Actions

Licenses issued by the NRC for new uranium mills as well as renewals of licenses for existing facilities incorporate conditions covering final tailings reclamation plans along with financial arrangements to insure completion of these plans. New mill licenses are issued after publication of a final environmental impact statement and completion of a safety evaluation report for each facility.

During fiscal year 1978, NRC issued a renewal for the Lucky Mc Corporation, Gas Hills, Wyo., plant and major amendments to licenses for: Federal American Partners, Shirley Basin, Wyo.; Petrotomics Company, Shirley Basin, Wyo.; and Union Carbide Corporation, Gas Hills, Wyo. A facility expansion was authorized for the Lucky Mc Corporation, Gas Hills, Wyo.

At year-end, environmental impact statements were being prepared and safety reviews conducted on license renewal applications for two other operating mills and on license applications for the following proposed new mills: Minerals Exploration Company, Sweetwater County, Wyo.; United Nuclear Corporation, Converse County, Wyo.; Energy Fuels Nuclear, Inc., White Mesa Project, San Juan County, Utah; Plateau Resources Limited Shootering Canyon Project, Garfield County, Utah; and Kerr-McGee Nuclear Corporation, Converse County, Wyo. Requests from four operating mills for major license amendments were under review. Environmental and safety reviews were being conducted on license applications for the following uranium ore-buying stations: Energy Fuels Nuclear, Inc., Blanding, Utah; Energy Fuels Nuclear, Inc., Hanksville, Utah; and Plateau Resources Limited, Blanding, Utah.

In Situ Solution Mining

The uranium industry is showing increased interest in applying *in situ* uranium solution mining techniques as a means of recovering uranium from low-grade ore deposits and small pockets of higher-grade ores. This technique is especially applicable where underground and/or open pit mining are not economically feasible or environmentally acceptable.

In this technique, uranium is leached from the ore, in its natural location, via an acid or basic leachant solution. The resultant uranium-bearing solution is pumped to the surface and the uranium recovered from the produced solution by standard mill operations. The barren (uranium-depleted) solution from the recovery unit operations is then reconstituted with chemical leachant additions and reinjected into the ore zone to repeat the cycle.

Twelve source material licenses have been issued authorizing *in situ* uranium solution mining research and development activities at various sites in Wyoming. In addition, production-scale operation by Wyoming Mineral Corporation was authorized in Wyoming. The final environmental impact statement for Exxon Mineral Company's commercial scale operation has been issued. Authorization for operation should be completed in early 1979.

Conversion to UF_6

Following the milling operation, uranium ore concentrates are shipped to a facility for purification and conversion to uranium hexafluoride (UF_6). This compound is fed into the gaseous diffusion plants where the uranium is enriched (see below).

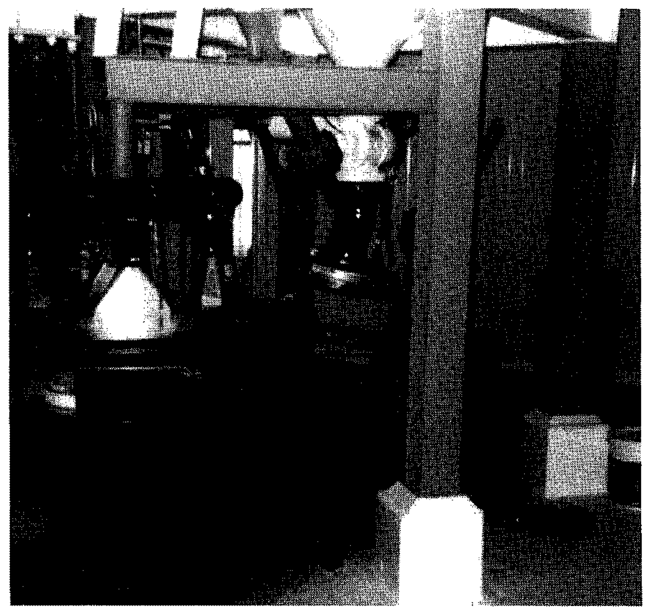
Two NRC-licensed facilities in the United States produce UF_6 from ore concentrates—the Allied Chemical plant at Metropolis, Ill., with a rated capacity of 14,000 tons of uranium per year, and the Kerr-McGee facility in Sequoyah County, Oklahoma, with a capacity of 10,000 tons of uranium per year.

Uranium Enrichment

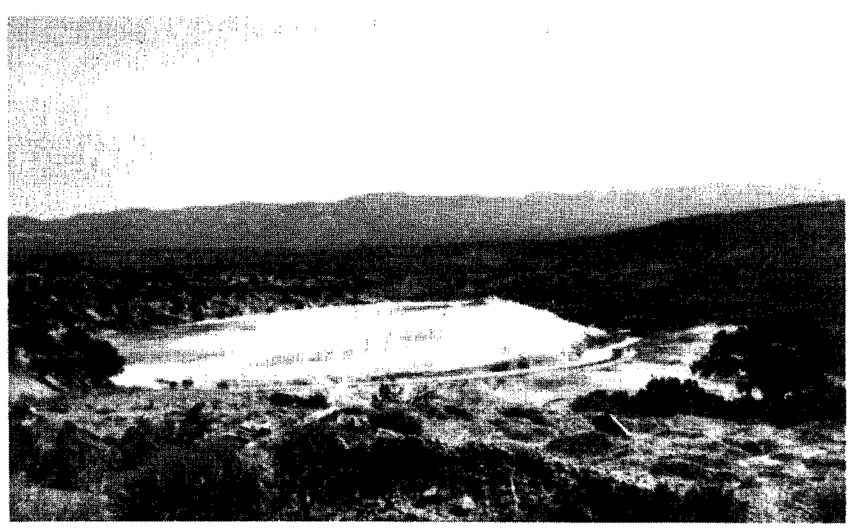
The enrichment of uranium to the degree needed to make it usable in reactor fuel continues to be the only major step in the nuclear fuel cycle not performed as a commercial enterprise. Three gaseous diffusion plants owned by the Department of Energy (DOE) constitute the entire U.S. enriching capacity. These plants are not regulated by NRC.

Fuel Fabrication

Uranium hexafluoride (UF_6) enriched to a maximum of five percent in the U-235 isotope is



Renewal of the NRC license for this Union Carbide Corporation uranium mill at Gas Hills, Wyoming, was one of 5 licensing actions for mills and related facilities during Fiscal Year 1978. Such mills convert raw uranium ore to "yellowcake" (U_3O_8), a uranium concentrate used as feed material for further conversion to uranium hexafluoride and ultimate refinement for reactor fuel or other uses. Shown in these photos are, above left, an exterior view of the Gas Hills Plant; above right, interior shot of part of the yellowcake process — packaging it for shipment — at a SOHIO plant in New Mexico, and, at right, a picture of an isolated Union Carbide tailings pile in Colorado.



shipped from enrichment facilities to fuel fabrication plants where it is converted to ceramic uranium dioxide (UO_2) pellets for encapsulation in long, pencil-like tubes made of "Zircaloy." These tubes are then sealed and assembled into fuel bundles for insertion into light water reactors. Currently, there are five such fuel fabrication plants.

In addition to having regulatory authority over the light-water reactor fuel fabrication plants described above, the NRC is responsible for the licensing of facilities engaged in the fabrication and assembly of high enriched fuel elements for naval reactors and of fuel plates for research and test reactors.

Licensing actions in 1978 included the issuance of license renewals for the Nuclear Fuel Services, Inc., fuel production facility (Erwin, Tenn.) and the Westinghouse Electric Company fuel fabrication facility (Columbia, S.C.).

Evaluation of Formerly Licensed Sites

In response to a General Accounting Office inquiry concerning potential radiation safety problems at sites previously operated under an AEC license, NRC committed to a reexamination of the files of licenses terminated prior to 1965. GAO indicated that files of licenses terminated since the mid-1960's contained adequate assurance of proper decontamination. The Oak Ridge National Laboratory is completing and evaluating the docket files to determine which sites, if any, may require surveying and possibly remedial action.

Environmental Review of Milling

The NRC is preparing a generic environmental impact statement (GEIS) on the U.S. uranium milling industry to the year 2000, with particular emphasis on mill tailings. This document is designed to lead to regulations covering management and disposal of mill tailings and to recommendations for institutional arrangements necessary for long-term isolation of the tailings waste. Alternative tailings disposal programs will be evaluated taking into consideration the risks to exposed individuals, health effects of population doses, effects of natural weathering forces, groundwater impacts and disposal costs. The GEIS is expected to be issued for public comment in early 1979. NRC also intends to publish for public comment proposed rules related to uranium milling no later than when the final GEIS is published, scheduled for August 1979.*

Tailings Management Proposals

During fiscal year 1977, the NRC embarked on an intensive program to resolve the tailings management issue. Performance objectives were established for the milling industry covering (a) the siting and design stability of tailings isolation areas, (b) operating criteria during the life of the mill, and (c) the final tailings area reclamation plan, including the requirement that financial arrangements be made to assure the availability of sufficient funds to complete the full reclamation project. The industry has responded by proposing various innovative schemes keyed to the specific geohydrological characteristics of the proposed tailings management sites. NRC's posture has been that the preferred tailings disposal procedure is burial below the natural grade, accepting burial above grade only in areas where the final reclamation configuration would result in erosion resistance characteristics comparable to that of below-grade burial.

Typical tailings management plans proposed by industry for new mills currently undergoing licensing review include:

- Disposal of slurried tailings in a mined-out open pit which will have been refilled with compacted overburden above the groundwater table and had its bottom and side walls lined with compacted clay. During operations, standing liquid would be decanted from above the tailings and evaporated from a lined pond constructed on the surface. Following drying, the tailings would be covered with sufficient compacted clay, overburden and topsoil to reduce gamma radiation essentially to background levels, reduce the radon flux to

*On November 8, 1978, the Uranium Mill Tailings Radiation Control Act (P.L. 95-604) became law, giving NRC direct regulatory authority over tailings. See Chapter 1.

no more than twice that of the surrounding environs and permit revegetation to the extent existing before the land was disturbed. Impoundment of tailings would take place in stages, thereby permitting staged reclamation of mined out areas during mill operations. When the mine ceases to operate, the evaporation pond dry solids and liner would be buried in the last impoundment area to be reclaimed.

- Dewatering of tailings to about 25 percent moisture before disposal in a mined-out pit which has been backfilled and lined on the bottom as described above. Because of the reduced amount of tailings solution available to migrate coupled with the decreased mobility of the toxic materials left in the tailings, lining the pit side walls would not be considered necessary. This would both eliminate a significant item of cost and increase the tailings disposal capacity of the pit. Staging of the operation and reclamation would be carried out as above.
- Staged discharge of tailings into cells previously excavated below the existing grade and lined on the bottom and sides with a synthetic lining material. The cells would be surrounded by above-grade embankments to provide adequate volume for an evaporation pond and to prevent surface runoff from reaching the cells. Tailings would be deposited only to a depth that would allow for covering it with sufficient overburden and topsoil to meet the gamma radiation and radon flux objectives without creating an above-ground mound. The area would be reclaimed by contouring to the natural ground level while excess material from the embankments would be used for reclaiming mine areas or disposed of on the mine waste dump.
- Discharging slurried tailings into a surface impoundment at the head end of a natural valley where the area is surrounded on three sides by natural hills and by a dam constructed on the lower fourth side. The basin floor would be lined with compacted clay which is keyed into the clay core of the dam. After mill shutdown and a drying out period, the waste would be covered with compacted clay, overburden, and topsoil as in the previously described plans. Final contouring of the reclaimed area would provide for a gentle slope away from the dam and toward a concrete spillway designed to divert water runoff away from the embankment and maintain surface integrity over the long term.

Assistance to Agreement States

The NRC is furnishing technical assistance to two States in assessing the environmental impacts of their uranium mill licensing actions, and expects to expand this activity as the result of an offer published by the Commission to extend such assistance to Agreement States on a trial basis. Section 274 i. of the Atomic Energy Act authorizes the Commission to provide technical assistance to any State or group of States "as the Commission deems appropriate." It is the Commission's belief that Agreement States which license uranium mills would benefit from NRC technical assistance designed to help the States conduct environmental assessments of their licensing actions. At present, most Agreement States do not prepare written assessments comparable to those of the NRC. A documented assessment for each major mill licensing action in Agreement States would be helpful in exploring the issues and alternative courses of action available in each case. While this document need not be identical in scope to those prepared for mills licensed by the NRC, they should, as a minimum, treat the most important environmental aspects of milling operation and tailings waste management and disposal, as well as siting and radiological assessment. Since licensing practices of States must be viewed in terms of their legislative underpinnings as well as the resources and expertise available to the States, the Commission, after evaluating options available to it, concluded that an offer of assistance was the most prudent course of action.

Technical assistance was provided by NRC to the State of Colorado in assessing the potential environmental impact of a heap leach operation conducted by Ranchers Exploration Company at Naturita, Colorado and the installation of a new tailings impoundment area at the Cotter Corporation uranium mill near Canon City, Colorado. In addition, at the request of both

Colorado and the U.S. Forest Service, NRC continued to provide technical assistance in the preparation of the environmental impact statement for Homestake Mining Company's new mill near Sargents, Colorado. The draft environmental statement for this project was issued in July 1978. NRC is also assisting in assessment of the potential impact of another Ranchers heap leach project proposed for Durango, Colorado.

Under a similar agreement with New Mexico, NRC is providing technical assistance in assessing potential environmental impacts of a mill proposed at Marquez by Bokum Resources Corporation and the Mount Taylor mill project proposed at San Mateo by Gulf Mineral Resources Company.

REPROCESSING-RECYCLE PROCEEDINGS TERMINATED

U.S. consideration of whether to permit recovery of plutonium from used water-cooled power reactor fuel and its recycling into fresh fuel was halted by a Commission decision of December 23, 1977 (see also 1977 NRC Annual Report, pp. 45-47). This action had a significant impact on domestic and international nuclear planning and projects as well as regulatory direction.

All U.S. light-water-cooled power reactors are fueled with uranium enriched slightly in the isotope uranium-235. During reactor operation, a quantity of the uranium is converted into plutonium. When the useful life of the fuel is over, considerable amounts of fissile uranium and plutonium remain which can be recovered by chemical reprocessing and manufactured into new fuel for recycling in light water reactors. However, objections have been raised against a "plutonium recycle economy," primarily concerning questions of national security and nonproliferation of nuclear weapons.

The NRC completed the first phase of its public hearings on the issue in February 1977. Before the next phase could be taken up by the GESMO hearing board, President Carter, on April 7, 1977, issued a statement in which he said the commercial reprocessing and recycling of plutonium produced in the U.S. nuclear power programs would be deferred indefinitely.

On October 4, 1977, the Commission was advised that the President believed his nonproliferation initiatives would be assisted both domestically and internationally if the GESMO proceedings were terminated. In light of events, and after receiving public comments on the President's views and on several specified alternative courses of action, the Commission decided at public meetings in December 1977 to terminate the GESMO proceeding. (A series of cases challenging the Commission's December 23, 1977 order have been consolidated in the Court of Appeals for the Third Circuit—see Chapter 13 under "GESMO Litigation.")

Effects on Reprocessing Plants

The Commission's order dated December 23, 1977, and Memorandum of Decision dated May 8, 1978, terminated the GESMO proceedings and actions on plutonium recycle-related license applications except for those portions of proceedings which involve spent fuel storage, disposal of existing waste and decontamination or decommissioning of existing plants. The actions had the following specific effects:

Exxon Application. The NRC staff ended its review of a 1976 application by Exxon Nuclear Company for licenses to construct and operate a large nuclear fuel recovery and recycling center on the Department of Energy's reservation at Oak Ridge, Tennessee. The proposed facility, which was in the preliminary design stage, had been planned to store up to 7,000 metric tons of spent fuel and to process up to 2,100 metric tons per year. The staff's action did not, however, deny the application.

Barnwell Plant. Licensing reviews being conducted by the staff were ended on Allied General Nuclear Services' Nuclear Plant Separations Facility at Barnwell, S.C., on which substantial construction had been completed under AEC permits dating to 1972. Among other things, this facility was designed to reprocess some 1,500 metric tons of spent fuel per year.

While undertaking only work needed to preserve the licensing effort already expended, the NRC staff terminated operating license actions on the separations facility, the uranium hexafluoride production facility, the waste solidification facility, and the plutonium product conversion facility.

The separations facility has been completed, except for several design changes which may be instituted as a result of preoperational testing, as has the uranium hexafluoride facility which is designed to receive uranyl nitrate separated from reprocessed fuel.

No final design had been received by the NRC for the plutonium product facility which was to convert plutonium nitrate into a form suitable for transportation and as a feed stock for fabrication into recycle LWR reactor fuel. Also, no final design had been developed for the facility that would be required to convert into solid form the high-level radioactive liquid wastes resulting from reprocessing.

NRC licensing activities regarding the fuel receiving and storage stations are essentially complete, with Safety and Environmental Reports having been issued in January 1976. Hearings were being held in a "pending" status, since Allied General Nuclear Services has indicated it no longer considers their operation to be a prudent commercial risk.

West Valley, N.Y., Plant. The GESMO decision had no practical effect on the inactive reprocessing plant of Nuclear Fuel Services, Inc., at West Valley, N.Y. This facility—the only commercial reprocessing plant to operate in the United States—has been shut down since 1972, and the licensee announced in 1976 its decision to withdraw from the reprocessing business. The provisional operating license will be modified to prohibit reprocessing of spent nuclear fuel. (See also section, "Other Fuel Cycle Activities," in this chapter.)

Effects on Spent Fuel Disposition

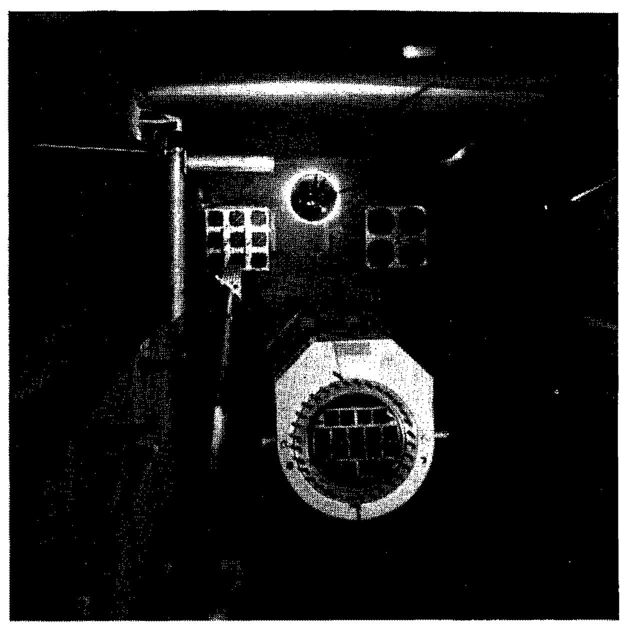
From its inception, the U.S. commercial nuclear power industry has provided storage pools at light-water power reactor sites with capacities for about one and one-third full reactor core loads. Thus, with a three-to-four year reactor reload cycle, onsite storage pools would be capable of holding the discharge from an annual refueling with sufficient room to unload all of the fuel if necessary. It had been planned that reactor fuel discharges after about six months of cooling would be transported to spent fuel reprocessing plants and the resulting wastes ultimately placed in a Federal repository.

Termination of the GESMO proceedings resulted in postponing indefinitely any reprocessing of commercial spent fuel. Also, the present Department of Energy target date for operating a national waste repository, which might accommodate spent fuel elements, has been set back from 1985 to some time in the period 1988 to 1993. Thus, for the immediate future, the growing accumulation of spent fuel discharged from nuclear power plants must be stored either in pools at the reactor sites or in new, independent storage installations.

SPENT FUEL STORAGE ACTIONS

Draft Environmental Statement

The interim spent fuel storage problem was addressed in a draft "Generic Environmental Impact Statement on Handling and Storage of



A shipping cask can be seen in the center of this picture of the storage pool at the General Electric facility in Morris, Ill. In the top left portion can be seen the rack for boiling-water reactor fuel bundles. The rack on the right holds pressurized-water reactor fuel bundles.

Spent Light Water Power Reactor Fuel" (NUREG-0404), issued by the NRC staff in March 1978.

The staff found that commercial spent fuel generated through the year 2000 can be accommodated in a safe and environmentally sound manner either by modification of storage pools at reactor sites or by providing independent spent fuel storage installations.

Extensive public comments will be taken into account in the final statement, scheduled for completion early in 1979. Meanwhile, a number of actions are being taken to authorize pool expansions and to prepare for licensing of offsite storage. By September 30, 1978, expansion applications had been received for 50 operating reactors and 36 had been approved.

Licensing Criteria

The development of regulatory guidance regarding the interim storage of spent fuel has received high priority. A proposed rule, 10 CFR Part 72, "Licensing Requirements for the Storage of Spent Fuel in an Independent Spent Fuel Storage Installation," was published for public comment in October 1978.

The proposed rule applies only to "aged" fuel, i.e., fuel which has decayed for more than one year since removal from the reactor. Such fuel, will have lost its short-lived radionuclides by decay. Therefore, the independent spent fuel storage facility can be designed to provide a level of protection of the public similar to that required at operating reactors without the need for a high degree of protection from such weather extremes as tornadoes, or from tornado-generated missiles. A principal feature of the proposed rule is that it does not require separate reviews for authorization of construction and operation. By thus providing for a single licensing action, the rule would increase licensing efficiency. Associated regulatory guides updating previously published staff positions are being prepared.

Licensing Reviews

The staff reviewed a topical report by Stone and Webster Engineering Corporation, "Independent Spent Fuel Storage Facility," containing a conceptual design for a standard installation to be located on the site of a parent facility such as a nuclear power station. A letter of approval for the conceptual design was issued in July 1978. The pool storage installation could hold up to 1,300 metric tons of uranium dioxide, equivalent to the volume of spent fuel which would be discharged during about 35 years of operation of a 1,000-MWe nuclear power station. The Stone and Webster design takes advantage of site data already acquired in connection with the construction of the parent facility; in addition, some logistical support from the parent facility would be available to the storage facility. The NUS Corporation is expected to submit a similar design in fiscal year 1979.

A *Federal Register* notice of opportunity for a hearing was published on August 18, 1977, concerning the General Electric Company's application for expansion of the storage capacity of its Midwest Fuel Recovery Plant at Morris, Ill., from 750 metric tons to 1,850 metric tons. This resulted in the filing of petitions to intervene by the Attorney General for the State of Illinois and the Natural Resources Defense Council. In the meantime, however, the Department of Energy (DOE) issued a policy statement on October 18, 1977, proposing that the Government accept spent nuclear fuel from utilities for interim storage and ultimate disposal. General Electric consequently requested that the Atomic Safety and Licensing Board suspend indefinitely further proceedings in the case until the company could determine its intentions for the future. This request was granted. General Electric has requested, however, that NRC proceed with its technical review of the application since work on design of the facility was continuing.

In accordance with its 1977 policy statement, DOE has requested, and NRC staff has provided, guidance regarding a potential license application for a DOE interim spent fuel storage installation. The NRC staff also has provided guidance to the Tennessee Valley Authority on licensing criteria and procedures in relation to a possible application next year for a similar facility which could potentially satisfy national requirements.

In addition to authorizations for pool expansions at reactor sites, some utility licensees have sought approval for the receipt and storage of spent fuel at one nuclear station from another to alleviate specific pool capacity problems. During 1978, approval was given Carolina Power & Light Company for receipt and storage at its Brunswick Station of spent fuel from its H.B. Robinson Plant Unit 2. At year-end, applications were under review from Commonwealth Edison Company for the intersite transfer and storage of spent fuel between its Dresden and Quad-Cities Stations, and from Duke Power Company for receipt and storage of Oconee Nuclear Station spent fuel at its McGuire Nuclear Sation.

OTHER FUEL CYCLE ACTIVITIES

NFS's West Valley Facility

The future of the Nuclear Fuel Services' West Valley site is yet to be determined. On February 25, 1978, the President signed Public Law 95-238 which, among other things, directed the Department of Energy to submit to Congress a study of the West Valley site. This study was conducted in cooperation with NRC and other Federal agencies. One of its key objectives is to recommend allocation of responsibility for the site among the Federal Government, the State of New York and present industrial participants. NRC has been providing the regulatory perspective on issues such as waste disposal and decommissioning. Late in the year, DOE issued a draft report on the study for comment by other agencies and the public.

NRC staff has continued its confirmatory studies of the effect of natural phenomena on the dormant West Valley plant. Analysis of the effect of an earthquake on the separations plant has confirmed the staff's previous conclusion that there would be no undue risk to the health and safety of the public or of employees. It was noted that additional analysis would be required for alternate uses of the plant. Analysis of the spent fuel and high level waste storage portions of the plant is nearing completion.

At the NRC's request, Nuclear Fuel Services has been compiling information which will be useful in decommissioning the facilities, should that become necessary.

Effects of Natural Phenomena On Plutonium Facilities

NRC regulations require that plutonium processing and fuel fabrication plants proposed for licensing must be evaluated to determine that there is reasonable assurance of protection against natural phenomena such as floods, hurricanes, earthquakes and tornadoes. Already licensed plutonium pilot plants and research and development plants must also be examined with the objective of improving their ability to withstand natural phenomena and to protect the health and safety of the public. Accordingly, the staff is evaluating six fuel fabrication facilities that are licensed to possess and process 5 kg (11 lb.) or more of unencapsulated plutonium.

Experts in seismology and geology, surface hydrology, normal and severe weather phenomena, structural analysis, source term characterization, meteorological dispersion, demography, ecology, and radiological impact are participating in the program. Site characterization regarding seismicity, flooding potential, and severe wind occurrence has been completed for four of the six sites. Engineering models have been completed which describe the three dimensional wind speeds in cyclonic storms and the dispersion characteristics of both high velocity straight-line winds and tornadic winds. These models have been used to compute dispersion from releases at two sites and are in place to process the wind speeds associated with damage scenarios at the remaining sites. Regions of the plant where structural failure would be most likely to cause significant release have been identified for five of the six plants, and structural analyses on two of the facilities have disclosed damage thresholds and damage scenarios associated with both earthquake and severe wind.

When assessment of the six facilities is complete, it will provide a basis for determining the extent of backfitting necessary to protect the public and for developing siting and general design criteria for future plants. Three of the six reviews are expected to be completed in early 1979.

Price-Anderson Study

A contractor study is being conducted to determine the quantities of a number of radioactive materials released in dispersible and respirable form that could cause about $140 million in damages. Losses of $140 million are currently the maximum amount covered by privately available nuclear liability insurance. Government indemnification under the Price-Anderson Act could be used to provide additional funds to compensate victims for damages sustained in a nuclear incident. The materials included in the study are those associated with the processing and fabrication of highly enriched uranium and plutonium, and with the preparation of large radioisotope sources. In addition, selected hypothetical accidents involving spent fuel in storage and transport are being examined.

The preliminary results of the study indicated that there would be no apparent need to indemnify licensees possessing and using highly enriched uranium, plutonium, and spent fuel, but were inconclusive about certain radioisotopes because of a lack of information regarding the specific operations and conditions involved in their use. The requisite information is being obtained.

Decommissioning of Babcock and Wilcox Facilities

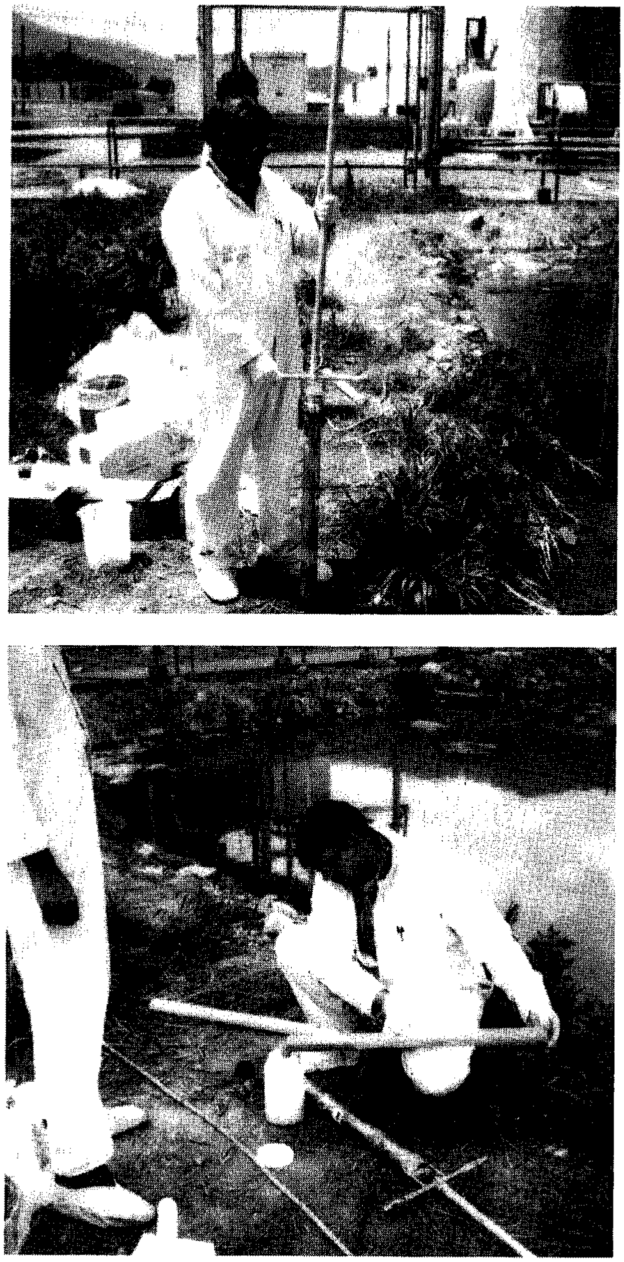
The Babcock and Wilcox Company (B&W) has submitted a plan for the decontamination and decommissioning of its high-enriched uranium fuel fabrication facility at Parks Township, Pa. B&W decided to terminate operations in the last quarter of 1977, and by June of 1978 had decontaminated and disposed of essentially all its equipment. The plan also includes decommissioning provisions for the scrap recovery operations for high-enriched uranium that were performed at B&W's nearby Apollo, Pa., operations. This facility is also being decontaminated and essentially all of its equipment is expected to be disposed of by the end of 1979. These activities are being conducted in accordance with the NRC-issued license.

ENVIRONMENTAL SURVEY OF THE URANIUM FUEL CYCLE

In 1974, the Commission published WASH-1248, "Environmental Survey of the Uranium Fuel Cycle," which assessed the environmental impacts associated with the nuclear fuel cycle in support of a typical 1,000-MWe light water reactor. The environmental impacts associated with the nuclear fuel cycle were summarized in the Commission's regulations, in Table S-3 of 10 CFR Part 51.2. In adopting this rule, the Commission noted that these environmental impacts would be re-examined from time to time to accommodate new technology and information. These values are used in environmental impact statements which are prepared in connection with light water reactor license proceedings.

The U.S. Court of Appeals, in 1976, acting on a suit filed by the Natural Resources Defense Council, held that rulemaking procedures used to promulgate the fuel cycle rule were inadequate and that the rule was inadequately supported by the record with respect to the environmental effects of reprocessing and radioactive waste disposal. In response to the D.C. Circuit Court's decision, NRC reopened the rulemaking hearing (Docket No. RM—50-3) on Table S-3 for reconsideration of environmental impacts associated with spent fuel reprocessing and radioactive waste disposal. In October 1976, the NRC published Supplement 1 to the original Environmental Survey report, giving the results of a new "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle" (NUREG-0116). Following public review of the Supplement, the NRC published report number NUREG-0216 as Supplement No. 2, giving the staff's responses to the many comments received on NUREG-0116. The NRC also promulgated an interim rule incorporating new environmental impact information into Table S-3. The interim rule was to remain in effect for 18 months until public hearings and other proceedings were completed and a final rule adopted. The public hearing on a proposed final rule was initiated in January 1978.

Meanwhile, in response to an NRC appeal, the U.S. Supreme Court found in April 1978 that the appellate court decision had "improperly intruded" on the decision-making process entrusted to the Atomic Energy Commission by the Congress and subsequently remanded the case to the appellate court for reconsideration of its decision regarding the inadequacy of the record on reprocessing and waste management. The NRC decided to complete its ongoing rulemaking proceeding on reprocessing and



An NRC specialist in nuclear chemistry, Dr. Dan Montgomery, takes environmental samples from settling ponds at the Nuclear Fuel Services, Inc., fuel processing facility at Erwin, Tenn.

waste management as part of its commitment to review and update Table S-3 as needed. All parties to this proceeding submitted concluding statements in May and June 1978. In September, the Commission extended the effectiveness of the interim rule to March 14, 1979.

Radon Estimates Increased. With the promulgation of the fuel cycle rule (Table S-3) in 1974, the Commission noted that the environmental impacts associated with the nuclear fuel cycle would be re-examined from time to time to accommodate changing technology and new or additional information. In this regard, and in response to a rulemaking petition filed by the New England Coalition on Nuclear Pollution, an amendment to the fuel cycle rule was announced in the *Federal Register* on April 14, 1978, giving notice that the Commission had decided to remove the value provided in Table S-3 for releases of naturally-occurring radioactive gas, radon, during mining and milling operations and to permit litigation of the issue in individual licensing proceedings. The NRC staff has revised upward its estimates of radon releases, and presented the higher estimates in testimony pertaining to several individual nuclear power plant licensing proceedings during the latter part of 1978. Different hearing boards conducting proceedings for three licensing cases all concluded that the increase in radon concentration above natural background is so small that the environmental impact and effects on human health cannot be significant. However, because of the expense and time of providing expert witnesses and responding to questions in each hearing, there is strong incentive for a rulemaking action to incorporate a radon environmental release estimate in Table S-3 at the earliest possible date.

Accordingly, NRC staff is seeking to develop better estimates of radon releases from mining and milling operations, including the long-term releases from mill tailings and from inactive uranium mines. The NUS Corporation, under contract to NRC, is investigating the radon emissions from inactive open pit mines. In addition, the NRC has awarded contracts to Battelle Pacific Northwest Laboratories to investigate radon releases from underground open pit mines.

Overall Updating of Survey Begun. Since the amendments to the fuel cycle rule described above are limited to specific portions of the fuel cycle or to individual effluents, they do not completely fulfill the intent of an overall periodic updating of the original rule. Therefore, the Commission has awarded a contract to the NUS Corporation for an overall updating of the environmental survey. It will re-evaluate the format and content of Table S-3 to determine the most effective way of characterizing environmental effects and will consider new concepts and technologies, such as centrifuge enrichment, mining by *in situ* leaching, or spent fuel disposal.

New information is expected to be available to permit a more detailed consideration of occupational exposure of workers, decommissioning of facilities, and the impact of nonradiological effluents. A draft updated Environmental Survey is scheduled for completion near the end of 1979. It will evaluate the environmental effects of providing the fuel to operate a nuclear power plant over its 30-40-year lifetime and of disposing of the spent fuel and radioactive wastes generated during this period. Because of current national policy, the study will assume that U.S. industry will not reprocess spent fuel during the period and the major study effort will be based upon the assumption that there will be interim storage and disposal of spent reactor fuel. However, an analysis adequate to bound the estimated environmental effects from spent fuel reprocessing has been carried out in the recent Hearing Board (Docket No. RM—50-3, noted above) and it is planned that the updated survey will also include some consideration of this alternative.

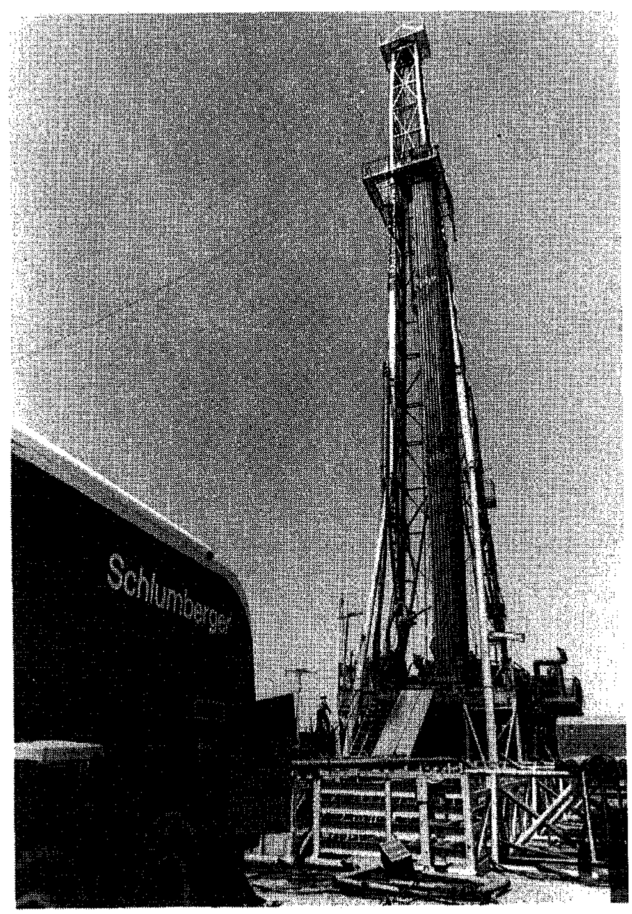
RADIOISOTOPES LICENSING

Radioactive materials are used widely in industrial applications, consumer products, medical diagnosis and treatment, basic and applied research, and in academic fields. The NRC administers approximately 8,000 licenses covering the above activities and processes 6,000 to 8,000 applications for new licenses, license amendments and license renewals per year. This represents about half of the approximately 19,000 nuclear material licenses in effect in the United States. The other half of the licenses in effect are administered by 25 States under regulatory agreements with the NRC (see Chapter 8). The NRC system of licensing the possession, use, transfer, and disposal of byproduct, source, and special nuclear material is designed to assure protection of public health and safety. At the same time, it is important that this licensing program be conducted in such a way as to be responsive to the large number of applications received per month without delay of needed services or economic losses to applicants.

In March 1978 a pilot study was begun to determine the advantages and disadvantages of decentralizing radioisotopes licensing to NRC regions. This study will continue for a two-year period.

Well Logging. Due to the country's energy situation, the largest number of oil and gas well drilling operations in the history of the industry were begun in 1978. In addition, the use of low energy gamma emitter sources and neutron sources for logging shallow bore holes for mineral deposits and the location of coal bearing deposits, a spin-off from the oil and gas well logging industry, also increased. Independent operators and small companies with one or two logging trucks have been licensed to engage in these activities in increasing numbers.

Nuclear techniques are used extensively in these explorations. The nuclear measurements fall into the general catagory known as "well logging." The "log" is a continuous recording



Well logging trucks in position to use nuclear sources for precision subsurface measurements of gas and oil wells. Such measurements, along with in-well measurements of electricity, sound and natural radiation are employed to obtain underground data essential to the search for energy.

of the value of physical parameters as a function of depth in the drill hole. The instrument package, i.e., well logging "probe" or "sonde," is lowered to the bottom of the hole at the end of a cable. The cable or "wire line" transmits power to the sonde and data signals to the surface. Small quantities of radioactive tracer materials and well logging devices containing sealed sources of gamma radiation and neutrons are used extensively in these operations. The porosity of the formation, the bulk density of the formation, salinity, etc., are examples of the information obtained from the use of sealed sources in underground formations. From these measurements, in combination with other measurements, one can obtain information about such things as liquid saturation, gas saturation, and the presence of coal and mineral deposits in a particular formation. Well logging techniques using small quantities of radioisotope tracer materials below ground provide information on such things as cement channel top locations when the well casing is cemented in place, fracture zone locations, well perforations, etc. The NRC and Agreement States license a large number of service companies to perform well logging and mineral logging operations. In September 1978, NRC published proposed regulations on procedures for dealing with radioactive sources lost down drill holes.

Industrial Radiography. Gamma radiation sources are used for nondestructive testing of materials used in the construction of power plants, ships, submarines, airplanes, bridges, pipelines, etc. There are about 300 NRC licensees with 15,000 radiographers involved in industrial radiography. To reduce the number of overexposures from industrial radiography activities, the NRC published for comment, on March 27, 1978, revisions to 10 CFR Part 34 which require additional safety features in the design and handling of radiography equipment. (See Chapter 10.)

Portable Gauges. With the expansion of activity in the construction industry there has been a proportionate increase in the use of portable moisture-density gauges containing gamma and neutron sources. The use of these gauges provides a rapid means of verifying quality control of construction in the field.

A technologist withdraws a radioactive drug from lead-shielded vial, using shielded syringe. Use of shielding minimizes radiation exposure to the technologist's hands. Note that the technologist is wearing thermoluminescent dosimeters on wrist and index finger to measure radiation exposure to the hands.

Nuclear Medicine

Radioactive materials are used in medicine to perform an estimated 40 million medical procedures per year at an estimated cost of $2.2 billion. These diagnostic and therapeutic procedures are performed by approximately 12,000 NRC and Agreement State licensees.

Medical Policy Statement. During 1978, the NRC released for public comment a proposed policy statement and rule changes concerning the medical uses of radioisotopes which are designed to further assure the safety of employees, patients and the public. (See Chapter 10.)

Other Licensing Matters. The NRC staff is working with the staff of the Food and Drug Administration (FDA) to develop a Memorandum of Understanding whereby medical devices containing byproduct, source, or special nuclear material would continue to be regulated effectively, but without duplication of effort by the two agencies.

Uses in Consumer Products

Some consumer products containing small amounts of certain radioactive materials may be distributed without the individual consumer having a specific license to possess and use the product. The NRC authorizes such general distribution only after determining that the product has sufficient benefit to the consumer, and that it presents little risk through normal use or misuse. Products reviewed and approved for such general distribution include certain smoke detectors for residential use, and timepieces.

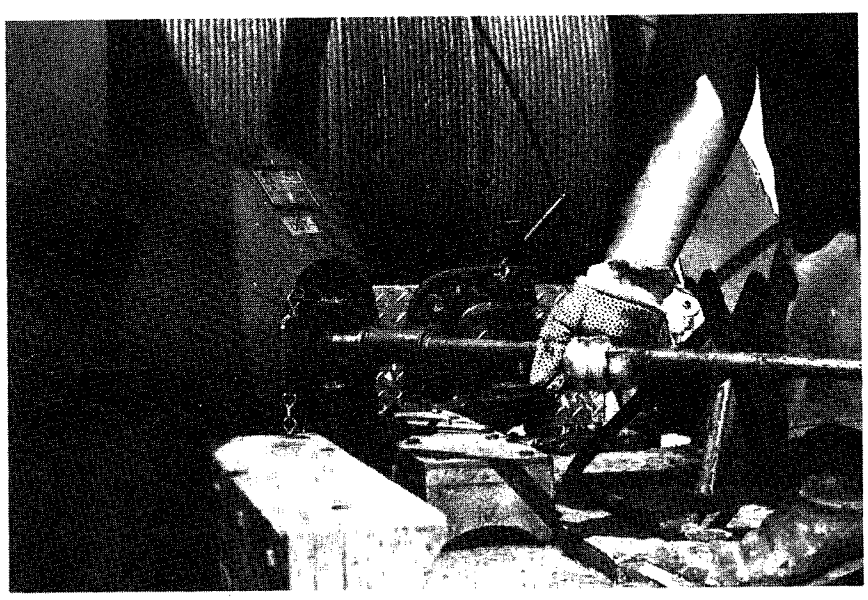
In September 1978 the NRC initiated a contract for a two-year study of consumer products containing radioactive materials. It will involve the issuance of a generic environmental impact statement and a re-evaluation of existing policy in light of findings in the environmental review. Initial efforts in the review will concentrate on the health and safety aspects of the use of ionization smoke detectors.

Smoke Detectors. There has been a tremendous growth in the use of ionization-type smoke detectors containing americium-241. More than seven million were distributed in 1977 alone. At the end of fiscal year 1978 more than 50 NRC licensees were authorized to distribute such products.

Continuous Liquid Display Watches. Since 1975, the NRC has authorized distribution of liquid crystal display (LCD) watches containing

At left, an engineer uses a portable nuclear moisture-density gauge to measure the density (compaction) of a road bed. Below, a workman on the bed of a logging truck uses an extension tool for the removal of a neutron source from the transport shield for placement in a logging tool.

A typical portable radiography device is shown above. The device is normally used to perform radiography at temporary job sites.



tritium. The LCD watches differ from other watches containing radioactive material in that the tritium is contained in sealed glass ampules and is used as a radioluminescent source to backlight the LCD. Approximately 12 licenses have been issued authorizing distribution of these watches and some 30 million have been distributed to retailers for public sale.

Transportation of Radioactive Materials

Transportation of radioactive materials is regulated at the Federal level principally by the NRC and the Department of Transportation (DOT). Under a memorandum of understanding with DOT (see 1977 NRC Annual Report, page 54), NRC is the standards-writing body for "Type B" packages (those whose content of radioactive materials requires that they be safely retained in their containers under both normal and accident conditions) and for packages containing fissile material. NRC also makes independent evaluations of package designs submitted by applicants and serves as a technical adviser to DOT regarding packages used for the import and export of radioactive materials.

Package designs used by contractors for the Department of Energy are reviewed and approved by that agency. An informal program under which the NRC has been reviewing such package designs has been conducted during the past year. These NRC reviews are not binding on the DOE.

NRC Certifies Safe Plutonium Package

On August 4, 1978, NRC certified to the Congress, in conformity with Public Law 94-79, that a safe plutonium container had been developed and tested which would not rupture under crash and blast testing equivalent to the crash and explosion of a high-flying aircraft. The law required that NRC prohibit its licensees from transporting plutonium by air until such certification could be made. (Exception was made for certain medical devices.) Development of the safe container was the result of intensive effort extending over three years, involving design and extensive testing at DOE's Sandia Laboratories and reviews by the NRC's Advisory Committee on Reactor Safeguards and the National Academy of Sciences' Assembly of Engineering (see also 1975 NRC Annual Report, page 66; 1976 NRC Annual Report, pages 54-59; and 1977 NRC Annual Report, pages 56 and 57.)

Low-Level Radioactive Shipments

A task force of NRC and DOT staff completed in July 1978 a draft report on the adequacy of existing requirements for the shipment of material containing a low level of radioactivity. The study followed a truck accident in September 1977 in which a shipment of uranium concentrate (yellow cake) was spilled onto a highway near Springfield, Colo. The preliminary findings of the task force were that:

(1) Carriers are basically responsible for the radioactive cargo in transit and should prepare an emergency response plan for controlling any spilled radioactive material, protecting the public, and cleaning up any spill site.

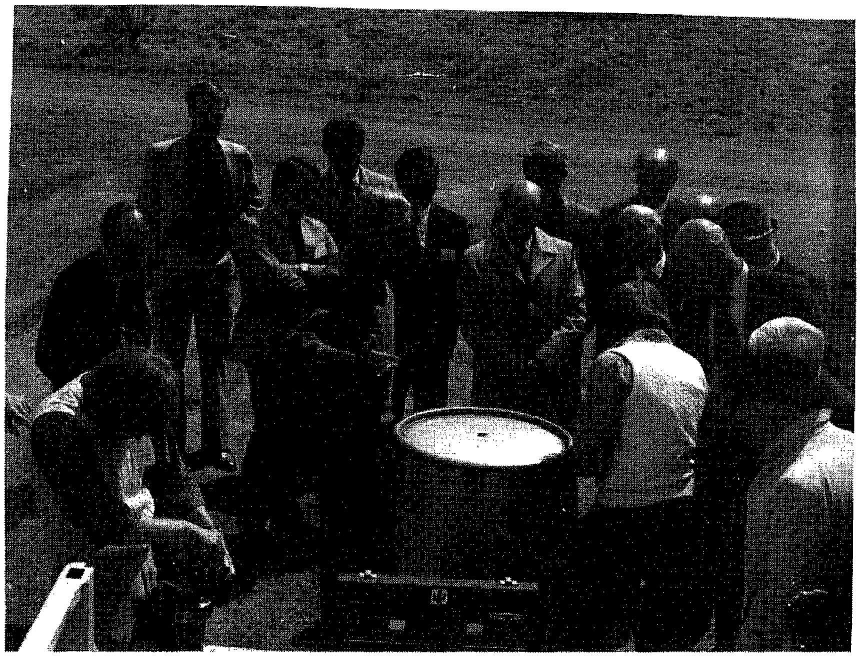
(2) Shippers are responsible for providing hazard information regarding their shipments, and should prepare an emergency response plan for conveying that information.

(3) Because of the low hazard associated with yellow cake, the cost of requiring more accident-resistant packaging or package closures for this material would not be matched by the benefits derived.

The draft report was to be submitted to the Commission in January 1979 with recommendation that it be published for public comment which will be taken into account in the final report.

Safety of Transportation Workers

Previous NRC studies indicate that some of the exposures received by transportation employees were attributable to unnecessary contact with the packages of radioactive material. In July 1978, NRC and DOT jointly issued two manuals and two posters instructing employees and their supervisors on how to avoid such contact. The manuals—"How to Handle Radioactive Material Packages—A Guide for Cargo Handlers" and "All About Radioactive Material



During the development of the package design to meet the NRC criteria for the safe air transport of plutonium, the ad hoc Committee on the Transportation of Plutonium by Air of the Assembly of Engineering of the National Academy of Sciences met at Sandia Laboratories to witness tests as part of their independent review of the program.

Packages—A Guide for Supervisors at Cargo Terminals"—also provide basic information on emergency procedures to be used following an accident involving radioactive materials packages.

Environmental Statement on Transportation

In December 1977, NRC released a final environmental statement (NUREG-0170) assessing the impacts associated with the transportation of radioactive materials, including the relative costs and benefits of various modes of transportation.

The study indicated that radioactive shipments are being conducted under the present regulatory system in an adequately safe manner. The environmental statement, letters of comment about it, and other documents are being evaluated to determine whether to terminate the public rulemaking proceeding initiated in June 1975 regarding air transport of nuclear materials.

Transportation Litigation

In *New York vs NRC et al.* (see 1977 Annual Report, p. 57), an amended complaint was submitted in September 1978. The NRC responded to this document, requesting that the complaint be dismissed.

In *United States vs New York City* (see 1977 Annual Report, p. 57), the most significant developments of 1978 were the finding by the DOT that the New York City ordinance was not incompatible with the Hazardous Materials Transportation Act, primarily because no Federal regulation of routing for transportation of radioactive materials had been established in accordance with that legislation. The DOT announced in August 1978 a rulemaking proceeding for routing restrictions of highway movements of radioactive materials, and the NRC is considering some forms of joint participation in that proceeding.

Transportation in Urban Areas

NRC plans to issue late in 1979 a draft generic environmental impact statement on the transportation of radioactive material in urban areas. The statement will be based in large part on a draft environmental assessment to be submitted to NRC by Sandia Laboratories in 1979. Work on this matter began in May 1976.

In-Transit Incidents

In fiscal year 1978, there were 19 transportation events which licensees were required to report to the NRC. These included eight instances of radioactive contamination or radiation levels above permissible levels on packages and 11 reports of lost or stolen material. (Seven of these shipments were recovered.) Two of the events were known to have caused exposures to radiation: a driver and six members of the general public were involved. None of the exposures exceeded 100 millirems.*

Sixty-six other events were called to the attention of NRC. These events either were not reportable or were reportable to DOT or to Agreement States. They included traffic accidents, containers incorrectly suspected of leakage, crushed packages in terminals, etc. None of these events contributed significantly to the risk to the public's health and safety.

Packaging Standards

In June 1978, the NRC amended 10 CFR Part 71 to extend until January 1, 1979, the date for licensees to file descriptions of quality assurance (QA) programs applicable to their transportation activities, including procurement of packaging. This extension responds to interested persons who raised questions about applicability to Agreement State licensees and requested a delay in the effective date for the QA requirements. The short-term delay will have no significant adverse effect on the public health and safety because regulatory provisions and licensing QA provisions are already in effect.

Revision 1 to Regulatory Guide 7.6 on design criteria for shipping cask containment vessels was issued March 1978 to reflect public comments.

In November 1978, NRC issued a revised compendium (NUREG-0383) of all the package designs for which current NRC Certificates of Compliance are in effect in accordance with the requirements of 10 CFR Part 71.

International Standards

During 1978, NRC and DOT continued their joint consideration of whether to adopt into their regulations recent revisions in transportation standards developed by the International Atomic Energy Agency. (See 1977 NRC Annual Report, page 55.) The revised regulations will be published for public comment in early 1979.

*Average annual doses from natural background radiation in the U.S. are in the range of 100 to 125 millirems, but vary from 90 to 200 millirems depending on elevation and amount of radioactive material in rocks, soil, etc. A millirem is one-thousandth of a rem—a measure of dose to body tissue from ionizing radiation biologically equivalent to an exposure of one roentgen of high-voltage X-rays.

Domestic Safeguards

The NRC has been directed by the Congress (PL 95-601, amending Sec. 209 of the Energy Reorganization Act of 1974) to submit a report on the status of the Commission's domestic safeguards program, as a separate document for fiscal year 1978 and as a separate chapter of the NRC Annual Report for succeeding fiscal years. The separate report for fiscal year 1978 is entitled "Annual Report to Congress on Domestic Safeguards, Fiscal Year 1978," which will be published as NUREG-0524. From its initial annual report in 1976 (for fiscal year 1975), the NRC has included a chapter on domestic safeguards and continues that practice with this chapter. It is largely drawn from the mandated separate report, though it is considerably less detailed. The reporting by NRC on domestic safeguards in future annual reports will constitute the mandated report on the matter.

SCOPE OF NRC SAFEGUARDS PROGRAM

Under the Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974, the NRC is responsible for the regulation of safeguards provided by certain of its licensees. NRC safeguards regulatory programs share the common goal of assuring that licensed activities do not pose undue risk to the public health and safety and are not inimical to the common defense and security. The NRC safeguards objective is to develop and require the implementation of measures designed to prevent, deter, detect and respond to: (1) the unauthorized possession or use of SNM; and (2) the sabotage of nuclear facilities. SNM includes plutonium, uranium-233, uranium enriched (to any degree) in uranium-235, or any other material determined by the NRC to be SNM, under Section 51 of the Atomic Energy Act of 1954. SNM does not include source material, such as the natural uranium or thorium from which nuclear fuel is produced, nor does it include by-product material, i.e., reactor-produced radioisotopes for medical or industrial applications. Some nuclear wastes may contain SNM.

The NRC currently has safeguards regulatory control over 19 fuel cycle facilities which are authorized to possess formula quantities of highly enriched uranium or plutonium, transportation activities involving highly enriched uranium or plutonium (about one shipment per month), 70 operating commercial power reactors, and 71 non-power reactors (for research, testing, training or the production of radioisotopes). "Formula quantities" refers to Strategic Special Nuclear Material (SSNM) in any combination, 5000 grams or more, computed by the formula: grams = [grams contained U-235 + 2.5 (grams U-233 + grams plutonium)]. SSNM includes uranium 235—contained in uranium enriched to 20 percent or more in the U-235 isotope—uranium-233, or plutonium.

DETERMINATION OF SAFEGUARDS ADEQUACY

The capability of NRC licensees' safeguards systems to defeat a hypothetical design threat is the test by which the NRC determines whether or not those systems are acceptable from a regulatory standpoint. This hypothetical threat is applied to safeguards associated with power reactor facilities and fuel cycle facilities involving significant quantities of SSNM. The threat comprises two distinct but potentially interrelated events:

- A determined, violent external assault, attack by stealth, or deceptive actions carried out by several persons, assisted by an insider.
- An internal threat of an insider, including an employee (in any position).

Safeguards regulatory requirements in the form of rules and license conditions are imposed on licensees by NRC to attain the desired level of protection. NRC performs licensing reviews (on-site in many cases) to judge the adequacy of licensee safeguards plans covering physical security or material control and accounting.

Safeguards are in place to protect the public against the possible theft or diversion of SNM or sabotage of nuclear facilities. Information available to the NRC does not indicate the existence of a significant near-term threat of theft or diversion involving strategic special nuclear material, or of sabotage.

The development and imposition of NRC safeguards requirements comes about in two ways: specific requirements are set forth in NRC rules and license conditions (including the safeguards plans of the licensee and contingency plans for responding to threats), and the correction of weaknesses discovered during inspections or special evaluations is required.

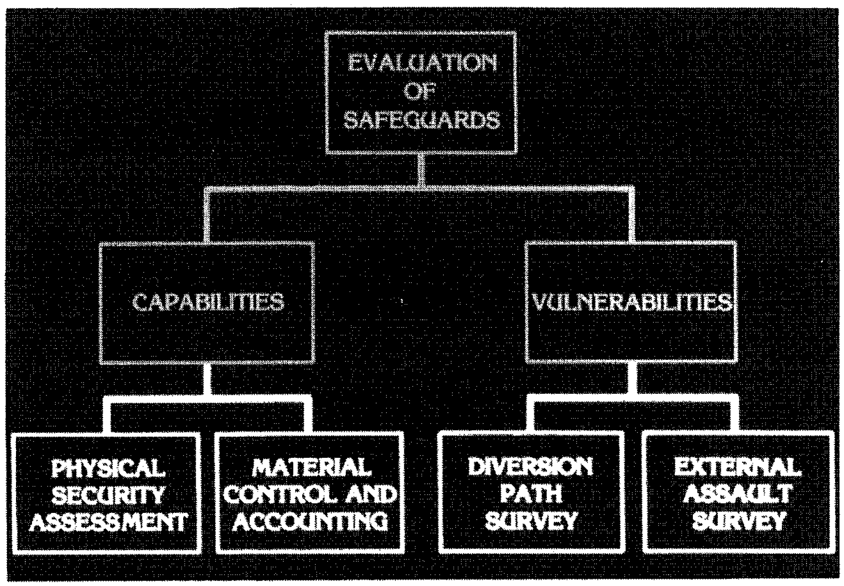
Fuel Cycle Facilities

NRC assesses safeguards adequacy at fuel cycle facilities through inspections and comprehensive evaluations. Inspections are conducted from four of the five Regional Offices, and the evaluations are conducted by special teams from the headquarters and regional inspection staffs. These evaluations determine the capability of licensee safeguards to protect against the hypothetical design threat.

NRC enforcement activities at the 14 fuel cycle facilities authorized to possess formula quantities of SSNM in unsealed, unirradiated form included plant shutdowns for reinventory and Immediate Action Letters identifying additional measures to be taken by the licensee both in material control and accounting and physical protection.

Based on the inspection and enforcement results, NRC concluded that the licensees' actions in response to identified items of noncompliance in their safeguards systems were acceptable. There is, however, one case involving possible falsification of guard training records which was under NRC investigation at the close of the report period.

During fiscal year 1978, inventory differences exceeding regulatory limits were experienced at three fuel cycle facilities. Inventory differences which exceed regulatory thresholds are examined by NRC to determine probable or actual cause. During fiscal year 1978 such examinations—which in some instances included reinventory and plant shutdown—did not identify any factual indication (other than the inventory differences which are of themselves inconclusive) that SSNM had been stolen or diverted during the report period.



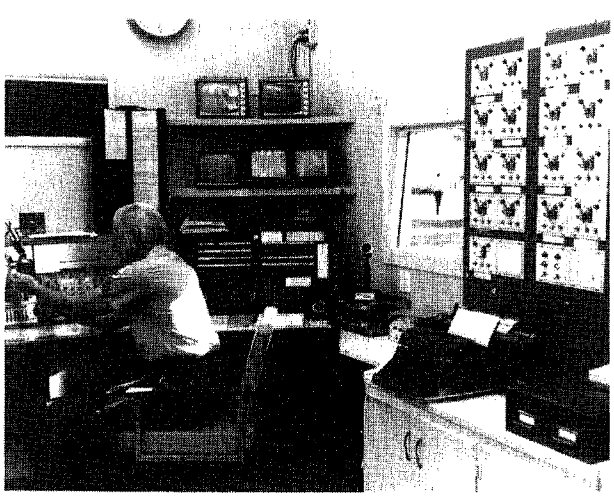
This diagram reflects the main elements of NRC's safeguards evaluation program for nuclear fuel cycle facilities. The goal is to achieve an integrated system of protection in which the elements of physical protection and material control and accounting are put in balance against postulated vulnerabilities and threats.

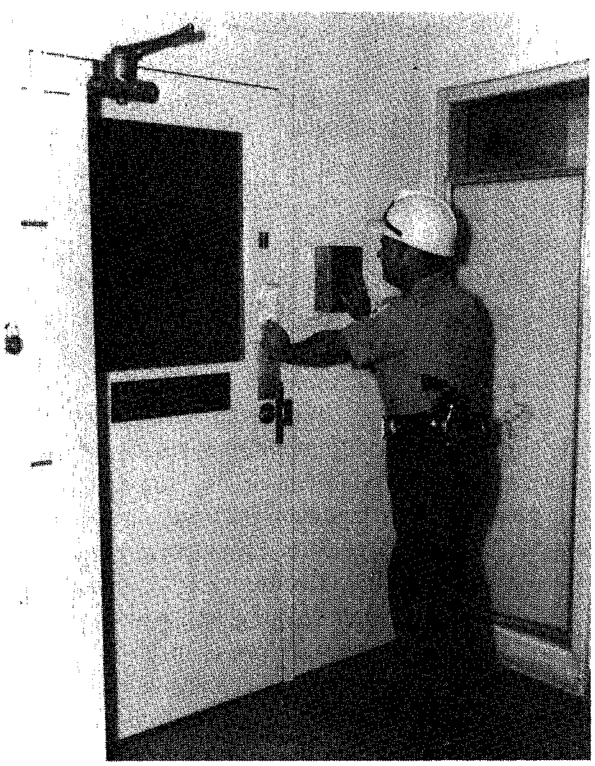
Transportation Activities

NRC ensures the adequacy of safeguards on SSNM shipments through its licensing and inspection programs. The in-transit portions of all domestic shipments of formula quantities of SSNM and the domestic segments of import and export SSNM shipments, including all storage and transfer points, are monitored by NRC inspectors. (Shipments of government-owned SSNM using DOE couriers are a DOE responsibility and as such are not inspected by NRC).

NRC inspectors keep each such shipment under surveillance during the entire period it is in transit. Inspection activities cover the broader areas of material control and accounting, physical protection, and health and safety. The thrust of the inspection activity is to ensure that

Guard forces and alarm systems are the main elements of any plant security system. Criteria for their application in protecting sensitive areas of nuclear plants are spelled out in detailed, rigorous NRC requirements. These photos show a security guard at a secure entrance/checkpoint, and the plant's central security control room. TV monitors in the control room afford continuous surveillance of key passages and doors throughout the plant.





the licensee is making the shipment in full compliance with NRC regulations and license conditions, and with his NRC-approved security plan.

During the period from January through October 1978, there were eight shipments of formula-quantity SSNM inspected, and NRC detected no items of non-compliance. They were all conducted without major incident.

Reactor Facilities

NRC is currently reviewing the adequacy of safeguards at all operating power reactors. The NRC staff has reviewed physical security plans for all of the 70 operating reactors, and, when the plans are fully implemented, these power reactor facilities will be capable of meeting the NRC safeguards adequacy standard. The implementation process is scheduled for completion in 1979. Continuing assessment and confirmation of safeguards adequacy will be assured through an inspection and enforcement program.

The NRC is in the process of evaluating non-power reactor safeguards, particularly with respect to the target attractiveness for theft of several types of fuel elements, the potential of various protective measures, and the physical security effectiveness at the various non-power reactor installations. The staff is also performing an in-depth evaluation of the sabotage potential at non-power reactor facilities, especially those with reactors operating at the higher end of the range of power levels (i.e., above 100 Kw).

NRC inspection and enforcement activities at reactor facilities also provide a means of judging the effectiveness of safeguards. (Starting in mid-1978, resident inspectors began to be deployed at power reactor sites.) NRC has issued a number of Immediate Action Letters which identified additional measures to be taken by the licensees to improve their safeguards systems, but it took no major enforcement actions such as orders or civil penalties during the year. Recent physical protection inspections and investigations of allegations concerning guard training have disclosed evidence of improper guard training record-keeping and possible falsification of training records. In addition, management audits of guard training have been found, in some cases, to be either non-existent or severely deficient. All licensees were informed of these conditions and were advised that NRC would be evaluating each licensee's program for guard qualification and training and would be inspecting their programs for compliance and adequacy.

CONTINGENCY PLANNING

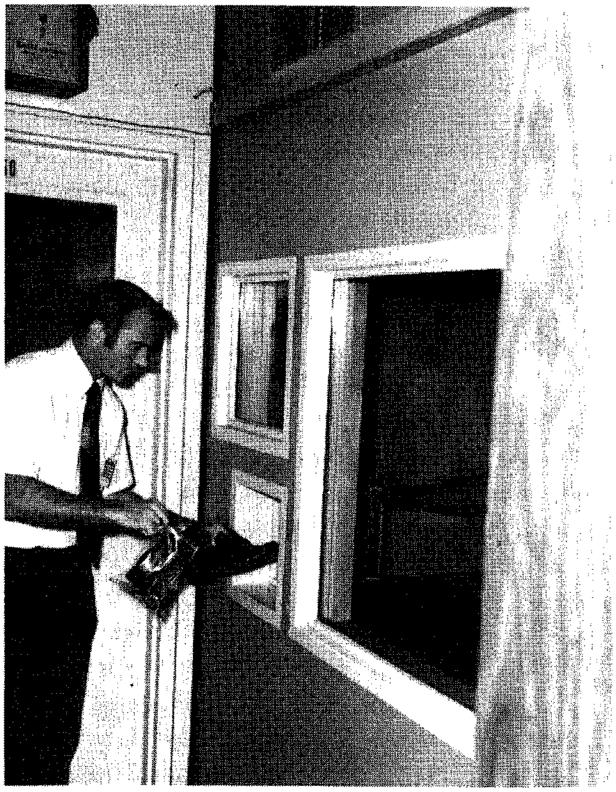
Safeguards contingency plans are developed to deal with threats, thefts, and sabotage relating to special nuclear materials, high-level radioactive wastes, and nuclear facilities. Contingency plans contain: (1) a predetermined set of decisions and actions required to satisfy stated objectives; (2) an identification of the data, criteria, procedures, and mechanisms necessary to make and carry out the decisions and actions efficiently; and (3) a specification of the individual, group, or organizational entity responsible for each decision and action.

During fiscal year 1978, the NRC staff effort was directed toward application of a previously developed contingency planning methodology. At the national level, contacts were made with 82 organizational elements of over 28 agencies and with three national associations. Those organizational elements that can provide useful information or response assistance have been identified, and inter-agency agreements are planned to formalize procedures for requesting information or assistance, communications channels, and other arrangements.

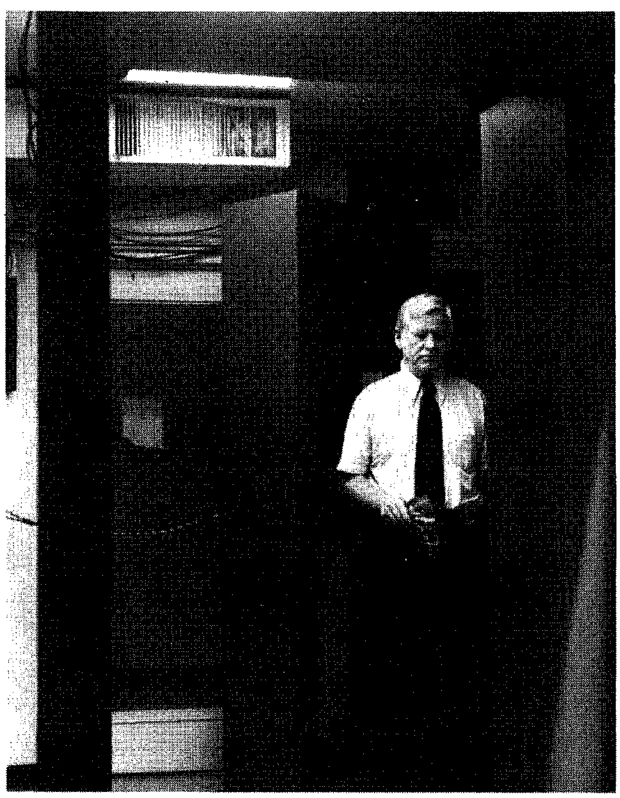
The amendments to 10 CFR Parts 50, 70, and 73 were published as a final rule in March 1978. These amendments require certain licensees to develop and implement acceptable contingency plans for responding to threats, thefts, and industrial sabotage of licensed nuclear materials and facilities.

SAFEGUARDS RESEARCH AND TECHNICAL ASSISTANCE

The NRC safeguards program includes both research (long term, comprehensive efforts) and technical assistance (short term efforts in support of operational assignments). In fiscal year 1978, about $10 million was spent on safeguards research and technical assistance, divided about



Entry/exit search and screening requirements were strengthened in 1978 following several inspections which revealed deficiencies in those areas. Shown here are two techniques used to ensure that unauthorized items are not taken into or removed from sensitive areas.



equally between these two categories. (The Commission approved all safeguards research programs, as the Congress requires.) During this period, the major efforts of the safeguards research program were directed to development of methods for evaluation of safeguards effectiveness. Technical assistance was provided to major program offices to support their current safeguards activities; projects ranged from aiding in the development of NRC's physical security upgrade rule to making improvements in nuclear measurement standards. (See Chapter 11.)

FUTURE SAFEGUARDS PROGRAM

To improve the safeguards protection at facilities and activities under its regulatory authority, the NRC is currently undertaking additional safeguards projects. These projects include:

- A new guard training upgrade rule (for fuel cycle facilities, transportation activities, and power reactors), which became effective in early fiscal year 1979 and which will be implemented over the next two years.
- A physical security upgrade rule for fuel cycle facilities, proposed by the NRC staff. If adopted in its proposed form, this rule would increase the level of protection required by increasing the postulated threat (including emphasis on internal conspiracies). This proposed rule would also require increased protection for nuclear shipments and certain non-power reactors. A definitive rule is scheduled to become effective in fiscal year 1979.
- A study of the results of hearings on a proposed personnel security clearance requirement, to help NRC decide whether such an effort will enhance safeguards protection.
- Possible development of a new rule for non-power reactor safeguards. This rule would cover non-power reactors not included in the fuel cycle facility upgrade rule mentioned above.
- A proposed rule, to be implemented in fiscal year 1979, that will specify physical protection measures for facilities processing less than strategic quantities of highly

enriched uranium and plutonium or certain specific quantities of low-enriched uranium. These materials are of moderate or low safeguards significance.

- The staff is also evaluating recommendations of an internal Task Force studying the role of material control and accounting in NRC's safeguards program. A plan has been prepared to implement those recommendations which are cost-effective. Recommendations involving the use of state-of-the-art technology will be carried out in fiscal year 1980; those recommendations requiring further research and development will be considered later.

(A more detailed discussion of new regulations is given below.)

An Integrated Safeguards Plan is under development which, when completed, will provide a formal, long-term plan which will, in its first phase, define the safeguards activities of all NRC offices engaged in the safeguards program and, in a later phase, specify objectives of the total program, set forth individual office responsibilities for achieving the objectives, and assure overall coordination.

New Safeguards Regulations

Performance-Oriented Regulation. The proposed amendments to 10 CFR Parts 70 and 73 to upgrade physical protection requirements for fuel cycle facilities and for transportation were published for comment in July 1977. In response to the extensive public comments received concerning, among other things, the conspiracy threat, package search requirements, level of threat, and need for public guard forces, the Commission decided to republish the revised proposed amendments for public comment on those changes that were made. In August 1978, the revised proposed amendments were published in the *Federal Register*.

The proposed rule describes the characteristics of a hypothetical external adversary group against which licensees would be required to design their safeguards systems. It also describes safeguards performance levels that nuclear facilities and transporters would be required to achieve but allows flexibility for the design of systems to meet the desired objective.

This approach acknowledges that there is more than one way to build a safeguards system. The proposed amendments do, however, identify elements and components that, if included in a physical protection program, would achieve the required performance. The NRC staff plans, at the time the regulation is issued in effective form, to issue final supplementary regulatory guides that further explain the intent of the regulation and provide design criteria for satisfying its requirements. The guides should help licensees in developing safeguards systems that satisfy the regulation.

Personnel Security Factors. In 1977, the NRC published for public comment two proposed regulations concerned with security clearances of personnel involved in licensed operations and qualifications of licensee guards and other security personnel that would be applicable to both nuclear fuel cycle activities and reactors.

Training, Qualification, and Equipping of Security Personnel. A proposed new Appendix B to 10 CFR Part 73 describes upgraded training, qualification, and equipment for security personnel who protect licensed nuclear facilities and transportation activities. The proposed rule, published in July 1977, is an outgrowth of the Security Agency Study, the findings of a joint ERDA-NRC task force on safeguards (NUREG-0095), and other deliberations. In response to extensive public comments, the final rule published in August 1978 was revised to specify performance-oriented requirements rather than detailed training requirements. Concurrent with the effective date of the rule, the following final guidance was published by the NRC to aid licensees in developing effective training and qualifications programs:

- NUREG-0219, Draft 2, "Nuclear Security Personnel for Power Plants."
- NUREG-0464, "Site Security Personnel Training Manual."
- NUREG-0465, "Transportation Security Personnel Training Manual."
- Revised chapters to Regulatory Guide 5.52, "Standard Format and Content for the Physical Protection Section of a Licensee Application (for Facilities Other Than Nuclear Power Plants)."

The regulation requires security personnel to meet minimum specified criteria for physical fitness, training, and other qualifications and to be requalified annually.

Material Access Authorization. In 1974, the United States Congress amended the Atomic Energy Act of 1954 to authorize the NRC to require a security clearance for persons involved in certain activities associated with special nuclear material (SNM). A staff proposal based on investigations conducted by the Civil Service Commission, was considered by the NRC in late 1976. This proposed program would be administered by the NRC, using procedures similar to those presently applied in clearing NRC employees.

In March 1977, the NRC published for comment proposed regulations (10 CFR Parts 10 and 11) that would require certain individuals involved in licensed nuclear activities to receive NRC authorization before being granted access to or control over SNM or vital areas at power reactors. In view of the extensive public comments concerning the proposed rule, a public hearing was held July 10, 11, and 12, 1978, to fully air all views. Pending submission of concluding statements by persons who participated in the hearing process and review by the hearing board, the NRC will determine final disposition of the rule.

The proposed rule would require certain individuals involved in licensed nuclear activities to receive authorization from the NRC before being granted access to or control over SNM. The proposed rule covers both fuel cycle activities and reactors. The purpose would be to provide a measure of assurance that those individuals would not use their positions to commit theft or sabotage. Authorization would be granted on the basis of background investigations.

The NRC proposal involves two clearance levels. The higher level, NRC-U, involves a "full-field" background investigation by the FBI and would be required for: (1) individuals who require unescorted access to SNM and to vital areas (areas that contain equipment vital to the protection of the public); (2) individuals whose positions make it possible, either alone or in conspiracy with another, to steal SNM or commit sabotage; and (3) drivers of motor vehicles and pilots of aircraft transporting certain quantities of SNM and those who escort SNM shipments. The lower clearance level, NRC-R, would be based on a Civil Service Commission check of Federal Government records for adverse information. It would apply to individuals who, while not being in any of the above categories necessitating an NRC-U clearance, do require unescorted access to protected areas.

The proposed program would be administered by the NRC, using the same procedures as are currently applied to clearing its own employees, e.g., use of the Civil Service Commission or FBI for all background investigations. Uniformity in the application of procedures and the availability of established avenues for appeal that would result from NRC's direct administration of the program should minimize the possibility that any individuals would suffer an undue loss of civil liberties such as the right of privacy from the personnel clearance process.

See Chapter 10 for a discussion of safeguards guides issued in fiscal year 1978.

5

Waste Management

NRC efforts in regulation of nuclear waste management activities during 1978 included the following:

- Work proceeded on a system for classifying wastes according to the type and duration of containment required for their safe disposal. A report setting forth the technical basis for the system was released for public comment.
- Studies were conducted to develop waste disposal performance objectives, including incorporation of societal attitudes.
- Studies were continued concerning the development of performance objectives and criteria for high-level, transuranic and military wastes during long-term storage in deep geological repositories.
- The National Academy of Sciences assisted NRC in evaluating potential criteria for assessing the suitability of sites for geologic waste repositories.
- The NRC staff (continued) preapplication interaction with the Department of Energy in anticipation of the possible submission of a license application for a geologic repository in New Mexico.
- A program to develop regulations on management of low-level waste was announced. A number of studies were conducted to develop the information base needed to establish these regulations.
- In late 1978, NRC published results of a screening of alternatives to shallow land burial for disposal of low-level waste.

Interagency Review Group

During 1978, the NRC staff participated in an Interagency Review Group (IRG) on Nuclear Waste Management. (Because of its status as an independent regulatory agency, NRC participated as a non-voting member. See also Chapter 1.) The IRG was instituted in March 1978 at the direction of the President to develop a strategy for dealing with the radioactive waste management problem. The primary objective of the plan is to provide assurance that existing and future nuclear waste from military and civilian activities can be isolated from the biosphere to protect public health and safety. The strategy developed by the IRG contains tentative policy and implementation recommendations, requirements for new legislation and work plans indicating key milestones for the involved Federal agencies. These plans and recommendations were published for public comment in a Draft Report to the President in October 1978. A Final Report, incorporating public comments received and additional agency reviews, was scheduled to be published in late 1978. (See Chapter 1.)

WASTE CLASSIFICATION

To provide a broad analytic basis for regulations governing the management and disposal of radioactive waste, the NRC is developing a system for categorizing wastes according to the type and duration of containment required for their safe disposal.

Three categories are currently proposed:

(1) *Class A*: Waste which, due to high or persistent radiotoxicity, requires isolation in a Federal repository or other disposal facility providing a high degree of isolation.

(2) *Class B*: Waste which is acceptable for disposal in near-surface facilities such as by shallow land burial.

(3) *Class C*: Waste which is nonradioactive or has such low levels of radioactivity that it can be disposed of routinely, as in sanitary landfills.

The classification system will present a systematic method for defining and quantifying the radioactivity concentration interfaces between the three categories.

In June 1978 the NRC published a report giving the technical basis for the classification system, "A Classification System for Radioactive Waste Disposal - What Waste Goes Where?" (NUREG–0456). In August, a Federal Register notice announced the availability of this report and requested public comments. An advisory panel with representatives of Federal and State governments, industry, universities, and a public interest group was convened in March and in December to review the progress of the study. A waste classification regulation, a supporting environmental impact statement, and a regulatory guide on complying with the regulation are scheduled for development in 1979.

PERFORMANCE OBJECTIVES

During fiscal year 1978 the NRC conducted two studies to develop performance objectives for radioactive waste disposal. The first of these, conducted jointly by NRC and by Lawrence Livermore Laboratory (LLL) under contract to NRC, surveyed current regulations and recommendations by scientific bodies regarding allowable levels of radiation exposure. From this information a set of objectives was developed which would limit the predicted radiological impacts from radioactive waste disposal to values likely to be considered acceptable by society.

The second study, conducted by LLL under contract to NRC, utilized a technique known as "multi-attribute decision analysis" to make a mathematical model of societal attitudes toward the risks associated with radioactive waste disposal. The major thrust of this study was to determine trade-offs between different types of risks (e.g., risks to the present generation versus risks to future generations) so that different repositories—or even totally different waste disposal concepts—can be compared.

The results of these studies (NUREG/CR-0540) are being evaluated by the NRC staff and will be used to further develop and refine NRC's waste disposal performance objectives. These objectives will, in turn, guide NRC's development of criteria for site suitability, repository design, and waste form performance, and will be used to evaluate the safety of proposed waste disposal projects.

Projecting Disposal Needs

During 1978, NRC-sponsored work was begun by Teknekron, Inc., on a computer model for projecting waste disposal needs. The model will consider the quantities of various classes of radioactive waste generated as a function of time and in a number of geographic regions of the country. This model will be used as a tool in making decisions about the need for licensing new sites. The project is scheduled to be completed in mid-1979.

HIGH-LEVEL AND TRANSURANIC WASTE

During fiscal year 1978 several studies were conducted by or for the NRC to provide a data base for regulations governing permanent repositories for high-level and transuranic waste. Proposed regulations are now scheduled to be published for public comment in the summer of 1979.

Waste Form Performance Criteria

Studies were continued by LLL under contract to NRC to investigate the performance of various forms of high-level and transuranic waste during long-term storage in deep geologic repositories. Investigations during fiscal year 1978 focused on storage in deep salt formations. Other media will be considered in the future. The high-level waste portion of the program was a continuation of fiscal year 1977 work. The commercial high-level waste study was terminated in February 1978 because of President Carter's decision deferring reprocessing. A report is being prepared by LLL which will summarize all the work performed on commercial high level waste through termination of the effort in February 1978. The report is expected to be completed in draft form in early 1979, at which time the report will undergo extensive review by the NRC staff and then be released for public comment.

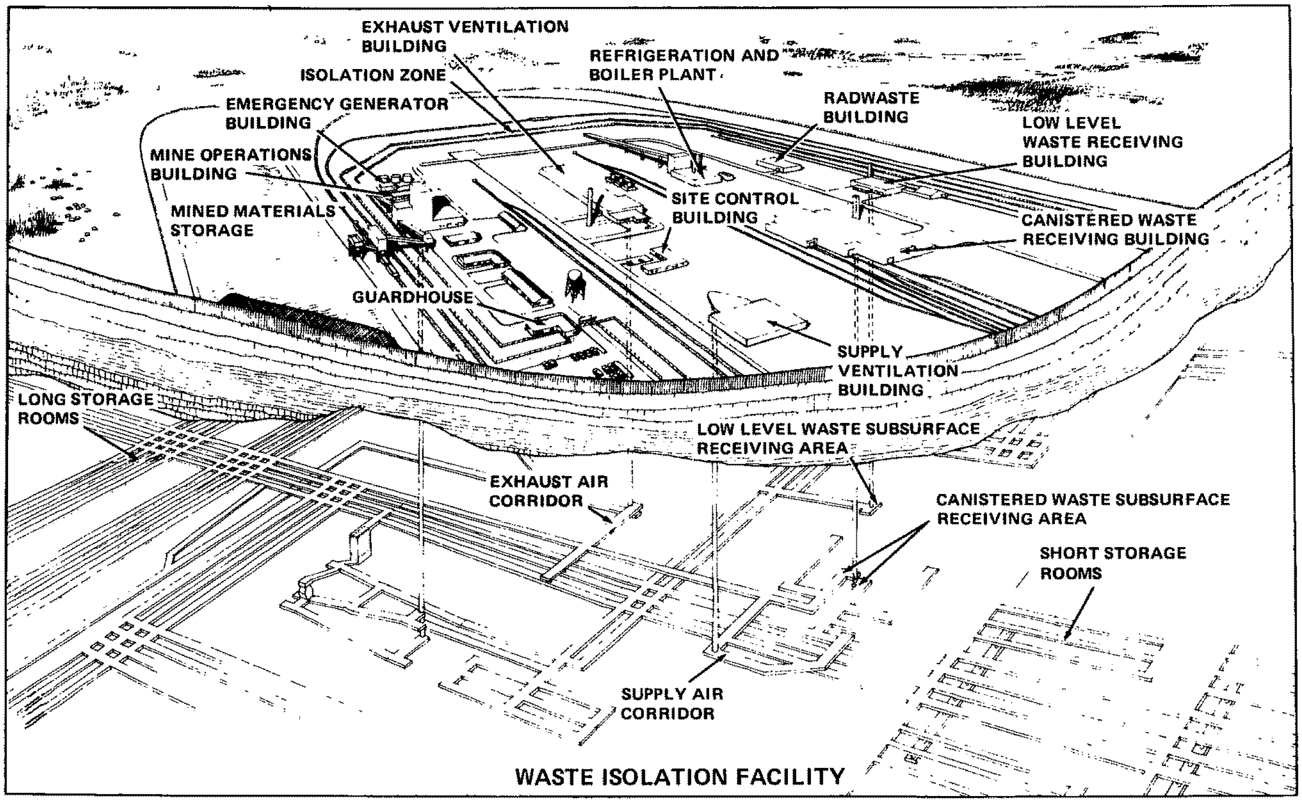
As the work on reprocessing high-level waste was phased out, work on spent fuel was initiated. Some of the models and mathematical codes utilized in the initial high-level waste studies were modified to apply to spent fuel. Investigations now are in the preliminary stage. They involve model development and modification, reference system definitions, and simplified analyses. The bulk of the study, also being conducted by LLL, is expected to be carried out in fiscal year 1979.

The long-term storage of transuranic waste is also being considered. (While transuranic waste is not considered high-level waste, it is thought to be necessary to dispose of it in the same manner as high-level waste because it maintains a hazardous level of radioactivity for long periods of time. The waste classification system will define those concentrations of transuranic waste which must be disposed of in this manner.) Earlier efforts in this area consisted of developing a working definition of transuranic waste and determining its inventory accordingly. As a result, LLL issued a draft report, "Inventory and Sources of Transuranic Solid Waste," in June 1978. The final version of this report is expected to be received by the NRC in the spring of 1979. Development of models for transuranic release mechanisms and rates has begun. Work planned for the next fiscal year includes identifying possible synergistic effects from placing transuranic waste in the same repository as high-level wastes or spent fuel. A report covering fiscal year 1978 work through July will be released in draft form in early 1979.

LLL also conducted an investigation of military waste mainly concerned with establishing the form and inventory of high-level defense-generated waste. This portion of the program was initiated and completed in fiscal year 1978. A draft report is to be issued in early 1979.

Repository Site Criteria

Under contract to the NRC, LLL has been conducting studies on the suitability of sites for geological repositories. The objectives of these studies are to identify those natural features which are most important to a geological repository's ability to isolate radioactive waste. In October 1977, LLL submitted an interim progress report to the NRC. In June 1977, the staff had drafted site suitability criteria based upon the study results at that time and on papers published by groups such as the International Atomic Energy Agency (IAEA). These criteria, and the interim study report, were



This is the Department of Energy's conceptual design of the probable layout of a bedded-salt repository for high-level and transuranic wastes. NRC will be responsible for the safety review and licensing of these facilities. As designed, the facility could handle both spent reactor fuel and high level waste from fuel reprocessing.

presented to a peer review panel on October 28 and 29, 1977. The panel's comments and suggestions, submitted to NRC in March 1978, were incorporated into a revision of the draft site suitability criteria and will be reflected in the regulations to be published for public comment in 1979.

In November 1977, the National Academy of Sciences convened a Panel on Geologic Site Criteria to assist the NRC by: (1) identifying the criteria needed in determining the suitability of a waste disposal site, (2) reviewing NRC's revised site suitability criteria, and (3) reviewing the LLL interim report. The panel's report was submitted to NRC in August 1978, and results will be incorporated in NRC staff position papers. The panel's comments on the LLL report were forwarded to the Laboratory for consideration in its continuing study.

Since submitting its interim report in October 1977, LLL has continued to refine the study. This has involved expansion and revision of the analytical model developed for waste transport in sedimentary basins, revision of the earth-sciences information used with that model, identification of areas where more research is needed, and determination of the effort required to study other geologic formations such as domed salt, basalt and granite.

The study for sedimentary basins is scheduled for completion by December 1979. The NRC staff will use its results as a basis for position papers on site suitability.

Repository Construction and Operation Requirements

The NRC staff is obtaining background information and developing regulations to govern performance of the engineered aspects of a geologic repository. All activities which might degrade the ability of an inherently suitable repository site to contain radioactive waste (e.g., mining, waste emplacement, mine closure) are being considered. Ongoing programs include:

- Identifying performance requirements for shaft and borehole seals.
- Defining performance requirements for equipment that will be operating in a repository.
- Identifying those interactions between wastes and the disposal media which would affect a repository's radionuclide containment capabilities, or adversely impact the ability to retrieve wastes.
- Analyzing the thermomechanical response of mine structure features.
- Identifying the decommissioning performance requirements.
- Analyzing the impacts of excavation on a repository's ability to contain radionuclides.

The NRC staff will use radionuclide transport and systems analysis models to determine which aspects of the design of a repository have the greatest impact on its performance.

Licensing Procedures for Repositories

The NRC staff is making preparations for the licensing review of geological repository applications to be submitted by the Department of Energy.

A statement of policy regarding administrative procedures to be followed by NRC and the applicant was expected to be issued for public comment in late 1978.

Technical papers are being prepared on the standard format and content of both environmental reports and license applications. Working drafts of these papers are undergoing internal review. They will provide early guidance to the Department of Energy (DOE) in its licensing activities.

Development of computer modeling techniques to assist in the evaluation of repository license applications continued at Sandia Laboratories, New Mexico, under contract with NRC's Office of Nuclear Regulatory Research. Preparation for the use of those techniques was initiated at NRC during the reporting period. This project is discussed under "Fuel Cycle Risk Assessment Research," Chapter 11.

NRC staff members have inspected potential repository sites under investigation by DOE in southeast New Mexico, at the Nevada weapons test site, and at the Hanford reservation in Washington. NRC inspection and enforcement procedures and quality assurance requirements were explained to DOE staff members at meetings held in April and June 1978, respectively. Docket files for the Waste Isolation Pilot Project (WIPP) have been established at the public document rooms in NRC Headquarters and in Albuquerque and Santa Fe, New Mexico, anticipating a possible license application by DOE for a waste repository in deep salt formations, near Carlsbad, New Mexico. An updated list of all docket material is maintained at three additional locations in New Mexico.

LOW-LEVEL WASTE DISPOSAL

Development of Regulations

In December 1977 the NRC announced in the *Federal Register* a program to develop regulations governing the management of low-level radioactive waste. The program was described in a document entitled "The Nuclear Regulatory Commission Low-Level Radioactive Waste Management Program" (NUREG-0240).

During fiscal year 1978, progress was made in developing the information base needed to establish these regulations. Approximately 40 percent of the radioactive waste shipped to the commercial shallow land burial sites is from sources not involved in the nuclear fuel cycle for commercial power reactors, such as hospitals, universities, radiopharmaceutical suppliers, and industrial users. Results of a study characterizing the sources, volumes, isotopic content and physical form of wastes from such non-fuel cycle waste generators were published in March 1978 as NUREG/CR-0028, "Institutional Radioactive Wastes." Other studies proceeding in 1978 related to the physical properties of solidified low-level wastes using commercially available solidification agents, the parameters important to obtaining an acceptable solid product, and the chemical toxicity of low-level wastes.

Field studies were initiated during fiscal year 1978 at licensed burial sites in West Valley, New York and Maxey Flats, Kentucky to identify potential pathways for radionuclide migration. Also, measurements of the radio-chemical compositions of trench leachate continued at licensed burial sites in cooperation with the U. S. Geological Survey. The results of these studies will be used to develop models to evaluate radionuclide migration and to establish criteria on the suitability of burial sites. Completion of the models and proposed regulations governing siting criteria for shallow land burial is planned for 1980. In October of 1978, the NRC staff published an advance notice of rulemaking in the *Federal Register* asking for public comment on the proposed rgulations and on the supporting environmental impact statement.

Limits On Disposal Capacity

Recent developments at the commercial low-level waste burial grounds have raised the question of whether adequate regionally distributed disposal capacity for the nation's low-level radioactive wastes will be available at currently operating facilities. Two of the six licensed commercial burial grounds (West Valley, New York and Maxey Flats, Kentucky) are closed. A third site, at Sheffield, Illinois, has reached its licensed capacity. A limit has been placed by the State of South Carolina on the volume which may be accepted at the Barnwell, S.C., site. Thus, a large fraction of the waste from reactors and other waste generators located in the Eastern and Midwestern United States must soon be transported to the burial sites at Beatty, Nevada and Hanford, Washington.

Battelle Pacific Northwest Laboratory personnel obtaining sediment samples from Cattaraugus Creek during winter sampling period at West Valley, New York, Nuclear Center as part of NRC sponsored program to model radionuclide migration by sediment transport.

It can thus be seen that the options available for disposal of low-level waste are now limited, especially if operational problems should develop at any of the functioning sites. The NRC believes that the situation can be addressed in the short term by having the industry work out cooperative arrangements for use of shielded casks, transport vehicles, interim storage and optimal utilization of the capacity of the operating sites. However, NRC also believes that additional standby capacity should be made available and has requested DOE to develop a contingency plan which would allow its burial sites to accept commercially generated wastes, should the need arise. The NRC has also requested DOE to consider disposing of radioactive wastes from its prime contractors at DOE sites rather than at commercial burial sites.

Alternatives to Shallow Land Burial

In 1978, the NRC continued a study of alternative methods to shallow land burial for disposal of low-level radioactive wastes. This study was initiated at the recommendation of an NRC Task Force set up to review the Federal/State program for regulation of commercial low-level radioactive waste burial grounds. The study was recommended because it was believed an alternative method could have advantages over shallow land burial and also because having more than one method would provide additional disposal capacity.

After a preliminary screening, NRC evaluated the following alternatives in some detail: (1)

emplacement of wastes in engineered structures, (2) disposal of wastes in ocean waters, (3) emplacement of wastes in mined cavities (existing mines or mines dug specifically for waste disposal), and (4) burial of wastes at an intermediate level (e.g., 30 feet of cover as compared to 4–6 feet of cover for shallow land burial). Preliminary results of the study were published in September 1978 (NUREG–CR–0308). The advance notice of proposed rulemaking, which was issued in October 1978 to solicit comments on development of the low-level waste disposal regulation and its supporting EIS, also requested comments on the development of a regulatory program for alternative disposal methods to the present practice of shallow land burial. (See Chapter 10 for discussion of the decommissioning of licensed facilities.)

(Developments on waste management occurring after the end of the fiscal year are discussed briefly in Chapter 1. Mill tailings management is discussed in Chapters 1 and 3.)

Inspection and Enforcement

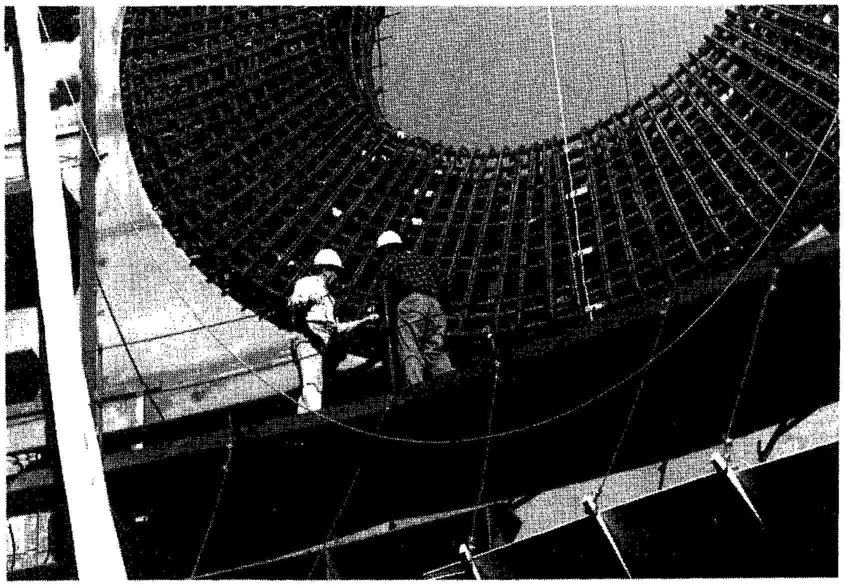
During 1978 the NRC continued to strengthen its inspection and enforcement program, the primary means for assuring that licensed nuclear activities comply with requirements designed to protect the health, safety and security of the public and the environment.

The total number of NRC inspections rose to almost 6,600 during fiscal year 1978, approximately double the annual rate of inspections being achieved at the time the NRC was created four years ago. Roughly one-half the inspections were conducted at nuclear reactor facilities, either under construction or in operation. In addition, 85 special investigations were carried out in response to allegations or reports of radiation incidents, equipment problems, complaints, and loss and theft of licensed materials.

One or more noncompliance items were disclosed in 36 percent of the inspections and in 44 percent of the 85 investigations. The more severe sanctions imposed in citations of licensees for failure to comply with NRC requirements included 14 civil monetary penalties and 10 orders to "cease and desist" operations, or for modification, suspension, or revocation of licenses.

In other inspection and enforcement developments of the year, the NRC:

- Stationed resident inspectors at the sites of 20 nuclear power stations, involving 45 power reactors under construction or in operation, and at 3 major nuclear fuel facility sites.
- Proposed legislation to Congress that would increase by twentyfold the amount of a fine that NRC could levy for a licensee violation as a measure to provide greater incentives for licensees to comply with requirements.
- Implemented a statutory requirement that officials of firms in the nuclear industry report to the NRC any defect that could create a substantial safety hazard, or a failure to comply with regulations relating to substantial safety hazards.
- Completed a specially-equipped Incident Response Center at NRC headquarters in Bethesda, Md.,



Inspectors Seth Folsom and Anthony Fasano of NRC's Region I Office near Philadelphia, Pa. examine reinforced steel cadwelding on a primary containment equipment hatch at the Beaver Valley Power Station, Unit 2, in Pennsylvania. More than 1300 such construction inspections were conducted during 1978.

improving the agency's ability to respond promptly to emergency situations.

The inspection and enforcement program is directed by NRC's Office of Inspection and Enforcement, with a headquarters staff located in Bethesda, Md., and a field staff deployed in NRC's five regional offices located in or near Philadelphia, Atlanta, Chicago, Dallas, and San Francisco. About 80 percent of the total office staff is assigned to the regions.

THE INSPECTION PROGRAM

The objectives of inspections are:

- To determine whether licensees are complying with NRC requirements.
- To identify conditions that may adversely affect public health and safety, the common defense and security, the environment or the safeguarding of nuclear materials and facilities.
- To provide information that may assist in developing a basis for issuance, denial, or amendment of an authorization, permit or license.
- To determine whether licensees and their contractors and suppliers have implemented adequate quality assurance programs.

When an inspection or investigation discloses events or conditions that present a potential or actual threat to public health and safety, the environment, or the safeguarding of nuclear materials and facilities, the NRC takes appropriate action and routinely communicates what it has found to other parts of Government, licensees and the public.

NRC's inspections are of two basic types: routine and reactive. In routine inspections, NRC inspectors concentrate on determining the effectiveness of quality assurance systems by observing work in progress, checking records, interviewing people, and, where appropriate, making direct measurements. Reactive inspections are conducted in response to information received by NRC regarding conditions or events affecting licensed facilities or material under NRC jurisdiction. Such information may come from routine NRC inspections; from an applicant, licensee, contractor or supplier; or from a licensee employee or other member of the public.

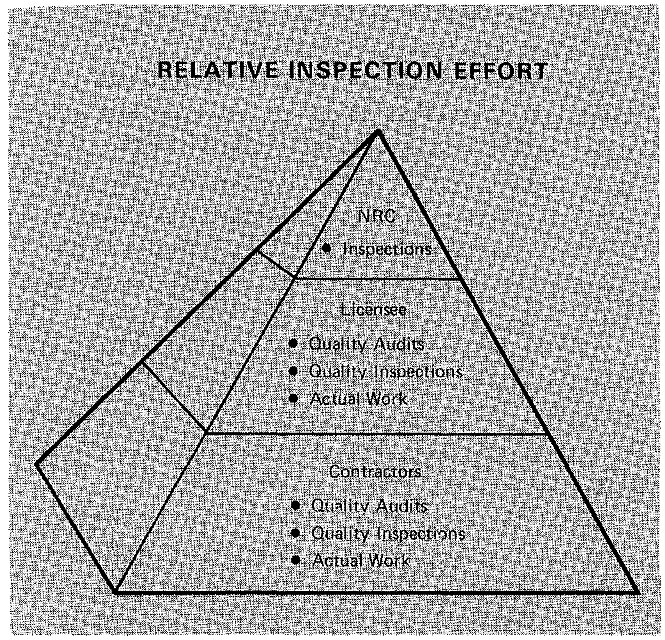
Inspections cover the entire range of NRC-licensed activities. Reactor-related inspections cover all phases of nuclear power plants (preconstruction activities, construction, preoperational testing and startup, operation, and decommissioning) and the operational phase of research and test reactors. In addition, NRC inspects the quality assurance programs of contractors and vendors who supply equipment, components and services to power reactors under construction or in operation. (This part of the program is centralized in the Dallas regional office.)

Inspections related to nuclear materials include inspection of the construction and operation of uranium mills; fuel fabrication, processing and reprocessing plants; waste disposal facilities; and the radiographic and medical uses of radioactive material. Measures for safeguarding nuclear material from theft and sabotage, for physical protection of reactors and fuel cycle facilities, and for transportation of nuclear materials are subject to NRC inspection.

The number of inspections carried out during fiscal year 1978 (ending September 30) for each of these activities is shown in Table 1.

Government-Industry Efforts

The NRC inspection program is based on the premise that the licensee is responsible for carrying out licensed activities safely and in compliance with NRC requirements. NRC verifies that the licensee has established the management control systems necessary to meet regulatory responsibilities. The inspection pattern for nuclear facilities is pyramidal (see accompanying diagram), with each level of activity verified, inspected or audited by those above. The NRC inspection effort is essentially the apex of the pyramid, i.e., NRC performs the last in the series of inspections and audits conducted by many different groups. NRC inspection manpower is usually far less than that of licensees and contractors. Because NRC inspectors cannot possibly inspect all components and activities, they probe the "pyramid" to the depth necessary to determine whether the licensee's and contractors' activities are properly performed.



Resident Inspectors Assigned

During 1978, the NRC completed the first stage of a program to station inspectors full time

Table 1. Inspections Conducted in Fiscal Year 1978

Program	*Number of Licenses*	*Number of Inspections*
Power reactor construction	175	1,310
Operating power reactors	72	1,703
Other reactors	94	148
Fuel facilities	39	194
Materials	8,863	2,456
Vendors	168	265
Safeguards	243	515

at the sites of nuclear power plants and major fuel cycle facilities.

By September 30, 14 inspectors had been deployed at the sites of 12 nuclear power stations and three nuclear fuel facilities. (By December 31, 1978, eight additional inspectors were to take up their assignments at another eight nuclear power stations. The 20 power stations involved have 45 power reactors in operation or under construction. See Table 2.)

The assignment of resident inspectors follows the completion and evaluation of a successful two-year trial program, ending in 1976, in which two resident inspectors were assigned to locations near four midwest reactor sites.

The resident inspector program is expected to improve inspection effectiveness in several ways, including: (1) providing more opportunities to observe licensed activities, verify compliance, identify safety-related problems, and respond to incidents; (2) affording the inspector greater knowledge of the plant, thus enhancing his ability to make prompt and accurate technical judgments; and (3) increasing the efficiency of inspections.

Other inspection goals are to increase the proportion of inspections that are unannounced, and to increase the number of inspectors as a percentage of the Office of Inspection and Enforcement's total staff. This percentage was increased from 56.5 percent in 1977 to 58.5 percent in 1978, and the target for 1979 is 60 percent.

Third Party Inspection Program

The trial program being undertaken with the American Society of Mechanical Engineers (ASME) to test the feasibility of using inspections by third parties was continued during the fiscal year. (See Annual Report for 1977, p. 88). An evaluation of the trial program will be submitted to the Commission in June 1979. The NRC has initiated discussions with the Institute of Electrical and Electronic Engineers (IEEE) on a similar program.

Reporting Defects and Noncompliance

On June 6, 1977, the NRC published in the *Federal Register* a regulation (10 CFR Part 21) setting forth the requirements for implementing Section 206 of the Energy Reorganization Act of 1974. Individual directors or responsible officers of a firm involved in the nuclear industry are required to report noncompliance with NRC regulations or the existence of defects which could create a substantial safety hazard. Any such person who fails to provide the required reports to the NRC is subject to a civil penalty not to exceed $5,000 for each failure and a total amount not to exceed $25,000 within any 30-day period.

The regulation became fully effective on January 6, 1978. Initial NRC inspections indicate that most affected major organizations have established measures for the reporting of defects and noncompliance. This was further evidenced by NRC's receipt during fiscal year 1978 of 77 reports from directors or responsible officers subject to the requirements. The reports were reviewed to assess the possibility of generic problems, and appropriate follow-up actions were taken.

The NRC's initial experience also revealed, however, that the regulation had an unintended adverse effect on procurement of commercial grade items, i.e., some construed the rule as applying to items available in general commerce. Accordingly, the Commission adopted an amendment to 10 CFR Part 21 to correct the situation which became effective in October 1978 (See Chapter 10).

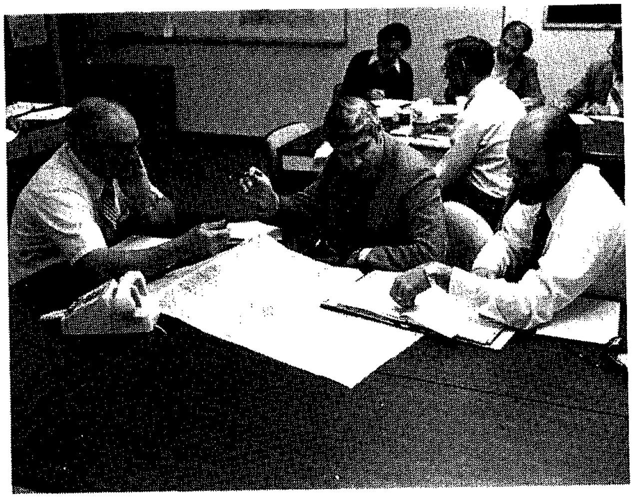
Response to Incidents and Emergencies

The NRC's ability to respond to situations that pose a significant threat, actual or potential, to the health and safety of the public has been augmented by the completion of an Incident Response Center at the Office of Inspection and Enforcement headquarters in Bethesda, Md. The 2,000 sq. ft. facility includes: a conference room for briefing NRC management; an operations room for monitoring and evaluating the incident; a secure communications room; word processing and computer support areas; and a library to house necessary information resources. The center is equipped with a specially-designed communications system and a variety of audio-visual aids. In addition, portable communications packages are being developed to assure

Table 2. Sites Manned by Resident Inspectors During 1978

Facility	*Location*	*Licensee*
Arkansas Nuclear Plant	Russelville, Arkansas	Arkansas Power & Light Co.
Browns Ferry Nuclear Power Plant	Decatur, Alabama	Tennessee Valley Authority
Commanche Peak Steam Electric Station	Glen Rose, Texas	Texas Power & Light, Dallas Power & Light, Texas Electric Service
Donald C. Cook Plant	Bridgman, Michigan	Indiana & Michigan Electric Company
Diablo Canyon Nuclear Power Plant	San Luis Obispo, California	Pacific Gas & Electric Co.
Dresden Nuclear Power Station	Morris, Illinois	Commonwealth Edison Co.
Edwin I. Hatch Plant	Baxley, Georgia	Georgia Power Co.
Indian Point Station	Indian Point, New York	Consolidated Edison Co.
Midland Nuclear Power Plant	Midland, Michigan	Consumers Power Co.
Millstone Nuclear Power Station	New London, Connecticut	Northeast Nuclear Energy Co.
North Anna Power Station	Mineral, Virginia	Virginia Electric & Power Co.
Oconee Nuclear Station	Seneca, South Carolina	Duke Power Co.
Peach Bottom Atomic Power Station	Peach Bottom, Pennsylvania	Philadelphia Electric Co.
Prairie Island Nuclear Generating Plant	Red Wing, Minnesota	Northern States Power Co.
Salem Nuclear Generating Station	Salem, New Jersey	Public Service Electric & Gas Co.
San Onofre Nuclear Generating Station	San Clemente, California	Southern California Edison Co. and San Diego Gas & Electric Co.
Surry Power Station	Gravel Neck, Virginia	Virginia Electric & Power Co.
Susquehanna Steam Electric Station	Berwick, Pennsylvania	Pennsylvania Power & Light Co.
Trojan Nuclear Plant	Prescott, Oregon	Portland General Electric Co.
Watts Bar Nuclear Plant	Spring City, Tennessee	Tennessee Valley Authority
B&W-Apollo & Leechburg* (Fuel facility)	Apollo, Pennsylvania	Babcock & Wilcox Co.
Westinghouse-Cheswick* (Fuel facility)	Parks Township, Pennsylvania	Westinghouse Electric Corp.
Nuclear Fuel Services (Fuel facility)	Erwin, Tennessee	Nuclear Fuel Services, Inc.

*Inspector stationed at Apollo, Pa., acts as resident inspector on a rotating basis at B&W's Apollo and Leechburg facilities and Westinghouse's Cheswick facility.



NRC's permanent Incident Response Center in Bethesda, Maryland, completed during Fiscal Year 1978, was the scene of several simulated emergency exercises. Shown above is the Center's operations room during the conduct of an exercise; the conference/briefing room is below.

that field personnel can transmit information to the regional offices and to Headquarters.

Early in 1978, before the permanent center was completed, an interim Incident Response Center was activated as a precautionary measure after the Public Service Company of Colorado reported what erroneously was believed to be a large gaseous release from its Fort St. Vrain Nuclear Generating Station. Later information and subsequent evaluations by the licensee, State and NRC showed that there was an accidental release of a small amount of radioactivity which could not be detected with radiation measuring equipment outside of the plant boundaries.

INVESTIGATIONS

Significant staff effort is put into responding to reports of radiation incidents, equipment problems, loss or theft of licensed materials, and other allegations and complaints received by NRC. Although many of these situations prove to be minor and of a sort that can be reviewed during scheduled inspections, some require special response. In these cases, a special inspection is scheduled or, when appropriate, an immediate, full investigation may be initiated. During fiscal year 1978, 85 investigations were conducted by inspection and enforcement personnel. Of these, 64 were prompted by allegations dealing with reactor construction or operational events at licensed facilities. Other investigations were conducted into events involving loss or theft of licensed material, overexposures and general public interest. In 37 of the 85 investigations, licensees were cited for failure to meet NRC requirements.

Three significant special investigations conducted during the year are described below.

D. C. Cook Nuclear Plant

The D. C. Cook Nuclear Plant (Indiana and Michigan Electric Company) is located about 11 miles south-southwest of Benton Harbor, Mich. In connection with NRC reviews concerning unqualified electrical connectors at several nuclear facilities, an investigation was initiated in December 1977 into specific testing and qualification practices at Cook Units 1 and 2. As a result of this investigation, it was determined that two materially false statements were made by the licensee in an application regarding testing of electrical penetrations and instrument cable for D. C. Cook Unit 2.

The NRC regulatory program is based on the premise that information provided by licensees will be factual, complete and well supported by data, records, calculations and judgments of technically qualified individuals. Information which does not meet these qualifications could result in decisions which adversely affect the health and safety of the public.

As a result of the investigation findings, a \$10,000 civil penalty was proposed, and collected from the licensee.

Technetium Generator Distribution

On August 5, 1977, officials of the State of Vermont notified the NRC that a number of radioctive technetium generators had been found at a foundry in Rutland. Used by hospitals as part of their nuclear medicine programs, the generators produce radioactive technetium-99m, a valuable diagnostic radioisotope with a relatively short half-life.

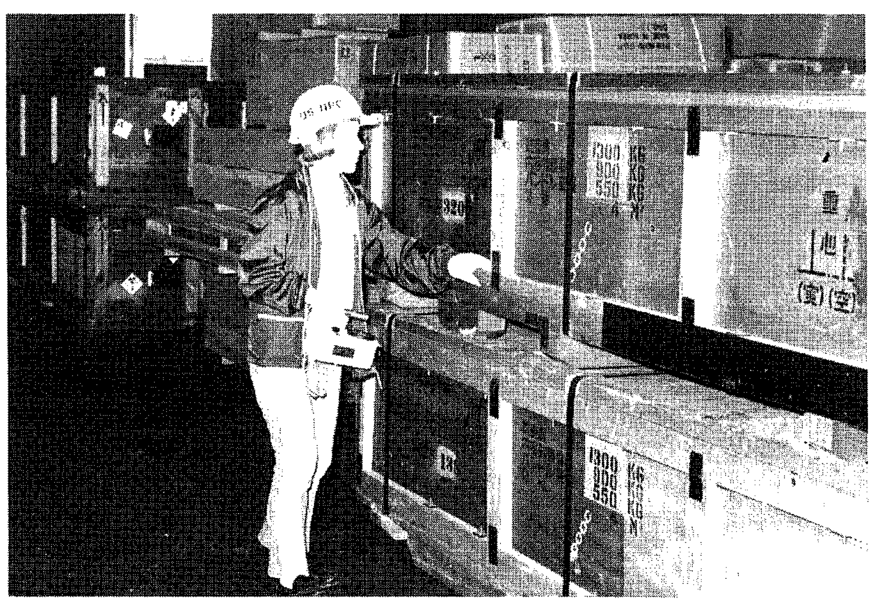
The NRC conducted an investigation at several hospitals in the Rutland area and determined that one of the hospitals was disposing of the lead shielding containers by returning them to the sales representative of a pharmaceutical supplier who had sold them the generators. The sales representative would then sell the containers as scrap metal to local salvage dealers. The problem arose from the fact that both the hospital and the sales representative assumed that the other party had removed the radioactive material contained therein and had made proper disposal.

Although this problem was rectified immediately and it was determined that the generators had posed no significant threat to the health and safety of the public, the NRC investigation uncovered a related problem. It was learned that generators containing large amounts of radioactivity were being purchased and used by larger hospitals until the material had decayed to a point where it could not be utilized effectively to handle their large patient loads. At this point, the generators were resold to hospitals with smaller patient loads for whom the amount of activity remaining would be adequate. Such unauthorized repackaging and redistribution of the technetium generators was halted immediately by the NRC. As a result of the investigation, NRC sent notices of violation to four Vermont hospitals.

Workman Fired—Alleges Reprisal

A construction man was discharged by the licensee's contractor some five months after the employee had made a series of allegations regarding what he considered to be unsafe practices and materials being used at the site of the Callaway plant (Missouri). All allegations have been investigated and resolved except one, which is still under technical review.

The licensee, the Union Electric Company, and its contractor indicated that the reason for the termination was that the individual had not followed orders. The workman requested that NRC "protect him" from what he considered to be retaliatory action on the part of the licensee. When the NRC attempted to investigate the facts surrounding the dismissal and was refused access to records or personnel by the licensee and his contractor, a Show Cause Order was issued. The order to show cause why the construction permits should not be suspended indicated that the investigation had been initiated to determine: (a) whether the allegations had caused or contributed to the dismissal of the



This NRC inspector, Radiation Specialist Beth Riedlinger of the Region V Office in Walnut Creek, Cal., is conducting a radiation survey of a shipment of fresh reactor fuel destined for export to Japan to assure that shipping containers are free of contamination.

employee; (b) whether the Commission's regulations should be amended to protect workers who communicate information related to public health and safety protection from retaliatory acts by their employers; and (c) whether the termination caused other workers to fear retaliation and, therefore, cut off the flow of safety-related information from the workers.

The utility requested direct action by the Commission, which referred the matter to the Atomic Safety and Licensing Board. The hearing process at this level upheld the action of the NRC in suspending the construction permit. The licensee appealed to the Atomic Safety and Licensing Appeal Board which had the case under review at the close of the report period. The construction worker has been reinstated with back pay, following his appeal through the union and under the union contract's provisions for arbitration.

In the NRC Authorization Act for Fiscal Year 1979 (Public Law 85-601, signed November 6, 1978), the Congress provided that employers cannot discharge or otherwise discriminate against employees for assisting the NRC enforcement process. (See Chapter 1.)

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

- Written Notices of Violation are provided for all noncompliance with NRC requirements.
- Civil monetary penalties are considered for licensees who evidence significant or repetitive items of noncompliance, particularly when a Notice of Violation has not been effective. Civil penalties may also be imposed for particularly significant first-of-a-kind violations.
- Orders to "cease and desist" operations, or for modification, suspension, or revocation of licenses, are used to deal swiftly and conclusively with licensees who do not respond to civil penalties or to deal with violations that constitute a significant threat to public health and safety or to the common defense and security.

Tables 3 and 4 summarize the enforcement actions taken during the report period.

ENFORCEMENT ACTIVITIES

The regulatory program is designed to assure that licensees perform in accordance with NRC regulations, licenses and permits and with applicable sections of Federal statutes. NRC is empowered to take enforcement action where licensees are not satisfying these requirements or are conducting operations that might endanger the public or the environment, or adversely affect the common defense and security.

Enforcement action is not usually taken regarding situations which are identified by a licensee's own inspection program, provided the licensee has adequately corrected the problem and the noncompliance is not significant. Enforcement action is more likely to be taken where the problem has escaped the licensee's attention and is first discovered in an NRC inspection. Such situations reflect on the effectiveness of the licensee's inspection program and the licensee is generally required, at the least, both to correct the particular problem and the deficiencies in his quality assurance program which allowed the problem to exist.

Enforcement Improvements

The Office of Inspection and Enforcement is seeking continued improvement in enforcement. The Commission recently forwarded to Congress a request to increase NRC's statutory authority to impose civil monetary penalties. If this request is implemented by amendment of the Atomic Energy Act, NRC's maximum allowable penalties will increase from \$5,000 to \$100,000 for a single violation and from \$25,000 to \$300,000 for all violations committed by a licensee within 30 days. Such an increase would provide greater incentives for major NRC licensees to comply with the regulatory requirements. A greater range would also permit the penalties to be imposed by NRC to reflect more equitably the different classes of licensees and the seriousness of offenses. A related enforcement initiative would establish a practice of informing State public utility commissions each time a civil penalty was imposed on an NRC licensee in that State.

NRC continues to develop better methods for the evaluation of the regulatory performance of

Table 3. Civil Penalties Imposed - Fiscal Year 1978

Licensee	*Amount*	*Reason*
J. G. Sylvester Associates, Inc. Rockland, Massachusetts (Radiographer)	$ 6,000	Head exposure to an individual. Permitting excessive radiation levels to exist in unrestricted areas. Inadequate supervision over radiographic operations.
CERAC, Incorporated Milwaukee, Wisconsin (Thorium processor)	$ 3,750	Excessive exposure to personnel to air concentrations of radioactive thorium. Failure to perform necessary radiation surveys and decontaminate equipment.
Dayton X-Ray Company Dayton, Ohio (Radiographer)	$ 6,100	Lack of training for radiographic personnel, improper calibration of radiation detection instruments by unauthorized personnel and failure to maintain records.
Pittsburgh-Des Moines Steel Pittsburgh, Pennsylvania (Radiographer)	$ 7,000	Radiation exposure to the hand of a radiographer during radiographic operations.
Boston Edison Company Boston, Massachusetts (Pilgrim Nuclear Power Station)	$16,000	Whole body exposure to an individual and failure to instruct workers.
Commonwealth Edison Company Chicago, Illinois (Dresden Units 1, 2 and 3)	$21,000	Numerous personnel and procedural errors relating to maintenance and surveillance.
Rochester Gas & Electric Rochester, New York (R.E. Ginna Nuclear Plant)	$24,000	Failure to follow radiation protection procedures. Failure to comply with high radiation area control requirements.
Amersham Corporation Arlington Heights, Illinois (Materials licensee)	$ 2,000	Distribution of various quantities of americium-241 to nine recipients in thirty-eight separate shipments without verifying that recipients were licensed to receive them.
Entronic Corporation Kingsville, Texas Earth City, Missouri (Materials licensee)	$ 6,000	Distribution of smoke detectors containing americium-241, was prohibited by their license. Failure to follow numerous procedures specified by license conditions.
Indiana & Michigan Electric Company New York, New York (D.C. Cook Unit 2)	$10,000	False statement made by licensee in an application submittal in October 1977 regarding testing of electrical penetrations and instrument cable for D.C. Cook, Unit 2.
Shelwell Services, Inc. Hebron, Ohio (Materials licensee)	$ 1,000	Loss and subsequent recovery in the public domain of a 2.8 curie americium-241 sealed source contained in a source holder, failure to report loss in time specified.
Wisconsin Public Service Corporation Green Bay, Wisconsin (Kewaunee Plant)	$ 7,000 (Pending)	Failure to perform a survey required by regulations to assure control of personnel exposures.
Portland General Electric Company Portland, Oregon (Trojan Nuclear Plant)	$20,500	Whole body exposures of two individuals. Failure to make adequate surveys, failure to notify NRC Regional Office of exposures and failure to provide proper barriers to restrict entry to a potentially high radiation area where the transfer tube penetrated containment while spent fuel was transferred.
Union Boiler Company Huntington, West Virginia (Radiographer)	$ 5,000	Extremity exposure of 123 rems to a radiographer, failure to perform an adequate radiation level survey, and failure to follow written instructions subsequent to radiographic operations.

Table 4. NRC Enforcement Orders — Fiscal Year 1978

Licensee	*Date*	*Type*
Bionic Instruments, Inc. Bala Cynwyd, Pennsylvania (Materials licensee)	12/16/77	Order rescinding order to show cause and order suspending license. *Reason:* The licensee disposed of all by-product material formerly held under the license and the Office of Nuclear Material Safety & Safeguards, Radioisotope Licensing Branch, terminated the license as of 10/26/77.
Radiation Technology, Inc. Rockaway, New Jersey (Radiation facility)	10/4/77	Order modifying order suspending license. *Reason:* The licensee requested relief from the Suspension Order to move certain cobalt-60 "pencils" stored in contact with an aluminum table in the irradiator storage pool to another, more suitable, location within the R&D pool.
	10/7/77	Second modification of order suspending license. *Reason:* The licensee requested relief from the Suspension Order to move 81 cobalt-60 "pencils" stored in 21 source tubes resting on the bottom of the irradiator storage pool to an upright position and place within a source tube basket.
	10/8/77	Third modification of order suspending license. *Reason:* The licensee requested relief from the September 23, 1977 order to reinstitute operation of the impregnated wood Irradiation Facility.
	10/11/77	Order rescinding order suspending license. *Reason:* The licensee organized a safety review committee, hired a new radiation safety officer. In response to an NRC letter dated 10/06/77 the licensee documented new procedures and modifications which were incorporated into the license by amendment dated 10/14/77.
Entronic Corporation Earth City, Missouri (Materials licensee)	12/29/77	Order to cease and desist. *Reason:* Findings indicated the licensee had distributed quantities of americium-241 as ionization sources in smoke detectors. The NRC had not authorized the company to commercially distribute the sources but had only licensed them for research and development.
Entronic Corporation Earth City, Missouri	1/12/78	Order to rescind a previous order. *Reason:* A meeting with licensee and modification of license to authorize and distribute americium-241 in smoke detectors.
Luminous Process, Inc. Ottawa, Illinois (Materials licensee)	2/17/78	Order of immediate suspension of license and order to show cause why license should not be revoked permanently. *Reason:* Findings during a followup inspection indicated that the licensee's evaluation of contamination levels continued to be inadequate.
Union Electric Company St. Louis, Missouri (Callaway Units 1 & 2)	4/3/78	Order to show cause why construction permits should not be suspended. *Reason:* Allegations of construction problems which could lead to unsafe conditions. Investigators denied access to the records. (In litigation.)
Radioassay System, Inc. Southfield, Michigan (Materials licensee)	7/13/78	Order to show cause why license should not be revoked and order suspending licenses. *Reason:* Licensee authorized storage only but was processing and distributing without authorization.

major licensees. By identifying licensees whose performance may require improvement, NRC hopes to anticipate potential safety and security problems and avert them through prompt remedial action. This would also improve the effectiveness of NRC's use of inspection resources. Identifying valid measures of licensee performance is a complex and controversial process. Measures considered to date include licensees' compliance records, evaluations of licensees by NRC inspectors, and detailed trend analysis of reportable licensee events. (See Chapter 1 for later developments.)

GAO Audit for Construction Inspection

During the past year, the GAO reviewed NRC's inspection activities dealing with the construction of nuclear power plants.

The GAO study, completed in September, 1978, concluded that:

"The Commission can improve the quality of nuclear power plant construction by adjusting its inspection and reporting practices. The Commission inspectors, in particular, need to be more aggressive in scrutinizing and following up on the items they select for review. Also, the Commission needs to increase the productivity of its inspectors by relieving them of many clerical duties. The Commission should seek additional staff and organizational units to investigate allegations of poor construction work without disrupting the routine inspection program."

The Office of Inspection and Enforcement is addressing the concerns of GAO in a Revised Inspection Program. The Revised Inspection Program was developed over the past two years and initial implementation was started this year. (See NRC Annual Reports for 1976 and 1977.)

The Revised Inspection Program, when fully implemented in 1981, will:

- Increase the time NRC inspectors are at the licensee sites, principally through use of resident inspectors at operating reactor sites and selected construction sites.
- Increase direct verification of licensee activities by NRC inspectors. This includes both independent measurement by NRC and direct observation by NRC.
- Provide for a performance appraisal program on a national level by NRC. This program will appraise licensee performance, the effectiveness of the NRC inspection program and inspector objectivity.
- Improve manpower management.

The Revised Inspection Program should provide more direct NRC independent measurements, more direct observations of activities by NRC, and the opportunity for more direct communications between NRC and licensee workers—matters of concern to GAO—without large increases in manpower. The basic goal behind the Revised Inspection Program was to increase NRC presence at sites. However, due to budget constraints, NRC does not, at the present time, plan to assign resident inspectors at construction sites until the last three years of construction. Other construction sites and vendors will continue to be inspected from NRC regional offices.

The GAO recommendation for more staff to increase effort in the construction areas is not consistent with growth limitations imposed by the Office of Management and Budget. The NRC is instituting methods to foster efficiency and effectiveness within all program areas and will address the question of allocation of more resources to construction in fiscal year 1979.

7

Operating Experience

By closely monitoring the actual operating experience of its licensees, the NRC is enabled to confirm the bases for existing regulation and to uncover areas where regulation may need to be altered, introduced or removed. The licensees themselves, of course, have a vital interest in the record of their activities, in tracking every aspect of their operation and in reporting and investigating any event significantly deviating from expectations. It should be noted that the unplanned and abnormal events which have occurred during the report period in nuclear power plants have taken place within the context of an overall safety record for these kinds of facilities of 415 reactor years of operation without any nuclear accident causing detectable injury to the general public, as of September 30, 1978.

Included in this chapter are: (1) a discussion of the NRC's continued development of the Reliability Data System (see 1977 NRC Annual Report, pp. 91–92); (2) a summary of occupational radiation exposures, i.e., exposures to employees in licensed facilities; and (3) a digest of the "abnormal occurrences" of fiscal year 1978—those unscheduled incidents or events which the Commission determines were significant from the standpoint of public health or safety.

RELIABILITY DATA SYSTEM

In 1978, further review was made of appropriate responses to the President's request in his April 1977 message on energy "to make mandatory the current voluntary reporting of minor mishaps and component failures at operating reactors, in order to develop the reliable data base needed to improve reactor design and operating practice." In November 1977, the Commission had expressed the view that any mandatory system, which was expected to incorporate the existing Nuclear Plant Reliability Data System (NPRDS), should be the subject of a rulemaking proceeding in which industry, the public and interested parties would be given the opportunity to express their views.

In March 1978, the final report of the NRC NPRDS Working Group was completed. The Working Group had been established by the Executive Director for Operations in June 1977 to evaluate user needs; to evaluate the Licensee Event Report and NPRDS programs and their relationship to each other; to evaluate use in regulatory programs; and to look at training requirements. The Group concluded that the NRC has a variety of uses for NPRDS-type data in its regulatory program. They also noted that many years of data collection would be required before a useful data base would be generated. The Working Group agreed that participation in NPRDS should be made mandatory provided that adequate resources were available for NRC to use the data to full advantage and that other NRC information needs were fulfilled.

Upon completion of the NPRDS Working Group Report, staff comments were received which prompted a complete review of the NPRDS mandatory issue by the Technical Advisor to Executive Director for Operations. His recommendation, made in April 1978, was "that a case has not been made for making participation in NPRDS mandatory for NRC power reactor licensees."

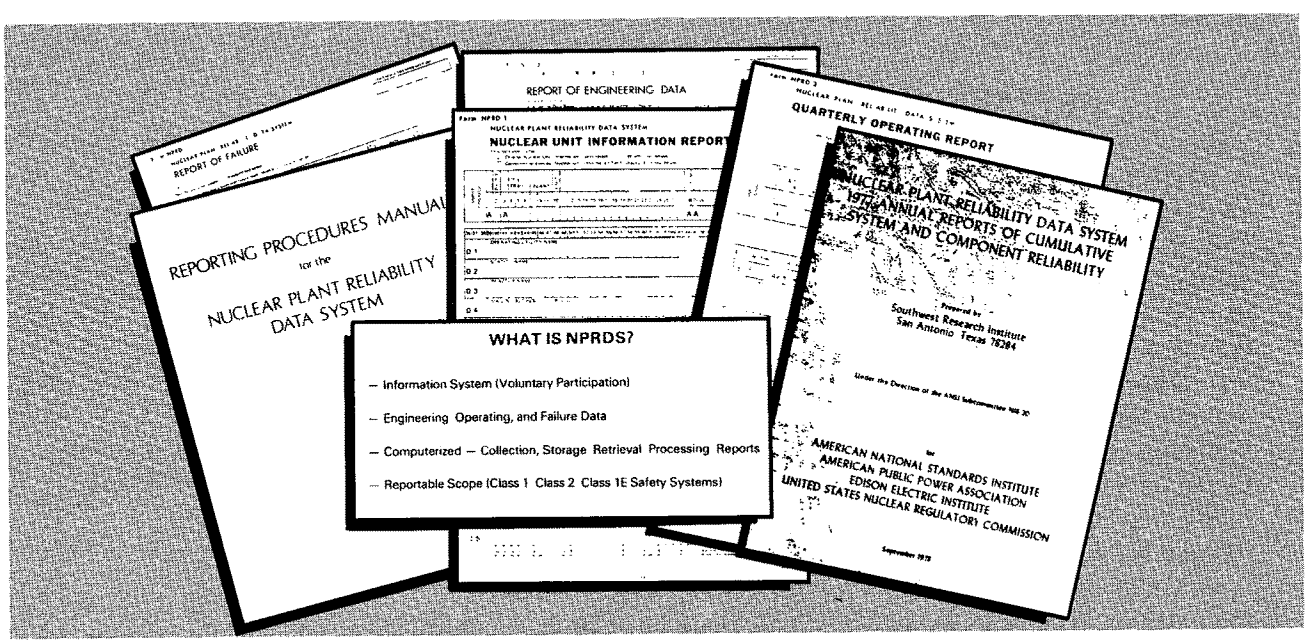
In May 1978, the ANSI N18-20 Subcommittee, which had been formed to direct NPRDS, provided the NRC with their comments on the NPRDS Working Group Report. Their report concluded that the "ANSI N18.20 Subcommittee does not consider the Working Group Report justifies mandatory reporting or NRC takeover and control."

In May and June 1978, presentations were made to the Advisory Committee on Reactor Safeguards (ACRS) full committee and the ACRS Subcommittee on Reliability and Accident Probabilities, respectively. These presentations discussed the mandatory reporting of NPRDS and included representatives from various NRC offices and from the ANSI N18.20 Subcommittee. In July 1978, the ACRS concluded that the "Committee sees no reason at this time to recommend that NPRDS reporting be made mandatory." The committee recommended that the staff and the industry continue to collect data, to improve the system, and to use the data and appropriate analysis as aids in effecting continuing improvements in reactor system safety and reliability.

In September 1978, the NRC requested of the ANSI N18.20 Subcommittee that some changes to NPRDS, as outlined in the NRC NPRDS Working Group Report, be evaluated for possible inclusion in the system. These changes were being evaluated by subcommittee members at the close of the report period to determine if 1979 funds should be available for this work.

OCCUPATIONAL EXPOSURES

Data on occupational exposures is collected from licensees in four categories required to submit annual and termination reports: power reactors, industrial radiographers, fuel fabricators



and processors, and certain processors and distributors of radioisotopes. The data for 1977 (the most recent available) indicated that 98,212 individuals were monitored by 457 licensees, showing a collective dose of 38,944 man-rems and an average individual dose of 0.40 rems. This is an increase over last years's average dose of 0.36 rems. Individual doses, however, continue to be well below the NRC's allowable limits, with only two reports of exposures exceeding the annual dose permitted by NRC regulations.

Beginning next year, all NRC licensees will be required to submit an annual statistical summary report. The NRC will then, for the first time, have data on the occupational radiation exposures being incurred by employees of all types of NRC licensees.

(See "Other Technical Issues," in Chapter 2.)

Reducing Radiography Overexposures

In the years 1971 through 1977, organizations licensed under 10 CFR Part 34 to perform industrial radiography accounted for 53 percent of the radiation overexposures greater than 5 rems whole body, or 75 rems to the extremities, reported by NRC licensees.

The average radiation dose received by workers in industrial radiography, however, is lower than that for five other classes of licensees, including power reactors. Thus, the NRC's principal concern in this area is a reduction in the number of overexposed individuals, and not in the buildup of long-term health effects in the worker population.

Under consideration to reduce the frequency of overexposures are possible regulatory actions for preventing causes of overexposures, as well as practical limitations on the extent to which industrial radiography safety can be improved by regulations and by equipment design changes. A paper entitled "Reduction of Radiography Overexposures," updated the previous "Action Plan to Reduce Radiography Overexposures," prepared by NRC staff and requested Commission approval to publish both proposed amendments to 10 CFR Part 34, dealing primarily with procedures for safe operation (see adjacent box), and an advance notice of proposed rulemaking. The latter identified for public comment certain design features to be considered as regulatory requirements, and announcement of a public meeting to discuss those design features.

Abnormal Occurrences Fiscal Year 1978

As required by law, the NRC reports to the Congress each calendar quarter on any "abnormal occurrence" that may have taken place involving facilities or activities regulated by the NRC. An "abnormal occurrence " is defined in Section 208 of the Energy Reorganization Act of 1974 as "an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety."

In making the decision that a given incident is or is not an abnormal occurrence, the NRC applies a criterion first promulgated in a policy statement issued February 24, 1977 (42 FR 10950), which provides that an incident or event which involves "a major reduction in the degree of protection of the public health or safety" shall be deemed an abnormal occurrence. The policy statement declares that such an event "would involve a moderate or more severe impact on the public health or safety and could include but need not be limited to:

"(1) Moderate exposure to, or release of, radioactive materials licensed by or otherwise regulated by the Commission;

"(2) Major degradation of essential safety-related equipment; or

"(3) Major deficiencies in design, construction, use of, or management controls for licensed facilities or material."

PROPOSED CHANGES TO PART 34 WOULD REQUIRE:

- Strengthened supervision and training
- Quarterly inspection of safety performance
- Improved procedures in use of radiography devices
- Audible and visible warning signals at permanent installations

During fiscal year 1978, a total of nine events were determined to be abnormal occurrences, and four events reported by Agreement States met the criteria for abnormal occurrences (see "Agreement State Occurrences," below). A summary of each of these 13 events is given below, following an update on three occurrences initially reported in earlier annual reports.

UPDATE OF EARLIER EVENTS

Fuel Rod Failure

This situation came to light on May 15, 1977 and was covered in the second quarterly report to the Congress for 1977; it is discussed on p. 101 of the 1977 NRC Annual Report.

As originally reported, a visual inspection of fuel assemblies at the La Crosse Boiling Water Reactor in Wisconsin showed that some sections of the fuel rods were missing from three assemblies. The licensee, the Dairyland Power Cooperative, calculated that about 51 total inches of fuel rod, containing about 742 grams of "elemental uranium," had broken away from the fuel elements. By the end of 1977, about 55 percent of the displaced material had, according to the licensee, been located and over half of that amount had been recovered.

Through a series of fuel recovery procedures—using recirculation through special filters and other techniques—the licensee was able to recover all but an estimated 220 grams of uranium (about four grams of U-235), which remained in the primary and secondary systems at the end of 1977. The licensee notified the NRC of the results of its efforts and of its conclusion that the remaining material would not interfere with the safe operation of mechanical components of the reactor.

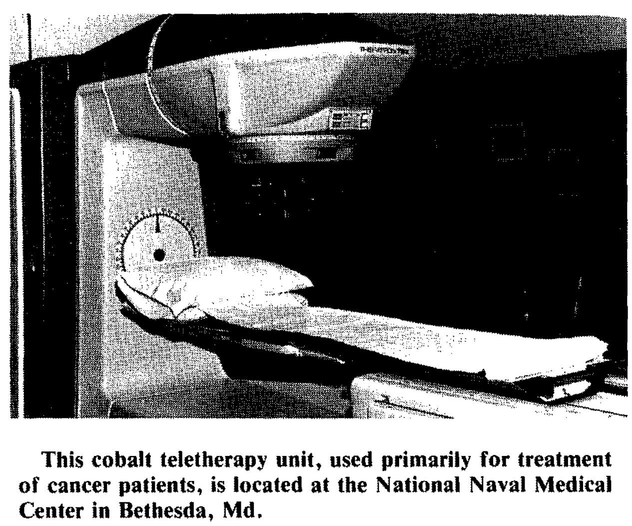
Further investigation of the problem by the licensee disclosed that, in the judgment of the licensee, the cause of the fuel failures at the facility was "fuel to cladding interaction and accelerated stress corrosion aggravated by the accumulative radiation exposure (burnup) of the fuel." To improve the integrity of the fuel elements and reduce the number and severity of fuel failures, operating restrictions were placed upon the rate of control rod movement, rate of reactor power increases, and the maximum allowable burnup limit for fuel assemblies within the reactor core.

Supplementing these restrictions were new limits imposed on the "off-gas" radioactivity release rates, since analysis revealed that previous fuel failures at the plant "correlated well" with those rates and they could thus be used as a monitor on fuel integrity. The unit has been operating at full power since restart in March 1978.

Overexposure of Teletherapy Patients

These incidents took place over the period of March 1, 1975 and January 30, 1976 and are considered a single abnormal occurrence, reported to the Congress in the second quarter report for 1976 and discussed in the 1976 NRC Annual Report, on pp. 109-110. Final actions on the occurrence were reported to Congress during fiscal year 1978 and are set forth below.

As reported before, about 400 patients at the Riverside Methodist Hospital in Columbus, Ohio, who were undergoing cobalt-60 teletherapy treatment there, received excessive doses of radiation, ranging from 10 percent to 40 percent more than the intended dose. The cause of the excessive doses was the incorrect calibration of the teletherapy unit, a situation which went unchecked and uncorrected for the period indicated. In July 1976, the NRC ordered the hospital to require periodic calibration of the unit by a qualified expert and improve management control of its operation. In August 1976, the NRC sent a bulletin to all licensees using



This cobalt teletherapy unit, used primarily for treatment of cancer patients, is located at the National Naval Medical Center in Bethesda, Md.

such units, directing them to perform comparison tests between the calculated and the actual output of their units and, if there were discrepancies found, to perform a full calibration of the instrument.

The situation at the Riverside Methodist Hospital was corrected to the satisfaction of the NRC. Both the hospital and local authorities are continuing to investigate the extent and implications of individual patient overexposures, however, and the NRC has taken specific actions to prevent a recurrence of the incident involving the unit at the hospital or similar incidents involving any of the approximately 500 units licensed by the NRC for medical use. These actions include:

- An extensive program conducted by NRC in which the radiation output of all licensed teletherapy units was evaluated. This program provided sufficient evidence to conclude that there were no other licensees with a calibration problem of the kind found at the Riverside Methodist Hospital. The calibration accuracy of other units was found to be satisfactory.
- A proposed amendment of NRC regulations to require annual calibrations of and more frequent periodic checks on teletherapy units licensed by NRC.
- The inclusion of direct physical measurements of the output of each teletherapy unit as part of the inspection of any licensed teletherapy facility.

NUCLEAR POWER PLANT EVENTS

The NRC reviewed events reported at the 70 nuclear power plants licensed to operate during fiscal year 1978 and determined that the following were abnormal occurrences.

Management Control Breakdown

This series of three events constituted a serious breakdown in procedural controls at the Zion Nuclear Power Station (Illinois), a two-unit power plant; they are treated as a single abnormal occurrence.

The three events took place on July 8, 10, and 12, 1977. The first two resulted in an inadvertent shutdown of the reactors at Unit 1 and Unit 2 respectively, and the third occurred at Unit 2 while the reactor was already shutdown.

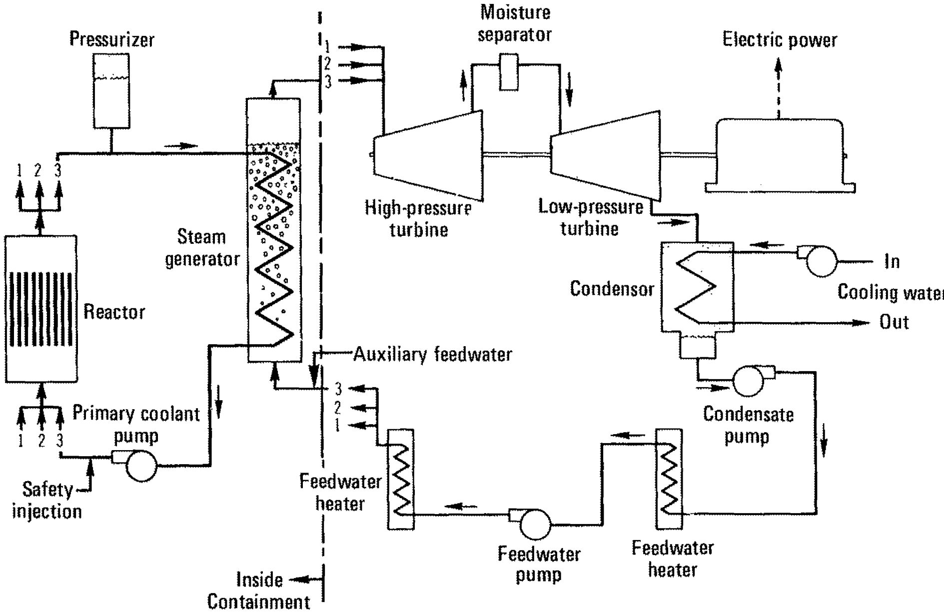
On July 8, during a periodic test of the reactor protection logic for Unit 1, the inadvertent omission of several procedural steps resulted in an automatic safe-shutdown of the reactor. The shutdown caused an automatic start-up of the auxiliary feedwater system; this system underwent a temporary pressure surge from "water hammer" (sudden steam condensation) in the line; vibration from the pressure surge activated safety signal transmitters in the area which resulted in activation of the safety injection system; operation of the safety injection system was terminated manually by plant personnel, prior to the 60-second operating time required under the system design.

This method of terminating operation of the safety injection system was not covered in operating procedures for this facility and there was insufficient evaluation of the total situation to justify the conclusion that safety injection was not needed. Subsequent analysis of the episode indicated that there was no damage to these systems or their components.

On July 10, 1977, at Zion Unit 2, a main feedwater pump failed because of lubrication problems. The pump failure led to a reactor shutdown, following which all auxiliary feedwater pumps automatically started up, according to design, and delivered the proper flow of water to the steam generators. About 20 minutes later, while the steam generator water levels were still below the feedwater spargers, the engineer on duty decided to start the motor-driven main feedwater pump. That action set up a water hammer in the line of sufficient magnitude to cause one or more transmitters in the area to initiate safety injection. The safety injection signal tripped the main feedwater pump and terminated the water hammer.

As in the incident two days before, the operations personnel again terminated the operation of the safety injection system in a manner not covered by the established procedures. In this instance, the motor casings of the two feedwater isolation valves were cracked. It was later determined that no stress limits in the piping had been exceeded and that the event was without consequence in terms of public health and safety.

On July 12, 1977, while Zion Unit 2 was in a "hot shutdown" condition (i.e., the unit was at



PRESSURIZED-WATER REACTOR SYSTEM

operating temperature and pressure, and the reactor at zero power level), management decided to perform a surveillance test of the reactor protection logic circuitry. To that end, artificial test signals were simultaneously inserted into 31 circuits, those connected with three "pressurizer water level" sensors; four "pressurizer pressure" sensors; three water level sensors in each of the four steam generators; and three flow sensors in each of the four primary coolant loops. These signals were supposed to be inserted only to the extent needed to simulate plant conditions during a "hot shutdown" of the reactor. Under conditions obtaining at the time, none of the signals need have been installed for the test. But all of them were installed.

The insertion of these signals had the effect of eliminating the ability of the "safety injection logic" (controlling automatic initiation of the Emergency Core Cooling System) to sense a loss of primary coolant from the pressurizer or a loss of heat-removal capacity in the steam generators. (The pressurizer maintains proper volume and pressure in the primary cooling system.) From an operational standpoint, the test signals resulted in false signals to the pressurizer level control system and to the visual displays used by the operator. Two automatic sensors remained in effect, however, and ready to activate the safety systems in the event of an accident; these were the containment high pressure sensor and the steam generator differential pressure sensor.

The signals had been installed for about 40 minutes when, because of unusual readings on pump seal flows and other signs, an operator requested that the test signals be removed. When this was done, the pressurizer level indication dropped below the range of the indicator, the result of a slight difference between the pressurizer water level test signal and the automatic pressurizer level control set point. In response to this condition, the charging pump flow was automatically reduced to the minimum pump flow rate and maintained there until the difference was removed. The consequence of the reduced flow rate was that coolant was being removed from the primary coolant system at a rate that was 75 gallons-per-minute greater than the rate at which coolant was being returned to the system. The pressurizer level was restored to an on-scale reading in about 10 minutes by the charging pumps.

Calculations showed that the lowest level reached during the event was in the surge line between the pressurizer and the piping of the primary coolant system. Water levels in the reactor remained normal and no fuel was uncovered. An estimated 3100 gallons of water were required to bring the pressurizer level back to its original level. No damage to plant equipment was detected.

All three of these events were brought about primarily by breakdowns in management and procedural controls, compounded to some extent by personnel error. Specifically, the causes assigned to the events were as follows:

July 8 event—the reactor trip was caused by the failure of the operator to follow each step of the test procedure. The water hammer and subsequent safety injection initiation were the result of using an obsolete procedure to regulate the auxiliary feedwater system flow rate; management had not seen to a proper distribution of the revised procedure.

July 10 event—start-up of the motor driven main feedwater pump should have been prohibited in established procedures while the steam generator feedwater spargers were uncovered. Past experience and the company's quality assurance procedures both dictate such a prohibition.

July 12 event—a review of this event disclosed that it occurred mainly because of an improper appraisal and approval of the request for surveillance testing on the part of management and plant operators. A contributing factor was inadequate communication between work groups.

Following a series of meetings between the licensee and the NRC, the former undertook a number of actions to prevent recurrence of these kinds of events. Assignment of a new plant manager and restructuring of operating organization, with a clear delineation of responsibility and authority, were effected. With respect to the water hammer and safety injection problems, revisions in procedure and personnel training were carried out, as well as certain modifications to the facility. In response to the event involving the pressurizer, the test procedure was changed to eliminate the use of dummy signals entirely, and any other tests requiring such signals are being reexamined to assure that they have been properly reviewed and approved before further use. The work request procedure has been modified to place major emphasis on its importance as a work control mechanism, and special training on system interactions was given the appropriate personnel.

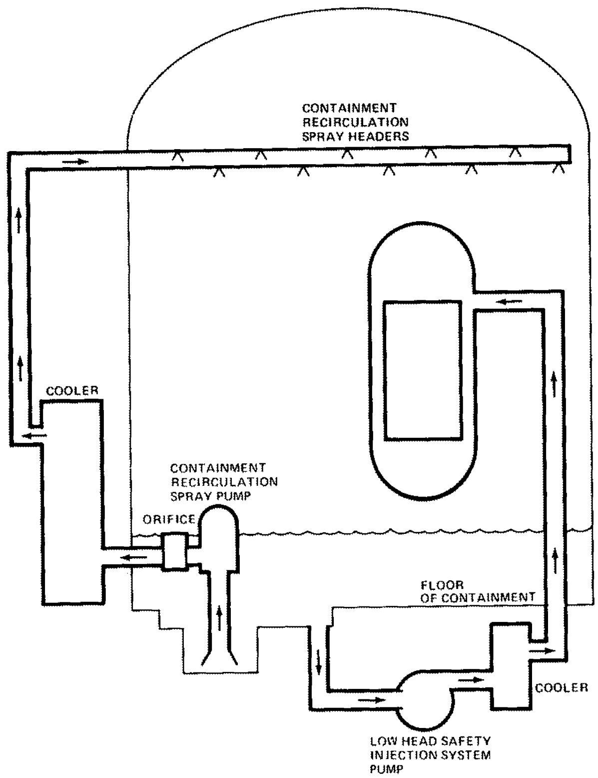
The NRC investigation of the incidents brought several items of noncompliance with regulations to light. Based on that investigation and the licensee's past history of noncompliance, the NRC issued the licensee a notice of violation on September 30, 1977, and proposed imposition of a civil penalty in the amount of $21,000. The penalty was predicated on six items of noncompliance, of which four were associated with the three incidents described above. One of these four, related to the July 12 event, was alleged by NRC staff to constitute a "violation," the most severe category of noncompliance. The licensee and NRC staff met to discuss the former's specific plans for prompt identification and correction of the kinds of factors which occasioned the failure of management control and permitted the errors which led to the three incidents and the items of noncompliance. A review of the management controls at the licensee's other facilities was also undertaken, and NRC inspection activities thereafter at all of the licensee's operating sites were augmented.

Generic Design Deficiency

On August 10, 1977, the NRC was informed by the Virginia Electric and Power Company that the architect-engineer for the utility's North Anna nuclear power plant, then under construction, had discovered a design deficiency affecting that proposed facility. Subsequent investigation showed that the same deficiency existed in "low head safety injection" (LHSI) system pumps and that the deficiency was generic to the following pressurized-water reactors with a subatmospheric containment design: North Anna Units 1 and 2 and Surry Units 1 and 2, facilities of the Virginia Electric and Power Company (VEPCO), and Beaver Valley Unit 1, a facility of the Duquesne Light Company.

The deficiency was found in the design of the containment recirculation spray system pumps (CRS pumps). It was determined that the "net positive suction head" available to the pumps in these systems was less than that specified by the pump manufacturer as necessary for the pump to operate as intended. The "suction head" has to do with the pressure at the inlet of a pump; if it is too low, the water will turn to steam and the pump may not operate correctly. The result may be too low a flow from the pump or damage to the pump.

Both the CRS and LHSI systems are safety features designed to mitigate the consequences of a postulated loss-of-coolant accident (LOCA), a low probability event. The CRS system is designed to remove heat from the containment in order to reduce the containment pressure to



A design deficiency affecting several pressurized water reactors concerned the Net Positive Suction Head (NPSH) at the inlet to emergency safety system pumps. The available NPSH must be greater than pump requirements to avoid cavitation (flashing to vapor in pump). The NPSH is a function of pump submergence below the containment water level, flow losses between pool and pump, water temperature, and containment pressure.

less than atmospheric pressure within one hour following a LOCA. The LHSI system is designed to inject cold borated water into the reactor core in the event of a LOCA. To fulfill their safety functions, the systems must be capable of providing the design flow rate under all postulated post-LOCA conditions of containment pressure and sump water temperature. Inadequate "net positive suction head" in the systems for extended periods could affect their ability to sustain the intended flow rate.

The deficiency was discovered when the architect-engineer undertook a reanalysis, at the licensee's request, of the containment pressure under accident conditions, using more conservative assumptions (i.e., assuming a minimum calculated containment pressure and a maximum sump water temperature).

Since the North Anna units had not been licensed for operation at the time the deficiency was discovered, the immediate safety concerns centered on the working power plants: Surry Units 1 and 2 and Beaver Valley Unit 1. In a meeting with the NRC staff on August 19, 1977, the licensees presented certain information in support of their contention that the affected facilities could continue safely in operation. The utilities maintained that:

(1) Based on information from the pump manufacturer, the CRS pump would continue to operate reliably at the calculated minimum available "net positive suction head," but at reduced flow and in a "cavitating mode," where vapor mixes with water and the pump's effectiveness is thereby reduced.

(2) All four CRS pumps in each affected plant had been determined to be operable, based on recent testing, and would not need to be removed from service while the deficiency was corrected.

(3) Predicated on the fact that the CRS pumps were operable at a reduced flow, the containment would, in the event of a LOCA, depressurize in less than one hour, thereby meeting the original design requirements.

(4) The probability of a LOCA requiring the operation of the CRS system was very small over the five-day interim needed to correct the deficiency.

The NRC staff found, on the basis of this information, that continued operation of the facilities was acceptable during the interim noted.

On August 24, 1977, Beaver Valley Unit 1 was shutdown for a maintenance outage, and the Duquesne Light Company made commitments to the NRC that operation would not be resumed until interim modifications had been made and approved by NRC staff. At this same time, VEPCO conveyed more data to the NRC regarding operation of the CRS pumps at the Surry station. According to the new data, the minimum "net positive suction head" required to assure satisfactory pump operation without cavitation was determined to be less than the specification of the pump maker, at the flow rate prescribed for the Surry (and North Anna) facility. The new tests also showed that the CRS pumps at Surry could be operated in the cavitating mode and at a reduced flow rate for at least 30 minutes without sustaining damage.

Considering these data, the utility proposed to make the following changes at the Surry facility to satisfy the intended performance level of the safety systems:

(1) Installation of flow-limiting orifices in the discharge lines of the two CRS pumps located outside the containment, reducing the flow and the required net positive suction head to a point less than that actually available. This alteration would assure continued pump operation without cavitation in the event of a LOCA. The combination of the reduced flow for the outside CRS pumps and the recirculation flow available from the remaining two pumps would be sufficient to serve their purpose under accident conditions.

(2) The CRS pumps located inside the containment would be required to operate in cavitating mode for a limited time (from 700 seconds to 2100 seconds after a postulated LOCA) and at a reduced flow rate. At all other times, the calculated available net positive suction head for these pumps would be greater than that required to preclude cavitation for the design flow rate.

(3) Limits would be set on certain operating parameters, such as service water temperature, containment temperature, and containment air pressure.

With regard to the LHSI pumps, a potential for pump cavitation was found to exist for a short period during the recirculation mode if the flow rate exceeds 3500 gallons-per-minute. To assure that this flow rate will not be exceeded, VEPCO proposed as an interim solution to throttle the valves in the pump discharge line while monitoring the flow rate in the control room to ensure that it is limited to 3500 gpm.

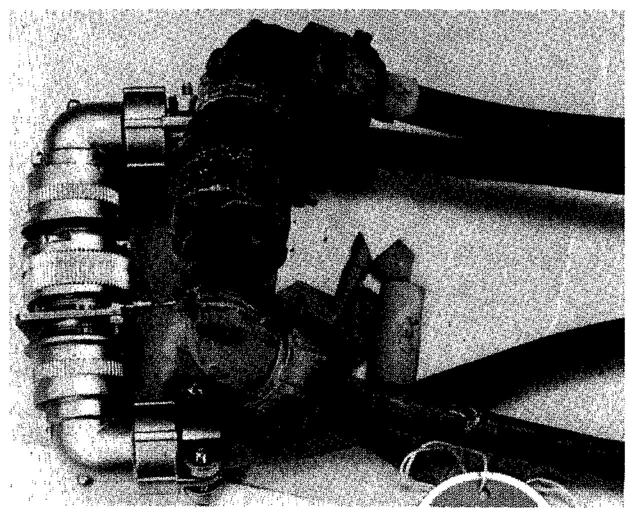
In similar fashion, actions were taken at Beaver Valley Unit 1 to restore the original margins intended in these systems.

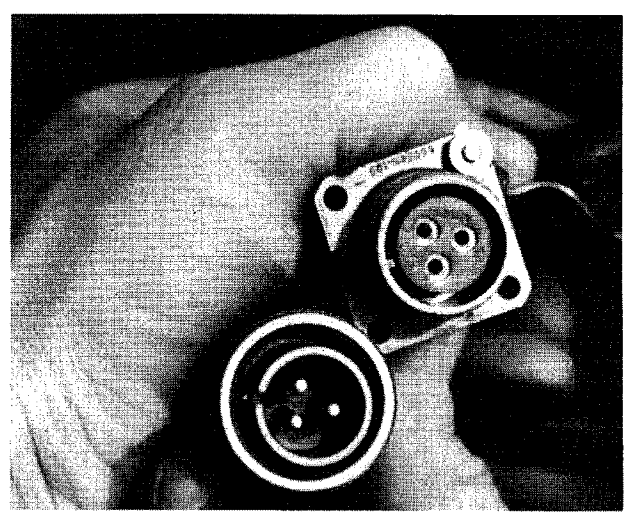
The NRC staff reviewed the design and operating changes proposed for the CRS system at the Surry station and found them to be acceptable on an interim basis. NRC directed the licensee to submit plans and schedules for realizing a final resolution of the deficiency within 90 days, beginning August 24, 1977. The proposed method for restoring the planned safety margin in the LHSI system was under study by NRC and subject to testing by VEPCO. Appropriate steps were taken to correct the problem at the North Anna units prior to the start of operations.

Qualifying Electrical Equipment

Early in the report period (fourth quarter of 1977), the possibility that some equipment in operating nuclear power plants had not been properly qualified in accordance with NRC requirements was identified as a potential safety concern.

All nuclear power power plants are required to have, in addition to the devices and procedures designed to prevent accidents, a number of safety systems whose purpose is to mitigate the consequences of postulated accidents, should they ever occur, such as a loss-of-coolant accident or a main steam line break accident. In the context of these safety systems, it is also required that any electrical equipment important





Electrical connectors were tested at the Sandia Laboratories under a program sponsored by the NRC's Office of Nuclear Regulatory Research.

to safety be "qualified" to function in the environment that might result from various postulated accident conditions. The potential safety concern at issue here is that some safety-related electrical equipment within the containments at some plants may not fully satisfy the regulatory criteria. The concern is significant because it could involve a major reduction in the reliability of engineered safety systems, such as the emergency core cooling system.

The Sandia Laboratories in New Mexico had been conducting tests on behalf of the NRC to obtain data by which to judge the suitability of NRC standards and regulatory guides pertaining to safety-related equipment. Specifically, the NRC and Sandia were concerned with the standards regarding qualification of such equipment to operate in a loss-of-coolant accident environment. The tests were focused on the adequacy of the testing methodology underlying the standards and guides and were not intended to verify the qualification of particular items of equipment.

Electrical Connectors. Twelve electrical connectors were tested under the program, selected because they were of a size that could be accommodated in the test facility. All 12 of the connectors failed at some point in the testing conducted during July 1977. Although the NRC staff's initial information did not indicate that such connectors were in use in reactor safety systems requiring continuity of service under loss-of-coolant accident conditions, the staff reexamined the matter upon receipt of a petition by the Union of Concerned Scientists (UCS) on November 4, 1977. The UCS petitioned the Commission to take certain emergency and remedial actions based in part on the qualification test results. A public briefing on the issue was conducted by the Commission on November 11 and additional affidavits and bulletins were put forward, as described below.

As a result of the Sandia tests, the NRC issued bulletin 77-05 on November 8, 1977, and supplemental bulletin 77-05A on November 14 to all 65 operating power reactor licensees, requiring them to determine whether their facilities were using connectors of the kind that failed at Sandia or whether the connectors that were in use had been properly qualified for operation under LOCA conditions.

After receipt of the UCS petition and prior to issuance of bulletin 77-05, the NRC conducted a telephone survey to ascertain from architect-engineering firms and others what kinds of connectors were in use in nuclear power plants, what use they were put to, and what grounds existed to believe they had been qualified for an accident environment.

The responses to the NRC bulletins and further inquiries led to the decision that further actions were required at 19 facilities, those with safety-related electrical connectors inside the containment which would be required to function in a LOCA environment. It was further determined that confirmatory testing and documentation of connector qualification would be necessary at an additional 16 plants.

Electrical Penetrations. Although the Sandia tests did not involve electrical penetrations (assemblies in the containment walls for the passage of electrical connectors), the UCS affidavits of November 10 and November 17, 1977, questioned the qualification of such penetrations on the basis of the Sandia tests and in light of problems with the penetrations at the Millstone Unit 2 facility. Because of electrical shorts in the penetrations at that plant, occurring during otherwise normal operation, the NRC had issued a bulletin on November 2, 1977, requiring licensees of operating reactors to provide oral and written information on their penetration assemblies. The event at Millstone Unit 2 was adjudged an abnormal occurrence (see section immediately following), and the NRC undertook a plant-by-plant review of the information provided in response to the November 2 bulletin.

In the course of its survey concerning electrical penetrations, the NRC uncovered instances in which certain other unqualified electrical components had been found and replaced by licensees. Further corrective actions will be required, if needed, on a plant-specific basis. The NRC also decided to require that the 11 facilities taking part in the Systematic Evaluation Program (see Chapter 2) evaluate the environmental qualification of all of the electrical equipment needed to mitigate the consequences of a design-basis accident. A decision as to whether or not such an assessment should be made at all other power plants would follow an appraisal of the results from these 11 facilities.

Commission Action. In its petition to the NRC of November 4, 1977, the UCS requested that the Commission shut down all operating power reactors, order the cessation of all construction involving electrical connectors and cables, and impose a moratorium on all power plant licensing until prospective licensees could demonstrate compliance with NRC regulations on system and component qualification. Seven plants did, in fact, shut down to test and, where necessary, replace connectors (or, in one instance, to seal certain connectors with epoxy pending the next refueling outage, at which time the NRC would require that full qualification of the sealed connectors be demonstrated or they be replaced). Two other plants extended regular outages in order to make necessary modifications.

On April 13, 1978, the Commission issued a Memorandum and Order which denied the emergency requests of the UCS petition insofar as they affect all licensed nuclear power plants. In response to non-emergency portions of the petition and to NRC staff and licensee conclusions, the Commission directed the staff to take certain actions related to environmental qualifications, including the following:

(1) Arrange for a repetition of the Sandia Laboratories' test program using a representative sampling of commercially available electrical connectors qualified in accordance with standards of the Institute of Electrical Engineers and in use in nuclear power plant safety systems.

(2) Provide the Commission with an analysis of alternatives for conducting independent testing to verify the qualification of safety-system equipment, including estimates of resource requirements and potential benefits.

(3) Carry out a comprehensive "lessons learned" evaluation, to include: (a) a review of all licensee responses on the subject, in order to determine their conformance to the applicable "quality assurance" requirements, as well as the accuracy and timeliness of the information provided in their responses (and appropriate enforcement actions to be taken, if that is indicated); (b) a review of how it came about that electrical equipment which was not fully qualified according to regulations was installed in some licensed power plants; (c) a review of NRC staff actions regarding one particular facility which was permitted to continue operation for some time following identification of the potential need to replace connectors, with a view to avoiding such delays in the future; and (d) a review of the need for further regulatory actions, including the possibility of an NRC policy statement, to re-emphasize the important safety responsibilities of licensees.

(4) Inform the Commission of the results of further qualification testing related to the three facilities for which fully documented test results were not yet available.

(5) Inform the Commission of the decision made on the question of whether or not nitrogen gas will be required for those containment penetrations which can accommodate such pressurization.

(6) Review the results of the first phase of the Systematic Evaluation Program, concentrating on the safety adequacy and environmental qualification of all class IE electrical equipment, and provide recommendations as to whether this review should be extended to other plants.

On May 2, 1978, the UCS requested that the Commission reconsider its decision of April 13 as a matter of discretion. On May 31, 1978, the Commission decided to entertain the request and asked for public comments on the UCS petition for reconsideration. At the same time, the Commission directed the NRC staff to perform an overall evaluation of the new petition, responding to certain issues raised in it and giving a complete and objective assessment of the petitioner's contentions.

With respect to directive number six, above, the staff completed its short-term safety assessment of the 11 facilities in the Systematic Evaluation Program and published the results in NUREG-0458, dated May 13, 1978. In the short-term review, the NRC staff did not identify any significant safety concerns that would require immediate remedial action at these 11 plants. Since these facilities include most older operating reactors—those likeliest to have a diminished level of environmental qualification for their safety-related equipment—the staff was of the belief that its conclusion could justifiably be generalized to include all operating reactor facilities.

Nevertheless, in light of previous problems associated with the qualification of electrical connectors, the staff issued a circular, dated May 31, 1978, to all licensees of operating reactor facilities, affirming the importance of assuring the qualification of safety-related components. The staff also took steps to incorporate the inspection of installed safety-related electrical equipment and an audit of the records for environmental qualification into regular NRC inspection activities.

(See discussion under "Action on Technical Problems," in Chapter 2.)

Failures in Insulation

This series of events is closely related to the foregoing account of concern with safety-related electrical equipment. Unlike the problems with qualification, however, these phenomena pointed up an actual, rather than a potential, condition reducing the margin of safety in the operation of a licensed nuclear facility. And the condition exposed was judged sufficiently serious to be dealt with as an abnormal occurrence.

The events transpired between September 30 and November 19, 1977, at the Millstone Nuclear Power Station, Unit 2, a pressurized water reactor plant in Waterford, Conn., licensed for operation by the Northeast Nuclear Energy Company.

Low-voltage wiring is routed through the wall of the containment building at this facility by means of electrical penetration assemblies. The wiring is part of various control or monitoring instrumentation. Within each assembly are "modules" carrying 85 wires, separated and insulated from one another by epoxy material at each end of the modules and by enamel on each of the wires within the modules.

On September 30, 1977, a valve in the "letdown system" (part of the reactor coolant purification system) closed unexpectedly and another valve in the "safety injection system" (part of the emergency core cooling system) was found to be open, though it is normally closed. Investigation of the condition revealed that the penetration module associated with controls for these valves showed low insulation resistance between several wire conductors—including those which govern these valves.

On October 14, 1977, a valve in the reactor coolant sample system failed to shut on electrical command; investigation showed that the wires in the penetration module associated with that valve were shorted together, the result of low insulation resistance.

On November 19, 1977, short circuiting of the wires in the "indicating and alarm" circuits for a reactor coolant pump caused false alarm signals in the control room of the plant, due again to low insulation resistance in a penetration module.

None of these events brought about an unsafe condition in the plant, but the potential for an unsafe condition existed because the safety-related equipment depends on wiring that functions properly and reliably.

The deterioration of the insulation in the penetration modules was originally thought to be caused by moisture seeping through cracks in the epoxy seals, but evidence from the lab tests showed that the failures were caused by heat at the connection splices within the modules. This heat, in turn, was caused by an intrusion of epoxy into spaces in the connection splices which were not insulated during the manufacturing process. The carbon deposits which resulted from the heating process created a conductive path and a short circuit between adjacent wires in the modules.

After each incident involving a malfunction in the penetration module, the conductor wires affected were replaced by others with acceptable insulation resistance. A test program was initiated to check on selected wires and other components periodically, and the modules were repressurized with nitrogen, producing some increase in insulation resistance. Following the last mentioned event, however, the plant was shut down (November 20, 1977) and the refueling outage scheduled for two weeks later was moved up to that date. During the outage the licensee replaced all 20 of the low voltage control power penetration modules with modules of a different design and undertook comprehensive testing of other penetration modules in use at the plant.

NRC staff had met with the licensee on November 7 to discuss all implications of the events at Millstone Unit 2. Subsequently the licensee agreed to the following: (1) penetrations at the plant would be continuously pressurized with nitrogen; (2) special surveillance of the modules would be performed; and (3) the plant would be shut down if any further degradation in the conductor wires was identified or any recurrence of insulation failure was experienced. It was because of that last provision that the plant was shut down on November 20.

Prior to permitting resumption of operations at the Millstone plant, the NRC reviewed and approved the design of the replacement penetration modules, as well as the results of tests conducted on the remaining modules. The replacement "feedthrough" modules were found to be environmentally qualified in accord with the appropriate IEEE standards, and the remaining modules were found to be suitable for continued service.

As a result of this abnormal occurrence, an NRC bulletin was issued on November 12, 1977: "Potential Problems with Containment Electrical Penetration Assemblies." The bulletin went to all licensees for operating reactors, requesting that they examine their installed penetrations and determine if the potential for failure such as occurred at Millstone existed, what methods were to be employed to detect possible degradation, and what corrective action was to be taken, if any.

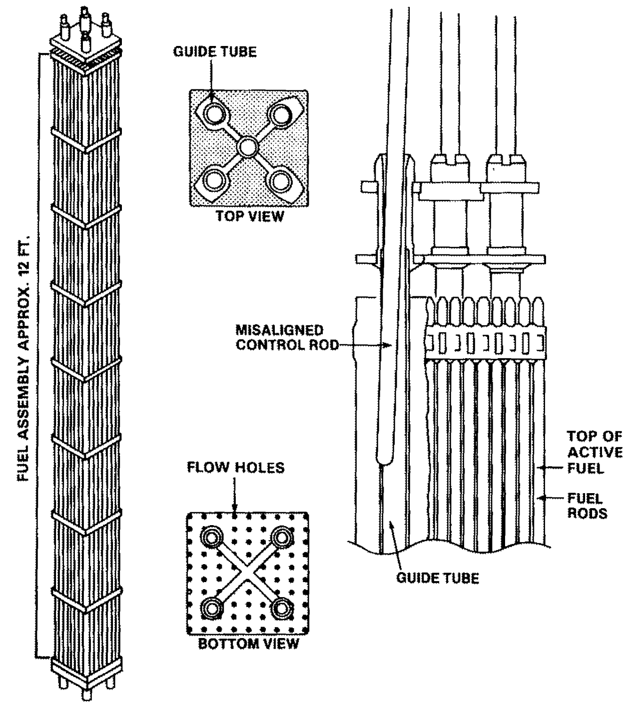
Information submitted in response to this bulletin was studied closely by NRC staff, which concluded that reasonable assurance was present that penetration assemblies in use at operating reactor facilities were capable of performing their design function in a LOCA environment. The qualification of penetrations in service was established by documented test results, while the assemblies that failed at Millstone Unit 2 were determined to be of a unique design (and were, as noted, replaced).

The generic problems associated with qualifying electrical components in safety-related systems continue to be of concern. (See "Qualifying Electrical Equipment," above.)

Worn Control Rod Guide Tubes

This generic problem first came to light on December 13, 1977, when holes were found in several control rod guide tubes at Millstone Unit 2, which was shut down at the time for refueling. Similar indications of tube wear in fuel assemblies were later discovered at other facilities designed by Combustion Engineering, Inc. (CE), viz., Calvert Cliffs Unit 1, St. Lucie Unit 1, and the Maine Yankee Atomic Power Plant. Wear was also suspected at Calvert Cliffs Unit 2 and an inspection was scheduled at the refueling outage. No significant wear was found in the CE-designed Fort Calhoun Station Unit 1.

The reason for the unexpected wear in the control rod guide tubes is believed to be flow-induced vibration of the control rods against the tubes. These tubes serve a dual function as both



structural members of the fuel assemblies and guide channels for movement of the control rods. Structural integrity of the tubes is essential under both normal and accident conditions to ensure that the reactor can be shut down and the reactor core maintained in a safe condition. The wear discovered at these plants occurs at the top of the guide tubes where the tips of the control rods are much of the time (in the fully withdrawn position), which supports the hypothesis that vibration of the tips is causing excessive wear. The safety function of the guide tubes has not been impaired as yet by this phenomenon, but it is obviously important to minimize the possibility of any impairment.

The fuel assemblies removed from the facilities have been subjected to extensive examination by CE and careful measurement of the amount of wear-induced erosion taking place in the tube wall was made. This analysis has provided the basis for concluding that continued operation at the affected plants is safe. Modifications to the fuel assemblies at Millstone Unit 2, St. Lucie Unit 1, and Calvert Cliffs Unit 1 included the installation of stainless steel sleeves in both worn and unworn guide tubes. The reinforcement provided by the sleeves is intended only as an interim remedy.

The NRC reviewed and approved the actions taken by affected licensees to assure the safety of continuing to operate the reactors. In particular, the NRC found that a 3-inch partial insertion of the control rods ("unsleeved" tubes) was an acceptable interim precaution, as was the use of steel sleeves. The partial insertion stops the wear at the portion of the tube corresponding with a fully withdrawn position and gives greater stiffness and resistance to vibration at that point. It was also determined that the sleeves do not hinder control rod operation nor significantly change core temperature or flow rates. NRC is requiring all affected licensees to conduct guide tube inspections during scheduled refueling outages and will continue to study the results of these inspections and take further action as necessary. Other reactor designs appear to be less susceptible to this kind of wear, probably because the control rods are supported differently. Nonetheless, the NRC is gathering information about other designs and actively exploring the generic implications of the problem.

Two Technicians Overexposed

On April 6, 1978, NRC received word from the Portland General Electric Company that two radiation protection personnel at the Trojan Nuclear Plant in Columbia County, Ore., had accidentally been exposed to radiation in excess of regulatory limits. The accident occurred on April 5, during refueling activities which had begun on April 1. The refueling procedure involves the removal of spent fuel elements by remote control from the reactor vessel. The elements are kept under water and moved to a "refueling cavity," a pool within the containment building. Later they are transferred through the containment wall into the spent fuel storage pool located in another building. The conduit through which the elements pass from the containment to the adjacent building is called the fuel transfer tube.

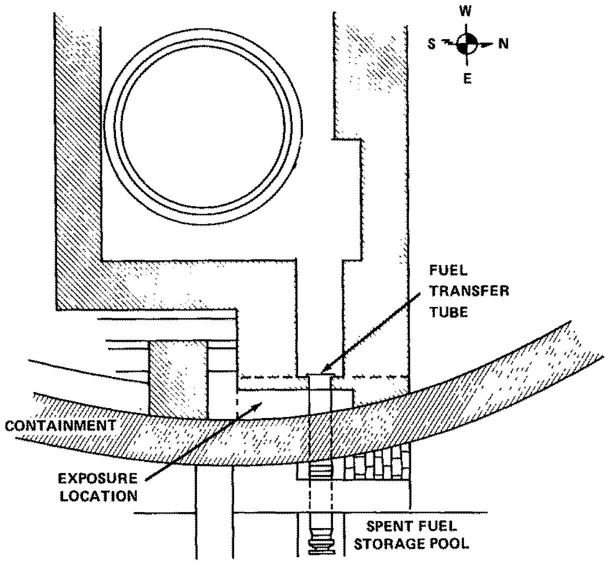
On April 1, personnel of the quality assurance (QA) staff notified the radiation protection staff that higher-than-expected readings had been obtained with pocket dosimeters on a 45-foot elevation of the containment building in the general area of the fuel transfer tube. The QA personnel were aware that the fuel transfer tube passes through a compartment at that elevation and is unshielded in that segment. The radiation protection personnel, however, were not aware of that fact.

On April 5, as part of continuing efforts to locate the source of the unexpected radiation, three radiation protection technicians entered the shielded area near the fuel transfer tube's "seismic bellows" compartment, and two of them climbed up into the compartment. The technicians believed that the tube which passed through the space was a ventilation duct, and they thought the fuel transfer tube was encased in concrete beyond the far wall of the compartment they occupied. The closed tube, about two feet in diameter, which passed through the compartment was the fuel transfer tube. The technicians had scheduled their presence in the compartment to coincide with the transfer of a spent fuel element through the fuel transfer tube. They expected radiation levels to run about 200 milliroentgens per hour.

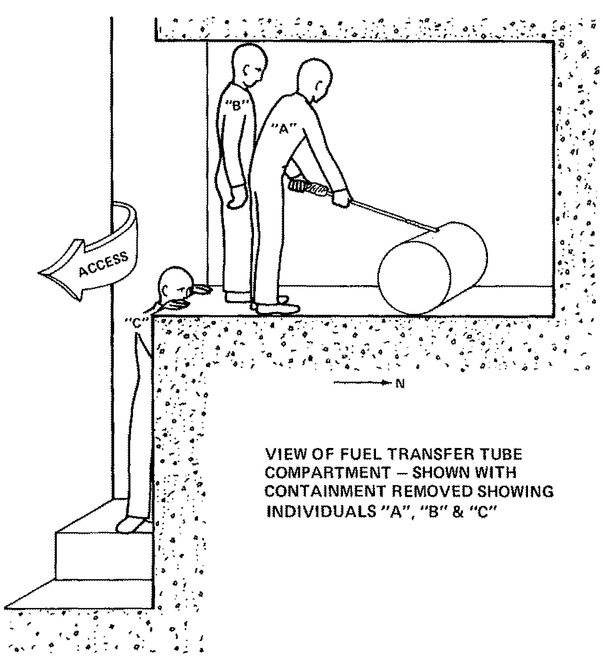
At about 3:30 p.m., the technicians observed their survey instrument go to full scale on the two-roentgens-per-hour scale. The one holding the instrument leaned over the transfer tube to bring the detector closer to the far wall. When the instrument was switched to the 50-roentgens-per-hour scale, the reading dropped off to zero (it was later learned that there was a malfunction in the instrument and it did not operate on the 50- or 100-roentgen scales). It required about 22 seconds for the spent fuel element to pass through the unshielded stretch of tube in the seismic bellows compartment.

On leaving the compartment, the two technicians found that their 200-milliroentgen and 1-roentgen pocket ionization chamber dosimeters were completely discharged to an off-scale reading. The third technician's dosimeter showed an exposure of 165 milliroentgens. The thermoluminescent detectors (TLDs) of the first two technicians were immediately sent to the vendor for processing.

On April 6, the TLD results came back, indicating that the technician nearest the fuel transfer tube had received a whole body dose of 12.9 rems and his companion a dose of 17.1 rems (the TLD of the first technician may have been shielded from the source of radiation by his body). The licensee later performed a special study to try to ascertain the actual doses to the technicians and concluded that the first technician had received a dose of 27.3 rems, the second a dose of 17.1 rems, and the third a dose



TROJAN – 45' ELEVATION



VIEW OF FUEL TRANSFER TUBE COMPARTMENT – SHOWN WITH CONTAINMENT REMOVED SHOWING INDIVIDUALS "A", "B" & "C"

Two technicians were exposed to excessive radiation as they worked in the fuel transfer tube compartment at the Trojan nuclear power plant. They were not aware that the tube passing through the compartment was an unshielded tube being used to transfer highly radioactive spent fuel through the containment building wall to the fuel storage pool.

that was not excessive. Blood tests on the three turned out negative. The two overexposed individuals were removed from work for the remainder of the calendar quarter.

The principal causes of the incident were identified as: structural design which failed to provide for access control and suitable shielding for the protection of plant personnel; a failure of communications between working groups; a failure on the part of radiation protection personnel to adequately assess the potential radiological hazard in the area. The malfunction of their survey instrument was not considered a significant contributor to the accident.

The licensee took the following corrective actions in response to the incident: new training for radiation protection personnel in selected plant systems; reorganization of the radiation protection group and the addition to it of a supervisor trained in both radiation protection and plant operation; testing of all radiation detectors in use at the plant in all ranges; and distribution of special instructions in the calibration and use of detection equipment to all chemistry and radiation protection technicians.

The NRC investigated the incident and evaluated the licensee's plans for preventing recurrence of the event; an NRC bulletin on the matter was sent to other licensees. Civil penalties in the amount of $20,500 were imposed in the case for the licensee's non-compliance with regulations.

Crack in Primary System Pipe

An abnormal condition in the primary system piping at the Duane Arnold Power Plant (Iowa), a boiling-water-reactor facility, was reported to the NRC on June 17, 1978, by the licensee, the Iowa Electric Light and Power Company. A crack had been found in the piping of the reactor coolant system, specifically in the fitting (a nickel-steel alloy) which joins the recirculation pipe to the reactor vessel.

The reactor had been shut down at the plant on June 17 because of a problem unrelated to this condition. The licensee decided to use the opportunity to check out the source of a leak in the primary coolant system which leakage monitoring equipment had detected but which was still within limits allowed by the technical specifications. The crack was found in the fitting on the recirculation pipe during inspection of the coolant system piping. The recirculation pipe is 10 inches in diameter and carries primary coolant to two jet pumps located inside the reactor vessel; the pumps in turn circulate the coolant through the reactor core. The crack was found to run eight inches long on the outer surface and about three-fourths around the circumference of the inner diameter of the nozzle transition piece, or "safe end," part of the recirculation line near an attachment weld. There are seven other such lines, each leading to two jet pumps, and tests showed that all seven had indications of potential cracks or weld irregularities, though none of them showed penetration through the pipe wall. The repair procedure was initiated while the reactor remained shut down and was continuing at the close of the report period. There was no threat to the public health and safety from the leakage. The recirculation line, however, is part of the primary system "pressure boundary," one of several barriers to prevent the release of radioactive material, and plant operations are not permitted if this boundary is degraded.

The licensee removed all eight safe ends from the system and sent the leaking piece to a metallurgical laboratory for analysis. The NRC sent a second cracked piece to a different laboratory for metallurgical analysis. Both analyses gave preliminary findings that the cracking originated at a point where another pipe, called a thermal sleeve, was welded to the inside of the safe end, and that the crack then propagated outward. The preliminary indication is that the cracks were caused by intergranular stress-assisted corrosion.

Because of similarities in design and material composition with the Duane Arnold system, the safe ends at the Brunswick Units 1 and 2 facilities (North Carolina) were examined by non-destructive techniques. Indication of a potential crack or weld irregularity was found in Unit 1.

The Iowa licensee obtained safe ends of a different design which is intended to minimize high stress points, and also developed an extensive training program to qualify welding personnel and certify welding practices and equipment. The significance of these events at the affected plants and for other BWR facilities is being pursued by NRC and the licensees.

(See discussion under "Action on Technical Problems," in Chapter 2.)

FUEL CYCLE FACILITY EVENTS

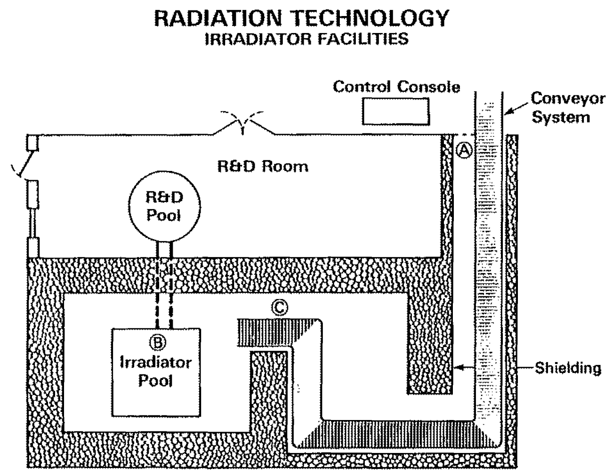
The NRC has reviewed events reported by these licensees during fiscal year 1978 and determined that none of the events was an abnormal occurrence.

OTHER NRC LICENSEE REPORTS

Overexposure at Irradiation Facility

On September 23, 1977, NRC received notice from its licensee, Radiation Technology, Inc., of an accidental overexposure to an employee at its "in-air irradiation" facility, a plant engaged in the radiation sterilization of various materials. During the week of September 19, the licensee had been modifying the conveyor system at the plant. On September 23, while material was "set up" in the facility, an operator entered the area, circumventing both a safety lock on the cell entrance and a lock to prevent withdrawal of the radiation source. The safety devices could be circumvented because a temporary entrance had been created for use during modification of the facility. The operator raised the 500,000-curie cobalt-60 array from the storage pool to begin irradiation of the material. That action happened to coincide with a scheduled shift change at 12:30 a.m., and a worker just arriving at that time was not warned that the array was in the exposed position. This worker entered the facility through the open door to adjust the position of the material to be irradiated and began to move the material about. After a few seconds he realized that the array was exposed and immediately left the area. It was estimated that he had spent 10 to 20 seconds in the radiation zone. From reenactment of the incident and the worker's film badge it was estimated that he had received a whole body dose of about 220 rems in the time he was exposed to the source.

The direct cause of the mishap was the decision by plant management to permit the source to be raised while the safety devices were inoperative. The initial negligence was compounded by the worker's failure to take a radiation survey of the area he was entering, his failure to follow procedures controlling access to a high radiation area, and management's failure to give thorough training in procedures to the employees.



An employee, entering the facility at (A), mistakenly assumed the sources in the irradiator pool (B) were in a locked, safe position. Just coming on shift, he did not know that a safety lock at the entrance and the lock to prevent withdrawal of the sources had been bypassed before his arrival. The worker became exposed (C) to radiation when he proceeded past the protective shielding of the maze without taking a radiation survey.

The NRC issued an "order suspending license" on September 23 and all operations at the facility were halted while the NRC investigated the accident. The licensee convened a panel of three consultants to review the matter and evaluate the operation of the facility in general. In a series of letters to the NRC, the licensee documented the new procedures it was adopting, the modifications to the facility, and management commitment to safety control. The license was restored on October 14, 1977, and operations were resumed. The revised procedures, facility modifications and management control commitments were incorporated into a license amendment; NRC inspectors confirmed that prescribed corrective actions had been taken and new license conditions were being observed.

Radiographer's Hand Exposed

This incident occurred at the Neville Island facility of the Pittsburgh-DesMoines Steel Company of Pittsburgh, Pa. At about 3 a.m. on November 12, 1977, a radiographer was engaged in taking radiographs of metal objects manufactured by the licensee with a 75-curie iridium-192 source. At one point in the operation, the radiographer neglected to retract the source before approaching the device to adjust the source guide tube in preparation for the next radiograph. Before adjusting the guide tube, he placed his radiation survey instrument on top of the shielded containment without checking it, and afterward picked it up and returned to the control crank. When he found the crank in a position indicating that the source was exposed, he retracted the source immediately and notified his supervisor.

From re-enactment of the event it was estimated that the fingers of the radiographer's left hand were in close proximity to the source for 3 to 5 seconds. That time-frame and examination of the film badge led to the estimate that the radiographer had received a whole body dose of about 0.6 rem and a dose to the fingers of the left hand of 300-600 rems. The individual was hospitalized for medical observation and the licensee retained a medical consultant to monitor the case. The radiographer suffered no clinical symptoms and returned to work soon thereafter.

The principal cause of the incident was the failure of the radiographer to retract the source back into the shielded position before approaching it; a contributory cause was his failure to take a radiation survey of the area he was entering.

The NRC inspected the licensee's operation and met with management to discuss the accident and the latter's plans for preventing any repetition of it. The licensee indicated that it would augment its internal audit program with audits by qualified persons outside the staff, that retraining of all personnel engaged in radiography would be undertaken (and a thorough study of this incident would be part of it), and that the trainees' level of understanding would be confirmed by written tests and on-the-job observation. Enforcement action was initiated by the NRC, including the proposed imposition of a civil penalty in the amount of $7,000.

AGREEMENT STATE OCCURRENCES

Under Section 274 of the Atomic Energy Act, as amended, the NRC is able to consummate agreements with the States whereby the latter—called Agreement States—assume regulatory authority over byproduct, source and special nuclear materials (in quantities less than that needed to sustain a chain reaction). While unplanned events at facilities licensed by Agreement States have been treated in publications of the NRC before that time, the Commission decided in early 1977 that events at such facilities which meet the criteria for abnormal occurrences should be included in the NRC quarterly report on that subject to the Congress. The following four such occurrences were included in reports of fiscal year 1978.

Radioactive Source Disconnected

On October 10, 1977, the Louisiana Nuclear Energy Division was notified by Riley-Beaird, Inc., an industrial radiography firm, that a radiographer in its employ had incurred an overexposure.

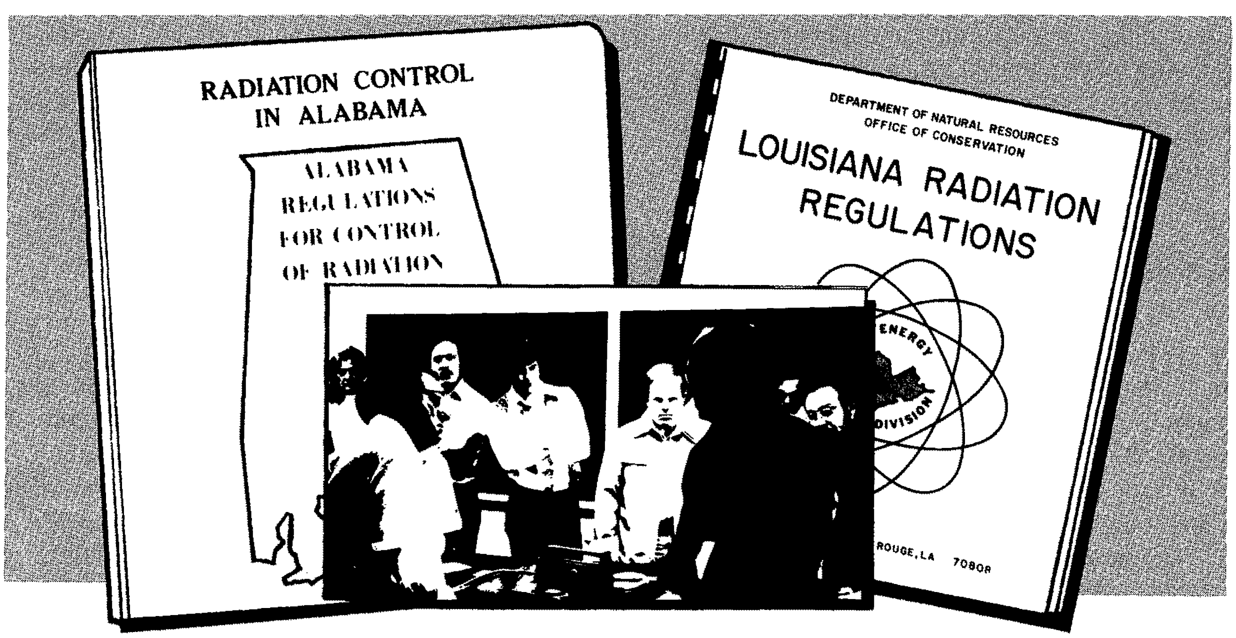
The accident came about in the following manner. When the shift ended for one of the firm's radiographers, he left the site of the exposure device—which contained 34 curies of cobalt-60—with only the drive cable and controls attached to the device, and not the source guide tube. When the second radiographer came on duty, he attempted to crank out the source, in the belief that the device was ready for use. Because the guide tube was disconnected, the source became disconnected from the drive cable. The radiographer contacted the chief radiographer who tried to retrieve the source and, in the effort, inadvertently touched the source capsule for about 0.2 second. The chief radiographer was wearing a direct-reading dosimeter but no film or thermoluminescent dosimeter badge. The direct-reading dosimeter was found to be discharged beyond the range of its indicator.

From a re-enactment of the incident it was calculated that the individual had received a dose of 6,000 rems to the fingertips and a whole-body dose of 8 rems. Blistering of the fingertips was evident about 10 days after the accident, and the radiographer was placed under the care of a physician.

The cause of the occurrence was the fact that the source guide tube was not attached to the device as it should have been and the fact that the victim of the overexposure took hold of the wrong end of the assembly. The licensee has modified the exposure device so that the source cannot be moved from the shielded position unless the source guide tube is attached and has made commitments to the Louisiana Nuclear Energy Division that it will not, in the future, attempt to retrieve a disconnected source, but will engage experienced consultants to do so.

Willful Violation of Regulations

On May 2, 1978, a routine inspection of SERICO, Inc., of Mobile, Ala., holder of Alabama Radioactive Material License No. 595, was begun. Eight days later, a former employee of the licensee submitted a set of written allegations



to the State concerning possible willful violations of regulations by the licensee; at that point, the routine inspection became a formal investigation.

The following violations were confirmed by the State health authorities:

(1) Inadequate instruction of radiographer's assistants in operating and emergency procedures.

(2) Inadequate training and supervision of radiographers.

(3) Allowing an individual to continue working in a radiation area after the individual's exposure to radiation had exceeded the quarterly limit.

(4) Allowing individuals to continue working with radioactive material or in a radiation area after their dosimeters had gone off-scale and prior to receiving the results of film badge processing.

(5) Submission of false reports to the Alabama Department of Public Health and the falsification of certain required reports by individuals in order to conceal information from the State.

(6) Allowing individuals to work in a radiation area without documenting their qualifications and training.

(7) Allowing individuals to take oral examinations without documenting their qualifications and training.

The hearing officer concluded that the licensee management knowingly permitted or required the willful violations of regulations in the manner described, risking unnecessary and possibly excessive exposure of individuals in the company, as well as employees of other companies and the general public. The company's license was revoked as of July 7, 1978.

This incident constitutes an example of the fact that serious deficiencies in management or procedural controls can be considered an abnormal occurrence.

Offshore Accident—Cause Unknown

On August 15, 1978, the NRC was notified by the Louisiana Nuclear Energy Division of evident overexposures incurred by radiographers and radiographers' assistants working on a barge 100 miles out in the Gulf of Mexico. The personnel affected were employees of the Monroe X-Ray Company and the overexposures took place between June 20 and July 8, 1978, during pipeline radiography. One radiographer's assistant was hospitalized for indications of serious overexposure to his hands. The film badge supplier reporting on this individual's badge dosimeter estimated that he had received a whole body dose of 5,450 millirems. It was estimated from clinical indications that the dose to his hands ranged from 3,000 to 10,000 rems. The employee was placed under medical supervision. Film badge reports on another six employees showed exposures to whole body doses up to 6.1 rems.

Neither the individual receiving the severe overexposure nor the other members of the radiography crew could recall any unusual event or circumstance that might account for the excessive dosage. State authorities concluded that the radiographer's assistant must have handled the source tube on the "crank-out" radiography device without retracting the source to the shielded position.

Investigation was completed and a hearing process had been initiated by the State at the close of the report period.

Radiography Cameras Stolen

On August 28, 1978, the Louisiana Nuclear Energy Division reported the theft of two radiography devices to the NRC. The devices are "crank-out" radiography cameras and they were taken from the Pittsburgh Testing Laboratory Storage Vault in Morgan City, La., sometime between the evening of August 24 and of August 25. The licensee contacted all employees who might have taken the equipment without filling out the utilization log before notifying State authorities of the theft.

The two devices contain 39 curies and 33 curies of iridium-192 respectively. They were, according to the licensee, properly locked and labeled, and the storage vault was locked during the period in which the theft presumably occurred. Given those facts, and the fact that nothing else in the vault was disturbed, there is reason to believe that the person or persons who took the devices knew what they were looking for and how to handle it. The possibility remained, however, that, if the devices got into the hands of persons unfamiliar with radioactive sources, accidental overexposures could happen.

The State has determined that the licensee did not contribute by neglect or lack of security to the theft of the devices, and thus, at the close of the report period, no specific cause had been assigned to the occurrence and the matter remained under investigation.

The licensee, besides contacting the State Nuclear Energy Division and the local police, issued warnings through the media that the devices could be dangerous if unlocked and that, if they are found, local authorities should be notified at once. Potential purchasers of this kind of equipment were alerted to the theft, as were regulatory agencies in adjacent States.

8

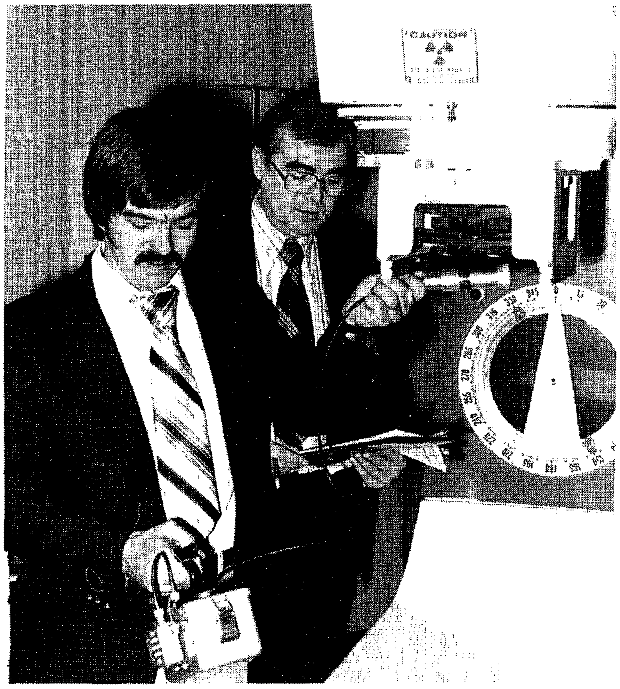
State Programs

During 1978, the NRC continued to expand its cooperative and working relationships with State and local governments in radiation control, licensing and siting of nuclear facilities, and planning appropriate actions in the event of radiological emergencies. Highlights of the fiscal year included the conclusion of formal agreements with four States regarding water quality and other matters relating to nuclear licensing actions, and NRC concurrence in six more State plans for radiological emergency response. NRC also began to assign liaison officers to its regional offices in order to establish closer contacts with the States.

While the NRC's consultations with the States are wide ranging, and involve activities of many of its larger program offices, the principal responsibility for NRC/State interaction is centered in the Office of State Programs. Several important areas which received particular attention during the report period in this chapter are (a) the State Agreements Program, providing for the relinquishment by NRC and assumption by the States of regulatory authority over nuclear materials; (b) assistance to State and local governments for radiological emergency response planning; and (c) cooperative activities on key NRC responsibilities affecting the States, such as siting, licensing, decommissioning, waste management, and transportation of radioactive materials.

STATE AGREEMENTS PROGRAM

Section 274 of the Atomic Energy Act of 1954, as amended, authorizes the Commission to enter into agreements providing for the assumption by qualified States of regulatory responsibility over byproduct and source material and small quantities of special nuclear material. At the end of 1978, there were 25 Agreement States which exercised regulatory authority over some 11,500 nuclear material licenses. The current Agreement States are: Alabama, Arizona, Arkansas, California, Colorado, Florida, Georgia, Idaho, Kansas, Kentucky, Louisiana, Maryland, Mississippi, Nebraska, Nevada, New Hampshire, New Mexico, New York, North Carolina, North Dakota, Oregon, South Carolina, Tennessee, Texas and



An NRC reviewer observes as an Agreement State inspector performs an independent radiation level measurement at the head of a teletherapy unit during inspection of a State licensee's facility.

Washington. No State has entered the NRC's Agreement Program since 1974; however, recent indications of interest have been received from Rhode Island, Indiana, Michigan and Illinois.

Reviews of State Regulation

The NRC conducts a formal annual review of the radiation control program of each Agreement State to determine whether it is adequate to protect the public health and safety and compatible with NRC's regulatory program. These reviews cover the organization, administration and staffing of the program; program regulations; licensing and compliance functions; and field evaluations of State inspectors. During fiscal year 1978, the NRC conducted 29 program reviews and one followup review. NRC staff also visited three State-licensed uranium mills during the year.

Adequacy and Compatibility Findings. During calendar year 1977, the NRC found that 24 of the 25 Agreement State radiation control programs were adequate to protect public health and safety. A determination of adequacy for the State of Washington was deferred on the basis of results of the annual review meeting with the State which was held October 31 to November 4, 1977. The review disclosed various program deficiencies, mostly attributable to a shortage of professional staff. Subsequently, the staff conferred with State officials concerning actions necessary to correct the deficiencies. A followup review, conducted March 28-31, 1978, determined that Washington's program was adequate to protect the public health and safety.

With respect to the criterion of compatibility of Agreement State programs with NRC's regulatory program, the NRC determined that 20 of the 25 States had compatible programs in calendar year 1977. Those with programs considered not fully compatible, in addition to the State of Washington, were Nebraska, New Mexico, Nevada and New York. Affirmative findings of compatibility for these four States were deferred because they had not adopted regulations equivalent to those of the NRC dealing with requirements for notices, instructions, and reports by licensees to workers (10 CFR Part 19).

The Department of Labor exempts State licensees from its regulation under the Occupational Health and Safety Act when the NRC certifies that the radiation control program of the State concerned is adequate to protect the public health and safety, and is compatible with NRC's program.

NRC Technical Assistance

The NRC provides a variety of technical assistance to the Agreement States, including evaluation of major licensing actions, review of amendments to State regulations, guidance in inspection and enforcement matters, and evaluation of complex health physics problems. NRC staff also assists in reviewing and evaluating the environmental impacts of major licensing actions. In 1978, for example, NRC assisted Colorado in evaluating the environmental impacts of two uranium mill operations, and is currently negotiating an agreement with Colorado whereby NRC will perform certain environmental analyses on future mill sites in that State. NRC has offered similar assistance to all other

Agreement States where mills are located and active negotiations are being conducted with New Mexico. (See discussion under "Assistance to Agreement States," Chapter 3.)

NRC also helps Agreement States in developing model legislation and regulations, and with reviews and comments on proposed changes to State regulations. During fiscal year 1978, NRC developed suggested State license fee legislation and provided comments on regulations to 12 States.

Training Offered by NRC

State regulatory employees regularly take advantage of training provided by NRC to increase technical and administrative skills. The training, which is available to employees of both Agreement and non-Agreement States, is provided at no cost to the States.

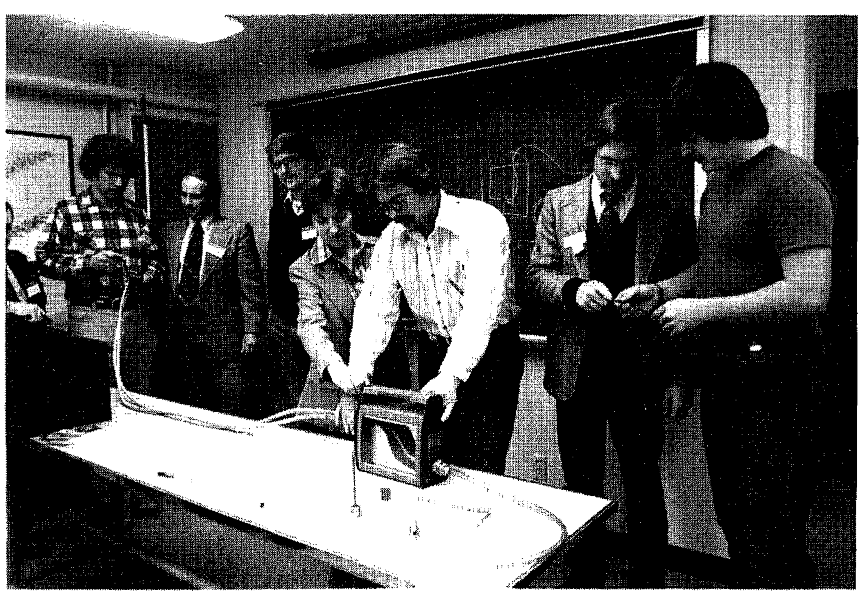
Eight training courses were presented last year: Safety Aspects of Industrial Radiography (twice), at Louisiana State University; Health Physics and Radiation Protection, at Oak Ridge Associated Universities; Medical Use of Radionuclides, at Baylor College of Medicine; Calibration of Teletherapy Machines (twice), at M.D. Anderson Hospital; Inspection Procedures, at NRC Region III Office; and Orientation in Regulatory Practices and Procedures, at NRC Headquarters. In all, 117 State employees received 282 student-weeks of training.

Annual Meeting

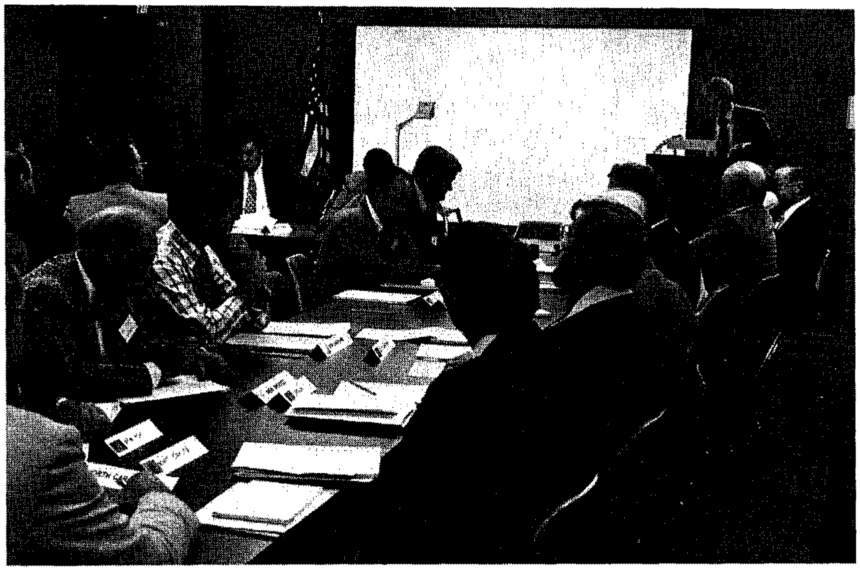
Radiation control program directors in the Agreement States participate in annual meetings at NRC Headquarters for discussions of a wide variety of issues. Topics discussed at the October 1977 meeting included the Clean Air Act Amendments of 1977, medical licensees, regulation of waste management, regulation of uranium mills, inspection and enforcement, equal employment opportunity, emergency planning and response, license fees, abnormal occurrence reports, and transportation. The meeting produced recommendations by State representatives to the NRC about training programs, inspection of generally licensed devices, NRC/Agreement State compatibility, funding of Agreement State programs, NRC medical licensing, and decommissioning of previously licensed facilities.

EMERGENCY RESPONSE PLANNING

The responsibilities of Federal agencies for assisting State and local governments in developing plans for responding to radiological emergencies are outlined in a *Federal Register* notice of December 24, 1975, promulgated by the Federal Preparedness Agency (FPA) of the General Services Administration. The notice, entitled "Radiological Incident Emergency



State students learn the operation of a radiography camera using a demonstration camera. This radiography course at the Nuclear Science Center, Louisiana State University, is sponsored by NRC's Office of State Programs.



NRC Chairman Joseph M. Hendrie addresses the annual meeting of Agreement State representatives in October 1977 at NRC headquarters in Bethesda, Md. Radiation control program managers in the Agreement States discussed regulatory matters of mutual interest with NRC officials and staff.

Response Planning; Fixed Facilities and Transportation," gives the "lead agency" role to NRC, while assigning specific support responsibilities to the Environmental Protection Agency (EPA); the Department of Energy (DOE); the Department of Transportation (DOT); the Department of Health, Education and Welfare (HEW); the Defense Civil Preparedness Agency (DCPA); and the Federal Disaster Assistance Administration (FDAA) of the Department of Housing and Urban Development. The entire effort is monitored by the FPA.

In carrying out its role, NRC has prepared and issued planning guidance, developed and conducted training courses, and provided field assistance to State and local governments to develop and test radiological emergency response plans. NRC also reviews and evaluates these plans, and determines the instrumentation requirements for measuring the offsite consequences of radiological incidents.

Planning Guidance to States

NRC has been working with the EPA to determine the types of accidents for which radiological emergency plans should be developed by State and local governments. A draft report on this subject (NUREG-0396) was prepared by an interagency task force and reviewed by several Federal agencies and by a committee of four representatives each from the Conference of (State) Radiation Control Program Directors (CRCPD), the U.S. Civil Defense Council and the National Association of State Directors for Disaster Preparedness. This inter-organization group, under the aegis of the CRCPD, gives these national organizations a common forum where they can review and comment on Federal policies on radiological emergency planning which affect State and local governments.

The task force concluded there was no specific accident sequence that could be used for emergency planning because each accident could have different consequences, both in nature and degree. Instead, the task force developed recommendations in an alternative form which would provide State and local governments with a basis on which to formulate emergency plans. The planning basis selected was a set of possible outcomes from a variety of accident scenarios. The distance, time characteristics, and release characteristics specified require response planning which involves at least a nucleus of necessary State and local emergency response organizations. The selection of the parameters involved an element of judgment supported by accident consequence and probability considerations.

The fundamental recommendations in the Task Force Report is that Emergency Planning Zones be established around each nuclear power plant for purposes of emergency planning. The zone would be 10 miles for the so-called plume exposure pathway and 50 miles for the ingestion exposure (milk and foodstuff) pathway. The Commission was briefed on the recommendations in the Task Force Report and has authorized its issuance for public comment.

Under a contract with DOE, the Sandia Laboratories are developing for NRC a set of accident scenarios which describe possible accidents at fixed nuclear facilities and project their likely consequences. The scenarios chosen range from relatively small accidents involving liquid releases to large ones involving a core meltdown. NRC plans to distribute "scripts" based on these scenarios to State and local government emergency planning organizations, giving them a basis against which to test their emergency plans.

Three of the Federal agencies involved in the interagency effort (NRC, DOT, EPA) agreed upon a plan to furnish more guidance to State and local governments regarding transportation accidents involving radioactive materials. Scenarios describing the nature and consequences of possible accidents are now being developed. They, too, will be translated into "scripts" and distributed for use in testing emergency plans.

Training Program for States

Several years ago, in cooperation with the States and other Federal agencies, NRC identified a number of areas where training was needed for State and local government personnel involved in radiological emergency planning and preparedness. Three courses are now being offered. Courses dealing with radioactive materials in transit are being developed by DOT, and courses in the medical area are under consideration. These courses are offered free of charge to qualified State and local government personnel.

The courses currently available are:

(1) *Radiological Emergency Response Operations:* This course is now conducted routinely at DOE's Nevada Test Site. It is designed for personnel who are, or will be, assigned to a State or local radiological emergency response team. In ten sessions conducted during fiscal year 1978, 200 State or local government employees received training.

(2) *Handling Radioactive Material in Transportation Accidents:* The Department of Transportation has developed a 20-hour course for first-at-the-scene emergency response personnel. It is general in nature and helps students recognize hazardous materials situations. DOT and NRC are sponsoring development of an 8-hour supplement to cover radioactive materials in transportation accidents. The course supplement is scheduled for completion in fiscal year 1979.

(3) *Radiological Emergency Response Coordination:* This course is designed to help the State coordinator make decisions on what protective actions to take in the event of an accidental release of radioactive material to the environment from a nuclear facility. It is divided into two parts: one for the plume exposure pathway, and the other for the ingestion exposure pathway. Part 1, which is approximately five days long, was presented five times in fiscal year 1977 and once in fiscal year 1978. Part 2 will be made available when protective action guides for food and animal feeds are completed by the Food and Drug Administration during fiscal year 1979.

(4) *Radiological Emergency Response Planning:* This was the original course developed to provide training needed to develop State and local radiological emergency response plans. It was presented 11 times in 1975-1976 for 366 State officials and then held in reserve until a specific need was apparent. The course was presented once in fiscal year 1978 to emergency planning officials of New York State at their request.

Field Assistance Program

NRC continues to lead and to coordinate Federal interagency field reviews and critiques of State radiological emergency response plans and exercises to test these plans. During fiscal year 1978, the regional advisory committees made 18 field reviews and assistance visits and critiqued six radiological emergency response exercises.

These photos were taken at the Nevada Test Site during an NRC-sponsored training course for State and local government personnel who might be involved in responding to a radiological emergency. Above left, students don self-contained breathing apparatus during an exercise designed to teach them the limitations of activities when wearing such gear. Above right, a student is dressed in full anti-contamination clothing prior to an exercise involving the rescue of casualties during a simulated reactor accident. Below left, students learn monitoring techniques at the gamma isodose facility.

Concurrence in State Plans

As lead agency, NRC is charged with reviewing and concurring in State and local government radiological emergency response plans. A checklist (NUREG-75/111) first published by NRC in 1974 gives basic guidance for development of these plans, and a 1977 supplement lists 70 planning elements which each plan must contain, at a minimum, before the NRC will concur in it. The list of essential elements is revised from year to year.

Six State plans received NRC concurrence (Delaware, South Carolina, Connecticut, Florida, Kansas, and California) during fiscal year 1978, bringing to eight the number of State plans so approved. (NRC concurred in the plans of Washington and New Jersey during fiscal year 1977.) Plans of about six other States will probably reach the NRC concurrence stage in 1979. In response to suggestions from State and local government offices, NRC recently changed its procedure to require the conduct of a successful test of each State plan as a condition for NRC concurrence.

OTHER LIAISON AND COOPERATIVE ACTIVITIES

Further clarification of the issues that confront the States and the Federal Government concerning siting issues was provided in a report by the National Governors' Association. The report, contracted by the NRC's Office of State Programs, was entitled "State Perspectives on Energy Facility Siting" (NUREG-0198, March 1978). It concludes that careful delineation of responsibilities between the Federal and State governments can be achieved and that direct, persistent actions can assist in bringing about a siting process more acceptable to all legitimate interests with less susceptibility to costly delay. It concluded further that clear procedures, increased planning competence, added trust in the capacity of States to participate, and improved management systems can combine to serve the public interest better and to produce environmentally sound sites.

Need for Power Study

NRC contracted with the Center for Natural Areas for a study of the factors entering into States' decisions on the need for the electricity to be provided by proposed power plants. The States chosen for review were the 38 with nuclear power stations in operation, under construction, or in the planning stage as of October 1977, and six additional States with siting laws pertaining to fossil-fueled power stations. The study tried to identify the factors considered and processes used by the States in determining the need for power. It also included summaries of the process by which States issue certificates for new power stations, and a discussion of how costs of construction and operation were treated for rate-making purposes.

The study, entitled "Need for Power: Determinants in the State Decisionmaking Processes" (NUREG/CR-0022, March 1978), concluded that there is a wide disparity in the criteria used by the States in deciding on the need for power. This conclusion points up the importance of agreement on more comprehensive and uniform criteria and methodologies as a tool in effective nuclear licensing and regional energy planning.

Transportation Surveillance

Between 1973 and 1976, the Atomic Energy Commission (predecessor of the NRC), DOT, nine States and New York City engaged in a joint pilot program to study certain of the health and safety aspects of the transportation of radioactive material through major air carrier and freight forwarder terminals. In particular, the study covered the shipping procedures, shipping modes, and the resulting radiation exposure to workers handling radioactive material shipments. The study was conducted over three-month periods in New York City and Illinois, Louisiana, Minnesota, Missouri, New Jersey, upstate New York, Oregon, South Carolina and Texas.

In fiscal year 1978, the Los Alamos Scientific Laboratory, under contract to NRC, completed a review of the 10 surveys conducted and issued a study, "Summary Report of the State Surveillance Program on the Transportation of Radioactive Materials" (NUREG-0393). The main findings were that, in general, the transportation of radioactive material does not present a significant health or safety hazard to cargo handlers or members of the public, but that there is a need for continued surveillance.

As a follow-on to the pilot program, agreements for one- to three-year expanded inspection programs were entered into with the radiological health bureaus of Pennsylvania, South Carolina, Illinois, Georgia, Michigan and Kentucky. Pennsylvania and South Carolina completed the first year of their programs in fiscal year 1978, and their findings were published as NRC documents, NUREG/CR-0286 and -0266. Illinois and Georgia also completed their first-year surveillance studies, and the results will be published early in fiscal year 1979. As in the pilot program, there were no findings of significant hazard to the health and safety of transportation workers or the general public.

Work with State Program Directors

The NRC continued its financial and technical assistance to the Conference of Radiation Control Program Directors, an organization comprised of the directors of radiation control programs in the 50 states, certain territories, and large municipal areas. The Conference serves as a forum for the exchange of radiation health and safety information and ideas between States and Federal agencies, as well as among the States themselves, to ensure that medical patients, radiation workers, consumers, and the general public receive the lowest possible radiation exposure consistent with the benefit derived.

The primary work of the Conference is carried on through 18 task forces which evaluate, discuss and recommend specific action to resolve identified problems. The NRC has assigned technical personnel to assist the task forces.

The Conference held its Tenth Annual Meeting from April 30 to May 5, 1978, in Harrisburg, Pennsylvania. Some 300 representatives of local, State and Federal radiation control agencies, professional organizations, industry and others with an interest in radiation protection attended. NRC personnel presented papers on radioactive waste management, transportation of radioactive materials, decommissioning of nuclear facilities, and radiological emergency response preparedness.

Regional Workshops

NRC recognizes that the States can have a significant interest in and can make substantial contribution to the development of policy and regulations. A workshop was held in November 1977 to discuss issues dealing with Federal and State regulation of uranium mills. Also, in September 1978, three regional workshops were held to solicit State comment on NRC's proposed plan for reevaluation of its decommissioning policy. More than 150 State officials from 44 States and Puerto Rico participated in the workshop.

National/State Organizations

Throughout 1978, NRC engaged in cooperative efforts with regional bodies such as the Western Interstate Energy Board and the Southern States Energy Board, and with national state organizations such as the National Governors' Association, National Conference of State Legislatures and National Association of Regulatory Utility Commissioners. NRC staff also met with State legislators several times during the year to discuss NRC's programs on radioactive waste management and the decommissioning of nuclear facilities.

An increasing amount of legislation dealing with nuclear power was active in State legislatures during the 1977-78 legislative year. For example, there were 41 bills dealing with high-level radioactive waste, 18 tying nuclear power plant siting to solution of the waste problem, and 30 dealing with transportation. NRC continued to provide comments on proposed legislation when requested and in several instances presented testimony before legislative committees.

Intergovernmental Personnel Assignments

It is the policy of NRC to permit and encourage temporary personnel assignments to or from State and local governments to enhance Federal, State and local cooperation. These assignments are made in accordance with the Intergovernmental Personnel Act of 1970.

During 1978, NRC employees were assigned to Arizona, Georgia, New Mexico and Oregon for periods up to two years, and a staff member of the Michigan Public Service Commission was assigned to the NRC. There will no doubt be additional activity under this Act in the months ahead.

Coordination Pacts with States

In January 1976, NRC and EPA entered into a Second Memorandum of Understanding regarding implementation of certain of their respective responsibilities under the Federal Water Pollution Control Act (FWPCA).

NRC has adopted the policy which encourages agreements with States to whom EPA has delegated the National Pollutant Discharge Elimination System (NPDES) under section 402 of the Federal Water Pollution Control Act Amendments of 1972 (FWPCA). Although there is considerable diversity in the individual agreements, they generally embody the following principles of the Second NRC/EPA Memorandum:

- The State and NRC will work together to identify and consolidate the environmental information needed by the State for the issuance of NPDES permits.
- The State will exercise its best efforts to issue NPDES permits prior to the planned date of issuance by NRC of the final environmental statement for the early site approval, construction permit or operating license for each nuclear power plant. The State will work closely with NRC to assure that water quality certifications under Section 401 of the FWPCA are issued in advance of NRC's final environmental impact statement.

- NRC and the State will consider the feasibility of holding joint or concurrent hearings on the State's NPDES permits and NRC's construction permits for nuclear power plants.
- The State and NRC will explore means, including NRC technical assistance to the State, for joint or cooperation preparation of parts of environmental impact statements for nuclear power plants.
- The State and NRC will maintain close contact on water quality matters throughout the entire environmental review process.

During fiscal year 1978, NRC entered into agreements with Virginia, New York, South Carolina and Washington. At year end, discussions expected to lead to further agreements were underway with several states.

The agreement with Virginia is limited to nuclear power stations and almost entirely to water quality matters. The New York agreement, called a Memorandum of Understanding, also is limited to nuclear operating facilities, but anticipates subagreements under which the State would prepare for NRC certain portions of environmental impact statements in areas of concurrent jurisdiction. Subagreements regarding water quality and need-for-facility are now being developed. The South Carolina agreement, while limited to water quality matters, applies not only to nuclear power plants but also to certain other facilities subject to regulation by NRC. The Washington agreement applies to all nuclear facilities subject to licensing by NRC or certification by the State; and contemplates specific subagreements in several areas.

It is NRC's view—one that is increasingly shared by the States—that agreements between NRC and EPA-permitting States have considerable merit in avoiding costly duplication, assuring timely action on section 402 permits and section 401 certifications, and providing maximum use of limited technical staff resources.

State Liaison Officers' Program

The Governors of all States except West Virginia have appointed liaison officers as an official communication channel with NRC. Puerto Rico also has designated such an officer.

On October 26 and 27, 1977, the State liaison officers met in Washington with NRC commissioners and senior management to discuss matters of mutual interest, including waste management, decommissioning and licensing. As a followup to this meeting, regional State liaison officer meetings will be held to aquaint the States with NRC regional office operations and to discuss major issues. A Regional meeting was held at Argonne National Laboratory (Region III) in June 1978. Regional meetings will be held in other NRC regional offices during fiscal year 1979.

In January 1978, NRC placed its own State liaison officers in the Philadelphia and San Francisco regional offices. When approved by the Commission, liaison officers will be assigned to the other NRC regional offices.

Governor Dixy Lee Ray and Lee V. Gossick, NRC's Executive Director for Operations, sign an NRC-State of Washington Memorandum of Agreement at Olympia, Wash., on September 6, 1978.

International Activities

During 1978, the NRC's international activities continued to intensify under the impetus of international concern over issues pertaining to health and safety in nuclear activities, including reactor exports, and nonproliferation. Enactment of the Nuclear Nonproliferation Act substantially expanded the agency's responsibilities regarding exports.

NRC's international efforts, coordinated by its Office of International Programs, cover a broad range of information exchange and cooperation with other countries, administration of nuclear export and import licensing, and the closely related area of international nuclear safeguards.

Developments in the sphere of international cooperation on health and safety matters included:

- Conclusion of bilateral arrangements with the nuclear authorities of five additional countries, bringing to 17 the number of NRC's regulatory information exchange and cooperation arrangements in effect.
- Two new agreements on nuclear safety research.
- Participation in U.S. technical support of the International Nuclear Fuel Cycle Evaluation (INFCE) which is studying ways to minimize nuclear proliferation risks without jeopardizing energy supplies.
- Completion by the NRC staff of a health and safety study related to reactor exports, presaging expanded NRC assistance in this area.

Highlights in NRC's export/import and international safeguards efforts included:

- Issuance of consolidated export/import regulations, including new requirements and scope mandated by the Nuclear Nonproliferation Act, Public Law 95-242, signed into law on March 10, 1978.
- Increased staffing and revised procedures to accommodate new export licensing responsibilities mandated by the Nuclear Nonproliferation Act.
- Work on procedures to expedite actions on minor export license applications.

An arrangement between the U.S. Nuclear Regulatory Commission and the Belgian Government for exchange of technical information in regulatory matters and for cooperation in safety research and in standards development was signed in Washington on June 6, 1978. From left: Dr. Louis Groven, Scientific Counselor, Embassy of Belgium; His Excellency Willy Van Cauwenberg, Belgian Ambassador to the U.S.; Dr. Joseph M. Hendrie, NRC Chairman; Dr. Joseph D. Lafleur, Jr., NRC Office of International Programs; and Richard T. Kennedy, NRC Commissioner.

- Establishment of an automated data system for the export licensing program.
- Intensified examination of the international safeguards and physical security aspects of proposed exports and increased support of U.S. actions to strengthen International Atomic Energy Agency (IAEA) safeguards.
- Publication of proposed regulations to implement the U.S./IAEA Safeguards Agreement, when ratified by the state.

Exchange of Information

Bilateral Arrangements

During the 12 months covered by this report, NRC entered into regulatory information exchange and cooperation arrangements with five additional countries: The Netherlands, the Republic of China, Israel, Belgium, and Greece, bringing to 17 the number of such arrangements currently in force. Previous parties to these bilaterals, which date from 1974, include the nuclear regulatory authorities of Brazil, Denmark, France, the Federal Republic of Germany, Iran, Italy, Japan, Korea, Spain, Sweden, Switzerland and the United Kingdom. Negotiations toward similar arrangements continued with Canada, Egypt, Mexico, the Philippines, Yugoslavia and the Union of Soviet Socialist Republics.

Objectives of these agreements are to:

(1) Establish a formal channel of communication with foreign regulatory organizations tó assure prompt and reciprocal notification of reactor safety problems that could apply to both U.S. and foreign nuclear facilities.

(2) Form a network for bilateral cooperation related to public health and safety, safeguards, and environmental protection.

(3) Assist in developing an international consensus on regulatory matters and safety standards and experiments.

(4) Provide assistance in improving health and safety practices of countries importing U.S. reactors.

Specific provisions of the arrangements call for the reciprocal exchange of regulatory information in the form of technical reports, correspondence, newsletters, meetings, training courses, and any other means agreed upon. In some cases, they also provide for future cooperation in reactor safety research and temporary assignments of personnel to agency headquarters and laboratory programs under the sponsorship of both parties.

Research Agreements

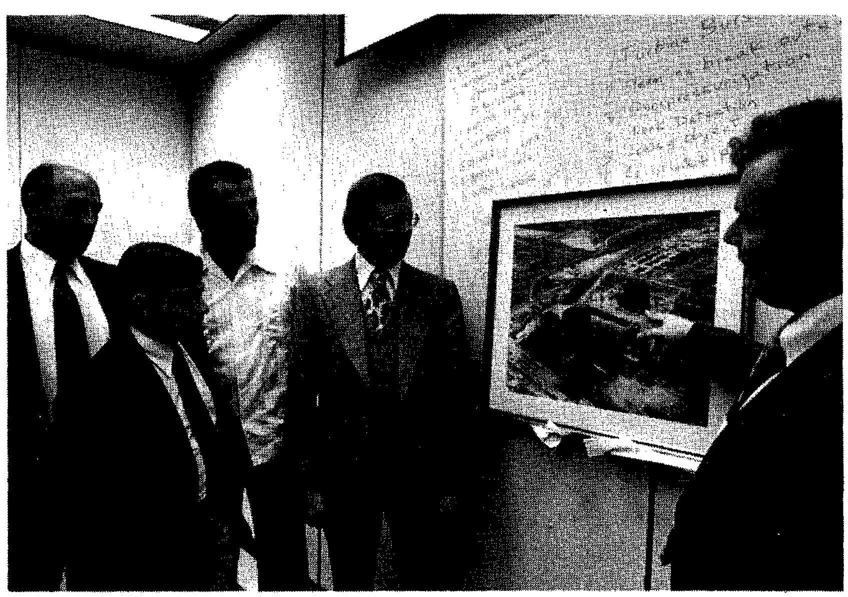
During the reporting period, the NRC executed two agreements in the area of nuclear safety research. A four-year agreement with the Netherlands Energy Research Foundation (ECN) provides for ECN's participation in the NRC Loss-of-Fluid Test (LOFT) program. In return for an annual payment of $160,000, the agreement grants ECN access to all experimental data and results of associated analyses and permits its direct participation in the conduct of LOFT experiments.

The second agreement is with the French Commissariat a l'Energie Atomique (CEA) and the West German Kernforschungszentrum Karlsruhe GmbH (KfK). It provides for the exchange of experimental data and technical information for the NRC Annular Core Pulsed Reactor (ACPR) program and the CEA-KfK CABRI program, and for participation by each party in the other parties' experiments. Both the ACPR and CABRI programs are related to the testing of advanced reactor fuels.

Bilateral Technical Exchange Meetings

NRC participates in regular nuclear safety exchange meetings with various countries. Usually, a visit to an operating, research or manufacturing facility is also included.

In November 1977, an NRC delegation of ACRS and staff members visited Tokyo for discussions with the Nuclear Safety Bureau and the Agency for Natural Resources and Energy. NRC hosted a reciprocal visit by the Japanese in June 1978 at its offices in Bethesda, Maryland. A delegation of reactor safety experts from the Federal Republic of Germany's (FRG) Ministry of the Interior and their technical contractor, the Gesellschaft fur Reaktorsicherheit, visited NRC in Bethesda in April 1978. Members of the FRG Reaktorsicherheitskomission (RSK) visited NRC in November 1978. A team from the French Groupe Permanent visited NRC for safety discussions in September 1978. Both the RSK and the Groupe Permanent are advisory technical safety groups having functions and responsibilities similar to those of the ACRS in the United States.



Three members of the French Groupe Permanent, (from left): Jean Stalz, Jean-Marc Oury, and Pierre Tanguy observe while two representatives of the Houston Lighting and Power Company describe features of the South Texas Nuclear Project, in the construction stage. Meetings with staff experts from this group as well as from the nuclear regulatory authority of the Federal Republic of Germany also were hosted by NRC's Advisory Committee on Reactor Safeguards.

FRG-U.S. Zircaloy Cladding Workshop. The third annual FRG-U.S. Workshop on Zircaloy Cladding Research was held at the Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany, in June 1978. Participants included NRC research and reactor regulation staff members, German staff from Project Nuclear Sicherheit (PNS) at KfK and Kraftwerk Union, and representatives from Oak Ridge National Laboratory, Battelle Columbus Laboratories, EG&G, the Electric Power Research Institute, and the Japan Atomic Energy Research Institute.

The focus of the meeting was on the comparison of data recently obtained in two burst tests of a 4X4 array of fuel rods at ORNL and two burst tests of a 3X3 array at KfK under different test conditions and procedures. Recent results of other fuel rod cladding research programs of NRC and PNS were summarized in short presentations.

INTERNATIONAL PROJECTS

IAEA Technical Assistance

Over the past three years, the NRC, in coordination with the IAEA Technical Assistance Program, has provided safety advice and assistance to regulatory and safety authorities of countries embarking on nuclear power programs.

In July 1978, NRC staff members presented a training course on "Pressurized Water Reactor Fundamentals" to the Korean Atomic Energy Bureau in Seoul, Korea, and a similar course on "Boiling Water Reactor Fundamentals" to the Chinese Atomic Energy Council in Taipei, Taiwan.

Over a period of 10 months beginning in May 1978, the NRC staff, on behalf of the IAEA, arranged various short-term reactor safety missions in support of the Brazilian National Nuclear Energy Committee (CNEN) in Rio de Janeiro. NRC advisors carried out missions in the areas of operator licensing, review of the Safety Analysis Report and radiation protection, preoperational tests, and start-up tests. Arrangements were also made for three Brazilian experts to have a two-week tour of duty at NRC's Region II office in order to gain knowledge of the NRC inspection program and to accompany inspectors during actual inspections. Four other Brazilian regulatory employees witnessed operator license examinations at the D.C. Cook Nuclear Power Facility in Michigan in October 1978.

Similar assistance was provided to several other developing countries during the year.

Also during fiscal year 1978, two NRC staff members were made available for one-year assignments as IAEA advisors in countries initiating or strengthening nuclear regulatory programs. An expert in quality assurance and licensing review was assigned to Mexico, and a health physicist to Korea. An expert in nuclear safety and licensing is slated to go to Turkey in 1979.

Spent Fuel Storage Conference. NRC, in cooperation with the IAEA, hosted a three-day international meeting on the expansion of storage facilities for spent nuclear fuel. The meeting, which was held from February 28 to March 2, 1978, emphasized the safety aspects of modifying existing storage pools at power reactor sites to increase their storage capacity. Representatives from 24 countries participated in the meeting.

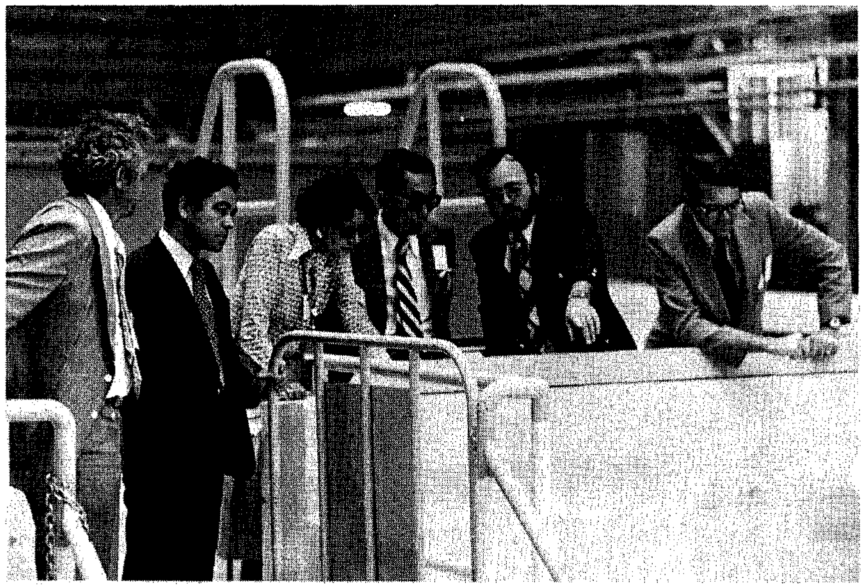
Argonne/IAEA Safety Course. NRC staff members presented a series of lectures at an eight-week course on Safety Analysis Review conducted on behalf of the IAEA by the Center for Educational Affairs at Argonne National Laboratory. The course was attended by 31 foreign nationals from 20 countries.

NRC will also participate in future safety-related courses at Argonne which will cover such subjects as quality assurance, siting for nuclear power plants, and safety and reliability in reactor operation.

Cooperation with the OECD

NRC has continued to participate in nuclear safety activities of the Organization for Economic Cooperation and Development (OECD), an organization of 20 Western European and other countries, including the U.S. and Japan, headquartered in Paris. NRC's work has been in support of two specialized OECD agencies—the Nuclear Energy Agency (NEA) and the International Energy Agency (IEA).

NRC senior staff members served on several standing committees of these agencies, including



Japanese Ambassador and Mrs. Togo visit the Idaho National Engineering Laboratory where several NRC experimental programs are under way. Shown second from left is Kazuo Suzuki, First Secretary, Embassy of Japan, then Mrs. Togo and the Ambassador. At far right is James R. Shea, Director of NRC's Office of International Programs.

the NEA Committee on the Safety of Nuclear Installations, the NEA Committee on Radiation Protection and Public Health, the NEA Waste Management Committee, and the IEA Working Group on Nuclear Safety. A new NEA working group on regulatory inspection, on which NRC is represented, was established during the year. In May, a member of NRC's Office of General Counsel represented the U.S. at a meeting of the NEA/OECD Group of Governmental Experts convened to revise the Paris and Brussels Conventions on Third Party Liability in the Field of Nuclear Energy.

Foreign Visitors to NRC

The increased pace of NRC international activities has been accompanied by an increase in the number of large technical delegations and individual visitors from foreign countries and organizations interested in holding in-depth discussions with the Commission and staff. During fiscal year 1978, the Office of International Programs scheduled NRC policy and technical meetings with 413 visitors from 31 countries and 4 international organizations. This included several week-long discussions with the foreign administrators of NRC bilateral regulatory information exchange and cooperation arrangements, as well as with their designated representatives, for the purpose of exchanging current operational safety, safeguards, and environmental protection information. These foreign visits typically included extended tours of various U.S. commercial nuclear facilities, both under construction and in operation, and of the national laboratories to observe ongoing NRC safety research programs.

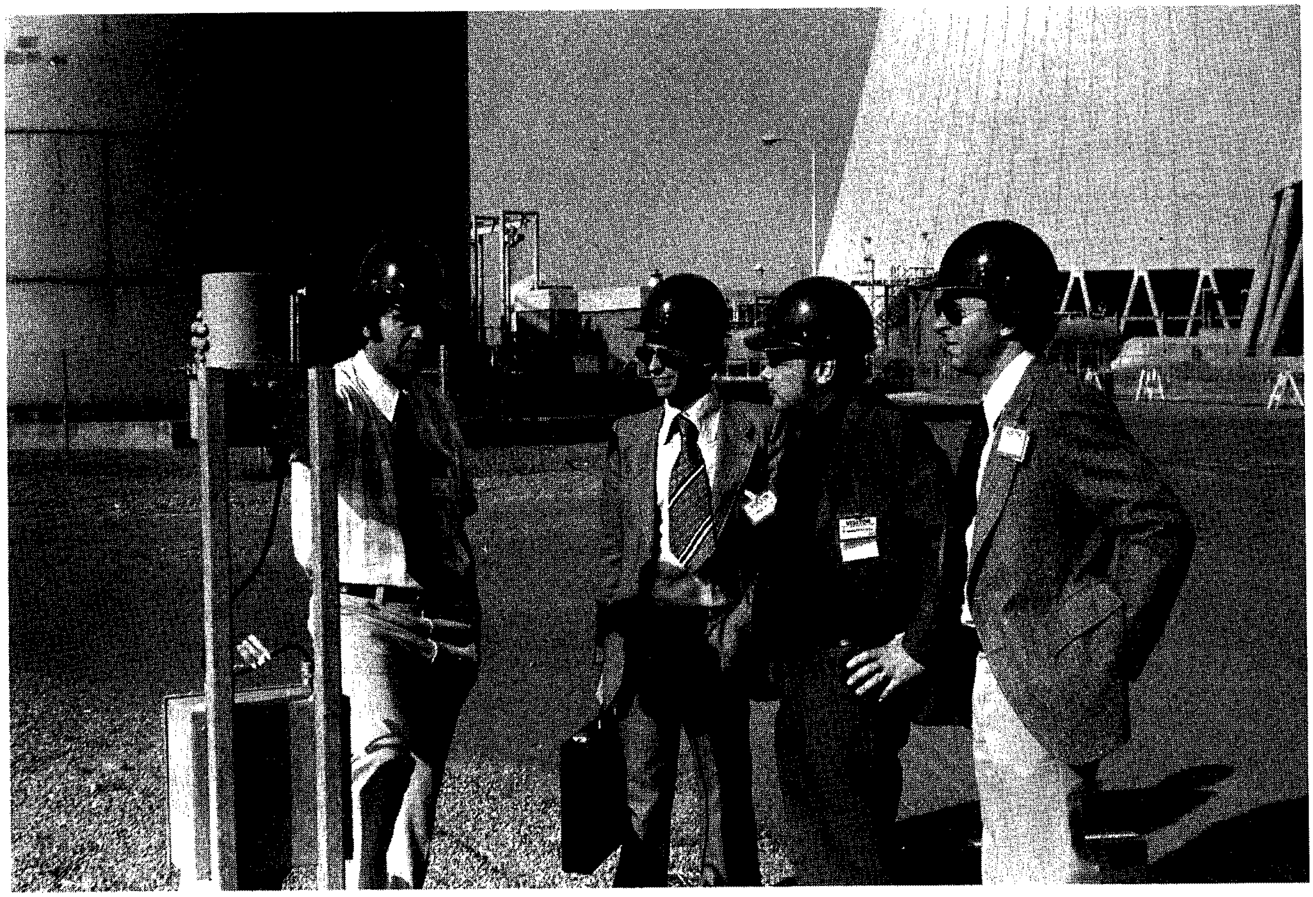
Nuclear Fuel Cycle Evaluations

The NRC participated throughout the year in support activities associated with both international and domestic evaluations of nuclear fuel cycle systems aimed at reducing proliferation risks. (Under provisions of Public Law 95-601, signed by the President in November 1978, NRC will broaden its efforts in this area. See Chapter 1.)*

International Program (INFCE). Following an organizational conference held in Washington in October 1977, fifty-three countries and four international organizations are conducting a two-year evaluation of means to develop and operate the nuclear fuel cycle in ways that minimize the risks of nuclear proliferation. The NRC is participating in U.S. technical support activities for this program, called the International Nuclear Fuel Cycle Evaluation (INFCE).

NRC staff members with appropriate expertise are serving in the U.S. support groups on several

*Section 9 of P.L. 95-601 also requires reports to Congress semiannually through calendar year 1980 and annually through calendar year 1982 on the status of INFCE and NASAP. A report on these activities will be submitted to Congress in mid-1979.



Two visitors from the Swiss Federal Office of Energy and the Swedish State Power Board (second and third from left) tour the Rancho Seco Nuclear Power Station near Sacramento, California, escorted by Andrew Robart (far right), NRC State Liaison Officer from NRC Region V.

of the basic INFCE study areas, namely: long-term fuel, technology and heavy water supply assurances; reprocessing, plutonium handling and recycle; fast breeders; spent fuel management; waste management and disposal; and advanced fuel cycle and reactor concepts.

In addition to these working groups, NRC staff also are participating in generic U.S. studies of INFCE-related subjects such as safeguards and proliferation resistance which cut across all of the basic study areas. The working groups were in an organizational phase until the early spring of 1978. Data collection then began and is expected to be completed in early 1979. The overall INFCE report will be prepared during 1979.

Domestic Assessment (NASAP). In a related effort, the DOE is conducting the Nonproliferation Alternative Systems Assessment Program (NASAP) which is complementary to INFCE and is providing substantial technical data and input to INFCE.

During fiscal year 1978, DOE provided to NRC preliminary descriptions of six basic reactor concepts and about twice as many fuel cycle variants being studied in NASAP. NRC staff members reviewed the preliminary reactor descriptions and identified health and safety, environmental, safeguards and licensing issues that needed to be addressed. At year-end, DOE was preparing preliminary safety and environmental information documents on fuel cycles and reactors which, among other things, will address the issues raised by NRC. These documents also will be reviewed by NRC in early 1979.

The NRC will report to the President and the Congress on findings of known or suspected licensing issues and problems associated with alternative technologies under serious consideration by DOE, including comparative evaluations of the safety, safeguards, environmental and licensing aspects.

Reactor Health and Safety Study

The health and safety implications of reactor exports is an issue that has attracted increasing attention both in Congress and among various U.S. and foreign public interest groups. In July 1978, the staff, in response to a Commission request, published the results of a one-year study on "Health and Safety Considerations in NRC Reactor Export Licensing and Nuclear Assistance Programs." Recommendations produced by the study will lead to an expanded NRC foreign health and safety assistance program.

The study identified a range of factors related to health and safety concerns, reviewed the statutory context of NRC's activities and responsibilities regarding foreign health and safety matters, and examined existing NRC international nuclear safety assistance activities. The analysis then focused on seven alternative measures that would be available to the NRC if it determined that it needed to expand its role in this field. The alternatives were reviewed in light of such factors as NRC's legal requirements to implement a given program, benefits obtained by the recipient country, foreign policy implications, estimated costs to the NRC, and implementation difficulties.

The Commission approved the following measures:

(1) Expand the level of the present NRC technical assistance program.

(2) Prepare, in conjunction with other Federal agencies, an action plan for U.S. strategy to upgrade national health and safety programs under the auspices of the IAEA.

(3) Review of foreign information needs for increased NRC efforts in transmittal of information on all safety-related modifications required of U.S. reactors.

(4) Provide assistance to other U.S. agencies that might be involved in evaluating the safety of U.S. reactors designated for export.

IAEA Symposium. NRC was represented at an international symposium held on March 6-10, 1978, in Vienna, Austria, dealing with the special problems associated with the export of nuclear power plants. NRC participants presented two papers, "NRC Advice and Assistance to Nuclear Power Regulatory Programs of Developing Countries" and "The Role of the USNRC in Power Reactor Exports: Legal and Procedural Aspects."

Export/Import Matters

Nuclear Nonproliferation Act of 1978

The Nuclear Nonproliferation Act of 1978 (NNPA), which was signed into law on March 10, addresses U.S. Government activities considered significant in deterring nuclear proliferation throughout the world. It provides a detailed policy framework for discharging the non-proliferation responsibilities of the NRC and the Executive Branch in the following areas: (a) ensuring nuclear export activities are conducted promptly and are consistent with national security and the specific NNPA criteria; (b) strengthening IAEA safeguards; (c) improving physical protection measures; (d) improving nuclear fuel assurances to other countries; (e) renegotiating Agreements for Cooperation; (f) evaluation of alternative nuclear fuel cycles; and (g) spent fuel disposition policy.

The specific criteria established by the NNPA for the export of nuclear commodities may be summarized as follows:

- Application of appropriate IAEA safeguards.
- No use for explosive purposes.
- Application of adequate physical security measures.
- Retransfers subject to U.S. approval.
- Reprocessing subject to U.S. approval.
- Material or equipment produced through the use of U.S. technology also subject to the foregoing export criteria.

In addition, the NNPA provides that, beginning with applications submitted after September 10, 1979, nuclear exports can only be permitted to countries accepting IAEA safeguards on all their peaceful nuclear activities.

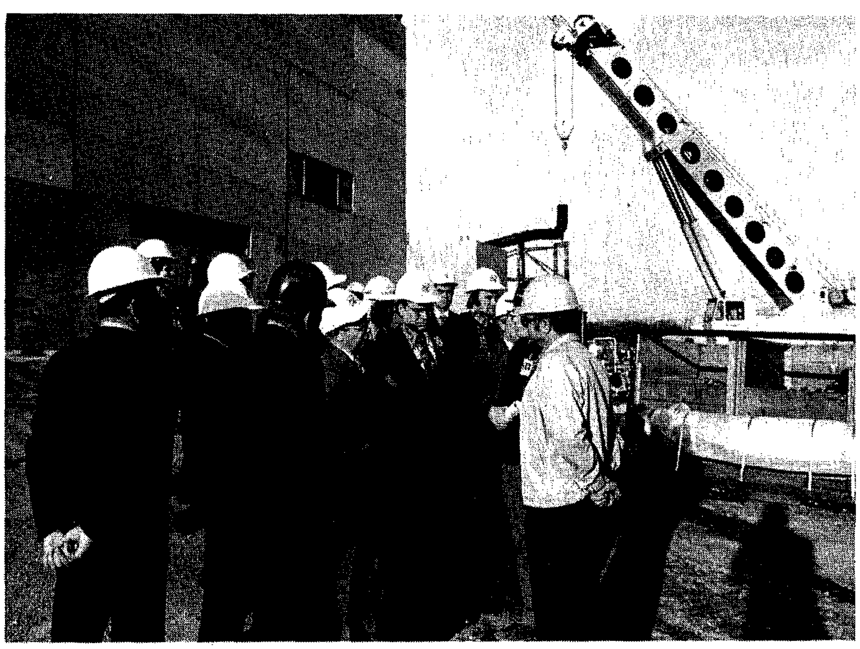
In licensing a nuclear export, NRC must determine, among other things, that it will not be inimical to the common defense and security and that it meets the applicable criteria of NNPA. The NNPA significantly expanded

NRC's export licensing responsibilities by adding nuclear facility components, substances and items of significance for nuclear explosive purposes to the list of commodities (special nuclear material, source material, byproduct material, production facilities, and nuclear reactors) already subject to NRC export licensing requirements.

In addition, the NNPA requires Executive Branch agencies to consult formally with NRC concerning nuclear export-related activities under their purview. These activities include (1) negotiation of new and revised agreements for cooperation (State/DOE); (2) nuclear technology exports (DOE); (3) foreign distribution of nuclear material (DOE); (4) negotiation of contracts for the supply of nuclear materials and equipment (including enrichment services) to foreign recipients (DOE); (5) consideration of requests to retransfer U.S.-supplied nuclear material and equipment (DOE); (6) consideration of requests to reprocess irradiated U.S.-supplied nuclear fuel (DOE); (7) other "subsequent arrangements" as defined in section 131 of the Atomic Energy Act of 1954, as amended; and (8) exports by the Commerce Department of nuclear-related commodities.

A particularly significant instance of consultation on proposed exports by the Executive Branch agencies with NRC took place during the year on the proposed subsequent arrangements involving shipping of US-origin spent fuel from Japan to the United Kingdom and France for reprocessing (the so-called TEPCO and Kansai cases). These cases were significant because they had a direct bearing on the President's general policy to defer commercial reprocessing worldwide during the conduct of INFCE. In the TEPCO case the justification for approval was based upon the physical need to remove the spent fuel from congested storage ponds. In Kansai, however, the justification was based upon the existence of reprocessing contracts entered into before the President's reprocessing policy was announced and also upon the commitments of the reactor operator not to store spent fuel on a long term basis at the reactor site. On the Kansai case, two Commissioners had no objections to the transfer, two recommended disapproval, and the fifth abstained. On the TEPCO case, four Commissioners indicated no objections to the transfer, and the fifth was not available for the vote. Both cases were eventually approved by the President. Congress also held public hearings on the matter.

To facilitate rapid interagency consultation on nuclear export activities, the interagency Subgroup on Nuclear Export Coordination (SNEC) was given responsibility to consider initially many significant or controversial nuclear export matters and for facilitating appropriate actions to achieve interagency agreement. NRC is a member of SNEC, but participates only in an observer capacity when the Executive Branch is formulating its position on individual export license applications filed with the NRC. Further



Soviet team delegation, escorted by NRC staff, visits construction site of the Limerick Generating Station (Philadelphia Electric Company facility) near Pottstown, Pa. During fiscal year 1978, a U.S. team consisting of several members of the NRC staff toured a number of nuclear facilities in the U.S.S.R.

interagency coordination procedures are spelled out in the "Procedures Established Pursuant to the Nuclear Nonproliferation Act of 1978," which was published in the *Federal Register* on June 9, 1978.

NRC Export/Import Regulations

On May 19, 1978, the NRC issued revised export/import regulations (10 CFR Part 110) which incorporate the pertinent export licensing criteria and requirements of the Nuclear Nonproliferation Act. NRC is currently reviewing several additional proposals from the public for improving the licensing procedures.

Source Material Exports

The question of continuing to export source materials destined for nonnuclear end use (e.g., depleted uranium used principally in aircraft counterweights, shielding, etc.) without subjecting them to an agreement for cooperation was reexamined by the staff in a report to the Commission. This analysis was performed because of the Commission's concern about the potential strategic significance of such material if it were to be diverted to enrichment or breeding/reprocessing facilities and subsequently converted to forms usable for a weapon.

An earlier (1977) analysis of this question had led the staff to conclude that such source material did not generally pose enough of a risk to justify subjecting it to an agreement for cooperation requirement, or to a formal certification or tracking provision. The reexamination generally supported the earlier conclusions except for certain rare instances where size of shipment or destination considerations might dictate a need for special restrictions. The Commission is considering the issues raised in the staff study.

EXPORT LICENSING ACTIONS

During the fiscal year ending September 30, 1978, the NRC issued 343 export licenses and received 517 new export license applications. The large number of new applications received — nearly double those received in fiscal year 1977 — reflects in large part NRC's new licensing responsibilities assumed from DOE and the Commerce Department under provisions of the NNPA.

Of the 343 licenses issued, 86 were major licenses which are listed in the accompanying table in three categories: special nuclear material, source material, and reactors. The 257 export licenses considered to be minor included 119 for small quantities of special nuclear material, 34 for source material, 94 for byproduct materials, and 10 for components. NRC issued 41 special nuclear material import licenses and received 50 new import license applications. Of the 41 licenses issued, 15 were major licenses which are listed in the accompanying table.

Fourteen different nations received U.S. shipments of special nuclear material under major export license during the year. In addition, four nations received major quantities of source material, and one nation received a reactor facility. No licenses were issued during the period for the export of large quantities of plutonium, although four applications for the export of kilogram quantities of this material were pending at year's end.

Tarapur (India) Case

The NRC Annual Reports for 1976 and 1977 set forth in considerable detail the circumstances surrounding the exports of low-enriched uranium to India for use in the Tarapur Atomic Power Station and the petition for intervention and request for hearing by the Natural Resources Defense Council, Inc., the Sierra Club, and the Union of Concerned Scientists with respect to three export applications, numbers XSNM-805, XSNM-845 and XSNM-1060.

The 1977 report (pp. 123, 126) gives the background which ultimately led to the issuance of licenses XSNM-805 and XSNM-845. The application for license XSNM-1060, involving 7,638 kilograms of low-enriched uranium, was still under Executive Branch review at the end of 1977.

In February 1978, the petitioners filed two motions with the Commission. One requested resumption of public hearings held by the Commission in 1976 on exports to India. The other requested that the Commission consolidate application XSNM-1060 with a new application for Tarapur fuel, XSNM-1222. The motion for consolidation was granted, but on April 20, 1978, the Commission denied the motion for a further public hearing on application XSNM-1060.

In their consideration of application XSNM-1060, the four Commissioners divided, 2-2, on the question of whether or not India met all of the criteria in the Nuclear Nonproliferation Act. Because of this inability to reach a majority decision on the application, the Commission, on April 25, 1978, referred the case to the President as provided for in Section 126b(2) of the Atomic Energy Act, as amended.

On April 27, 1978, the President issued Executive Order 12055, authorizing the export under XSNM-1060, after having determined that withholding the material would be seriously prejudicial to the achievement of the nonproliferation goals of the United States. The export was subject to a 60-day Congressional review period, as required by the NNPA.

The 60-day period expired on July 15, 1978, without the adoption by Congress of a resolution disapproving the proposed export. The material was exported by Edlow International Company, as agent for the Government of India, on July 20, 1978.

The Commission, in considering application XSNM-1222, covering export to India of 16,804 kilograms of low-enriched uranium, has solicited written expressions of views on certain questions connected with this proposed export.

Automated Data System

During the year, the NRC implemented previously announced plans to establish a data processing program to provide current and quickly retrievable information on the status of both completed and pending export and import licensing cases (see 1977 NRC Annual Report, p. 122). It is designed not only to provide access to information by remote computer stations at the NRC regional offices and other potential users, but also to connect with the Nuclear Materials Management System maintained by DOE.

The system also is designed to permit modification and expansion as may be required. For example, it is currently being upgraded to accommodate the additional licensing responsibilities assumed by the NRC under the Nuclear Nonproliferation Act of 1978.

AGREEMENTS FOR COOPERATION

The Nuclear Nonproliferation Act calls for an immediate program to renegotiate existing U.S. agreements for nuclear cooperation with other countries to reflect the new requirements of the NNPA. Work has begun on this effort under the lead of the Department of State in consultation with other U.S. agencies, including the NRC. In some cases, agreements may be accompanied by an exchange of notes to indicate the manner in which the U.S. intends to implement certain provisions of these agreements.

One of the aims of the renegotiated agreements is to make reciprocal for both the U.S. and its trading partners the provisions regarding physical security and the storage, retransfer and reprocessing of nuclear material. As a result, NRC will be developing measures aimed at permitting the tracking of foreign nuclear material in the U.S. licensed sector, and the application of physical security measures for categories II and III nuclear materials that are consistent with international standards.

Agreements with Israel and Egypt were initialed in 1976. These renegotiated agreements are being revised to reflect the new requirements of the NNPA.

Agreements with the IAEA, Canada, Iran and Australia were in various stages of completion at the end of the fiscal year.

The NRC is preparing regulations needed to bring U.S. licensees into compliance with the requirements imposed by these agreements.

Views on Nonproliferation Role

Section 602 of the Nuclear Nonproliferation Act requires the Commission and DOE to include in their annual reports to the Congress "views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibility of those agencies..."

In general, the Commission's experience in discharging its new responsibilities under the

Act, while still limited, has not yet disclosed any insuperable difficulties. Improvements in the processing of individual export license applications are anticipated as greater understanding of the complex new procedures is gained and the carrying out of the formal interagency coordination requirements of the Act becomes more routine. The Commission and the Executive Branch are well along in developing and implementing means of expediting future export application reviews in accordance with NNPA provisions regarding export determinations based on findings of "no material changed circumstances." This will avoid unnecessary repetition of analyses for countries where no significant changes have occurred since previous U.S. export approvals and allow efforts to be focused on those complex cases requiring detailed review of compliance with NNPA provisions.

The Commission has placed a high priority on developing a responsive export licensing system that does not unnecessarily delay approvals for the great majority of nuclear export applications which clearly meet the new criteria. Maintaining reliability of supply in support of legitimate nuclear commerce while ensuring that exports are consistent with U.S. national security remains a key element of the U.S. effort to reduce worldwide proliferation concerns.

With a view toward further improvements in carrying out its nonproliferation responsibilities, the Commission is focusing on the following areas:

(1) Improvement of IAEA safeguards (see discussion below).

(2) Establishment of a general license for exporting reactor components for approved facilities.

(3) Further clarification of NRC's role in consulting with the Executive Branch on nuclear export matters as required by the NNPA (e.g., subsequent arrangements).

(4) Revisions of NRC export regulations concerning minor export applications or those presenting no significant proliferation concerns.

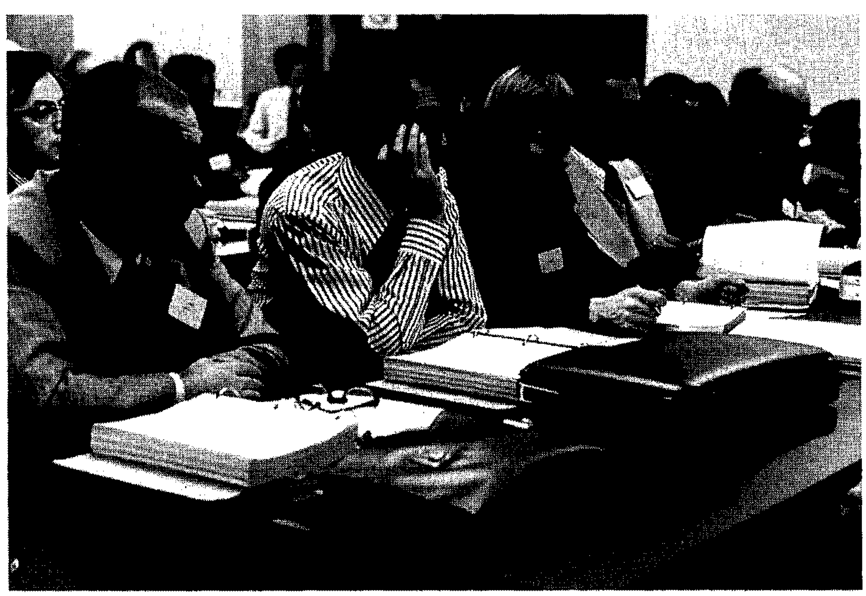
(5) Establishment of general policy guidelines for licensing exports of multiple fuel reloads.

(6) Establishment of a standard format for Executive Branch analysis of export license applications filed with NRC.

The Commission believes that progress in these areas will contribute significantly to the handling of nuclear exports in a manner that properly addresses U.S. proliferation concerns while providing for legitimate commercial transactions.

INTERNATIONAL SAFEGUARDS

International safeguards was a major focus of NRC interest in fiscal year 1978. The IAEA Secretariat's Special Safeguards Implementation



Specialists from 25 countries attended a meeting in Bethesda, Md., on February 28 - March 2, 1978, on the storage of spent fuel elements. Safety considerations related to increasing the capacity of existing storage pools were emphasized. The meeting was hosted by the NRC in cooperation with the International Atomic Energy Agency.

Report (SSIR), submitted to the IAEA Board of Governors in June 1977, was designed to describe how well the IAEA safeguards were functioning as an aid to assessing the need for improvements. The SSIR provided valuable statistics and descriptive information regarding the implementation of IAEA safeguards and concluded that there were no significant diversions of strategic nuclear materials in any of the 40 nations where the agency conducted inspections in calendar year 1976. The report indicated, however, that there were still problems associated with safeguards implementation in many countries (which were not named because of the IAEA's restrictions on release of such information) and it listed several recommendations for correcting these problems. In February 1978, the IAEA Board of Governors voted to retain the "restricted" classification on the SSIR.

The NRC, along with other U.S. Government agencies, encouraged preparation of the SSIR. As a result of its analysis of the document, however, the Commission determined that it was necessary to strengthen IAEA safeguards and that the Commission should reexamine the role of international safeguards in making its independent determinations regarding proposed nuclear exports. By September 1977, the NRC safeguards staff began to indicate in its review of nuclear export license applications that it did not have sufficient information to evaluate the effectiveness of IAEA safeguards on a country-by-country basis. In this regard, representatives of the Department of State have expressed the view that for the U.S. Government to insist now on obtaining country-specific inspection reports from the IAEA or, alternatively, on U.S. on-site inspections in other countries, would undermine the basic international consensus supporting IAEA safeguards.

In February 1978, the Commission informed the cognizant committees of Congress of NRC's views regarding the safeguards deficiencies identified in the SSIR. Subsequently, the Commission received the second annual IAEA Safeguards Implementation Report (SIR), issued in final form in August 1978, which indicated that many of the problems identified in the implementation report for 1976 still persisted in 1977. During the course of fiscal year 1978, the Commission actively participated in the development of an interagency U.S. Government action plan to strengthen IAEA safeguards. NRC will provide continuing support during implementation of that plan. Meanwhile, in analyzing export license applications in accordance with the requirements of the Atomic Energy Act of 1954, as amended, the NRC will continue to review available information on whether IAEA safeguards are being implemented adequately.

In other 1978 actions related to international safeguards and physical security matters, the NRC:

(1) Published a proposed rule (10 CFR Part 75), "Safeguards on Nuclear Materials," designed to implement the US/IAEA Safeguards Agreement, which is now before the Senate for ratification as a treaty.

(2) Participated with other U.S. agencies in support of phase II of the "Program Plan for Technical Assistance to IAEA Safeguards," as administered by the Department of Energy (DOE). NRC has been involved in both the planning phases of this effort and in providing experts directly to the IAEA at no cost to assist in developing safeguards technology.

(3) Participated with other U.S. agencies in analyzing the safeguards aspects of the International Nuclear Fuel Cycle Evaluation (INFCE). The NRC provided experts to the INFCE "Crosscut Safeguards Group" in support of this effort.

(4) Participated in meetings with international safeguards experts, both in the U.S. and overseas, to exchange views on safeguards and physical security, and made long-term assignments of NRC safeguards specialists to the IAEA staff in Vienna.

(5) Participated in a visit by a U.S. physical security review team, headed by DOE, to Mexico.

(6) Participated with other U.S. agencies in drafting the proposed International Physical Security Convention which, under the auspices of the IAEA, seeks common agreement regarding the physical security of nuclear materials.

U.S./IAEA Safeguards Agreement. In fiscal year 1978, further steps were taken toward implementing the voluntary U.S. offer to permit application to its civil nuclear facilities of IAEA

Table 1: Major Nuclear Export Licenses

(Major Licensing Actions Taken by NRC - October 1, 1977 through September 30, 1978)

SPECIAL NUCLEAR MATERIAL (One or More "effective kilograms" as defined in 10 CFR 70.4(t))

Licensee	*Kilograms of Uranium*	*Enrichment %*	*Country of Destination*	*Date Issued*
Mitsubishi	16,080	3.15	Japan	11/10/77
Transnuclear	18,981.415	3.0	W. Germany	11/10/77
Marubeni	12,836	2.87	Japan	11/10/77
Union Carbide	7.85	93.15	France	12/8/77
Transnuclear	1,833	3.06	Japan	12/14/77
Mitsui	24,577	3.01	Japan	12/14/77
Exxon	43,340	2.90	Sweden	12/14/77
General Electric	Additional 70,000	4.0	Switzerland	12/14/77
Mitsui	7,652	3.01	Japan	12/14/77
Transnuclear	15,528.25	3.35	Netherlands	12/22/77
Transnuclear	17.164	93.3	W. Germany	12/22/77
Transnuclear	11,234	3.0	W. Germany	12/30/77
Transnuclear	11,057	5.73	France	12/30/77
Transnuclear	36,338	3.40	W. Germany	12/30/77
Transnuclear	24,243.615	3.35	Belgium	12/30/77
Transnuclear	217,363.4	3.30	France	12/30/77
Transnuclear	10,712	3.65	Belgium	12/30/77
Exxon Nuclear	11,088	2.80	W. Germany	12/30/77
Westinghouse	51.086	(Plutonium)	Switzerland	2/16/78
Transnuclear	93.208	93.3	France	2/17/78
Westinghouse	Additional 60,580	3.15	Japan	2/27/78
Transnuclear	1,398.5	7.180	France	3/3/78
Westinghouse	51,889	3.15	Brazil	3/9/78
Transnuclear	16.077	93.3	France	4/7/78
Transnuclear	18.045	93.3	Denmark	4/7/78
Transnuclear	121.3	93.3	W. Germany	4/7/78
Transnuclear	74.759	93.3	France	4/7/78
Transnuclear	30.075	93.3	W. Germany	4/7/78
Transnuclear	23.056	93.3	W. Germany	4/7/78
Transnuclear	27.610	93.3	France	4/7/78
Transnuclear	10.025	93.3	W. Germany	4/7/78
Transnuclear	1,367.8	3.35	W. Germany	4/7/78
Transnuclear	119.298	93.3	W. Germany	4/7/78
General Electric	120,050	3.1	Spain	5/19/78
Transnuclear	21.053	93.30	Sweden	6/12/78
General Electric	Additional 200	3.1	Spain	6/14/78
Mitsubishi International	48,000	3.25	Japan	6/23/78
Mitsubishi International	18,440	3.33	Japan	6/23/78
Westinghouse	Additional 3,729	3.15	Sweden	6/29/78
Mitsubishi International	53,118	3.45	Japan	7/5/78
Mitsubishi International	13,065	2.85	Japan	7/5/78
Westinghouse	6,000	3.4	United Kingdom	7/14/78
GE Tech Services	24,675	3.1	Japan	7/24/78
Mitsui & Co., Inc.	32,592	3.80	Japan	7/28/78
Mitsui & Co., Inc.	11,252	2.87	Japan	7/28/78
Edlow International	Add Intermediate Consignee		Sweden	8/3/78
Marubeni America	12,784	2.87	Japan	8/3/78
Mitsui & Co.	7,751	3.07	Japan	8/3/78
General Electric	5,624	3.1	Japan	8/3/78
Edlow International	39,600	2.85	Japan	8/3/78
Babcock & Wilcox	1,376,000	5.00	W. Germany	8/3/78
Mitsui & Co., Inc.	28,582	3.01	Japan	8/3/78
Mitsui & Co., Inc.	10,527	3.01	Japan	8/3/78
Transnuclear	814.050	12.18	Japan	8/18/78

Table 1: Nuclear Export Licenses (Continued)

Licensee	*Kilograms of Uranium*	*Enrichment %*	*Country of Destination*	*Date Issued*
Transnuclear	25,456.027	3.4	Austria	8/4/78
Transnuclear	74.759	93.3	France	8/17/78
Edlow International	89,800	3.55	Sweden	8/31/78
Edlow International	52,688	3.55	Sweden	8/31/78
Marubeni America	22,084	3.80	Japan	9/1/78
Transnuclear	18,846	3.55	W. Germany	9/1/78
Marubeni America	22,084	3.80	Japan	9/1/78
Transnuclear	10,326	3.65	Belgium	9/8/78
General Atomic	15.6	70	S. Korea	9/11/78
General Atomic	3.28	70	Korea	9/11/78
	5.7 gms	93		
Transnuclear	5,573	3.0	W. Germany	9/12/78
	4.	3.0		
Transnuclear	34,851	3.25	W. Germany	9/13/78
Exxon Nuclear	55,440	3.8	Sweden	9/13/78
Transnuclear	7.0	93.3	Greece	9/18/78
Transnuclear	12,714	3.25	W. Germany	9/20/78
Transnuclear	17,472	3.25	Sweden	9/21/78
Mitsui & Company	Increase maximum enrichment		Japan	9/25/78
Edlow International	Additional			
	14,955	3.55	Sweden	9/27/78

SOURCE MATERIAL

Licensee		*Material*	*Country of Destination*	*Date Issued*
Fansteel	10,572	kgs. uranium & thorium	W. Germany	1/20/78
Edlow International	38,465.028	kgs. uranium	Japan	2/16/78
Edlow International	634,673	kgs. uranium	Canada	3/15/78
Aerojet Ordnance & Manufacturing Company	45,043	kgs. depleted uranium	Canada	8/16/78
Mitsubishi International Corporation	230,427.288	kgs. uranium concentrate	Canada	8/16/78
NL Industries		Extend expiration date from 8/01/78 to 8/01/80	Canada	8/25/78
Transnuclear	10,025	kgs. depleted uranium	France	9/18/78

REACTORS

Licensee	*Facility Description*	*Country of Destination*	*Date Issued*
General Electric San Jose, California	Extend expiration date	Japan	12/1/77
General Electric San Jose, California	Extend expiration date	Japan	12/29/77
Westinghouse Electric Pittsburgh, Pennsylvania	Change address of licensee, increase value to $26,000,000, and extend expiration date	Sweden	2/6/78
Westinghouse Electric Pittsburgh, Pennsylvania	Add other party to export	Spain	2/27/78
General Atomic Company San Diego, California	Extend expiration date to 7/01/79	Romania	7/12/78
Westinghouse Electric Pittsburgh, Pennsylvania	Two 2,785 PWR Kori-3 and Kori-4 Value of items $200,000,000.	S. Korea	10/4/78

safeguards. The NRC published a proposed rule that would implement IAEA safeguards for its licensees. The NRC also participated in concluding discussions with the IAEA regarding the general part of the "Subsidiary Arrangements" to the U.S./IAEA Safeguards Agreement (which outlines how international safeguards will be carried out in the U.S.) and began assisting the IAEA in developing model "Facility Attachments" (which defines the safeguards to be applied at specific facilities). Once the Agreement enters into force, the IAEA will receive safeguards information from all U.S. civil nuclear facilities "not of direct national security significance" and it is expected to select a small number of facilities for full safeguards inspections by IAEA inspectors. Activation of the Agreement will fulfill a 1967 Presidential offer to apply IAEA safeguards to U.S. civil nuclear facilities in order to demonstrate to other nations, particularly the nonnuclear weapon states, that application of international safeguards measures would not impose commercial disadvantages. Both the United Kingdom and France, nuclear weapon states like the U.S., have made similar voluntary offers.

10

Standards Development

NRC standards provide for protection of the public and nuclear industry workers from radiation, the safeguarding of nuclear materials and facilities from theft and sabotage, and protection of the quality of the environment in nuclear activities. Thus, the development of standards cuts across the range of the NRC's activities and requires close interaction between the Office of Standards Development and the agency's other program offices.

While many of the standards issued or worked on during fiscal year 1978 are discussed in this chapter, some are discussed elsewhere in this Annual Report under the topics to which they relate (e.g., transportation in Chapter 3 and safeguards in Chapter 4).

CONCERNS OF HIGH PRIORITY

Current issues of high priority in standards development which are discussed in this chapter include:

Decommissioning. NRC policy is being reevaluated in this area with a view toward improving standards for all nuclear facilities. Major technical studies are continuing on the engineering methodology, radiation risks, and estimated costs of decommissioning light water reactors and fuel cycle facilities.

Spent Fuel Storage. Proposed licensing requirements were issued in October 1978 for independent spent nuclear fuel storage installations to supplement the capacity of pools at reactor sites. (See also Chapter 3.)

Nuclear Medicine. A proposed policy statement and rule changes provide for NRC regulation of the radiation safety of workers and the general public and of the radiation safety of patients with minimal intrusion into medical judgments affecting patients.

Occupational Exposure. The NRC is considering rule changes to strengthen and make more inspectable and enforceable its requirements that workers' exposures to radiation be kept not only within regulatory limits, but " as low as is reasonably achievable" within such limits. In addition, the Commission is committed to conduct a public hearing in 1979 on the adequacy of present radiation protection occupational standards.

Early Site Review. Detailed guidance on procedures and possible technical review options is being prepared to make nuclear plant siting decisions possible before the site is needed and before large commitments of resources are made.

Low-Level Radiation Effects. NRC has mounted a substantial effort in studying potential health effects of low-level radiation on humans. A public meeting was held on low-level radiation risk. NRC assisted the Department of Health, Education and Welfare in its Presidential assignment to develop a program responding to concern about the effects of exposure on workers in nuclear-related projects.

Seismic and Geologic Criteria. Revision of NRC's seismic and geologic siting criteria for nuclear power plants is underway to reflect advances in scientific knowledge and experience in licensing.

High-priority standards activities discussed in other chapters include the following:

Transportation. NRC issued a final environmental statement on transportation of radioactive materials by all modes and held an informal public "workshop" meeting on a study of transport through urban areas. The latter assessment will be the subject of a draft environmental statement planned for issuance in 1979. (See Chapter 3.)

Smoke Detectors. An assessment of the environmental impact of consumer products containing radioactive material will concentrate, as a priority issue, on the health aspects of the increasing use of ionization chamber smoke detectors containing americium-241. (See Chapter 3.)

Safeguards. The NRC issued a final rule and guidance for upgrading the training and qualification of personnel who guard nuclear facilities and strategic special nuclear material (SNM) shipments, and amended requirements for licensees' safeguards contingency plans. Proposed rules were published concerning (a) upgraded physical protection requirements for fuel cycle facilities and transportation, (b) upgrading of safeguards for SNM of moderate

REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth in Title 10, Chapter I, 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 a new or amended regulation is proposed, it is first published in the *Federal Register* to allow interested citizens time for comment before final adoption, in accordance with the Administrative Procedure Act. Following the public comment period, proposed regulations are revised, as needed, to reflect the comments received. If the regulation is adopted by the NRC, it is published in the *Federal Register* in final form with the date it becomes effective. After that publication, rules are codified for inclusion in the annual publication of the Code of Federal Regulations.

Some regulatory guides delineate techniques used by the staff to evaluate specific situations. Others provide guidance to applicants concerning information needed by the staff in its review of applications for permits and licenses. Many NRC guides refer to or endorse consensus standards (also called "national 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 of the standard has been made 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 at all times, and they are revised to take account of appropriate comments and suggestions and to reflect new information or experience. Newly issued guides have a comment period of about two months after issuance, following which the staff reviews the comments received and revises the guides, as appropriate.

Copies of regulatory guides are also mailed for comment to many individuals and organizations. When a guide is issued, a staff analysis of it is placed in NRC's Public Document Room in Washington, D.C. The analysis indicates the objective of the guide, its expected effectiveness compared to alternative ways of achieving the objective, and expected impacts on other safety systems, NRC operations, other Government agencies, industry, and the public.

Proposed and effective regulations published during fiscal year 1978 are summarized in Appendix 4. Regulatory guides issued, revised, or withdrawn are listed in Appendix 5.

and low strategic significance, (c) application of International Atomic Energy Agency safeguards to U.S. facilities, and (d) licensing of SNM carriers. A public hearing was held on proposed regulations to require NRC clearance of personnel for access to or control over SNM or vital areas at nuclear facilities. (See Chapter 4.)

High-Level Waste. The NRC is developing a rule, backed by extensive safety and environmental research, to establish licensing requirements for high-level radioactive waste repositories. (See Chapter 5.)

In other areas of high priority in standards development, the NRC is:

- Defining a set of "design basis" tornado missiles to help ensure that nuclear structures, systems and components important to safety are designed to withstand tornado environments.
- Seeking to upgrade the capability of inservice reactor inspection methods to reliably detect and characterize flaws in components of the primary coolant and other safety-related systems. Research is underway and guides are being developed regarding inspections of welds in pressure vessels and austenitic piping.
- Taking steps to ensure that petitions to the Commission for rulemaking are handled in an efficient and timely manner.

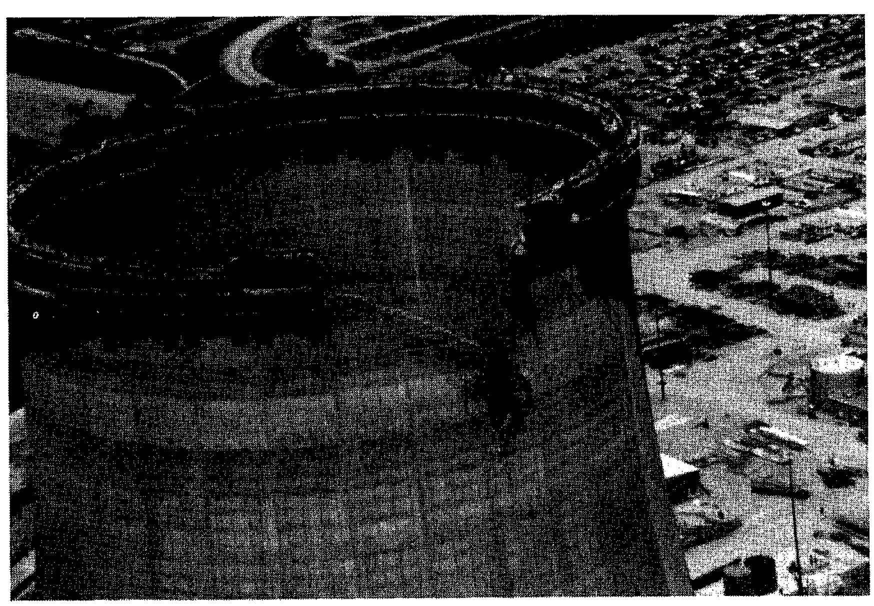
POWER REACTOR STANDARDS

Development of power reactor standards continued during fiscal year 1978 to be aimed primarily at protecting the health and safety of the public and secondarily at reducing the regulatory burden.

Reporting Defects and Noncompliance

A rule (10 CFR Part 21) requiring certain persons to report to the NRC defects that could create a substantial safety hazard or failures to comply with regulations relating to substantial safety hazards became fully effective in January 1978. The rule, which implements Section 206 of the Energy Reorganization Act of 1974, established a reporting system that, to some extent, anticipates problems before they occur.

NRC's experience in implementing the regulation in the early months of 1978, however, disclosed an unintended impact due to interpretations of the term "basic component" as defined in the rule. Some construed the rule as applying to orders for items available in general commerce such as "standard stock," "off-the-shelf," or "commercial grade" equipment. Thus, 10 CFR Part 21 was being applied by organizations within its scope to an extent not contemplated by NRC, causing problems in the supplying of equipment.



NRC's Standards Development Office revised its tornado design classification guide in April 1978 to prescribe a more acceptable method for identifying features of reactor plants needing special protection from tornadoes and tornado missiles. This cooling tower, under construction at the Grand Gulf, Miss., nuclear power plant was damaged by a tornado on April 17.

Amendments to correct the situation were adopted by the Commission in October 1978. They provide that an item available in general commerce which has no unique requirements imposed for nuclear application will not be within the scope of the revised rule until the item is designated for use as a basic component of a regulated facility or activity.

(Implementation of the rule is discussed in Chapter 6.)

Surveillance and Inservice Inspection

Revision 1 to Guide 1.133, which recommends a program for detecting loose parts in the primary system of light-water-cooled reactors is being developed to reflect public comments, including those from two public meetings held specifically for this guide by an ACRS subcommittee.

The staff, working closely with a committee of the American Society of Mechanical Engineers, has developed supplementary criteria to the Section XI, "Rules for Inservice Inspection of Nuclear Power Plants" of the ASME Boiler and Pressure Vessel Code to permit adoption by NRC in its regulations of the recent edition and addenda to these inservice inspection rules. The last edition and addenda to Section XI adopted was the 1974 edition and the Summer 1975 Addenda. The supplementary criteria have been incorporated in the Summer 1978 Addenda to the 1977 edition of the Code and an amendment to NRC regulations will be proposed to incorporate these changes with appropriate modifications.

Accident Analysis

The NRC is considering modifying the Emergency Core Cooling System (ECCS) Rule (Section 50.46 of 10 CFR Part 50 and Appendix K to 10 CFR Part 50) to take into account experience with the rule in the licensing process, new research information, and reactor operating experience gained since the rule was implemented. Various alternatives for rule modification have been studied by the NRC staff, and an action plan has been prepared. In December 1978, the NRC published in the *Federal Register* an advance notice of the proposed rulemaking action and invited public advice and recommendations.

Protection Against Fire

The fire protection guidelines for nuclear power plants published in Guide 1.120 in June 1976 were revised in response to comments received. The guide was reissued in November 1977 for an extended one-year comment period due to (1) the extent of revisions to accommodate public comments and (2) a suggestion of the Advisory Committee on Reactor Safeguards that the staff consider a "dedicated shutdown system" instead of some of the individual fire protection items called for in the guide. The guide describes how to implement NRC's requirement that the probability and effects of fire must be minimized through fire prevention, detection, and suppression. It also provides guidelines for designing fire safety features into nuclear power plants.

Sandia Laboratories, under NRC contract, is continuing to develop the technical bases for guidance in ventilation, fire detection, barriers, and fire hazards analysis.

Protection Against Extreme Loadings

Revision 1 to Guide 1.91 was issued for comment in February 1978. It describes acceptable methods for determining whether the risk of damage due to an explosion on a nearby transportation route is sufficiently high to warrant a detailed investigation. Acceptable methods for evaluating structural adequacy when an investigation is warranted are also described. The scope of this guide is limited to solid explosives and hydrocarbons liquefied under pressure and is not applicable to cryogenically liquefied hydrocarbons, e.g., liquefied natural gas (LNG). The effects of airblasts on highway, rail, and water routes are considered, but pipelines and fixed facilities are excluded.

Guide 1.142, issued for comment in April 1978, describes an acceptable method for complying with NRC regulations related to ensuring that concrete structures important to safety are designed to withstand the effects of postulated accidents and environmental conditions.

Seismic Design

Revision 1 to Guide 1.122, describing acceptable methods for developing the two horizontal and one vertical floor design response spectra at various floors or other equipment-support locations of interest, was issued in February 1978.

The guide uses the time-history motions resulting from the dynamic analysis of the supporting structure. The floor design response spectra are needed for the dynamic analysis of the systems or equipment supported at various locations of the supporting structure.

Reactor Containment

Containment Design. Guide 1.141, issued for comment in April 1978, describes an acceptable method for complying with NRC regulations on isolation capabilities for piping systems penetrating the primary reactor containment.

In October 1978, the NRC published a regulation that will reduce significantly the number of the plants required to have inert containment atmospheres in order to prevent hydrogen explosions under certain accident conditions. This change takes account of increased conservatism in the revised emergency core cooling system requirements. Revision 2 to Guide 1.7, which describes acceptable methods for implementing the new rule, was issued in December.

Concrete Containment and Structures. Guide 1.136, issued for comment in November 1977, endorses the ASME Boiler and Pressure Vessel Code rules for materials for concrete containments.

System and Component Criteria

General Design Guidance. The Codes and Standards Rule (Section 50.55a of 10 CFR Part 50) was amended again to incorporate new nuclear addenda of the ASME Boiler and Pressure Vessel Code.

Modifications to the ASME Code are often introduced through "Code Cases," a document published by the ASME Boiler and Pressure Vessel Committee. Generally, the individual sections of this document explain the intent of Code rules. NRC provides the industry with a timely indication of its approval or disapproval of such code cases through the prompt revision of Guides 1.84 and 1.85. Following procedures for revising these guides after ASME Council meetings that approve new code cases, the NRC issued three revisions of each guide during the year.

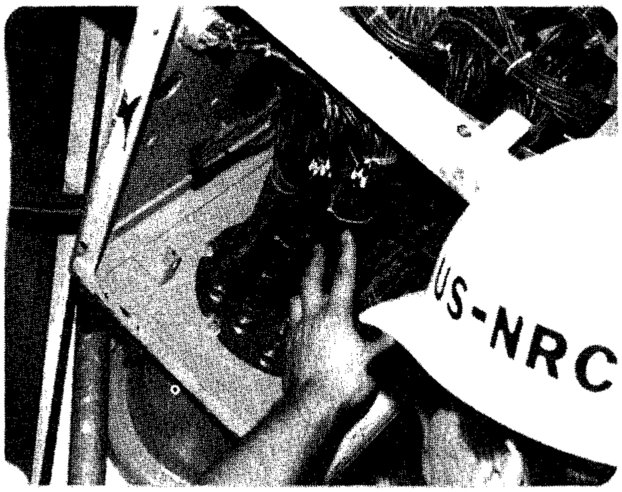
Revision 1 to the tornado design classification guide, Guide 1.117, was issued in April 1978. This guide describes an acceptable method for identifying those structures, systems, and components of light-water-cooled reactors that should be protected from the effects of the design basis tornado, including tornado missiles, and remain functional.

Guidance on Systems and Components. The following guide revisions were issued to reflect public comments: Revision 3 to Guide 1.31, on control of ferrite content in stainless steel weld metal, in April 1978; Revision 1 to Guide 1.126, on methods for the analysis of fuel densification, in March 1978; and Revision 1 to Guide 1.124, on service limits and loading combinations for ASME Class 1 linear-type component supports, in January 1978.

Revision 1 to Guide 1.72, on fiberglass-reinforced spray pond piping, was issued for comment in January 1978. Revision 1 to Guide 1.56, on maintenance of water purity in boiling water reactors, was issued for comment in July 1978. It incorporates operating experience in the methods provided for minimizing the probability of corrosion of reactor coolant pressure boundary components.

During fiscal year 1978, a review of NRC fracture prevention requirements for reactor vessels resulted in the draft of revisions to relevant regulations. Also, NRC's research program on radiation damage to materials has been expanded with greater emphasis on development of surveillance and fracture analysis methods.

Revision 2 to Guide 1.68, which describes acceptable methods for complying with NRC regulations on preoperational and initial startup testing programs for water-cooled power reactors, was issued in August 1978. Another guide, Revision 1 to Guide 1.68.2, in the series of



In its continuing program to upgrade the safety and effectiveness of reactor-plant electrical systems, NRC applies experience factors, as reflected in a variety of operational and inspection reports, to the improvement of regulatory guides and standards. Here an NRC inspector examines electrical connectors at the Salem (N.J.) Nuclear Power Plant, Unit 2.

guides being developed to provide more detailed guidance concerning specific areas of the preoperational and initial startup testing program was issued in July 1978. This guide describes the initial startup test program to demonstrate remote shutdown capabilities for water-cooled nuclear power plants.

Guide 1.137, issued for comment in January 1978, describes an acceptable method for complying with NRC regulations regarding fuel-oil systems for standby diesel generators and assurance of adequate fuel-oil quality.

Guide 1.139, issued for comment in May 1978, describes an acceptable method for complying with the NRC regulations with regard to the removal of decay heat and sensible heat after a reactor shutdown.

Revision 2 to Guide 1.52, issued in March 1978, and Guide 1.140, issued for comment in March 1978, address design, testing, and maintenance criteria for air filtration and adsorption units for postaccident engineered-safety-feature atmosphere cleanup systems and for normal ventilation exhaust systems, respectively.

Guide 1.143, issued for comment in July 1978, provides design guidance for radioactive waste management system components.

Electric Systems and Components

Emphasis was placed on the reliability of direct current systems and components for nuclear power plants. Two related regulatory guides pertaining to nuclear power plant station batteries were issued: Revision 1 to Guide 1.128, on installation design and installation of large lead storage batteries, in October 1978, and Revision 1 to Guide 1.129, on maintenance, testing, and replacement of large lead storage batteries, in February 1978.

Revisions updating the following systems-oriented guides were also issued: Revision 1 to Guide 1.118, on periodic testing of electric power and protection systems, in November 1977, and Revision 2 in June 1978; and Revision 2 to Guide 1.75, on physical independence of electric systems, in September 1978.

Qualification Testing

Electrical. Work continued on the development of standards and guides for the qualification testing of electric equipment. Supporting research continues at Sandia Laboratories on test source equivalence, synergistic effects, and aging. Underwriters Laboratories continue the NRC-sponsored study of the adequacy of IEEE Standard 383-1974 on flammability testing.

Revision 2 to Guide 1.63, on electric penetration assemblies, was issued in July 1978.

The NRC staff continued to participate with national standards committees in developing criteria for qualifying specific electric components that are important to safety, including modules, connectors, battery chargers, penetration fire stops, and motor control centers, as well as a general standard for qualifying both electric and mechanical equipment. NRC also participated in the updating of existing national qualification standards, including those for qualifying electric valve operators, cables, and continuous duty motors.

Mechanical. The staff is working closely with two national standards groups that are developing standards for qualification tests to make sure that safety-related pumps and valves will operate in their appropriate environments when called upon. The NRC staff is currently developing a guide to endorse an American National Standards Institute (ANSI) standard on functional specifications for self-operated and power-operated safety-related valves for applications in nuclear power plants.

A guide is being developed on qualification tests for safety-related snubbers (the components in piping systems intended to resist excessive motion under severe loads, e.g., during earthquakes, while allowing normal motion during operation) to provide design and test methods for ensuring proper snubber operation during normal and abnormal plant conditions.

Quality Assurance

Quality assurance requirements for the design, construction, and operation of safety-related structures, systems, and components of nuclear power plants are established in Appendix B to 10 CFR Part 50. During the past fiscal year, the NRC issued revised guides concerning the implementation of these requirements: In March 1978, Revision 2 to Guide 1.33, on quality assurance program requirements for the operation of nuclear power plants, was issued, and Revision 1 to Guide 1.28, on quality assurance program requirements for the design and construction of nuclear power plants, was issued for comment.

In addition to these guides, Revision 1 to Guide 2.5, which describes an acceptable method for establishing and executing a quality assurance program for the design, construction, testing, modification, and maintenance of research reactors, was issued in November 1977.

Water Control Structures

Nuclear power plants use water control structures such as dams and canals for a variety of purposes. In March 1978, the NRC issued to reflect public comment Revision 1 to Guide 1.127, which covers the inspection of water control structures associated with nuclear power plants.

Maintaining Safety at Multiunit Sites

In August 1978, the NRC published a rule that would require applicants for construction permits and operating licenses for multiunit reactor sites to take proper precautions to ensure the integrity of structures, systems, and components important to the safety of any operating unit while construction goes forward on other units. The rule was considered in response to a petition for rulemaking filed by the Business and Professional People for the Public Interest.

Control Room Observer

The Commission denied a portion of a petition for rulemaking concerning stationing a full-time Federal employee in a reactor's control room with full authority to shut down the plant in case of an operational abnormality. The Commission concluded that the current inspection program, which will be improved by increasing onsite presence and capabilities to perform independent verification, adequately provides for fulfillment of NRC responsibilities with respect to audit and inspection of nuclear power plants.

In its revised inspection program, an NRC resident inspector will be assigned to each operating reactor site and to selected construction sites. (See Chapter 6.) The NRC regional office will provide technical support. There will be increased capability for independent verification of licensee action.

Underground Siting of Reactors

The Commission denied two other portions of the above petition that are related to placing reactors underground and in heavy vacuum containments. These parts of the petition were denied because there is insufficient supporting material to indicate that such design provisions should be made mandatory to the exclusion of all other nuclear power plant designs.

FUEL CYCLE PLANT STANDARDS

The NRC devoted substantial effort during fiscal year 1978 to the development of standards concerning the safety and environmental impacts of fuel cycle plants.

NRC published a detailed plan and schedule for the reevaluation of its policy on decommissioning nuclear facilities in March 1978 and submitted it to State governments and interested segments of the public for comment. In September, regional workshops were held in three cities to discuss the plan with State officials. In this picture, NRC's Sheldon A. Schwartz, Assistant Director for State Program Development, briefs a Philadelphia regional workshop audience.

Decommissioning

The NRC is giving increased attention to the proper retirement or decommissioning of nuclear facilities. Technical studies for NRC are continuing at the Battelle Pacific Northwest Laboratory (PNL) to develop a decommissioning information base for light water reactors and fuel cycle facilities. This base will be used in developing appropriate regulations and guides. The PNL reports on the technology, safety and cost of decommissioning a reference nuclear fuel reprocessing plant (NUREG-0278) and a reference pressurized water reactor power station (NUREG/CR-0130) were published in October 1977 and in June 1978, respectively. Another PNL report on the decommissioning of small mixed-oxide plants (NUREG/CR-0129) was nearing completion at the end of the fiscal year.

These reports are part of the comprehensive reevaluation of NRC policy relating to decommissioning nuclear facilities. The detailed plan and schedule for this reevaluation is described in an NRC staff report entitled "Plan for Reevaluation of NRC Policy for Decommissioning of Nuclear Facilities" (NUREG-0436), which was published in March 1978 and sent to all States for comment. Three regional workshops were held in September 1978 to review the specifics of the NRC plan with State officials.

During the year, work proceeded on a response to a petition by the Public Interest Research Group et al., to initiate rulemaking to promulgate regulations for nuclear power plant decommissioning that would require plant operators to post bonds, to be held in escrow, to ensure that funds will be available for proper and adequate isolation of radioactive material upon each plant's decommissioning. Factors being considered in the response include, among other things, the present unavailability of long-term bonds and whether other alternatives offer reasonable financial assurance for achieving essentially the same results as proposed by the petitioners but in a more economical and flexible manner. One major component of the overall reevaluation described in NUREG-0436 is an extensive examination of the financial assurance needed to cover decommissioning costs. It is intended during this examination to assess the relative merits of several different financial assurance techniques to weigh and judge the financial assurance needed regarding decommissioning the various classes of nuclear facilities.

During 1978, the staff participated in hearings (discussed in Chapter 3) on a proposed revision to Table S-3 of 10 CFR Part 51 concerning uranium fuel cycle environmental impacts from spent fuel reprocessing and radioactive waste management. Testimony was provided on estimates of environmental impacts that would occur from the decommissioning of fuel cycle facilities.

Spent Fuel Storage

As a result of the need to accommodate some of the accumulating spent fuel from commercial reactors, the NRC issued for public comment in October 1978 a proposed new regulation covering the requirements for extended storage at installations built specifically for this purpose that are not coupled to either a nuclear power plant or a fuel reprocessing plant. In addition to general provisions, the proposed regulation contains siting requirements, general design criteria, and certain operational aspects of such installations. Guides on license application and design requirements for these facilities are being developed, and work continues on other guides for facility siting and plant protection. (See Chapter 3.)

Nuclear Criticality Safety

Several objectives for providing guidance to applicants on nuclear criticality safety were realized. Revision 1 to Guide 3.4, on acceptable procedures for the prevention of criticality accidents during operations with fissionable materials outside reactors, was reissued in February 1978. Guide 3.43, issued for comment in August 1978, provides guidance on nuclear criticality safety in the storage of fissile materials.

Plant Safety

Several guides address safety issues other than nuclear criticality safety (discussed above). Revision 1 to Guide 3.5, on the content of applications for uranium milling licenses, was issued for comment in November 1977. Revision 2 to Guide 3.11, on design, construction, and inspection practices and methods for embankments systems to retain mill tailings at uranium mills, was issued in December 1977. Revision 1 to Guide 3.40, which characterizes floods to be used as a basis for the design of fuel reprocessing and plutonium processing and fuel fabrication plants, was also issued in December 1977.

ANSI continued its emphasis on the preparation of standards and guides for fuel cycle facilities in the areas of quality assurance and radiological and safety-related features. NRC staff members participated in the work on ANSI committees.

Waste Management

The NRC is giving increased attention to the development of regulations needed to ensure the safe disposal of both high-level and other radioactive wastes. Under development are proposed rules for licensing of high-level and low-level radioactive waste management facilities and supporting guides on license application requirements. (See Chapter 5.)

General Site Suitability Criteria

The staff is developing guidance for siting fuel cycle facilities. Since no general guidance exists on site suitability for these facilities, the staff has drawn on guidance prepared for the siting of nuclear power plants. Initially such guidance will use a format similar to that used for nuclear power plants, but the criteria applied will be modified specifically for fuel cycle facilities.

In support of this effort, the staff has contracted with the Environmental Impact Division of Argonne National Laboratory to collect and analyze data on occurrences (both accidents and natural phenomena) that bear on the impact of nuclear fuel cycle facilities at existing sites.

SITING STANDARDS

The standards on the siting of nuclear plants deal with procedures for site review, site safety, and protection of the environment.

Site Review Procedures

Early Site Reviews. A draft revision to NUREG-0180, describing in more detail the procedures and possible technical review options, was issued for comment in February 1978. The staff continued developing detailed descriptions of the review options possible for the remaining technical issues. (See Chapter 2.)

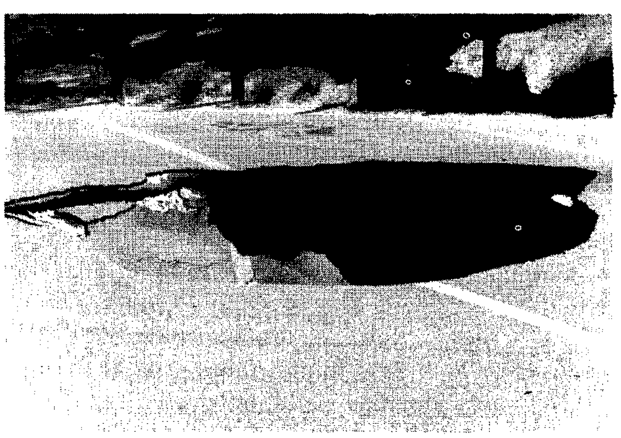
Siting Policy and Practice. The staff completed preparation of draft policy statements on

(1) alternative site evaluations under NEPA for nuclear generating stations and (2) emergency planning. In connection with the emergency planning policy statement, NRC issued for comment a proposed amendment to Appendix E to 10 CFR Part 50 on emergency planning outside the low population zone.

A staff review of current accident evaluation practices in siting and licensing of nuclear power plants was completed. Also completed was a review of methodology for accident consequence assessments as part of an overall review of accident evaluation practices being conducted under NRC contract by Battelle Pacific Northwest Laboratory.

The NRC has contracted with the Brookhaven National Laboratory for a two-year study to assist in developing staff procedures for evaluating methods for selecting sites for nuclear power plants.

NRC/State Cooperation. Technical siting issues that are of concern to the States as well as the NRC, such as need for power, alternative site selection, water resources management, regional geology, and socioeconomic effects, are being addressed in a demonstration program with the member States of the Southern States Energy Board (SSEB). The program, which began in fiscal year 1977, is designed to develop procedures and standards that will resolve the siting issues that arise as a part of the site selection and regulatory process. South Carolina, North Carolina, and neighboring States are working together to address technical issues of common concern and provide procedural and technical information for use in improving NRC standards for the site selection and evaluating the site selections, particularly in early site review. The other member States of the SSEB are reviewing the process for compatibility with their own institutional arrangements.



Standards used for the selection of sites for nuclear facilities and activities must take account of both natural and man-made hazards, including the type of geologic and meteorologic hazards that produced this parking-lot ground collapse. The lot was built on limestone which later dissolved as a result of groundwater erosion.

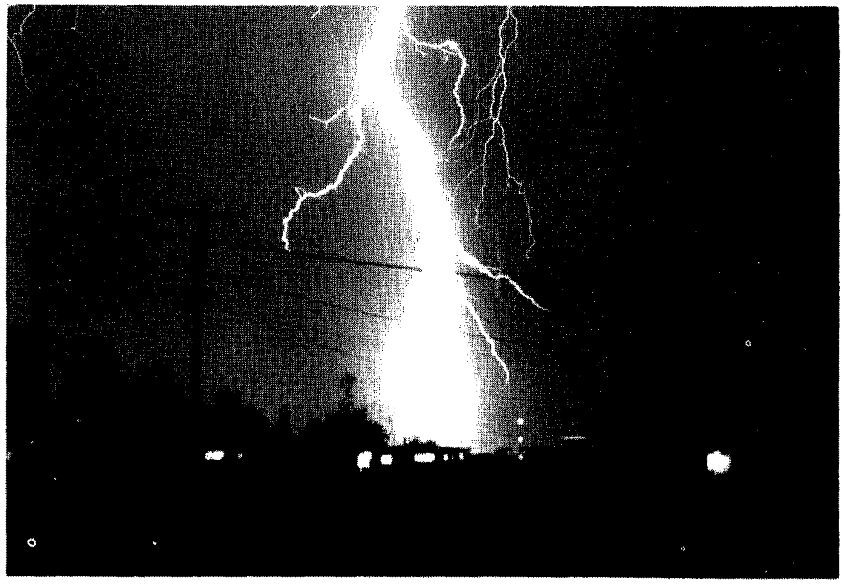
Coastal Zone Management. Several coastal states have submitted their Coastal Zone Management (CZM) programs for review by key Federal agencies and ultimately for approval by the Federal Office of Coastal Zone Management in the Department of Commerce. Since NRC actions must be consistent with approved State CZM programs, the NRC staff is participating in the review of these programs to promote compatibility and consistency with the existing and developing procedures by which the NRC carries out its mission.

Site Safety

NRC site safety standards are rules and guides for assessing and mitigating adverse effects associated with natural events such as earthquakes, floods, and extreme meteorological conditions and man's activities at and near nuclear sites.

In the field of meteorology, the staff is continuing data evaluation for the development of standards on extreme wind speeds for coastal areas, extreme snow and ice accumulations, extreme temperatures, and the hazards associated with lightning. A regulatory guide on atmospheric dispersion models for potential accident consequence assessments at nuclear power plants is nearing completion. In addition, two other meteorological guides are being revised.

In the geology and seismology area, review continued of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100. The review is being carried out in light of the experience gained since adoption of the regulation in 1973. In late 1977 and early 1978, public meetings were held with the Seismic Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) to obtain their views, and public comments were requested in a *Federal Register* notice. The staff is assessing the comments received. In other earth science activities, NRC published



NRC's development of meteorological standards governing hazards from lightning entails the recording of and compilation of data from phenomena such as this cloud-to-ground lightning strike. The photo was taken about two miles from the strike.

NUREG-0406, "Methods for Prediction of Strong Earthquake Ground Motion," and is analyzing technical data for guides on (1) methods used for dating fault movement, (2) characterization and classification of geologic faults and fractures in the Appalachian foldbelt, and (3) the siting of nuclear facilities in areas susceptible to ground collapse. Data are also being evaluated for use in standards for high-level radioactive waste disposal.

A proposed regulation, 10 CFR Part 60, is being developed for licensing geologic repositories for the disposal of high-level radioactive wastes. The rule will include subparts on general procedures, performance objectives, and general technical criteria. The procedural portion is scheduled to be published for comment in early 1979. (See Chapter 5.)

In the area of geotechnical engineering, Guide 1.132, on site investigations for foundations of nuclear power plants, is being revised to reflect public comments. Guide 1.138, on laboratory investigations of soils for engineering analysis and design of nuclear power plants, was issued for comment in April 1978. Guides for nuclear power plants have been developed and are under staff review on the following subjects: procedures and criteria for assessing seismic stability of soils, quality control and assurance of foundation and earthwork construction, and geologic mapping of excavations.

In the hydrology area, contract work was completed on "Probable Maximum Flood Estimates, Ohio River," which will serve as the technical base for a revision to Guide 1.59 regarding design basis floods for nuclear power plants. Revision 1 to Guide 1.125, on physical models for the design and operation of hydraulic structures at nuclear power plants, was issued in October 1978. A revision to Guide 1.135, on normal water level and discharge, is in progress. ANSI committee work on surface water and ground water supply is also nearing completion. Work was begun on the hydrologic assessment of siting criteria for high-level radioactive waste repositories.

ENVIRONMENTAL STANDARDS

Environmental standards are concerned with the protection of the public and the environment from both radiological and nonradiological impacts of nuclear facilities. This includes assessment of environmental impacts, control of effluents, and monitoring of the environment around the facilities. In the past, emphasis has been placed on development of environmental standards for nuclear power plants. Currently, greater emphasis is being placed on developing standards for other nuclear facilities.

During fiscal year 1978, the following regulatory guides were issued for comment: Revision 1 to Guide 3.8, on preparing environmental reports for uranium mills, and Guide 4.16, on measuring radioactive materials released from fuel fabrication plants. A regulation was proposed to revoke Section 20.304 of

10 CFR Part 20, which currently allows licensees to bury small quantities of radionuclides without notifying NRC.

The NRC received two petitions in the environmental standards area. In one petition, the New England Coalition on Nuclear Pollution requested amendments to Table S-3 of 10 CFR Part 51, which quantifies environmental impacts of fuel cycle facilities supporting nuclear power plants. The NRC amended Table S-3, but differently from the way in which the petitioner requested. The entry for radon-222 from uranium mill tailings was removed from the table as substantially understated. Further work to update the table is in progress. (See Chapter 3 under "Environmental Survey of the Uranium Fuel Cycle.") In the other petition, the State of New Jersey asked NRC to publish additional regulations specifying what quantities of radioactive material in effluents would be considered "as low as is reasonably achievable" for large radioisotope facilities.

A substantial effort is now being devoted to the environmental aspects of uranium milling, decommissioning and decontaminating nuclear facilities, radioactive waste disposal, and continued consideration of the health effects of low-level radiation.

Health Effects of Low-Level Radiation

The NRC has expended substantial effort in studying potential health effects of low-level radiation on humans. Efforts during this fiscal year included funding of research on the effects of specific radioactive isotopes and of epidemiology studies, the analyses of current research in the field of radiobiology and epidemiology, the drawing up of preliminary plans to study the feasibility of a large-scale epidemiology investigation on low-level effects, and the convening of a public meeting on the health risks of exposure to low-level radiation.

The work in the area of low-level effects will continue at an expanded level to ensure that NRC health-related radiation regulations reflect the most recent scientific data.

The Director of the Office of Standards Development presented testimony before the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works in April 1978. He discussed the responsibilities for setting radiation standards, the bases used for setting them, NRC activities concerning radiation standards and exposure limits for workers, and NRC staff views on the conduct of a major epidemiological study of the effects of low-level radiation.

NRC provided assistance to the Department of Health, Education, and Welfare (HEW), assigned by the President to develop a program responding to concern about the effects of radiation exposure on workers in nuclear-related projects. In this regard, representatives of NRC and HEW met, with NRC providing information to aid in the preparation of a Presidential report scheduled for completion in early 1979.

Interagency Coordination

NRC has the responsibility for implementing both EPA's guidance and generally applicable environmental standards for protection against radiation. During 1977, EPA published standards (40 CFR Part 190) which limit releases of radioactive material and resulting doses to the public from the operation of various nuclear facilities associated with the uranium fuel cycle. An NRC task force, which includes EPA staff members, is establishing the program for implementing these standards.

The NRC became a member, along with 15 other major Federal agencies, of the Toxic Substances Strategy Committee, formed under the leadership of the Council on Environmental Quality. NRC staff served on seven of the task groups of this committee, which was to submit a report to the President by the end of 1978 recommending strategies to be used by the Federal Government for the control of toxic and hazardous substances. Although radioactive materials have been excluded from this report, the principles for controlling cancer-causing materials would be expected to affect radiation control strategies, and the expertise gained by NRC in controlling radiation is directly applicable to some aspects of controlling other carcinogens.

International Activities

The NRC staff is participating in activities of the Nuclear Energy Agency (NEA) of the

Organization for Economic Cooperation and Development (OECD). The NRC is represented on the NEA Radiation Protection Committee and on NEA expert groups on the control of long-lived radionuclide emissions from the nuclear fuel cycle, the control of ionization chamber smoke detectors, and the control of naturally occurring radioactive materials.

The staff has been working with the Department of Energy and EPA to develop the U.S. position on a recent draft of revised IAEA basic radiation safety standards. The NRC was represented on a working group that prepared these revised standards.

The IAEA is preparing guidance on principles and procedures for establishing effluent limits for the release of radionuclides into the environment. A report on the principles for establishing limits is scheduled for issuance by the end of 1978. The NRC staff is represented on the IAEA advisory group preparing guidance on procedures to implement these principles.

In April 1978, the NRC staff also participated in an IAEA advisory group preparing guidance on reactor decommissioning.

SAFEGUARDS STANDARDS

The NRC devoted substantial standards efforts during fiscal year 1978 to the safeguarding of nuclear materials and facilities against theft and diversion. Development of regulations in this area is discussed in Chapter 4.

Physical Protection

In support of the newly adopted and proposed safeguards regulations discussed in Chapter 4 and existing regulations, the NRC issued several reports and regulatory guides. They include:

(1) NUREG-0320, "Interior Intrusion Alarm Systems." In meeting the requirements for safeguarding of special nuclear material and for physical protection of licensed facilities, the licensee is required to design a physical security system that will meet minimum performance requirements. An integral part of any physical security system is the interior intrusion alarm system. The purpose of this report is to provide information on the various types, components, and performance capabilities available to enable the user to design and install the appropriate alarm system. In addition, this report discusses and recommends maintenance and testing procedures that, if followed, will help the user obtain optimum results.

(2) NUREG-0464, "Site Security Personnel Training Manual" (published for comment) and NUREG-0465, "Transportation Security Training Manual" (published for comment). Both these training manuals were developed to assist licensees in developing effective security personnel training and qualifications programs, as required by 10 CFR Part 73. The manuals typify the level and scope of training for security personnel assigned to perform specific tasks and job duties to protect special nuclear material, nuclear facilities, and shipments.

(3) Guides 5.54, 5.55, and 5.56. The Energy Reorganization Act of 1974, which established the NRC, directed the NRC, among other things, to develop contingency plans for dealing with threats, thefts, and sabotage relating to SNM, high-level radioactive wastes, and nuclear facilities resulting from all activities licensed under the Atomic Energy Act of 1954, as amended. These guides assist licensees in developing contingency plans acceptable to the NRC.

Material Control and Accounting

During fiscal year 1978, the NRC issued the following reports for improving SNM control and accounting.

(1) NUREG/CR-0019, "Value Impact of Vault Automation in Special Nuclear Material Storage," which presents the result of a cost-benefit study. The report indicates that automation of SNM storage vaults can significantly improve safeguards over material not in process in nuclear facilities.

(2) NUREG/CR-0014, "An Evaluation of the Use of Calorimetry for Shipper-Receiver Measurements of Plutonium." Three modes of use are discussed: (a) calorimetry alone, (b) calorimetry plus chemical assay, and (c) calorimetry plus gamma-ray spectrometry. The report indicates that calorimetry can be used in conjunction with another assay technique to substantially reduce shipper-receiver differences for plutonium.

(3) NUREG/CR-0033, "Procedures for Rounding Measurement Results in Nuclear Materials Control and Accounting," which discusses applications of the procedures presented. Rounding of data contributes to the uncertainty of measurement results and must be taken into account when calculating limits of error.

(4) NUREG/CR-0087, "Considerations for Sampling Nuclear Materials for SNM Accounting Measurement," which presents principles and guidelines for sampling nuclear material to measure the chemical and isotopic content of the material. Emphasis is placed on development of sampling plans and procedures that maintain the random and systematic errors of sampling within acceptable limits for SNM accounting purposes.

RADIOISOTOPES IN MEDICINE AND INDUSTRY

Nuclear Medicine

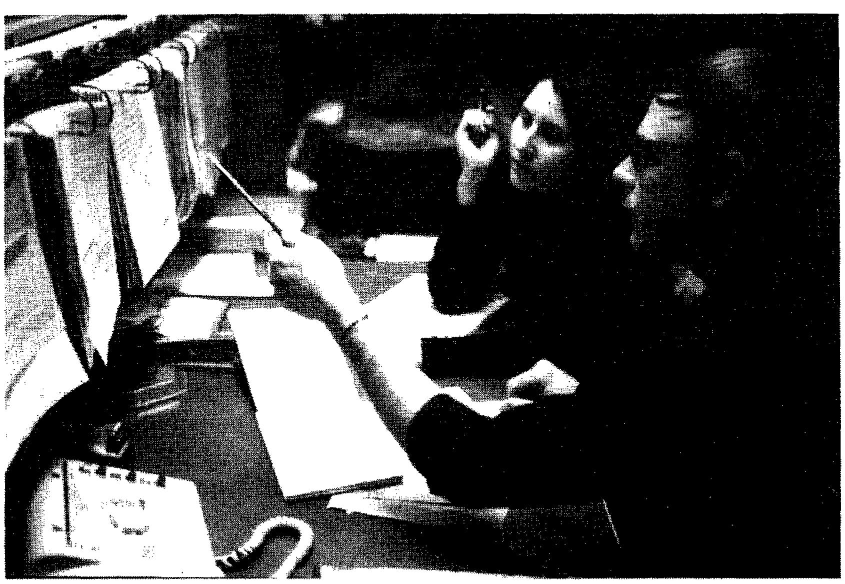
Several objectives were achieved during fiscal year 1978 in developing the NRC's regulations on use of nuclear materials to diagnose and treat human illnesses.

In March 1978, the NRC published a proposed policy statement on the regulation of the medical uses of radioisotopes. In essence, the policy provides that:

- The NRC will continue to regulate the medical uses of radioisotopes as necessary to provide for the radiation safety of workers and the general public.
- The NRC will regulate the radiation safety of patients where justified by the risk to patients and where voluntary standards or compliance with these standards are inadequate.
- The NRC will minimize intrusion into medical judgments affecting patients and into other areas traditionally considered to be a part of the practice of medicine.

Concurrently with publication of the draft policy statement, the NRC proposed to permit physicians greater latitude when they use certain low-dose diagnostic radiopharmaceuticals by no longer designating authorized clinical procedures.

In June 1978, the NRC proposed a rule change to require persons holding NRC specific licenses for human use of byproduct material to ensure that patients treated with cobalt-60, cesium-137, or iridium-192 implants remain hospitalized until a source count and a radiation



Improvement in control and accounting methods used in nuclear materials facilities is a continuing objective. In this photo, an NRC evaluation team member (left) checks revised NRC measurement procedures with the material control manager at a nuclear fuel cycle plant.

survey of the patient confirm that all implants have been removed.

Consistent with the draft nuclear medicine policy statement, the NRC published in July 1978 a proposed rule change to require nuclear medicine licensees to keep records of all misadministrations of radioactive material or radiation from radioactive material. Misadministrations include the administration of a radiopharmaceutical or radiation from a wrong source, to a wrong patient, or by a route other than that intended by the prescribing physician. The proposed rule would also require prompt reporting of potentially dangerous misadministrations to the NRC, to the patient's referring physician, and to the patient or the patient's responsible relative.

Products Containing Radioactive Materials

In January 1978, the NRC exempted from licensing requirements persons using small quantities of cobalt-60 near the spark gap of oil-fired furnaces to prevent ignition problems. The final environmental statement (NUREG-0137) prepared in connection with the exemption concluded that, in order to protect the environment, each spark gap irradiator must contain no more than one microcurie of cobalt-60 and must be used in electrically ignited fuel-oil burners having a firing rate of at least 3 gallons (11.4 liters) per hour.

In response to a petition for rulemaking, the NRC amended its general license for use of small quantities of source material to include operational use of source material by Federal, State, and local governmental agencies. The immediate benefit of the rule change was to lessen the existing administrative burden of specific licensing required for infrared lenses coated with a thin layer of thorium fluoride.

Addressing Petitions for Rulemaking

In July 1978, the NRC amended its statement of organization to delegate to the Executive Director for Operations (EDO) additional functions for dealing with petitions for rulemaking. One delegation authorizes the EDO to deny any petition for rulemaking where the grounds of denial do not substantially modify existing precedent. The other delegation authorizes the EDO to propose, in response to a petition for rulemaking, an exemption from licensing requirements for certain radioactive products when existing policy provides background or precedent. The rule change resulted from a staff study of improvements in efficiency and timeliness in dealing with petitions for rulemaking.

Licensing Matters

In July 1978, the NRC proposed to amend its regulations to require specific licensees to notify the NRC when they decide to permanently discontinue all activities involving byproduct, source, or special nuclear material. This would allow the NRC to communicate with licensees on a timely basis regarding disposition of licensed materals and cleanup of facilities.

In March 1978, the NRC published an advance notice of proposed rulemaking on safety design requirements for radiographic exposure devices. To aid in this undertaking, the NRC invited interested persons to submit information, comments, and suggestions on the requirements in writing or orally at an informal public meeting held April 18, 1978, in Bethesda, Maryland. The new requirements are intended to reduce radiation overexposures by radiography equipment failures.

Space Applications

In early 1978, at the request of the U.S. General Accounting Office and the Office of Science and Technology Policy, the NRC agreed to participate in all relevant nuclear safety evaluation processes for space launches. Accordingly, an NRC plan is under development for the safety review of nuclear systems for future space programs.

OCCUPATIONAL HEALTH STANDARDS

Respiratory Protection

The NRC's new requirement governing the use of respiratory protective equipment (in 10 CFR Part 20) became fully effective in December 1977. During 1978, additional information was provided to licensees who use such equipment. In March 1978, a notice was sent to licensees concerning the current status of requirements for the medical surveillance of people who wear respirators at licensed facilities. Since there is no currently developed standard method for medical surveillance of this type, the NRC has contracted with the Los Alamos Scientific Laboratory (LASL) to review the problems of medical surveillance of respirator users and to assist in the development of more definitive guidance.

Respirator performance tests conducted at Los Alamos Scientific Laboratory in 1978 are helping NRC develop new guidance on protective equipment used by workers in nuclear facilities. This photo, showing a standard commercial respirator protruding over the user's nose and, thus, failing to "seal" against her smaller-than-average face to offer protection, shows why new guidance is needed.

LASL also continued to provide measurements of the amount of protection provided by respirators. This information is used to revise and update guidance to licensees on the amount of allowance that may be made for the protection provided when respirators are used to limit the internal radiation doses to workers who are exposed to airborne radioactive materials. LASL submitted a progress report, LA-7089-PR, to the NRC on measurements that were completed on all approved airline supplied-air respirators. A significant finding was that such respirators, when operated in the "demand" mode, do not provide as much protection as was previously estimated. This information was made known to licensees in an Inspection and Enforcement Office Bulletin (No. 78-07) in June 1978. Research was also continued at LASL to develop acceptable performance criteria and test methods for air-purifying respirators to protect against airborne radioiodines. There are no approved air-purifying respirators of this type, and the development of the criteria and test methods would permit the testing and certification of such a respirator.

The NRC continued to cooperate with other governmental and nongovernmental agencies toward the development of needed improvements in occupational respiratory protection.

Exposures at Nuclear Power Stations

In June 1978, the NRC issued Revision 3 to Guide 8.8, which provides information and guidance on planning, designing, and operating a light-water nuclear power station to meet the objective that exposures of workers to radiation during operations will be maintained as low as is reasonably achievable (ALARA).

Transient Worker Radiation Protection

The NRC is considering whether to change its regulations to reduce the probability that short-term "transient" workers, particularly those employed by more than one NRC licensee in a calendar quarter, may receive radiation doses that exceed the NRC standards.

Under the present regulations, a licensee is required to control the use of licensed material so that radiation doses to workers in the licensee's facilities do not exceed the standards. However, a worker could work for more than one licensee in a quarter and receive a dose within the standards at each location, even though the combined dosage might exceed the standards. The proposed regulations are designed to prevent such exposure from happening.

The proposed amendments would require licensees to control the total occupational radiation doses to their workers. Licensees would be required to obtain information from a prospective worker on occupational doses already received during the calendar quarter in which the individual is assigned to work if there is a chance that the worker might subsequently exceed 25 percent of the standards. This information is to be used to prevent workers from receiving more radiation dose than the standards permit, regardless of the number of licensed facilities in which they work. Notice of proposed rule-making was published in February 1978. At year-end, the staff was considering the public comments received and other factors involved.

Medical Institutions

Guide 8.18 and a companion report giving more detailed information and references (NUREG-0267) were issued for comment in January 1978. These two documents provide broad guidance and information for establishing acceptable occupational radiation safety programs in medical institutions. Both documents have received a generally favorable reception from the medical and medical physics communities, but there have been a number of suggestions for improvement, additions, or deletions that will require careful balancing of various viewpoints in reaching the final versions. Several suggestions indicate that the NUREG report should be broadened in cooperation with other agencies to cover all sources of radiation in medical institutions, not just NRC-licensed materials.

Two other guides specific to radiation safety in medical institutions will be issued for comment early in fiscal year 1979: Guide 8.23, on radiation surveys in medical institutions, and Guide 10.8, on medical licensing. The licensing guide will explain the information to be submitted in an application for a license to use byproduct radioactive materials in diagnostic and therapeutic medical applications; will provide a simpler Form NRC-313M for completing the required entries; and will provide acceptable methods and statements related to radiation safety and user qualifications. The acceptable methods may be merely checked on the new form to indicate that the applicant agrees to follow the indicated procedures.

Bioassays

In order to give more uniform and definitive guidance to licensees and applicants planning surveillançe programs for internal radiation exposure, the NRC staff issued Guides 8.20 and 8.22 for comment during fiscal year 1978. These guides provide guidance for I-125 and I-131 bioassay and for bioassay at uranium mills, respectively. They supplement two previous guides that gave information on acceptable methods of interpreting bioassay results in general and specific guidance on interpreting uranium bioassays. The new documents provide guidance to management on the levels of radioactivity or working conditions under which bioassay should be performed. They also specify on whom the assays should be performed and the action levels at which appropriate investigative or corrective measures should be taken. The iodine bioassay guidance in Guide 8.20 takes into consideration the amounts of I-125 and I-131 above which exposure potential becomes appreciable, as indicated by industrial and medical experience.

In addition, a staff position on guidance for establishing tritium bioassay programs was prepared at the request of the Office of Nuclear Materials Safety and Safeguards. It is being used to inform license applicants of acceptable bioassay programs for installations that use various chemical and physical forms of tritium.

Health Physics Surveys at Manufacturing Plants

Guide 8.21, issued for comment in May 1978, identifies the types and frequencies of radiation surveys that are acceptable to the NRC staff in plants licensed to manufacture or process byproduct material for distribution. The guide tailors frequencies of surveys and permissible contamination levels to the relative radiotoxicity of the nuclides involved and to the relative hazards of the process in order to ensure that both internal and external exposures of employees will be maintained as low as is reasonably achievable.

Health Protection at Uranium Mills

In July 1978, Guide 8.22 was issued for comment. It recommends urinalysis and *in vivo* counting to determine intakes of uranium among workers who have the greatest potential for exposure. These bioassays serve to independently confirm intakes based on measurements of airborne uranium concentrations. A separate regulatory guide on health physics surveys at uranium mills is in preparation.

A memorandum of understanding to assure consistency of regulatory actions is being developed between the NRC and the Mine Safety and Health Administration (MSHA) of the Department of Labor. The Federal Mine Safety and Health Act of 1977 gives MSHA jurisdiction with respect to protection of uranium mill workers similar to that given NRC under the Atomic Energy Act of 1954, as amended.

Industrial Radiography Safety

In March 1978, proposed amendments to 10 CFR Part 34 on safety in industrial radiography were published for public comment. The staff is considering the comments in drafting the final amendments.

A petition for rulemaking to have the NRC license individual radiographers is under consideration. The petition states that safety in industrial radiography could be improved by making individual radiographers more directly responsible for their actions.

The staff is also considering whether the use of audible-alarm dosimeters would improve radiography safety. Battelle Pacific Northwest Laboratories is conducting tests of such dosimeters to determine their reliability.

Gamma Irradiators

A rule change that became effective in March 1978 established new requirements in 10 CFR Part 20 both to improve safety in the use of sealed radioactive sources that produce very high intensities of radiation and to reduce the probability of accidental exposures of workers to such sources. The rule applies to those high-intensity radiation sources used in devices (called irradiators) to irradiate materials for various purposes (e.g., sterilization of medical products studies of radiation effects on materials, polymerization of plastics).

The intense radiation from an irradiator source could be immediately lethal to people who might accidentally be exposed to it. The new rule requires automatically functioning entry and warning controls (lockout, shutdown, and signaling devices) as well as procedural control to reduce the likelihood of exposures.

Effecting Occupational ALARA

The NRC staff has developed proposed amendments to its regulations that would strengthen the implementation of the "as low as is reasonably achievable" (ALARA) concept in the control of occupational exposures to radiation and radioactive materials in licensed activities. These amendments would require certain licensees to develop and implement individual programs for maintaining occupational radiation exposures ALARA. Each program would then become part of the licensee's mandatory health protection program, subject to inspection and enforcement. The rule changes would be applicable to all licensees who are required by the NRC to perform personnel dosimetry, air sampling, or bioassays for worker protection.

In addition to these amendments, the NRC is proposing to eliminate the use of the 5(N–18) (where, for workers over 18, N is the worker's age in years) dose limit formula that permits workers to receive radiation exposures as high as 12 rems per year under certain conditions. Instead, an annual dose limit of 5 rems would be established and would be accompanied by a quarterly dose limit of 3 rems. The present NRC occupational dose limits are 1.25 rems per quarter if the worker's exposure history is unknown. If the exposure history is known, the limit is 3 rems per quarter, provided the lifetime accumulated dose does not exceed 5(N-18) rems.

Personnel Monitoring Reports

The NRC amended its regulations to extend to all licensees the requirements for annual statistical summary reports on workers' radiation exposures. Under the previous regulation, only four categories of licensees were required to submit an annual statistical summary of monitored whole-body exposures, i.e., the number of people in each of 18 prescribed ranges of radiation exposure.

The amendment to Part 20 extends this statistical summary reporting requirement to all NRC specific licensees for a period of two years. After evaluating the data for 1978 and 1979, the NRC will consider whether or not to extend or modify the reporting requirement. The four categories of licensees previously covered will continue to be required to report in any event. The amendment does not affect existing requirements for the provision and use of personnel monitoring equipment or for the records of personnel monitoring data that must be kept, but relates solely to the reporting of data already recorded.

The rule change was originally proposed in May 1975; however, in an effort to determine the cost to licensees and to obtain data for one year for evaluation, the NRC requested the voluntary submission of reports for 1975. The personnel monitoring and cost data that were collected for the 1975 period were published in March 1978 in NUREG-0419, "Occupational Radiation Exposure at NRC-Licensed Facilities — 1975."

Testing for Personnel Dosimetry

Evaluations of the degree of accuracy that is provided by personnel dosimetry processors in the United States indicate that improved performance of some processors is needed. Personnel dosimetry devices are used to measure the radiation dose received by workers in NRC-licensed facilities. To obtain more accurate processing of dosimeters, the NRC staff is considering a requirement that personnel dosimetry results be accepted only from a processor who has successfully passed certain prescribed accuracy tests. The test criteria would be adapted from a consensus standard being developed by ANSI.

In preparation for the new regulation, the NRC is funding a two-year pilot study being conducted by the University of Michigan. The objectives of the pilot study are:

(1) To provide processors an opportunity to correct any process problems that they may have prior to publication of the new regulation in effective form.

(2) To test the consensus standard for practicality as well as for degree of difficulty.

(3) To develop a detailed procedures manual for use by future testing laboratories.

By the end of 1978, the study was more than 50 percent completed. Fifty-seven dosimetry processors are participating. Early results indicate that, while some participants are performing with acceptable accuracy, considerable improvements will be required on the part of others.

NATIONAL STANDARDS PROGRAM

The national standards program is conducted under the aegis of 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.

The actual drafting of standards is done by experts, most of whom are members of the pertinent technical and professional societies. Approximately 230 NRC staff members serve on working groups organized by technical and professional societies. These societies are listed in the accompanying table. National standards are used in the regulatory process only after independent review for suitability by the NRC staff and after public comments on their intended use have been solicited and considered.

IAEA REACTOR SAFETY STANDARDS

NRC has continued its lead role in organizing and carrying out U.S. participation in the IAEA program to develop safety codes of practice and safety guides for nuclear power plants. The NRC coordinates U.S. technical activities associated with this program. The codes and guides will provide a basis for national regulation by developing countries of the design, construction, and operation of nuclear power plants. NRC staff members continued to represent the United States on the IAEA Senior Advisory Group (SAG) that oversees the program and on the Technical Review Committees working in the five areas of primary interest: governmental organization, siting, design, operation, and quality assurance. Dr. J. M. Hendrie, Chairman of the NRC, is the U.S. member of the SAG and has served in this capacity since the inception of the program in late 1974.

During 1978, the Senior Advisory Group, Technical Review Committees, and working groups under them drafted nine new guides and completed five safety guides that were forwarded to the Director General of the IAEA with the recommendation that they be issued. About 40 of the approximately 50 safety guides planned to date have been drafted and are undergoing review. During the drafting process, the NRC standards staff coordinated the reviews within the U.S., soliciting comments from interested members of the public, industry, and other government agencies.

SOCIETIES SPONSORING NUCLEAR STANDARDS DEVELOPMENT ACTIVITIES IN WHICH NRC STAFF MEMBERS PARTICIPATE

American Association of Physicists in Medicine
American Concrete Institute
American Conference of Governmental Industrial Hygienists
American Institute of Chemical Engineers
American Institute of Steel Construction
American Insurance Association
American National Standards Institute
American Nuclear Society
American Society of Civil Engineers
American Society of Mechanical Engineers
American Society for Nondestructive Testing
American Society for Testing and Materials
American Welding Society
Health Physics Society
Institute of Electrical and Electronics Engineers
Institute of Nuclear Materials Management
Instrument Society of America
Metals Properties Council
National Council on Radiation Protection and Measurements
National Fire Protection Association
National Sanitation Foundation
Society of Naval Architects and Marine Engineers
Welding Research Council

Regulatory Research

NRC's regulatory research program expanded considerably during fiscal year 1978 in both scope and productivity. Under a provision of the 1978 Appropriations Authorization Act (P.L. 95-209), the Commission's Office of Nuclear Regulatory Research undertook to move its reactor research efforts a step beyond the prior confines of "confirmatory research" as stipulated in the Energy Reorganization Act of 1974. As mandated, an initial long-term plan was developed in the form of a report to the Congress (NUREG-0438) in April 1978, dealing with development of new or improved safety systems for nuclear power plants. The research described in the report was carefully defined so as to ensure that its purpose is the improvement of reactor safety and not the enhancement of the economic attractiveness of nuclear power as opposed to alternative energy sources. A required annual update of that report is contained in the final section of this chapter.

In 1978 NRC's research program was broadened by the development of plans for cooperative efforts with Germany and Japan, and the initiation of negotiations for formal agreements on the use of facilities in those countries, together with advanced U.S. computer codes and instrumentation. (International research exchanges are discussed in Chapter 9.)

Benefits deriving from 1977 agreements between NRC and DOE, dealing with the coordination and management of research facilities and projects, began to emerge during 1978. Advanced reactor safety research programs have been carefully coordinated in continuing discussions to ensure that DOE programs, aimed principally at accident prevention, and NRC programs, which primarily address the course of events if accidents occur, are not duplicative. Similar coordination in safeguards research assures that DOE responsibility for developing cost-effective safeguards systems is kept clearly separate from NRC's responsibility to create models for evaluating the effectiveness of those systems. In the fields of fuel cycle safety research and research on waste management, transportation, the environment, and health, liaison between NRC and DOE technical staffs and exchanges of research plans and other information also are aimed at preventing overlap and effecting economics.

A report by the Advisory Committee on Reactor Safeguards (NUREG-0392), which reviewed and evaluated NRC's research program as part of an annual requirement set forth in PL-95-209, supported the need for NRC research on improved safety concepts, and made additional recommendations to guide the research program.

Research highlights in 1978 included (1) the initiation of a new program to evaluate safety margins in seismic design methodology for power reactors; (2) a major step forward in water reactor safety research by bringing the Loss-of-Fluid-Test (LOFT) facility in Idaho to full power; (3) the availability of production versions of major systems, component and containment codes; (4) development of the technical bases for NRC certification of plutonium air-shipment containers; (5) the first integral systems test in the full-length MOD 3 Semiscale facility; (6) initial operation of the newly upgraded Annular Core Research Reactor (ACRR) (formerly the Annular Core Pulse Reactor) at its upgraded design power; and (7) completion of a program to resolve issues raised by ACRS regarding pressure vessel loading. Additionally, in the risk assessment area, a seven-member independent review group issued a report to the Commission (NUREG/CR-0400) on its year-long evaluation of the Reactor Safety Study (WASH-1400).

These and other research activities and experiments conducted by the NRC are discussed in this chapter under the following sub-program headings:

- *Water Reactor Safety Research* — to provide additional or independent information regarding margins of safety used, or recommended to be used, in licensing actions on current families of light water reactors.
- *Advanced Reactor Safety Research* — to develop an independent NRC capability, through use of a family of safety codes, to assess the safety of advanced reactor concepts (i.e., breeder reactors, gas-cooled reactors, etc.).
- *General Reactor Safety Research* — three research activities not specific to other specialized programs or cutting across two or more programs: Site Safety Research, Mechanical Engineering Research, and Structural Engineering Research.
- *Fuel Cycle, Environmental and Waste Management Research* — to produce computer models which address or confirm basic data on such varied fields as fuel facility operations, transportation of radioactive materials, routine reactor operations, and disposal of radioactive wastes.
- *Safeguards Research* — to develop data for the assessment of alternative policy options as well as strategies and procedures dealing with safeguards regulation, and for the evaluation of safeguards proposals from applicants or licensees.
- *Risk Assessment Research* — to develop and improve risk assessment methodology for application in regulatory decision-making.
- *Improvement of Reactor Safety* — to plan for the development of new or improved safety systems for nuclear power plants.

RESEARCH OBJECTIVES

The NRC research program provides technical information, independent of the nuclear industry, to define with greater precision the safety margins provided in nuclear facilities. The broad objectives of the program are:

- To provide objectively evaluated safety data and analytical methods that serve the needs of regulatory activities.
- To provide better quantified estimates of the margins of safety for reactor systems, fuel cycle facilities, and transportation systems.
- To establish a broad and coherent exchange of safety research information with other government agencies, with industry, and with foreign governments and organizations.

Safety research is largely directed toward defining precisely the safety margins imposed in the licensing process. In general, these divide into two types: (1) engineering safety margins which allow for both normal and abnormal variations in operating parameters plus an ample degree of conservatism, and (2) the additional safety margins which allow for lack of knowledge of accident processes due to lack of accident experience.

The former type of research encompasses the common sense protections which are specified in codes and regulations imposed by state and Federal governments, including those of the NRC. The NRC research program works to provide additional definition to both types of margins but places heavy concentration on the latter type.

Water Reactor Safety Research

The NRC's confirmatory safety research program on light-water reactors (LWRs) can be generally divided into five principal categories:

• Systems Engineering	Thermal-hydraulic tests of postulated accidents* and the effectiveness of engineered safety features
• Fuel Behavior	Fuel-rod behavior in postulated accidents and associated failure limits
• Computer Code Development	Computer code development for accurately predicting the consequences of postulated reactor accidents
• Metallurgy and Materials	Safety design and protection of integrity of reactor pressure vessels and piping
• Research Support	Operational safety aspects of nuclear power-plant operation

Principal achievements in water reactor safety research during 1978 under each of these programs are given in the sections that follow. Detailed plans for each program are provided in a report titled "Water Reactor Safety Research Program - A Description of Current and Planned Research" (NUREG-0006).

*The term "postulated accident" is used here to describe a range of "design-basis" accidents (DBAs) which license applicants must consider in the design of a plant.

SYSTEMS ENGINEERING

The systems engineering program employs experiments to provide measured physical data for the development and assessment of computer codes for analyzing the performance of emergency core cooling systems (ECCS), which are provided to keep the nuclear fuel cooled in the event of a loss-of-coolant accident (LOCA).

A LOCA can be divided into several phases: blowdown (depressurization), refill (ECC water entering the lower plenum), and reflood (ECC water entering the core). An understanding of thermal-hydraulic behavior in all these phases is necessary to assess LOCA/ECCS behavior. In systems engineering specific attention is focused on determining (a) heat transfer phenomena (critical heat flux-CHF-in pressurized water reactors and boiling transition-BT-in boiling water reactors) during blowdown and (b) the subsequent heat transfer after CHF/BT and during reflood. (CHF or BT is the condition at which there is a marked decrease in the heat transfer between fuel-rod cladding and cooling water, and this raises the fuel-rod surface temperature.) If the surface temperature of the fuel rod becomes too high, the cladding may be damaged to the extent that a release of radioactive material could occur. Systems engineering experiments (which address thermal-hydraulic behavior, heat transfer, fluid-flow, pressure phenomena, etc.), during a LOCA, are of two major types: Integral Systems Tests and Separate-Effects Experiments.

Integral Systems Tests

Loss of Fluid Test (LOFT) Program. The LOFT project investigates the integral thermal-hydraulic and nuclear fuel behavior aspects of LOCAs to permit the validation of analytical models developed for reactor safety analysis and the evaluation of ECC systems.

Fiscal Year 1978 marked the most significant year to date for LOFT. The reactor achieved initial criticality in March, completed the L1-5 LOCA experiment in April, and went through power range testing, achieving full power in October. Many new systems, most notably the secondary system which removes heat from the reactor coolant system, were completed and satisfactorily tested. Substantial improvements were made in existing experimental instrumentation, and new instrument concepts were developed for later installation in the facility. Analysis of the LOFT nonnuclear experiments culminated in a Research Information Letter (RIL #37) summarizing the results, which will aid in the improvement of LOCA analytical models. Foreign participation continued with scientists from Austria, Germany, Japan, the Netherlands and Scandinavia working on LOFT in Idaho. The LOFT program leading to the first nuclear experiment was completed two months ahead of the schedule established in 1976.

NRC's LOFT PROGRAM

Unique among its research projects, NRC's Loss of Fluid Test (LOFT) program features a complete pressurized water reactor (PWR) designed to operate at a power level of 50 MWt. LOFT was developed to provide experimental information pertinent to licensing criteria for large commercial PWR's. A major portion of the program is aimed at improving technical understanding of the loss-of-coolant accident (LOCA) and the performance of emergency core cooling systems.

Fiscal Year 1978 marked the completion of preparatory non-nuclear LOFT experiments, and final preparation for the first in a series of about 20 nuclear tests in facility. That first experiment in the test reactor was conducted successfully on December 9, 1978. It permitted, for the first time, the direct measurement of fuel temperature in a reactor during a simulated loss-of-coolant accident and, thus, a comparison of predicted temperatures with measured temperatures.

The experiment began with the rapid opening (18/1000 of one second) of blowdown valves to simulate the instant shearing of a major coolant pipe. Steam and water were rapidly discharged through the break to a suppression tank where the steam was condensed. The experiment was conducted at a power level about l/120th that of a commercial power reactor, yet core power density was nearly two-thirds that of a commercial reactor.

Preliminary evaluation of test results indicate that the emergency core cooling system functioned as expected, and that measured temperatures of fuel cladding were significantly lower than predicted peak temperatures.

Austrian, Dutch, Finnish, German, and Japanese scientists, who observed the December 9 experiment, will assist their U.S. counterparts in the detailed analysis of test data. Test results also will be useful in analyzing the adequacy and accuracy of computer codes used by the NRC to evaluate power plant safety.

LOFT nuclear experiments in 1979 will feature higher power levels and densities and a variety of pipe break sizes and locations and with alternate emergency cooling systems. Tests are expected to continue into the 1980's.

Semiscale Program. During the first half of fiscal year 1978, the Semiscale facility was converted to a two-loop system (Mod-3) which includes such improvements as a full-length (3.66 meter) core and a complete, active broken loop, with pump, piping, and steam generator scaled primarily to PWR counterparts. (These and other improvements over the earlier Mod-1 were described and diagrammed on page 150 of last year's Annual Report.)

Six tests, divided into three groups, were conducted to establish the baseline performance of the Mod-3 system in evaluating different phases of LOCA transients. The first group consisted of three blowdown tests, the second investigated Mod-3 behavior during core reflood, and the third established the characteristics of integral blowdown/core reflood response to emergency cooling water injection. The next Semiscale investigation will address upper head injection, and follow-on tests of alternate ECC injection systems are planned for the future.

Other highlights of the Semiscale Program during 1978 included:

(1) The completion of tests in Karlsruhe, Germany to determine error bands associated with certain Semiscale measurements and to establish criteria for the design of improved Semiscale instrumentation.

(2) The design and fabrication of a prototype probe which will aid in characterizing flow regimes, and development of other state-of-the-art instrumentation to measure flow, density and liquid levels.

(3) Analysis of Mod-1 steam generator tube rupture tests including publication of a final report. A RIL will be published early in 1979.

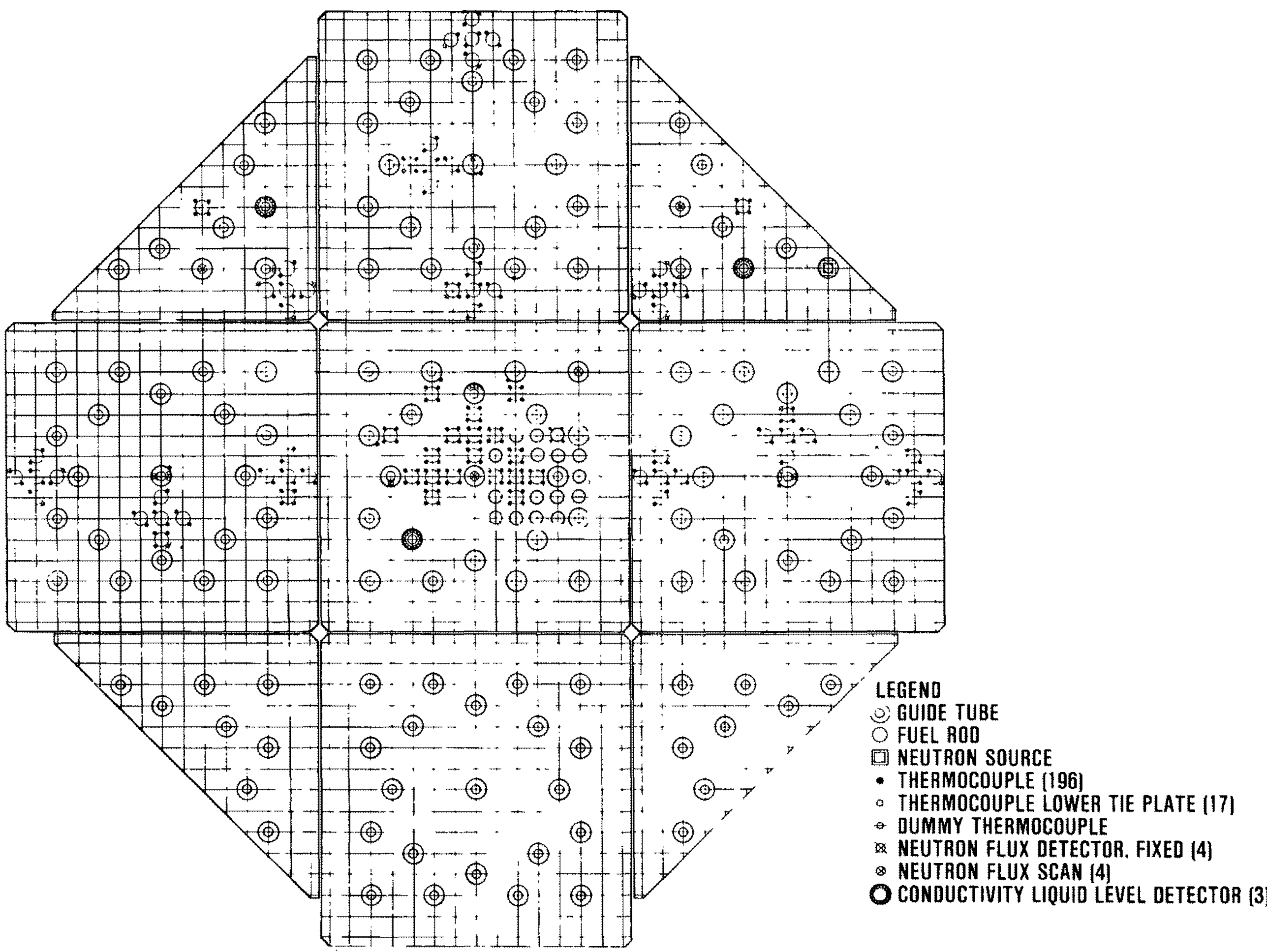
(4) An exchange of research staff personnel was accomplished between NRC's Semiscale project and the Italian LOBI (Loop for Blowdown Investigation) project at the Joint Research Center of the Communion of European Communities (CEC) in Ispra, Italy. The LOBI and Semiscale facilities are similar enough to permit data comparisons, and the resulting data can then be incorporated into a Semiscale-to-LOFT-to-PWR scaling study scheduled for completion in 1979.

Separate Effects Experiments

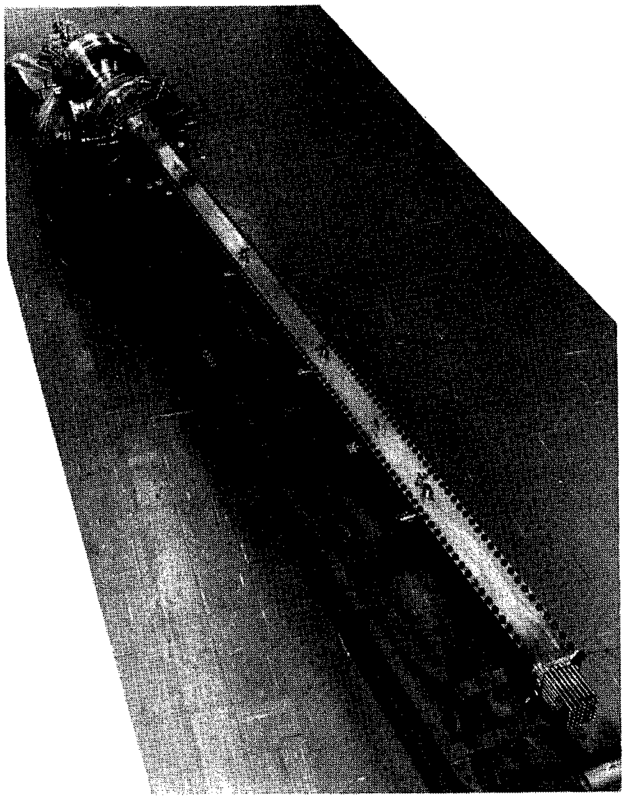
PWR Blowdown Heat Transfer Program. The PWR Blowdown Heat Transfer Program at Oak Ridge National Laboratory (ORNL) is an experimental separate effects study of transient two-phase flow and heat transfer in rod bundles. Results are obtained primarily from the Thermal Hydraulic Test Facility (THTF). (See pp. 151 and 152, 1977 NRC Annual Report)

During 1978, testing continued with THTF Bundle No. 1 (a 49-rod bundle of full-scale electric fuel rod simulators). Bundle No. 2, which is similar to Bundle No. 1, was fabricated, assembled, and placed on standby. Thirty-two tests have been conducted with Bundle No. 1. In general, analytical code predictions of surface temperatures in the lower half of the test bundles were in agreement with experimental results. However, problems involving predictions related to the upper half of the bundle, including a premature prediction of time to critical heat flux, were being studied at year's end.

Two-Loop Test Apparatus. For a description of this apparatus, refer to page 152, 1977 Annual Report. During 1978, a series of blowdown tests without ECC injection were completed, using an 8 x 8 electrically simulated BWR fuel rod bundle. Some integral blowdown tests with ECC



The heart of the Loss-of-Fluid Test Facility at the Idaho National Engineering Laboratory is its nuclear core, made up of more than 1300 fuel rods, assembled in nine bundles. Five fuel Bundles are 15 x 15 rod arrays, with guide tubes filling 21 of the locations to provide structural rigidity and to guide control rods in the tubes to control reactor power. The fuel rods are made up of 0.422 inch diameter zircaloy tubes filled with a 5 1/2 foot long stack of uranium dioxide pellets 0.600 inch long and 0.366 inch in diameter. The uranium is enriched to 4% U^{235}. The fuel rods and guide tubes are held in place by two end plates and five "spacer grids." The fuel bundle is attached to an upper structure providing pressure penetration for instruments. Four bundles are attached to control rod drive mechanisms. The core is heavily instrumented, as shown.



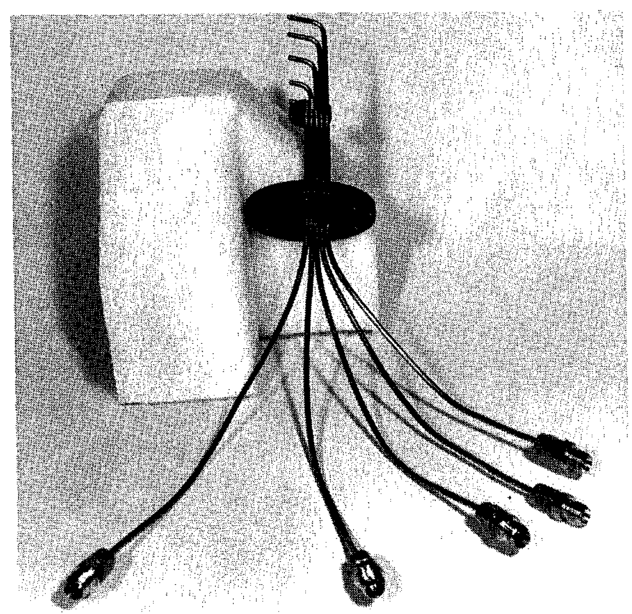
"Bundle 2"—This 49-rod bundle of full-scale (12-foot long) electrically heated fuel rod simulators was fabricated for installation in the Thermal Hydraulic Test Facility at Oak Ridge. A rectangular shroud box holds the 7 x 7 electrical rod array (seen protruding at lower right) which simulates the 15 x 15 nuclear rod configuration of a full-size PWR fuel assembly.

injection also were completed and these showed that ECC injection systems are effective during a design-basis LOCA as simulated in the apparatus.

Steam-Water Mixing and System Hydrodynamics Program. Analytical and experimental studies of steam-water mixing effects on the penetration of cooling water in PWR vessels continued in 1978 at Battelle-Columbus Laboratories (BCL) in Ohio, with analysis concentrated on the relationship between coolant penetration and facility size. A preliminary scaling formula was used to predict experimental penetration results in the 1/15- and 2/15-scale models of a PWR (see pp. 152 and 153, 1977 Annual Report). Two different techniques were being developed for predicting the filling of the lower plenum as a result of coolant penetration. The simpler of the two predicts overall filling rates reasonably well. The more complicated model will attempt to detail the discontinuous plenum filling process.

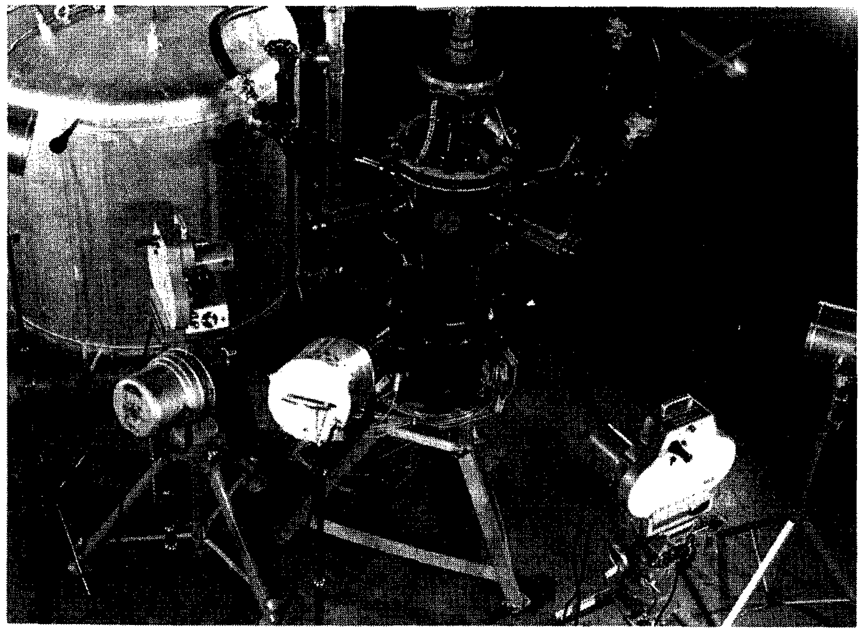
At Creare Incorporated in New Hampshire, experiments were conducted to study the timing and rate of lower plenum refilling. The primary thrust has been the development of a separate effects model which includes counter-current flow, superheated walls, lower plenum voiding, cold leg effects, and break effects, and which can analyze both steady and transient conditions.

FLECHT-SEASET Program. The Full Length Emergency-Cooling Heat-Transfer Separate Effects and Systems Effects Tests (FLECHT-SEASET) is an extension of the joint NRC — Electric Power Research Institute (EPRI) — Westinghouse FLECHT program (See p. 153, 1977 Annual Report) to study certain effects occurring in the reflooding phase of a PWR LOCA. The program now includes a modified FLECHT facility and four new facilities in which two major research tasks are addressed: (1) the study of heat transfer and liquid-steam flow effects in rod bundles, with and without flow blockage, and (2) the study of those same



The instrument pictured above is a pitot tube "rake" which measures pressures in such a way that relative velocities of ECC fluids can be determined. Pressure transducers are attached to the threaded fittings, and the multiple tips (hence the name "rake") permit fluid velocities to be determined at several points in the ECC flow stream. This instrument, which was developed for the LOFT facility, has proven an accurate and versatile flow-measurement technique.

This 1/15-scale transparent apparatus was developed at Battelle Columbus Laboratories in Ohio to facilitate the study of accident phenomena using high-speed motion picture photography of emergency cooling water behavior during a simulated PWR loss-of-coolant accident. Cameras and lights in the foreground are focused on a 12-inch-diameter vessel model to film a "pipe-break" accident sequence in which steam entering the vessel from the top (flanged pipe), will mix with "coolant" water entering through the three intact, lateral inlet pipes, and steam/water will be expelled through the simulated broken pipe (left rear of vessel), into the containment vesesl at upper left.



effects in a scaled representation of a PWR, with and without alternate emergency core cooling injection.

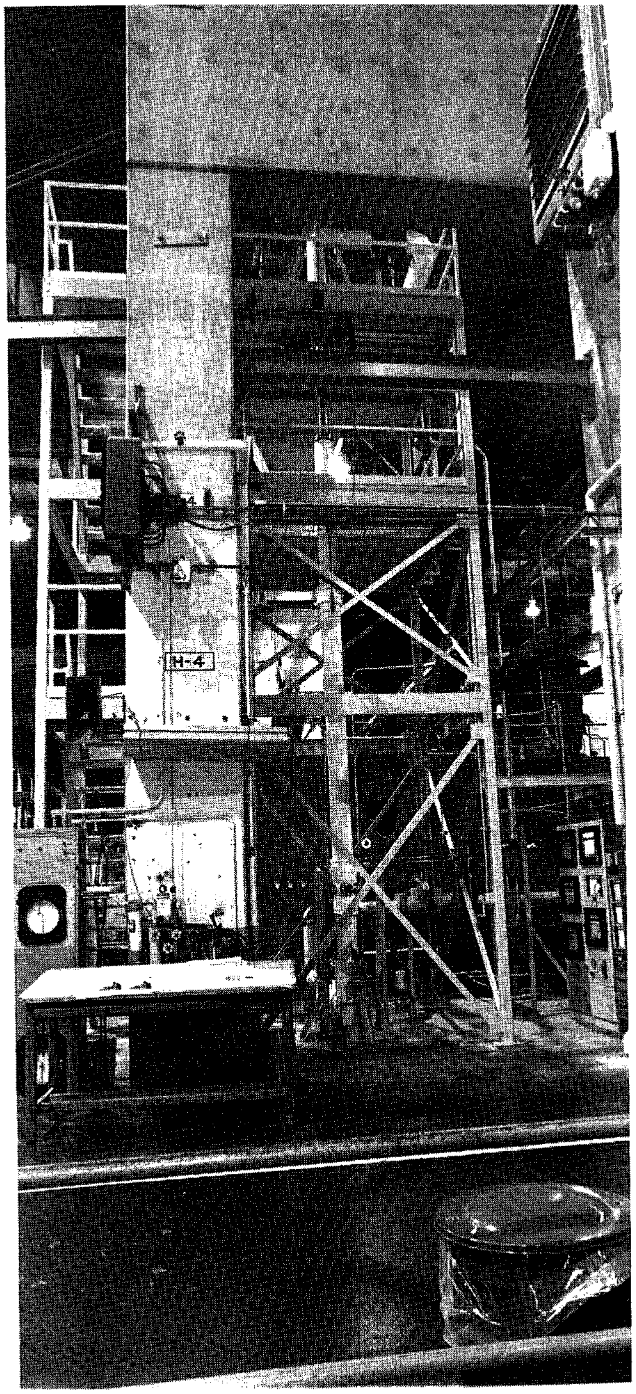
Progress during the year on the flow blockage task included completing the test facility itself, the beginning of shakedown testing, and, at the end of the fiscal year, the issuance of a task plan. Progress on the second task included completion of the test facility (a steam-generator and a separate-effects test apparatus, reinstrumented to provide more accurate heat transfer information) and the issuance, in August 1978, of a task plan.

Instrumentation Development. NRC's LOFT project and tri-national program called "3D" require particularly sophisticated instrumentation for the interpretation of two-phase flow phenonmena, and a program was initiated at ORNL to help meet this need. The program, entitled "Advanced Instrumentation for Reflood Studies (AIRS)," was aimed, in part, at producing measurement systems for use in reflood experiments under the 3D Program. (This program is described in the Research Support section, later in this chapter.) Two types of instrumentation are being developed: the impedance probe, which measures the velocity and volume of vapor in the vapor-fluid mixtures in the core and upper plenum areas of these test facilities; and the film probe, which measures the thickness and velocity of liquid films on the internals of the reflood facilities.

ORNL's impedance probe progressed during 1978 to the point where designs were sufficiently validated to permit their inclusion in the PKL (Primarkreislauf) Core II Facility in West Germany. Film probes, using techniques developed at Lehigh University, also were adapted for the severe environment of the PKL facility, and a steam-water test stand was designed, constructed, and placed in operation at ORNL during the year to verify and calibrate both instrument systems prior to their shipment abroad. Other instrumentation for the 3D program, developed by EG&G in Idaho, includes liquid level detectors, drag disks, turbine meters and thermometry and optical devices. An initial shipment of liquid level detectors for the 3D facility in Japan was made during the year.

A variety of other advanced instruments were developed, or under development, at several laboratories across the United States. Brookhaven National Laboratory, as well as Rensselaer Polytechnic Institute and the State University of New York at Stony Brook, all were working on instrumentation to measure void-fractions and droplet populations and velocities of two-phase (steam/water) flow phenomena. A neutron pulse generator under development at Sandia Laboratories appeared to promise both greater accuracy in two-phase flow measurements and the capability to calibrate other instruments, as well. At the end of the fiscal year, Argonne National Laboratory was developing techniques for its use.

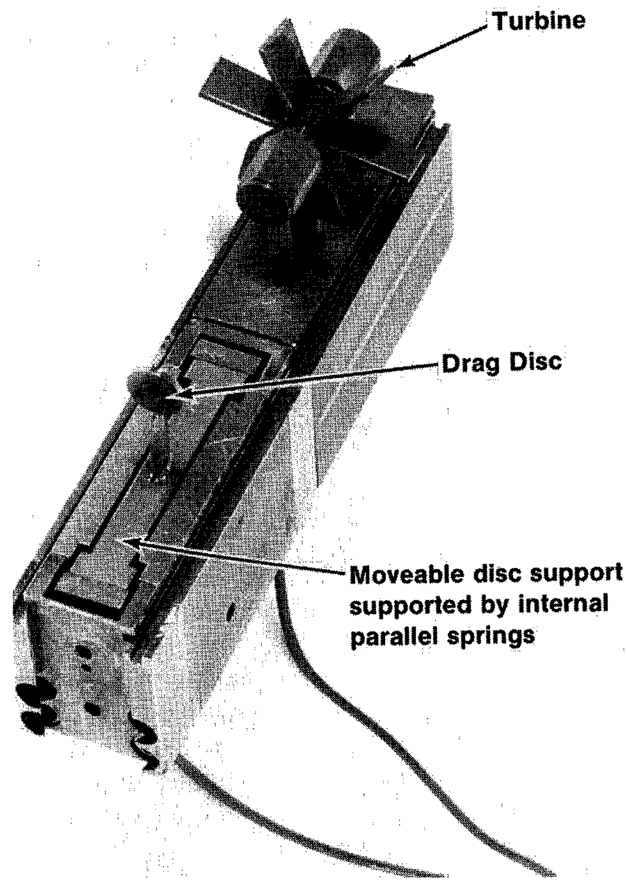
In a separate activity, Creare developed an instrumentation system to record and display the



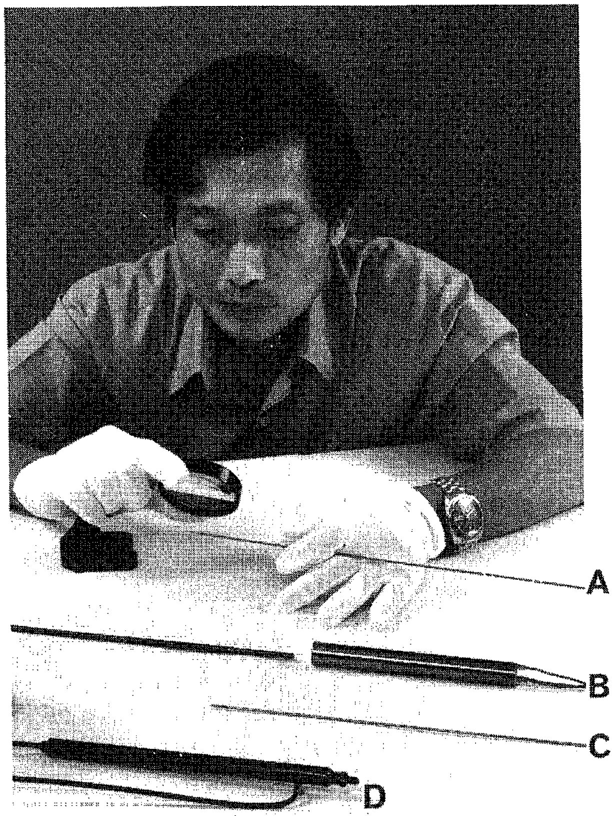
This Steam-Water Test Stand generates a range of steam-water flow conditions for testing advanced in-vessel instrumentation. It is located at Oak Ridge and is used in NRC's Advanced Instrumentation for Reflood Studies Program. The overall height of the stand is about 9 meters (30 feet). The instrumented test section, which measures 19 feet in length and 3 1/2 inches in diameter, is shown on the right of the post marked "H-4." Viewports can be seen at the bottom and a removable section containing instruments is at the top. Adjusting the flow rates of incoming steam and water permits the evaluation of different types of two-phase flow measuring instruments.

two-phase flow phenomena occurring in a reactor. "Snapshots" of steam/water distribution can be taken at about 100 frames per second to produce an array of about 300 conductivity sensors. These can be displayed on hard copy, compiled as movies, or manipulated numerically by computer for quantitative analysis. Resulting data are expected to be useful in developing advanced codes.

The Hanford (Washington) Engineering Development Laboratory continued its development of advanced fuel rod instrumentation for the LOFT reactor, with emphasis on sensors to measure fuel rod axial motion, centerline temperature, internal gas pressure and temperature during a LOCA. In 1978 a set of qualification test sensors was fabricated which, if successful in performance tests, will lead to procurement of commercial production units for installation in LOFT.



The modular drag-disc turbine transducer shown above is an improved version of the LOFT drag-disc turbine used in the LOFT nonnuclear test series. The parallelogram spring arrangement for drag disc support improves the reliability and accuracy of the device, and its modular construction improves maintainability.



Special instruments have been developed at the Hanford Engineering Development Laboratory, in Washington, to measure fuel rod characteristics during loss of coolant accidents simulated in the LOFT reactor. The 4-foot long fuel centerline thermocouple (shown at A) is designed to operate at temperatures in excess of 2200°F. Fuel rod length changes will be measured by a small-diameter displacement sensor (B). Fuel-rod internal gas temperature and pressure will be monitored by a fast thermocouple (C) and pressure sensor (D). A special test pressure vessel that simulates the fuel rod during tests in autoclaves, surrounds the 0.35-inch diameter pressure sensor in this photograph.

FUEL BEHAVIOR

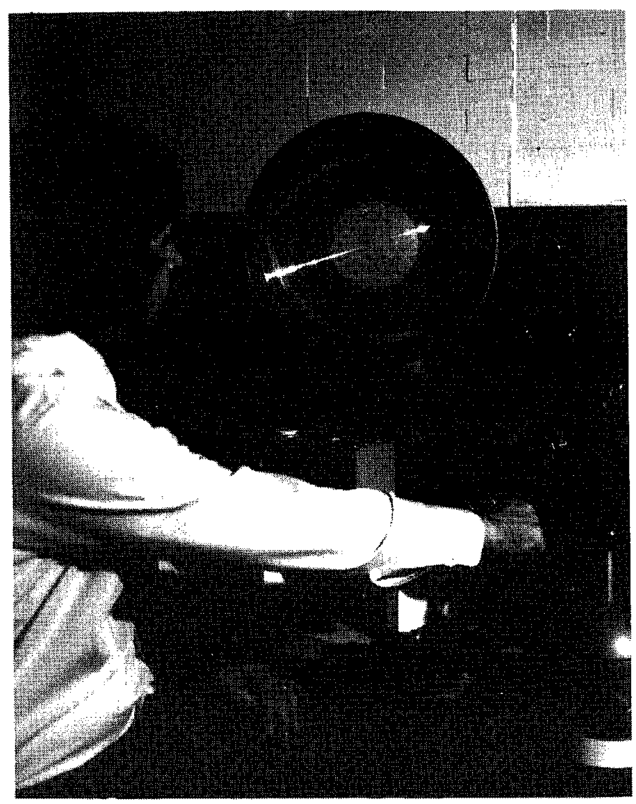
NRC's fuel behavior research program provides experimental data needed for the independent assessment of fuel behavior during accidents. Research in 1978 included cladding experiments, in-reactor testing, fuel meltdown and fission product transport testing, and fuel code development.

Cladding Experiments

Zircaloy Deformation Experiments. Three 16-rod bundles of simulated PWR fuel elements were tested at ORNL to determine how Zircaloy fuel element cladding deforms in the post-blowdown part of a LOCA and how much this deformation restricts the flow of emergency core cooling water. Two of the bundles have been examined and the examination of the third has just begun. In these bundles, bulges and bursts occurred almost randomly over the test sections. Inspection of the bundles and analysis of the test data were continuing at the end of the period.

Irradiated Zircaloy Ductility Experiments. Measurements continued at Battelle Columbus Laboratories which permit comparisons of unirradiated Zircaloy cladding with that which has been irradiated in a reactor. Tests in 1978 showed that at 1400°F (a predicted temperature in an accident) changes in cladding properties due to irradiation disappear and the cladding behaves as through no irradiation had occurred.

Zircaloy Embrittlement Experiments. In 1978, Argonne National Laboratory experiments to study changes in Zircaloy properties when it



In this photo an examination technique called "Hot Cell Metallography" is shown in use on an Irradiated Zircaloy Tube Burst Specimen at Battelle Columbus Laboratories. The specimen, located behind an irradiation shield, is examined remotely at approximately 16X magnification via a multi-angled optical path. On the viewscreen, the burst tube specimen is shown as the outer broken band.

reacts with oxygen at high temperatures led to development of quantitative measurements of cladding embrittlement when it reacts with steam and fuel at temperatures predicted for reactor accidents. Only qualitative criteria had been used previously.

Cladding Creepdown Studies. An NRC cooperative international program, undertaken in 1978 with the Netherlands Nuclear Research Foundation (ECN), involves the design and fabrication of experimental equipment at Oak Ridge National Laboratory and its irradiation in the ECN's Petten Reactor in Holland. Over a long period of time under the high pressure conditions in a PWR, the cladding "creeps" down on the fuel. This closes the fuel-cladding gap which affects heat transfer and causes changes in the fuel geometry. These changes, in turn, make it difficult to predict how fuel will behave in accident situations. The ORNL/Petten experiment permits researchers to measure the rate of creepdown while it is occurring in a reactor, and results are expected to significantly improve predictions of the accident behavior of fuel which has been exposed to creepdown conditions.

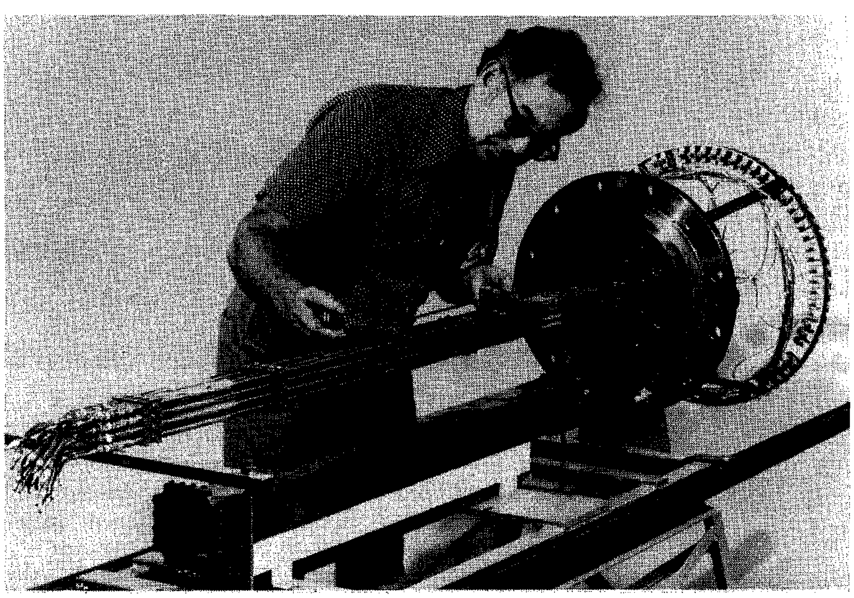
Mechanical Properties. In 1978 the University of Florida developed unique experimental techniques, based on an optical sensing system and closed loop controller, to measure the deformation of cladding materials under constant true strain rate. Conventional testing methods had failed to produce data under controlled strain rate conditions. The importance of such control was demonstrated in tests on "1008" steel and "304" stainless steel, and work is proceeding on extending the system to characterize Zircaloy.

In-Reactor Testing

Power Burst Facility Tests. The Power Burst Facility (PBF) at INEL is a principal tool for determining the behavior of fuel rods under various operating and accident conditions. (See description on page 154, 1977 Annual Report).

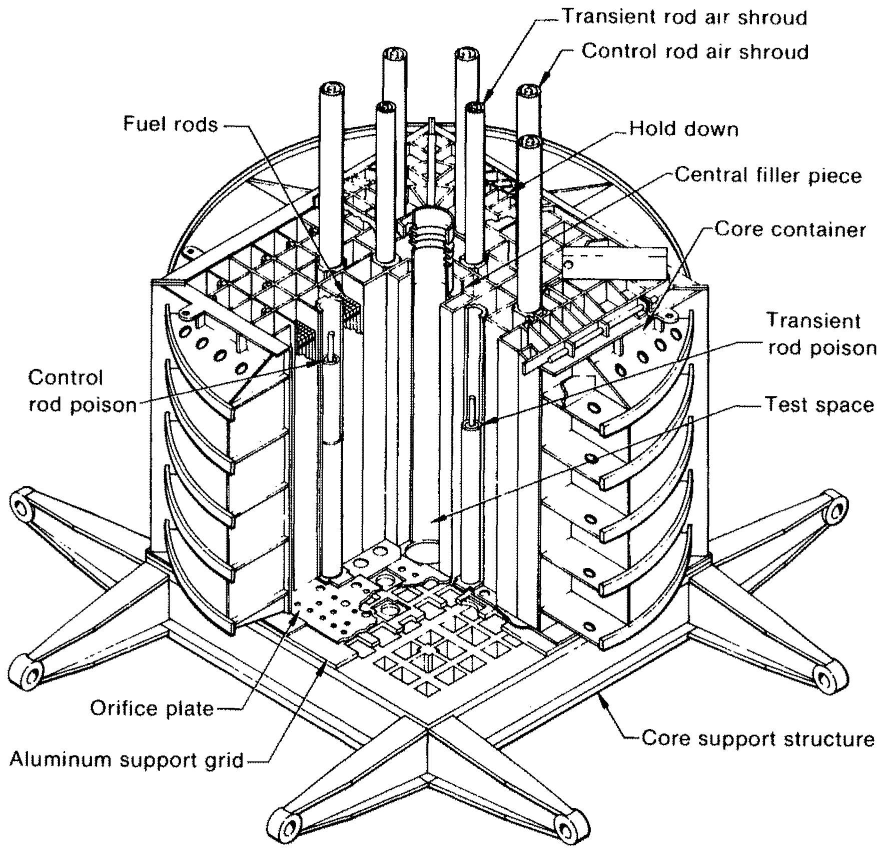
Near the end of fiscal year 1977, two major modifications had increased the overall utility and efficiency of the PBF. In 1978, fuel behavior testing was resumed with an experiment comprising three nuclear blowdown tests, followed by two power-cooling-mismatch tests (PCM-1 and PCM-5) and a Reactivity Initiated Accident (RIA) Scoping Test. Prior to the RIA test, 40 tests were performed to determine the characteristics of the PBF core and to qualify it for the experiment.

The nuclear blowdown experiment was the first ever to be performed at hot PWR coolant temperatures. Calculations of coolant behavior agreed reasonably well with the measured coolant behavior. Calculated cladding surface temperatures were slightly greater than measured temperatures; however, the fuel rods did not fail and exhibited less damage or loss of function than anticipated.



This bundle of 16 test rods has been exposed to conditions simulating the post-blowdown phase of a PWR loss of coolant accident. An Oak Ridge researcher examines the Zircaloy cladding for bulges, bursts and other deformations as part of continuing NRC-sponsored cladding studies at ORNL. The Zircaloy tubes contain electric heating elements which permit the simulation of cladding temperatures under LOCA-like conditions.

Shown at right is a cutaway view of the core of the Power Burst Facility (PBF) in Idaho. The PBF is a principal tool in NRC's program to study the behaviro of nuclear fuel rods during both normal and abnormal conditions. Small clusters of instrumented fuel rods are installed within a thick-walled tube, then that assembly is inserted in the test space (shown at center of diagram) for testing. The core can be operated in a variety of modes to permit the simulation of hypothesized loss-of-coolant, power-cooling-mismatch, inlet-flow-blockage or reactivity-initiated accident conditions. The core is 1.3m (52 in.) in diameter and 0.9m (36 in.) high, enclosing a vertical test space 0.21m (8.25 in.) in diameter. PBF fuel consists of stainless steel rods containing ceramic fuel pellets. The rods are grouped in fuel canisters containing from 28 to 63 fuel rods, which form a close-packed, square fuel lattice. The reactor is controlled with eight control rods and four transient rods. The transient rods are used to initiate and control natural and shaped power bursts, whereas steady state operation is maintained with only the control rods.



The PCM-1 test showed that fuel and cladding exposed to very high power and very poor cooling conditions maintained its integrity longer than expected. PCM-5, a test with 9 rods, showed a general equivalence of data from earlier single rod and cluster PCM tests.

The RIA test series will determine the behavior of fuel rods under special conditions—those caused by the ejection of a control rod from a power reactor core, for example. The first RIA scoping experiment comprised five single-rod tests. Three were low-energy tests to examine the fuel rod failure threshold and to evaluate calorimetry techniques. The other two tests were performed at high energy inputs, and the results showed very little conversion of thermal or chemical energy to the kinds of mechanical energy which could damage core components.

Halden Program. As noted in Chapter 9, "International Activities," NRC participates in the multinational OECD Halden nuclear fuel performance project in Norway. Results of NRC-sponsored activities under this project in 1978 included post-irradiation examination and the issuance of a final report on mixed oxide (UO_2-PuO_2) fuel rod experiments in the Halden reactor. Substantial data on the thermal and mechanical response of fuel rods to long periods of irradiation were produced. In another NRC/INEL-sponsored experiment, irradiation of PWR-type rods at Halden continued after three fuel rods were removed for instrument readings and postirradiation examination. Indications were that only a small amount of the helium (used in the fuel rods to pre-pressurize them) is lost by absorption in the UO_2.

Battelle Pacific-Northwest Laboratories (PNL) is analyzing data from tests of the instrumented six-rod test assemblies irradiated at Halden (See page 157 of 1977 Annual Report). Analysis of one such high-power, long-exposure assembly showed that 20 of 28 detectors survived the 2-1/2-year life of the assembly, contributing data of unusual reliability. Conclusions drawn from these experiments are that fuel densification in a reactor does not necessarily lead to significantly higher temperatures in the fuel as had been thought, and that some movement of cracked pellet pieces and consequent reduction in fuel-cladding gap does occur in a reactor, and does lower the fuel temperatures, although the extent and speed of this fuel outward relocation vary greatly along the rod length.

LOCA Simulation in NRU. In another international cooperative effort, this one with Canada, NRC initiated a major new program in 1978 in which experimental equipment designed and fabricated by Battelle Pacific Northwest Laboratories (PNL) will be tested in the U-2 loop of Canada's NRU reactor at Chalk River Ontario. This Canadian facility can accommodate bundles of commercial-length fuel rods and provide sufficient power to simulate conditions predicted during accidents in large pressurized water reactors. The objective of the experiments is to provide information for comparison with the results of cheaper electrically heated experiments, and for testing the ability of codes to predict fuel behavior in loss-of-coolant accidents.

Plans are to irradiate six assemblies over a period of approximately three years, and both virgin and pre-irradiated rods will be tested. In addition to code verification, the experiments are designed to confirm various electrically driven tests (FLECHT, Semiscale, etc.) and to quantify the scaling procedures to be used in applying results of PBF and LOFT tests.

Fuel Behavior Codes

Fuel behavior information from the PBF, Halden and LOFT programs is used in developing and assessing the basic NRC fuel rod analysis codes, FRAP-T and FRAPCON. (FRAP stands for Fuel Rod Analysis Program.) FRAPCON is used for the longtime steady-state analysis of fuel rod response during normal reactor operation. FRAP-T is used for the transient analysis of fuel rod response during off-normal reactor conditions. During 1978, models and correlations for the prediction of fuel rod response to a LOCA were completed, giving FRAP-T a capability to predict fuel rod temperature, ballooning and possible failure dur ing all phases of a LOCA. Another major capability developed for the FRAP codes during the fiscal year is an option permitting prediction of both the nominal fuel rod response, and the error bands on the response. This option is fully automated to give FRAP codes a unique capability among fuel rod analysis codes.

FRAP-T requires, as input, the coolant conditions during a transient. In order to predict the behavior of fuel rods and their interaction with the coolant, FRAP-T should be linked with a thermal-hydraulics code. The first step towards giving it this capability was taken in 1978 when FRAP-T was linked to the COBRA-IV Code (See p. 178, 1976 Annual Report).

Fuel Meltdown and Fission Product Behavior

Fission Product Release and Transport. ORNL is conductng a program to determine the quantity and type of fission products which might escape from irradiated fuel rods during a shipping cask accident or a reactor accident. Experiments conducted in 1978, using segments of fully irradiated commercial fuel rods, have led to the development of a semiempirical model for the prediction of cesium and iodine release from failed fuel rods. At the same time, Battelle Columbus Laboratories continued work on a computer code (TRAP-MELT) capable of assessing radionuclide (fission product) transport and deposition within the primary system (reactor vessel and piping) following hypothetical accidents, including meltdown accidents.

Explosive Interactions Between Water and Core Materials. As noted in NRC's Annual Report for 1977 (see p. 157), the Reactor Safety Study (WASH-1400) concluded that a large release of radioactivity can only result from a fuel meltdown, and indicated that accidents in which the core melts at nominal operating pressures are important contributors to overall risk. Simulant fluid experiments (mineral oil/Freon 22) have demonstrated that high system pressures reduce the potential for explosive vapor formation. These results were substantiated in small-scale "corium"/water experiments in which vigorous explosions occurred at atmospheric pressure but not at pressures approximately seven times higher. Large-scale experiments at the Joint Research Center at Ispra, Italy, during 1978, have thus far provided additional evidence that high system pressures curtail vapor explosions.

Fuel Meltdown Studies. A program at Sandia Laboratories to investigate phenomena associated with the interaction of molten core materials and concrete has produced important data on the rate and nature of gases and aerosols generated during the interaction, the rate of penetration of the melt into the concrete, and the rate of fission product evolution from the melt. The information will be used in developing an improved code model for such phenomena (CORCON).

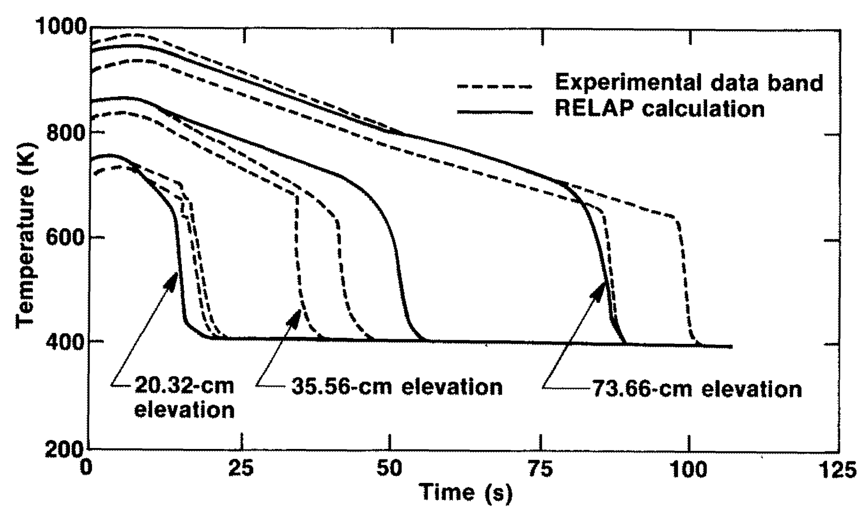
COMPUTER CODE DEVELOPMENT

Improved Existing Codes

RELAP-4. This systems LOCA code was first developed in the early 1970's and subsequent modifications have added certain desirable features. For example, RELAP-4/MOD-5, issued in 1976, could only handle the blowdown stage of light water reactor LOCA. RELAP-4/MOD-6, issued in 1977, could handle this, plus a reflood stage of LOCA in a PWR, but in a discontinuous fashion. Current efforts at INEL are on RELAP-4/MOD-7, applicable to a continuous best estimate analysis of a PWR LOCA. Completion of this final version of the RELAP-4 series, including extensive checkout and comparisons with test data, is expected towards the end of 1979. During 1978 an extensive assessment of RELAP-4/MOD-6 was undertaken by an independent group at INEL. Results thus far indicate that the code performs well for the blowdown phase of a LOCA, not so well for the refill phase, and spottily in the reflood regime. The code is being applied at Sandia Laboratory for initial statistical studies of PWR LOCA consequences.

WRAP. The Water Reactor Analysis Program (WRAP) designation covers a package of evaluation- model codes used in the licensing review process. It provides for automatic linkage of existing codes through an executive data management system. WRAP is under development at the Savannah River Laboratory where a BWR LOCA version was completed in July 1978. That version links the fuel behavior code GAPCON-THERMAL-2, the blowdown code RELAP-4/MOD-5, the reflood code NORCOOL-1 (developed in Denmark for NRC), and the hot channel code MOXY. A similar package for PWR LOCA licensing review is slated for completion in January 1979.

COBRA. The NRC 1977 Annual Report (p. 158) described a code developed by Pacific Northwest Laboratories to analyze thermal-hydraulic behavior in nuclear reactor cores and vessels. Like RELAP, the COBRA code has undergone both major and minor modifications over the years. One major modification introduced in 1977 resulted in COBRA-DF which was designed for adaptation to PWR vessels equipped with upper head injection ECCS. The suffix DF denotes that a model called "drift flux" was used to account for certain thermal and mechanical effects. During 1978, PNL concluded that the drift flux model was less suitable



This illustration shows a comparison between temperatures measured at various core elevations during a Semiscale MOD-1 reflood experiment and those calculated using the RELAP4/MOD6 code. The peak temperatures, cooling rates, and ultimate quenching, in this instance, were predicted with encouraging accuracy.

for multidimensional analyses than a more complex "two-fluid" (TF) model, and the code was again modified to the current COBRA-TF version. This version is now being applied to its intended task, encountering and resolving modeling or programming errors in the process.

Advanced Systems Codes

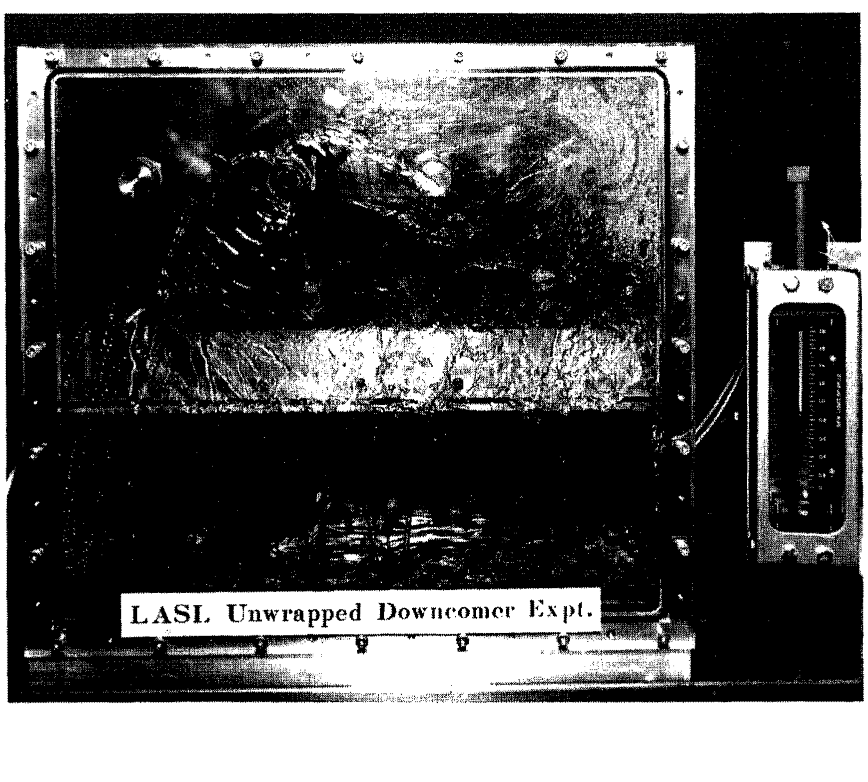
TRAC. The Transient Reactor Analysis Code (TRAC) developed at Los Alamos Scientific Laboratory (LASL) is an advanced best-estimate computer program designed to predict the thermal and hydraulic response of LWRs to LOCAs and other transients. (For a detailed description of TRAC, see p. 159 of the 1977 Annual Report).

The first PWR-LOCA version, TRAC-P1, was released to the public at an international workshop held at LASL in March 1978. A faster-running version, TRAC-PlA, is being released in December 1978, and a version with improved modeling, TRAC-P2, is scheduled for completion in March 1979. Work is starting on the first BWR-LOCA version, TRAC-B1, which should be operational at LASL in early 1979. TRAC is also used at integral systems test facilities at INEL and at Gesellschaft Reaktorsicherheit (GRS) in Germany. Both INEL and GRS are performing experiments of interest to NRC. TRAC is being used by LASL to analyze these experiments as part of the TRAC assessment.

In addition, TRAC has been used to analyze data from such separate effects activities as CREARE and BCL downcomer experiments, FLECHT single bundle reflood experiments, MARVIKEN (Sweden) blowdown tests, and INEL single-bundle air/water flooding tests.

As TRAC evolves, it is being applied to an increasing range of reduced scale water-reactor safety experiments. In a cooperative NRC reflood experimental program with Germany and Japan, TRAC is used to analyze and synthesize large-scale upper-plenum (Germany) and reactor core (Japan) experiments. Pre-test predictions of these experiments will be used to validate the treatment of multidimensional and scale effects in TRAC.

In conjunction with data comparisons, TRAC is also being used to analyze LOCA behavior in full-scale PWR's. The important result here is that a current best-estimate calculation shows the peak cladding temperature occurring during blowdown rather than later in reflood, as predicted by conservative evaluation-model calculations. This means a lowering of the failure rate, since the cladding is not subjected to high temperature for long periods.



NRC's continuing development of TRAC models is supported by small-scale experiments such as this one shown in progress at LASL. The "unwrapped downcomer" is a 2' x 3' transparent lexon slab which simulates, in a flat configuration, the cylindrical downcomer of a PWR. This device permits researchers to observe two-phase flow (steam-water) phenomena and will enable them to quantify the rates of water entrainment or de-entrainment during a LOCA.

Other LASL Codes. In addition to TRAC, the LASL program during 1978 included development and application of other advanced codes in computing thermal-hydraulic processes in reactor components. The K-TIF code has successfully simulated laboratory tests by Creare, Inc. and Battelle-Columbus Laboratories involving cooling water injection into a PWR downcomer. The SOLA-FLX code has been developed to predict the coupled-fluid and structural dynamics of a PWR core barrel following a postulated pipe break. (The core barrel is a three-inch-thick cylinder surrounding the core to separate heated, upflowing water from incoming, down-flowing water.)

In addition, studies employing K-FIX and SOLA-DF have provided new insights into two-phase flow processes crucial to the accurate prediction of PWR blowdown transients. The K-TIF, K-FIX and SOLA-DF codes were released to the National Energy Software Center (NESC) at DOE's Argonne National Laboratory during 1978.

THOR. The THOR Code at Brookhaven National Laboratory is described on page 161 of the 1977 Annual Report. Near the end of fiscal year 1977 it was determined that THOR modeling then employed was unstable. As a result, each component module as well as the THOR system itself was reprogrammed during 1978. The rewritten component modules were integrated with a new operating system and developmental verification was begun. Both multiple component assemblies and integral system assemblies (compatible with existing experimental information) were used. Further considerations affecting THOR are described below.

RELAP. Researchers at INEL, aware for some time of limitations in the reference code RELAP-4, began in 1977 studying more advanced modeling techniques. Those techniques were incorporated in a 1977 pilot code for individual system components, and the knowledge gained with that code was applied during 1978 in the framework of a LOCA code named RELAP-5. Both the mission and the capabilities of RELAP-5 and THOR are very similar (only the methods of achieving these capabilities are significantly different) and NRC must decide whether one of these two codes meets projected NRC needs and warrants further support. The code selected (from among RELAP-5, THOR, and a simplified TRAC) will serve as a basis for the advanced version of the evaluation-model code used in licensing audit.

Advanced Containment Codes

INEL continued work during 1978 on the development of a multidimensional computer code called BEACON for use in the analysis of containment systems during postulated accidents. BEACON/MOD-2, which incorporated an air-steam-water mass and energy transport model, was completed and sent to the National Energy Software Center in December 1977. The latest version of the code, BEACON/MOD-2A, is slated for completion in December 1978. It will add a wall and structure heat transfer analysis capability, as well as models for wall liquid film formation and liquid pool formation.

The Lawrence Livermore Laboratory in California is developing a best-estimate code to model pool dynamics in the wetwell of a pressure-suppression containment. An initial version of a finite element shell structure code named PELE-IC has been completed for use in analyzing fluid-structure interactions in wetwells during steam condensation. The code was able to simulate results of BWR pool dynamics experiments conducted at UCLA and MIT.

Code Assessment

Over the last four years INEL has developed a procedure for assessing, in detail, the accuracy of computer codes used to predict nuclear reactor response during normal and off-normal operations. RELAP-4/MOD-6 was being assessed with respect to its predictive capability at year's end, and comparisons were being made for pertinent variables such as maximum cladding temperatures, mass flow rates through the system, temperatures during reflood, and quench times (time at which cladding temperature falls rapidly to normal).

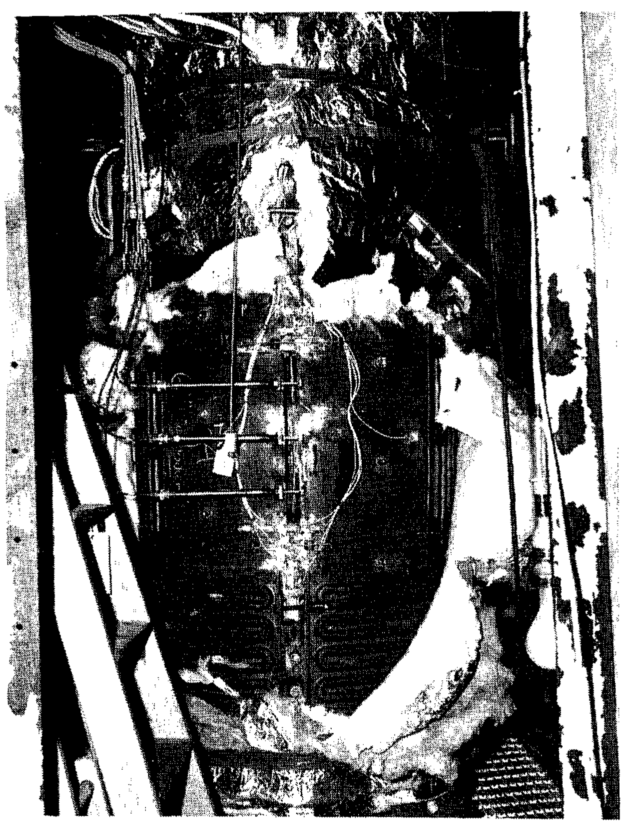
METALLURGY AND MATERIALS

Metallurgy and materials research led to the issuance of three important RIL's in this area during 1978. Accomplishments in these and other areas are discussed below.

Fracture Mechanics

Heavy Section Steel Technology (HSST) Program. Simulated in-service weld repairs were performed successfully at ORNL in 1978 on thick-walled intermediate pressure vessels, using procedures recommended in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. If a defect is discovered in a pressure vessel, use of this method precludes certain potential problems, such as warpage, which were previously associated with such procedures.

In another ORNL test, an intermediate-size pressure vessel with a large flaw, which had been subjected to enough pressure to cause a leak, was repaired by welding. Then the weld was deliberately flawed with a similar defect. The vessel was retested to essentially the same pressure overload, and leakage occurred without disruptive failure, as predicted. During the reporting period, ORNL compiled a great deal of new information on welding techniques, material strength and toughness. In addition, Battelle Columbus Laboratories completed a computational model to predict the residual stresses due to weld repairs. The information improves NRC's ability to evaluate weld repair methods for nuclear reactor systems.

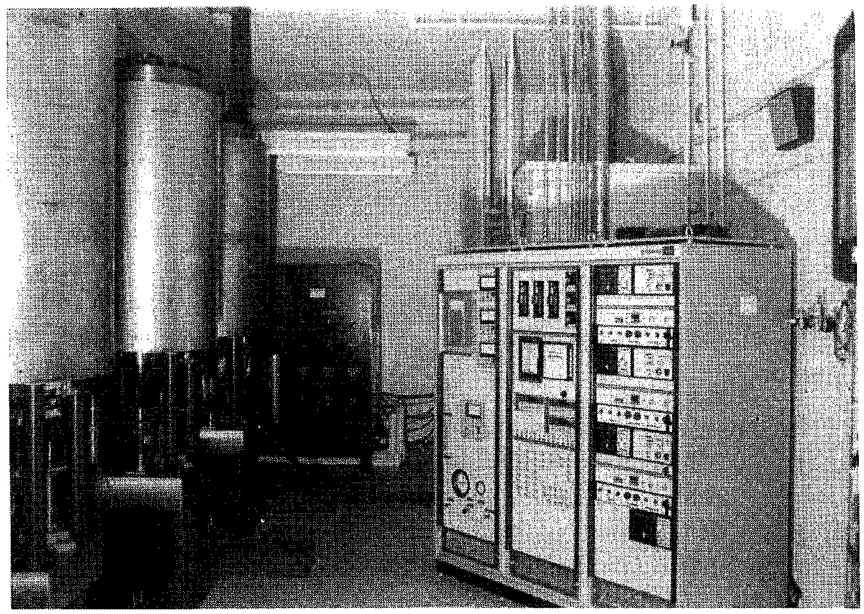


The protective insulation has been pulled aside following the testing of an 18-inch weld more than 5 inches deep in a six-inch-thick vessel. Pressure overloads more than double the design pressure were sustained in the instrumented weld area without disruptive failure. Such experiments are conducted at ORNL as part of NRC's Heavy Section Steel Technology Program.

Design Criteria for Piping and Nozzles. During 1978, a computer program was developed at ORNL for the stress analysis of pressure vessel nozzle arrangements in a variety of reactor vessel designs. Computed stresses from internal pressure loads were compared with known stress limits to evaluate margins of safety in current design criteria. The studies showed that the rules governing the design details of nozzle penetrations, and the spacing between them, are conservative. As a result, modifications to the rule have now been developed to assure that the full range of permitted geometries are covered. The computer programs were also extended to consider pressure vessel loads from attached piping. At the end of the period, detailed studies for these loadings were being prepared.

Cyclic Crack Growth Rate in Reactor Vessel Steels. Studies are being conducted by the Naval Research Laboratory (NRL) and ORNL to establish a cyclic crack growth rate data base for reactor vessel steels. The data will form the basis for updating rules contained in American Society of Mechanical Engineers (ASME) Code for evaluating flaws discovered during inspections. The tests are conducted in autoclaves (high pressure chambers) which simulate the loading conditions and water environment encountered in operating reactors. The number of variables being considered is so large that the NRL and HSST programs are coordinated with related studies at other laboratories through an international cooperative program conducted by the U.S., six European countries and Japan. With these several groups concentrating their research in selected areas, the total time required to characterize the crack growth phenomena can be shortened by several years, with the quantity of data greatly increased, and the cost reduced perhaps as much as $1 million.

Crack Arrest. If a running crack is to be stopped before the vessel wall is breached, the



These high pressure chambers (left), or "autoclaves," at the U.S. Naval Research Laboratory in Maryland, are used to test reactor-vessel metals to the temperature, pressure and water chemistry environments of a reactor. Tests on both irradiated and unirradiated specimens are run to measure crack-growth as a function of various loading conditions. Data from these experiments have contributed to improved accuracy in predicting crack propagation.

pressure vessel steel must have sufficient fracture toughness to arrest it. The object of a research project at Battelle-Columbus and the University of Maryland is to define how much toughness is required to stop the crack and to measure the actual toughness levels of typical steel and welds. During fiscal year 1978, a mathematical analysis of the LOCA was performed and successfully compared to experimental results obtained from Oak Ridge National Laboratory.

A Cooperative Test Program was initiated in November 1977 with 30 laboratories in the U.S. and abroad, to gain experience with the test procedure and test specimen. Using this specimen and procedure, BCL has developed a data base on crack-arrest toughness for unirradiated steel, and other specimens are being irradiated to determine the loss in toughness caused by reactor operating conditions.

Steam Generator Tube Integrity

Stress Corrosion Cracking of PWR Tubing. Under normal service conditions, the stress corrosion resistance of one type of tubing used in steam generators is known to be adequate. However, under abnormal conditions (such as corrosion and excessive deformation), cracking appears in such tubing. A research program was started at Brookhaven National Laboratory to examine the factors involved, and researchers there have demonstrated that accelerated tests can reproduce corrosion cracking quickly in the laboratory. This is a distinct advantage, since such testing previously required thousands of hours. At the end of fiscal year 1978, further research was under way to use this and other techniques to define relationships between the exposure conditions and the environment and metallurgical factors affecting crack resistance. When fully worked out, these relationships may be useful in arriving at licensing decisions concerning the performance of tubing.

Integrity of Flawed Tubing. Another important research project was conducted at Battelle Pacific Northwest Laboratories to investigate the behavior of defected tubing and to assess a currently accepted inspection method, the "single frequency eddy current method." Test results are used to validate the margin of safety against the bursting of defected tubing under both operating and accident conditions in a PWR steam generator. Specimens used in the tests then are subjected to the single frequency inspection.

Because single-frequency eddy-currents can produce ambiguity due to independent variables which affect the signals, a new program was initiated in 1978 at ORNL to develop improved eddy current techniques. The program uses mathematical models and computer programs to design optimum examination instruments and techniques. At year's end, considerable progress was being made in establishing the necessary computer codes, in computer design of the improved eddy-current test probes and instrumentation, and in the acquisition of tubing with selected flaws.

Radiation Embrittlement

Irradiation-Anneal-Reirradiation Program. As reported in the 1977 Annual Report (p. 164), NRL has been investigating the merits of periodic heat treatment (annealing) in reducing radiation embrittlement in older reactor pressure vessels, and has demonstrated the usefulness of such treatment. With this basis, studies were undertaken in 1978 to project the rate at which the toughness of the vessel wall will again decrease with irradiation following annealing treatment, and to determine if the heat treatment may be necessary more than once in a system's lifetime. Thus far, the tests have shown that periodic treatment has a high potential for reducing embrittlement. It also appears particularly promising in restoring toughness in welds of older vessels where long-term property projections fall below code minimums. NRC expects the program to provide a basis for deciding on the suitability of the method for industry-wide use.

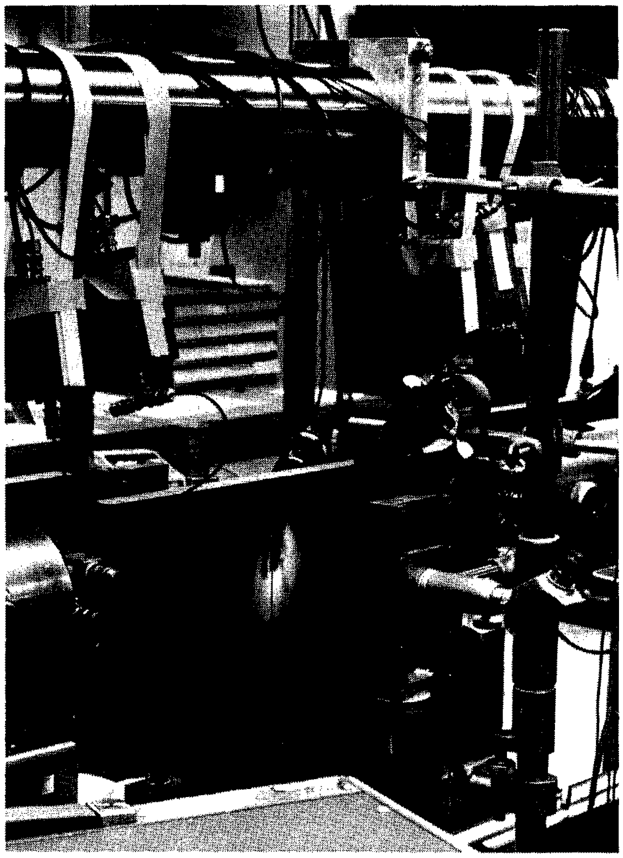
Irradiation Surveillance Program. Radiation embrittlement is caused by neutrons emitted by reactor fuel during operation, and in order to assess its significance, it is necessary to know the number of neutrons causing the phenomenon. This is being studied in a coordinated program involving Hanford Engineering Development Lab., Oak Ridge National Laboratory and the National Bureau of Standards, with important contributions from other research labs and organizations in the U.S., Belgium, Germany, the United Kingdom and Italy. Excellent progress was made this year in obtaining calculations and measurements of the numbers and energy of neutrons to be expected in typical reactor environments—measurements which will support the development of extensive ASTM procedures for further defining and measuring embrittlement and its causes.

Flaw Detection

Detection of Sensitization. As reported in the 1977 Annual Report (See p. 166), researchers at General Electric are developing a technique for measuring sensitization in stainless steel. Sensitization is a microstructural change in stainless steel which occurs when heated to certain temperatures. It sometimes occurs during welding if the welding heat is not carefully controlled, and it can lead to corrosion and cracking of key pipes. Visual examination cannot detect sensitization from welding, however the change can be measured using a technique called Electrochemical Potentiokinetic Reactivation (EPR). Work in 1978 was directed toward the fabrication of a portable instrument for nondestructive sensitization testing of piping and other reactor components, and on developing procedures for field application of the EPR method. Round-robin testing of the EPR has been initiated to help obtain its adoption by ASTM as a standard technique for detecting sensitization in stainless steels.

Ultrasonic Testing. An improved technique for ultrasonic testing, now in its fourth year of development at the University of Michigan, has proved capable of producing high-resolution images of flaws in heavy-section steel. As in conventional ultrasonic testing methods, an area of a pressure vessel is scanned with a transducer which sends sound waves through the metal. A "synthetic aperture" process then is employed to synthesize signals seen by the tranducer from its various scanning positions to produce an accurate picture of the flaw location and shape. This laboratory development was being transferred to a field inspection unit at Southwest Research Institute at the end of the reporting period.

Work by the two contractors during 1978 included development of a new computer plotting system to produce three-dimensional line simulations of flaws from the ultrasonic data. The drawings will enable fracture-mechanics experts to more accurately determine the safety of components. Another 1978 development was a new "spotlight mode" scanning technique which allows imaging at greater depths. Finally, studies were made to determine requirements for computer and other hardware to improve imaging speeds for quicker tests results. Imaging in the



In this test at Pacific Northwest Laboratories, a typical fatigue crack growth specimen of pressure vessel steel is being tested to study acoustic emissions. As the specimen is stretched and relaxed, the crack (illuminated by spotlight) grows and the resulting acoustic signals are picked up by the transducers mounted on the ends of the specimen (two shown on left, three on right). By looking through the eyepiece and manipulating the calibrated handle of the device (shown in foreground), the growth of the crack can be measured.

laboratory is relatively slow now, and these developments in real-time imaging will enhance future inspections.

Acoustic Emission. The acoustic emission program is designed to improve our ability to detect, locate and characterize flaws during the actual welding of nuclear piping and pressure vessels. An NRC contract research firm, GARD (for General American Research Division),Inc., has developed a two-channel acoustic system specifically designed for monitoring unattended multi-pass welds and to identify and locate defects in single-pass welds. It was tested successfully in nuclear fabrication facilities during 1978, and, by the end of the year, more than 13,000 feet of weld passes had been monitored.

RESEARCH SUPPORT

NRC's Research Support Program, sometimes referred to as "operational safety research," encompasses activities which affect more than one of the specialized research programs. These activities, discussed in detail below, include fire protection research, qualification testing evaluation, noise diagnostics, human factors, information and code dissemination and the international thermal-hydraulics program designated "3D."

Fire Protection

On September 15, 1978, Underwriters Laboratories conducted a full-scale fire test of vertically oriented cable trays, using a configuration of cable-tray protective barriers and an automatic sprinkler system. The source of the fire was a small amount of flammable liquid. There was melting of cable insulation at the lower end of the trays and one electrical short. The system failed to respond to the fire sprinkler. The test raised questions with regard to the effectiveness of both the automatic sprinkler system and the protective barrier used in the test configuration to protect against a flammable liquid spill.

At years end, the data were being assessed and a report was being prepared.

Other activities in fire protection research in 1978 included: (1) series of tests on fire retardant coatings which indicate that coatings offer some protection against fire propagation; (2) fire shield tests which added to the knowledge of ceramic insulating blankets, solid bottom steel cable trays, solid cover steel cable trays and ceramic boards, and which showed that, for qualified cable, they all will prevent the spread of a fully developed fire; (3) recommendations concerning experimental procedures for flammability tests on vertical cables; and (4) the design by Sandia Laboratories of a fire suppression facility which will be built in 1979.

Qualification Testing Evaluation

The Qualification Testing Evaluation Program is designed to test methodologies used in equipment qualification tests toward better simulating the effects of environmental factors such as radiation and temperature. In a LOCA synergistic test series completed at Sandia Laboratories, researchers found no significant differences between sequential applications of test environments (temperature, radiation, LOCA steam/chemical spray) and their simultaneous application. At the end of the year, the researchers were preparing for more realistic testing of the methodology using a much improved test facility. A list of generic safety related equipment to be considered for testing already had been compiled.

Also at Sandia, calculations were performed to evaluate the ability of simulators to duplicate the radiation effects expected from various accident scenarios. These were under study as the period ended, and a methodology was developed to assess the effects of the different aging environments on reactor hardware. The methodology will lead to equipment tests which can determine how long the hardware will last in an operational nuclear plant.

Noise Diagnostics

"Noise-diagnostics" is used for diagnosing and monitoring reactor system behavior, and for identifying malfunctions such as excessive vibrations, power oscillations and loose parts. During 1978, researchers at ORNL took the first step in developing analytical methods using "neutron noise" (small variations in the signals from reactor instrumentation). They also assessed various methods of mounting detectors to sense the presence of loose parts in a reactor system. In addition, efforts were under way to develop a model to better assess BWR stability.

Human Factors

Research on human factors is directed toward assessing the role of human errors in reactor safety and to the study of means by which they can be reduced. In 1978, training programs for use by NRC's Office of Inspection and Enforcement were initiated. A study also was initiated to assist in setting safety-related criteria for operator actions in nuclear power plants.

Safety Information and Computer Code Dissemination

Nuclear Safety Information Center. The Nuclear Safety Information Center (NSIC) at Oak Ridge, added more than 12,000 accessions to the 135,000 items in its computer file, published eleven reports for sale through the National Technical Information Services (NTIS) in Springfield, Virginia, and had several other reports in preparation at the end of 1978. The Center also responded to more than 1000 requests, received more than 100 visitors, and continued its abstract service and its bimonthly review, *Nuclear Safety*, now in its nineteenth year.

NRC Software Exchange and Information Activity. On July 5, 1978, the Argonne Code Center was renamed the National Energy Software Center (NESC) to better reflect the present scope of the Center's program. In the 12 months from October 1977 through September 1978, the Center distributed 942 copies of the software packages and authorizations for their use in response to requests from NRC and DOE offices and contractors, other government agencies, universities and commercial and industrial organizations.

Faculty Institute. At NRC's request, the Argonne Center for Educational Affairs presents each year an institute for the academic community on the general subject of reactor safety with the aim of encouraging university people to prepare and use curriculum material on the subject. The second such institute, devoted to LMFBR safety, took place in February and drew attendance by 24 faculty members. More than 200 pages of abstracts and teaching aids were made available to U.S. universities with nuclear engineering departments or programs.

Water Reactor Safety Research Meeting. The NRC Division of Reactor Safety Research held its fifth Water Reactor Safety Research Information Meeting on November 7- 11, 1977 at the National Bureau of Standards, Gaithersburg, Md. 126 papers were presented describing the latest results and significant research achievements in: (1) loss-of-coolant accident studies, (2) metallurgy and materials, (3) fuel behavior research, (4) analysis development, and (5) reactor operational safety.

More than 700 persons participated in the four-day meeting, including 161 foreign visitors. Summaries of papers presented at the meeting have been made available for review at NRC's Public Document Room in Washington, D.C.

3D Program

The NRC has entered into a cooperative arrangement with the Ministry of Research and Technology (BMFT), Federal Republic of Germany (FRG), and the Japan Atomic Energy Research Institute (JAERI) to undertake a coordinated analytical and experimental study of the thermalhydraulic behavior of emergency core cooling during the refill and reflood phase of a postulated LOCA in a PWR. The objectives of the program are to study: (1) whether steam formed inside a reactor core during a postulated LOCA can cause sufficient back pressure to inhibit the movement of emergency coolant into the center of the core, (2) how the emergency coolant redistributes itself horizontally as it moves through the heated core, and (3) how the steam and water move and interact in the core and upper plenum during the period when the ECC water fills the lower plenum and starts to cool the core. (These objectives deal with flow patterns in three dimensions, hence the "3D Program" title).

The cooperative arrangement provides that NRC programs at LASL will apply the advanced, three-dimensional TRAC code to a series of non-nuclear experiments to be conducted in Japan and the FRG, and those at INEL and ORNL will loan advanced instruments to BMFT and JAERI to permit the measurements needed for TRAC computer code studies.

The Japanese experiments will emphasize the behavior of steam and water in an electrically heated core. The FRG experiments will focus on steam and water behavior in the upper plenum and downcomer of a mock-PWR. Measurements using common US instrumentation will provide a valid comparison of results, and the TRAC code will permit the integration of Japanese core tests with FRG upper plenum tests. From this program, NRC will be able to get the data needed for tests of the TRAC code without having to build expensive test facilities, and this will round out the results obtained from such "one dimensional" test facilities as LOFT and Semiscale.

Advanced Reactor Safety Research

NRC's Advanced Reactor Safety Research Program provides tools for evaluating the safety of advanced reactors at the time of their commercial introduction. Because advanced reactors are not expected to achieve commercial status until at least the last years of this century, a 15-to 20-year NRC safety research program is foreseen. Research to support licensing evaluation for demonstration and prototype plants will be accomplished, as required. The principal candidates for advanced nuclear power generation are the fast breeder reactor and the high-temperature gas-cooled reactor; however, attention also may be given to the heavy water reactor and the light water breeder reactor.

FAST REACTOR RESEARCH

The president announced in April 1977, a plan to cancel the Clinch River Breeder Reactor (CRBR) and to defer decisions on the commercialization of fast breeder reactors. Although his announcement provided for continuation of a strong program of breeder technology research, a shift of emphasis in NRC research was required to accommodate the deferral of fast breeder commercialization. In addition, in late 1977 and early 1978 new programs were initiated by the Executive Branch — the International Nuclear Fuel Cycle Evaluation (INFCE), National Uranium Resources Evaluation (NURE) and Nonproliferation Alternative Systems Assessment Program (NASAP) — and these also affected NRC research planning. (See Chapter 9.) In response to these policy changes and initiatives, NRC's research in fast breeder safety was cut back in 1978. Specifically, work related to CRBR licensing was discontinued and the program was refocused on generic safety problems and on monitoring fast reactor research sponsored by the Department of Energy.

At the end of fiscal year 1978, bilateral information exchange agreements had been established with major foreign sources of safety information (except the U.S.S.R). Included in these was an exchange agreement with the CABRI Project, a fast reactor in France, jointly funded by France and Germany.

NRC's fast reactor safety research program is made up of five sub-programs: Analysis Development, Safety Test Facility, Material Interactions, Aerosol Release and Transport, and Systems Integrity. (These subprograms are summarized on pp. 186 and 187 of the 1976 NRC Annual Report.) Because of budget constraints, not all the objectives for fiscal year 1978 were met; however, some substantial accomplishments can be reported:

- In the analysis development area, several accident codes were brought to operational status. A key item in the package is the development of the Los Alamos Fuel Model (LAFM) to predict the time and place of clad failure under test conditions.
- A major safety test facility at Sandia, New Mexico, the Annular Core Pulse Reactor (ACPR), was modified to vastly extend its research capabilities, and its name was changed to the Annular Core Research Reactor (ACRR) to reflect the improvement.
- A study of the generic safety proof tests needed to verify accident codes was completed and, from it, a list of safety test facility requirements was deduced.
- Work in materials interactions included completion of a series of tests in the ACPR at Sandia Laboratories with results that provided important new insights into fast reactor core disruptive accident phenomena.
- In the aerosol transport area, both evaluation-model and best-estimate codes were extensively tested and verified, although work on best-estimate codes was slowed by budget cuts. An evaluation-model code was developed to describe the fraction of the core which might be vaporized (the "source term") in a generic core disruptive accident, and a new facility (FAST) was developed and tested to establish the best-estimate model. Similar work dealing with a postulated core melt was deferred due to budget cuts.
- In systems integrity research, a quantitative predictive method was developed to analyze the interactions of sodium and concrete under extreme conditions, a significant concern in licensing review.

These and other activities of NRC's advanced reactor safety research program during 1978 are discussed below.

Analysis Program

Considerable progress was made in 1978 in NRC's development of computer codes and mathematical models. The work was performed primarily at the Argonne, Los Alamos, Brookhaven and Sandia Laboratories.

Argonne National Laboratory (ANL). The ANL computer model (EPIC), which describes how fuel and coolant move in and around one fuel pin (a "fuel pin" is a fuel rod used in test reactors) during a hypothetical core disruptive accident (HCDA), has been integrated into the Argonne Safety Analysis System (SAS) computer code. The SAS code models the behavior of an entire fast breeder reactor core during the initial phase of an HCDA.

THE NRC CODE DEVELOPMENT PROGRAM

Computer codes form the basis of nearly all research methodologies employed by NRC, and can be used to predict the course of postulated accidents and their potential consequences.

The credibility of codes used in reactor safety assessment depends on how accurately they can predict the outcome of safety research experiments and on the validity of the experiments themselves in simulating actual reactor structures, systems, or components under postulated accident conditions.

Codes used for predictions of safety research experiments and for evaluation of the margin of safety of PWR and BWR plants are called "best estimate" codes. Most of NRC's code development effort, to date, has dealt with codes describing thermal-hydraulic transients in light water reactor systems following postulated loss-of-coolant accidents (LOCA). As these codes are being completed, a redirected emphasis is being placed on their adaptation to non-LOCA conditions or incidents. These include Anticipated Transients Without Scram, Reactivity Initiated Accidents, steam pipe break accidents in PWRs, and other operational transients in both PWRs and BWRs.

In addition to the "best estimate" codes the so called "evaluation model" code version are being developed and/or improved. These codes provide for the deliberately conservative analyses used in NRC licensing reviews.

Another Argonne program is aimed at developing methodologies and techniques for predicting three-dimensional flow patterns and temperature distribution in LMFBR components. Currently, three codes are under development: (1) COMMIX-1, a transient, single-phase (liquid only) computer code, (2) COMMIX-2, a transient, two-phase (liquid and vapor) computer code, and (3) BODYFIT-1, a transient, single-phase code which transforms complex geometries to simple ones. COMMIX-1 and COMMIX-2 can treat sodium coolant systems in both the reactor plenum and rod bundle regions. Documentation of COMMIX-1 was completed in 1978 and is available to the public on request from Argonne.

A series of LMFBR safety-related critical experiments was completed with ANL's Zero Power Reactor-9 (ZPR-9) during 1978, and the experimental results were being analyzed at year's end. The tests were designed to provide an experimental basis for comparison and validation of the neutron physics part of the LMFBR accident analysis.

Los Alamos Scientific Laboratory (LASL). The LMFBR safety program at LASL produced important information related to hypothetical core disruptive accidents most notably in fuel pin failure mechanics, material motion during core melting, recriticality, primary system damage, identification of core materials which might escape ("Radiological source term" or the source of radiological hazard) and initial conditions for post-accident heat removal. The Los Alamos Fuel Model (LAFM) code, released in 1978, addresses the problem of predicting time and location of failure of irradiated fuel pins in unprotected transient overpower accidents (those in which the failure of control rod to insert results in an excessive power surge). This is an important first step toward defining the consequences of such accidents. The code was satisfactorily tested through comparison with 13 in-reactor experiments performed in the DOE Transient Reactor Test (TREAT) facility at the Idaho National Engineering Laboratory.

In another LASL activity, efforts to model the motions of fuel, steel, and sodium in a meltdown which might lead to the rearrangement of core material into a super-critical configuration met with success in the SIMMER-II code. SIMMER-II, which was released in 1978, is the first calculational tool capable of resolving these material motions, and initial calculations using it were being evaluated at year's end. It appears that the code will become a reliable predictive tool for this complex problem.

The SIMMER code has been used to predict the results of experiments performed at other laboratories. At LASL, experiments also were conducted which were designed specifically to check the adequacy of SIMMER models. Complex experiments have led both to minor improvements in the models, and to increased confidence in the ability of SIMMER to predict experiments.

Brookhaven National Laboratory (BNL). Work on the "Super System Code" (SSC) at BNL continued during 1978 (see p. 172, 1977 NRC Annual Report). This family of computer programs predicts variables such as maximum coolant and cladding tube temperatures in a reactor following a variety of operational and safety-related disturbances in an LMFBR plant.

Sandia Laboratories. Analysis development at Sandia in 1978 focused on three major areas:

Accident Delineation— Work which defines accident sequences using event tree methodology and identifies dominant accident sequences and key phenomenological uncertainties in the events.

Benchmark Containment Code— The development of a systems code to determine the radiological source term (leakage potential) from a containment following a core disruptive accident in an LMFBR.

Phenomenological Modeling — Those models and codes which have been developed to support tests, both in and out-of-reactor, that clarify basic phenomena associated with severe accidents.

Safety Test Facility Program

Highlights of NRC's safety test facility program at Sandia Laboratories in 1978 included the successful completion of a project to upgrade the performance of the Annular Core Pulse Reactor (ACPR), completion of the development of a high-precision fuel-motion diagnostics system, implementation of an agreement to participate in activities at a major foreign facility, and the termination of efforts to define future safety test facility needs.

ACPR Upgrade. Experiments in the Annular Core Pulse Reactor were suspended early in 1978 when ACPR was shut down for major modifications. This included the installation of a new control system and cooling system, the loading of the new high-performance core, development of a new high heat-capacity fuel, and a new fuel element design. The upgraded reactor achieved initial criticality on April 27, 1978, and became available for experiments at the end of September 1978. The upgrade exceeded all initial objectives and significantly improved both the pulsed and steady-state performance of ACPR without materially affecting the unique "short pulse" dynamic characteristics. In dedication ceremonies on September 14, 1978, the ACPR was renamed the Annular Core Research Reactor (ACRR) to better reflect its upgraded capabilities.

Fuel-Motion Diagnostics System. Development of a high-precision diagnostics system for observing fuel motion during test reactor operation was completed by Sandia Laboratories in 1978 for installation next year in the ACRR. This system employs coded-aperture imaging of fission gamma rays emitted by the fuel as it moves, a completely different technique from those previously used in fuel-motion diagnostics. Thus far, high precision has been demonstrated using a mock-up system.

ACPR/CABRI Exchange Program. Considerable progress was made in 1978 in the multinational fast reactor safety research exchange program, called ACPR/CABRI, with the approval of the exchange agreement and the assignment of U.S. and foreign staff representatives. U.S. participation in CABRI included the development of calculations to predict the results of initial CABRI test, the "A" series, a series of transient overpower (TOP) experiments. These calculations then were compared to pretest code predictions made by France, Germany, Japan, and the United Kingdom, and, in 1979, they will be compared to the test results for code verification.

Future Facility Planning. Development of the technical bases for NRC input to DOE planning for new safety research facilities was halted at the end of fiscal year 1978 because of DOE program deferrals and overall restrictions in NRC's funding. At year's end, the program was dormant.

Materials Interactions Program

Studies of the interactions of materials under core disruptive accident conditions continued in in-reactor experiments in the ACPR at Sandia and in out-of-reactor experiments at several other laboratories. Work included experiments and model development in prompt burst energetics, system changes following the loss of original core geometry, and the melt-through penetration of post-accident core debris. These are discussed below:

Sandia Laboratories. Materials interaction projects at Sandia included:

Prompt Burst Energetics (PBE). The initial ACPR/PBE test series, completed in 1978, was the first study of the response of fuel pins in both sodium and nonsodium environments to rapid heating conditions in what are believed to be potentially the most damaging hypothetical core disruptive accidents (HCDA's). The program involves the study of pressure generation and the damage potential of an HCDA. The experiments are leading to more realistic estimates of the amount of mechanical energy available to damage the reactor primary system containment.

Equation of State. The relationship between the pressure, volume and temperature generated by vaporization of reactor fuel is described in an "equation of state." (See p. 175, 1977 Annual Report, for added information.) In 1978, experiments to determine the vapor pressure of fresh UO_2 fuel up to 7000°K (about 12,000°F) were completed using electron beam and pulsed reactor techniques, and work at Sandia was redirected into the area of irradiated fuel.

Fuel Disruption. At Sandia, experiments were conducted in the ACPR to study the disruption of irradiated fuel under loss of-flow conditions. These are important processes, not yet well understood. The tests permitted visual observation of these phenomena for the first time. Some conclusions from these experiments are that (1) very rapid swelling of irradiated fuel can occur and may be the dominant initial mode of fuel disruption; (2) the cladding separated from the fuel due either to buckling or to internal gas pressure; (3) fuel swelling is not predicted by standard fission gas modeling. Improved modeling in the new Sandia Code, FISGAS--for "fission gas"--yields results closer to values determined in ACPR experiments, and (4) no evidence has been observed of the "dust cloud" disassembly or "fuel frothing" assumed in present accident analysis codes.

Brookhaven National Laboratory (BNL). Out-of-reactor laboratory experiments on the transition and post-accident heat removal phases of HCDAs continued at BNL in 1978 as summarized below:

Dispersal of Fuel in Boiling Pools. In 1978 an analytical model was developed to describe the dynamics of fluid motion in boiling systems. A test apparatus, using water as the simulant was designed and fabricated. Initial data were obtained using special fast-response instrumentation.

Heat Transfer from Internally-Heated Pools. Two topical reports (BNL-NUREG-50722 and BNL-NUREG-50759), issued in 1978, provided the experimental and analytical results of internally heated boiling pool experiments. A major conclusion of this work was that heat transfer from the boiling pool to vertical walls of the container can be described very well, using well known heat-transfer correlations with modified properties.

Flow-Freezing Experiments. Two reports were issued describing experiments investigating the solidification (freezing) of liquid systems (BNL-NUREG-23149) and mixed liquid/gas systems (BNL-NUREG-24486). The experiments used Wood's metal and paraffin as the liquid simulants and nitrogen gas. A major conclusion was that the presence of the gas significantly reduces the time required for the liquid to solidify and "plug" its own flow.

Aerosol Release and Transport Programs

Models of aerosol behavior are being developed to allow predictions of airborne radioactive particle concentrations in containment buildings of fast breeder reactors that might leak to the environment. Experiments at ORNL (see p. 173, 1977 NRC Annual Report) were designed to define the key properties of aerosols and to verify or develop analytical methods for predicting aerosol behavior. The equipment used is suitable for the study of aerosols which may be generated in accidents in either sodium or gas-cooled breeder reactors, using either uranium or thorium-based fuels. Activities during 1978 at ORNL and at other laboratories involved in aerosol work are summarized below:

Oak Ridge National Laboratory. The effort to quantify fuel and fission product escape from a primary containment in a postulated accident includes investigation of how discharged materials are suspended (as aerosols) within the secondary containment and how concentrations there change as the particles agglomerate and settle out. In the ORNL experiments sample fuel pins are vaporized in instrumented laboratory vessels. One vessel, called FAST (Fuel Aerosol Simulant Test), includes a sodium pool to simulate an LMFBR thermal environment. The facility was completed in 1978 and initial tests were conducted in which an argon environment was maintained in the vessel. Follow-on tests in water will begin in fiscal year 1979, and at some future date tests in sodium will be conducted.

The ORNL program on the behavior of aerosols in secondary containments was continued in 1978 using the Nuclear Safety Pilot Plant, a vessel measuring about 10 feet by 20 feet, in which sodium and uranium are burned to produce aerosol mixtures. At the end of the reporting period, the data on these aerosols and the mixtures were being used to test codes that predict aerosol transport within the containment.

Sandia Laboratories. During 1978, Sandia Laboratories analyzed the results of its 1977 tests to create uranium dioxide aerosols similar to those expected in a core disruptive accident. The experiments used neutron-induced fission in the ACPR to vaporize the fuel. Diagnostics included sampling the velocity and size distribution of the aerosol particles and visual observation of the vapor cloud. Photographs of aerosol particles produced in the reactor will be compared to those produced in out-of-pile experiments at ORNL.

Systems Integrity

A research program to develop the data NRC needs to assess the structural integrity of breeder reactors and certain plant systems continued in fiscal year 1978. Studies, conducted largely at

Sandia, addressed the integrity of containment systems in the event that a melting core might penetrate the primary reactor vessel. These, and the results from high temperature tests on reactor and containment materials, are described below.

Debris Bed Studies. A series of three unique in-reactor experiments was completed in 1978. For the first time, the behavior of a large quantity of post-accident core debris was simulated. (Previously, such experiments either used substitute materials such as sand and water, or proper materials uncharacteristically heated. Fission-heating of uranium oxide fuel in sodium is used so that the internal decay heat is properly simulated. This makes it possible to determine how debris immersed in liquid can be cooled. Results of the initial tests indicated that the transition to melt takes longer than expected or that the beds can accommodate significantly higher decay heat levels before all the liquid evaporates. If these and other results are confirmed by further research, accommodating post-accident heat removal should be easier than previously estimated.

Molten Core Technology. Large scale experiments continued in 1978 on the interaction between molten core materials and containment materials outside the reactor pressure vessel. Results indicate rapid deterioration of concrete in the presence of molten materials, as well as the release of combustible gases and radioactive aerosols. This information has been incorporated into NRC review criteria for LMFBR plants.

Sodium Containment. A large-scale facility was completed at Sandia in 1978 which will heat and contain up to 224 kg of sodium to temperatures approaching boiling. The sodium will be used in a variety of compatibility and interaction tests, as well as in sodium chemical fire tests. Several preliminary sodium-concrete tests have been performed and the results differ considerably from earlier, smaller-scale tests at other laboratories. Concrete erosion rates appear larger, the sodium consumption is complete, large amounts of carbon are formed, some concrete fractures are observed, and considerable heat is generated in the sodium/concrete interaction. Computer code models were developed in 1978 to describe the interaction and separate effects tests are being planned to verify resulting hypotheses.

CONVERTER REACTOR RESEARCH

Advanced reactor concepts other than fast breeder reactors are addressed in a variety of NRC research activities under the program title "Advanced Converter Reactor Research." The program goals are stated in terms broad enough to accommodate a variety of advanced concepts, including high-temperature gas-cooled reactors (HTGR), advanced HTGRs, heavy water reactors, and others. The research itself, although initially concerned almost exclusively with the Fort St. Vrain-type HTGR, has turned more to generic issues. The reoriented program is expected to provide sound bases for the development of licensing criteria for any type of advanced, gas-cooled reactors when they become commercially available.

Specific accomplishments in NRC's safety-related HTGR operational and accident-potential areas were conducted largely at four DOE facilities. These activities are summarized below.

Brookhaven National Laboratory (BNL)

A major part of NRC's experimental HTGR research program is carried out at BNL. Work was conducted in seven research categories, as follows:

Fission Product Transport. Laboratory experiments carried out in 1978 suggest that radioactive iodine will condense on internal reactor surfaces at any temperature below about 500 °C (930 °F). In an accident, temperatures in this range can be expected.

Core Heatup. The BNL experimental program continued in 1978 to evaluate the behavior of HTGR core materials during accidents at temperatures between 2200° and 3600 °C (4000° and 6500 °F). Preliminary tests have indicated that the conventional assumptions concerning material transport in graphite are not valid. These experiments are significant in that such transport governs the rate at which fission products may be released during an accident.

Primary Coolant Interactions. The Material Test Loop which was constructed in late 1977, operated without interruption in 1978 and provided a high temperature helium environment, containing controlled concentrations of contaminants, to the testing machines used in the materials test program. (See below.)

Materials Test Program. To gain insights into the metallurgical changes and possible degradation of various structural alloys in HTGR operations, the fatigue and creep properties at high temperatures were evaluated for alloys used in HTGR steam generators, helium circulators and thermal barrier systems.

Characterization of PGX Graphite. Two kinds of experiments were aimed at an improved understanding of the roles and behavior of the masses of graphite used as moderators in HTGRs. In one type, the gas permeability of a relatively impure graphite (PGX) was found to be 10 to 30 times higher than that of the purer graphite used in the fuel region of a reactor. It also was found that gas diffusion coefficients of the two graphites are quite similar and are in excellent agreement with approximations used in existing computer codes.

HTGR Accident Analyses. Concerning the mixing of primary and secondary containment gases during rapid depressurization of the HTGR during an accident, a continuing program has been undertaken to identify and study the relevant processes involved in depressurization.

An experimental program has been under way wherein helium depressurization into an enclosed volume can be observed. Visual observations made with a subscaled experimental apparatus are used to provide information needed in the unit-problem approach.

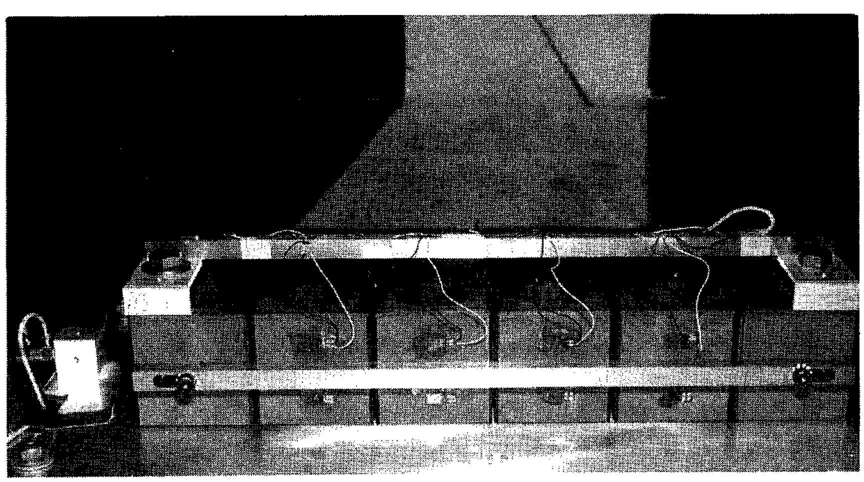
Los Alamos Scientific Laboratory

LASL continued its program of research in several areas of HTGR safety technology. Analytical work included the development of computer codes for calculating fission product transport in various parts of a reactor. Both theoretical and experimental studies were conducted to define transport mechanisms and supply data for calculations. In 1978, a new code (DASH) was developed to facilitate studies of transport of fission products from fuel particles to graphite-coolant surfaces in the reactor core.

Another task involved identification of potential accident sequences and development of a complete system-analysis code, CHAP, for their analysis. CHAP consists of a model-independent systems-analysis program with both steady-state and transient solution capabilities and a set of modules each representing a component or subsystem of the overall plant model.

The HTGR structural task includes the development of a code, NONSAP-C, for analyses of the behavior of prestressed concrete reactor vessels (PCRV) and an experimental seismic program concerned primarily with HTGR core block scale modeling. However, the experiments indicate a considerable reserve load-carrying capacity and a much greater vessel ductility than the calculations. In an attempt to simulate the observed postelastic behavior more

Seismic testing of HTGR core-block material (graphite) is conducted by LASL as part of NRC's gas-cooled reactor safety research. Shown here is a one-dimensional graphite block system, instrumented and mounted on a servohydraulic shaker. The response of this system was predicted from tests on a scale model made of plexiglass blocks. Tests are conducted in a LASL facility at White Sands Missile Range, New Mexico.



accurately, other models are being studied and implemented in the NONSAP-C code. A users' manual and a set of test problems for NONSAP-C have been prepared as part of the code release package.

Battelle Pacific Northwest Laboratories

The PNL program is to establish a method of monitoring the strength of the graphite support structure of HTGRs.

In 1978, experiments were conducted to determine the relationship of mechanical strength to sonic sound wave velocity for several graphite types and also the relationship of changes in sonic velocity to changes in the strength during oxidation under a variety of conditions. The results, together with other data, indicate that measurement of sonic velocity might be used to determine the strength of the HTGR support structure in periodic inspections, and thus help ensure continued safe operation.

Oak Ridge National Laboratory

At ORNL, the HTGR safety program is limited to studies of the Fort St. Vrain reactor in Platteville, Colorado. In 1978, further work was done to refine ORNL computer simulations of postulated accident scenarios. The program furnished analyses relating to NRC's approval of a 100 percent power operating license for the Fort St. Vrain reactor. Detailed transient data from tests at Fort St. Vrain were analyzed and used to confirm simulator predictions and to improve the models (see Chapter 2 for more on Fort St. Vrain).

General Reactor Safety Research

General Reactor Safety Research comprises three areas: Site Safety Research transferred from the Water Reactor Safety program in 1978, and two programs created in 1978, Mechanical Engineering Research and Structural Engineering Research, which just commenced toward the end of the period.

SITE SAFETY RESEARCH

The Site Safety Research Program consists of generic research directed toward estimating the effects of earthquakes, floods, and tornadoes and other severe storms; understanding the distribution of those severe natural phenomena in both space and time; providing information on meteorology affecting the atmospheric dispersion of radionuclides under hypothetical accident conditions, and developing new information on alternative siting concepts, such as floating and underground sites.

Geology and Seismology

Geology and seismology research concentrates on regional geology and seismology. It includes intensive study of areas in the eastern U.S. where large earthquakes have occurred and of the distribution of faulting and earthquakes in parts of the western U.S. which may offer potential nuclear sites. Studies of faulting processes and of methodologies for dating movements on faults are also included. For example, NRC supported a U.S. Geological Survey (USGS) paper in 1978, summarizing different aspects and the state of knowledge of the earth sciences bearing on causes of a large earthquake that occurred near Charleston, S.C. in the year 1886.

Seismic network coverage also increased slightly in the Northeastern U.S. and St. Lawrence Valley. Patterns of small earthquakes (magnitude 2 to 4, Richter scale) which are emerging resemble those of historic earthquakes based on records of the past 280 years. This is potentially significant since it is assumed, but has not been demonstrated, that the locations of large events can be predicted from knowledge of strain-release patterns defined by small events ('microearthquakes').

The NRC supports several State and university programs and some USGS research in a 200 mile radius of New Madrid, Missouri, where the largest recorded eastern U.S. earthquakes occurred in 1811 and 1812. The most significant result in 1978 was the identification of strongly linear patterns of small earthquakes which almost certainly define one or more buried faults. The rate of earthquake activity is much greater here than in any other eastern U.S. earthquake zone, with the seismic energy released annually exceeding that from all other eastern zones combined.

A region of moderate to low level seismic activity extends through Oklahoma, Kansas and Nebraska, then along a geophysical feature (the Midcontinent Gravity Anomaly) to Minnesota. Results of NRC-sponsored research to date include compilations of seismic, geological and geophysical data from a very widely spaced network of seismic monitoring stations which has recorded several small earthquakes in the magnitude 2.5 - 4.5 Richter scale.

A study of high quality instrumentation records, not previously analyzed for local earthquakes, was undertaken jointly with the U.S. Army Corps of Engineers in response to a 1977 recommendation by the ACRS. No results were available at the close of the report period; however, past analyses of areas around dam sites demonstrated that many previously unknown events can be located, and some of these probably can be associated with known faults.

NRC geotechnical engineering studies in 1978 resulted in the publication of detailed reports on the properties of subsurface materials at accelerograph station sites, the records from which are widely used in seismic analyses of nuclear power plants.

Meteorology and Hydrology

Severe Storms. The project to upgrade tornado data, in which existing intensity data was reevaluated for all U.S. tornadoes reported since 1950, was completed in 1978. Comparison and reconciliation of this new data to data on file at the University of Chicago has been initiated, and should provide the best tornado records available. In other NRC storm research, two studies were initiated to better quantify tornado-structure interactions. One involves measurements of pressures on scale-model structures (containments, cooling towers, etc.) from simulated tornado vortices. The other attempts to measure maximum windspeeds in the type of multiple vortex associated with the most severe tornadoes. Preliminary attempts to define the probability of tornado windspeeds as a function of geographic location also were undertaken in late 1978.

Flooding. NRC supports research in the fields of hurricanes and tsunamis (tidal waves) because they are significant to site selection and plant design. An NRC report, to be issued in 1979, reassesses the history of hurricanes in the Gulf of Mexico and in the Western Atlantic Ocean to determine the probable maximum hurricane. The University of Hawaii initiated a study to simulate numerically the Hawaiian tsunami of November 1975. (For detailed descriptions of these studies, see pp. 167-8, 1977 Annual Report.)

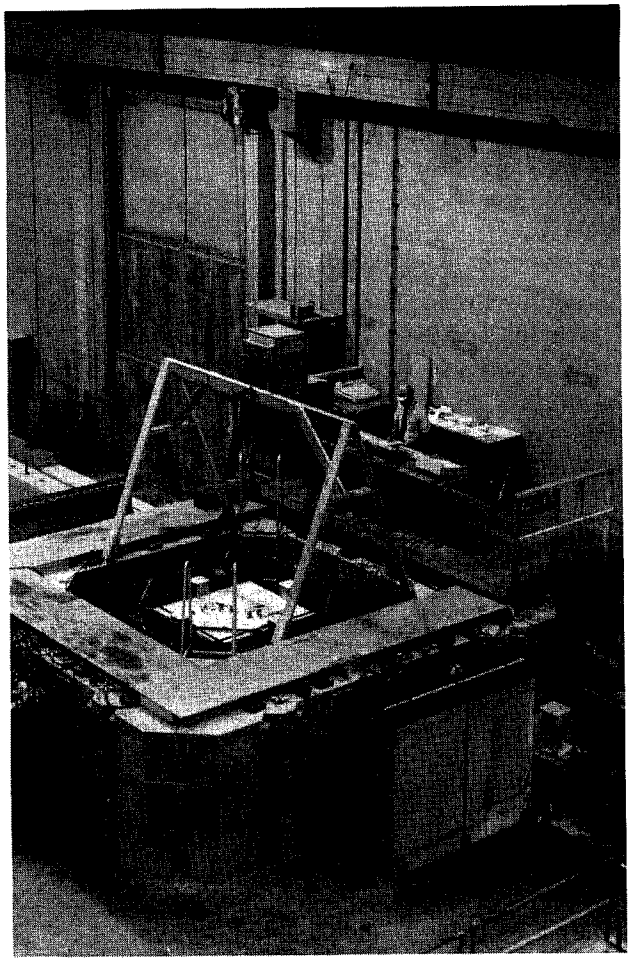
Atmospheric Diffusion. Some fifty full-scale field experiments were completed at the Rancho Seco power station in California and at the Idaho National Engineering Laboratory, to determine dispersion of suspended particles in the airflow patterns around nuclear complexes. Field measurements were initiated to determine vertical diffusion parameters. These incorporate "lidar" technology (similar to radar but based on light waves rather than radio waves) and vertical sampling techniques.

Mechanical Engineering Research

This new activity in reactor safety research addresses such diverse fields as seismic effects on LWR's, including the dynamic loads involving both seismic and loss of coolant impacts; prevention of and protection against the whip-action of pipes; the performance of snubbers and other restraining devices, and impacts on pump and valve operability. The testing and analysis of mechanical components and the modeling and scaling of systems behavior will be part of this research. A new program in this area was initiated in February, with the objective of developing mathematical models to predict the probability of seismically induced radioactive releases from power plants.

Structural Engineering Research

Another new area is Structural Engineering (SE) Research, aimed at establishing methodologies to assess quantitatively the safety



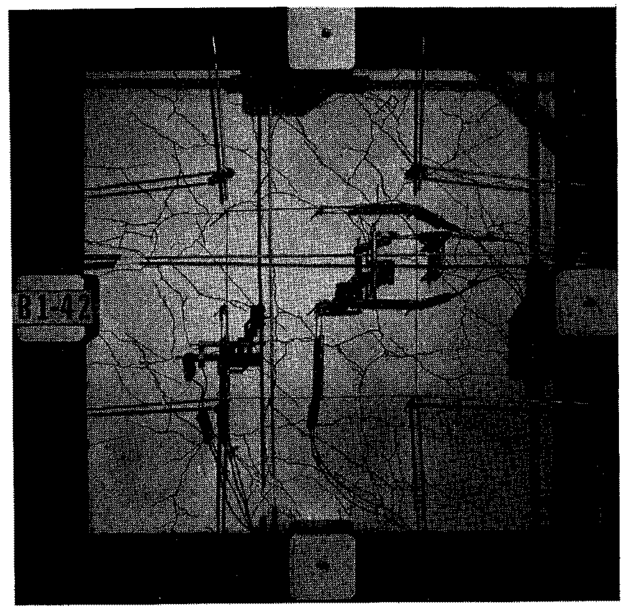


Photo at left shows a testing frame at the Portland Cement Association (PCA) Concrete Laboratory, Skokie, Ill., during a stress test on a large specimen of reinforced concrete, similar to that used in reactor containments. Tests such as this are used to permit assessment of shear force transfer in a biaxial tension field, subjected to external shear loading. Actual-size (#18) reinforcing bars are used, and loads up to 200 tons can be applied to each bar by the hydraulic rams shown. The photo above shows the instrumented concrete specimen after stress testing.

of nuclear reactor plants. This program devotes a major effort to the study of plant structures during and after earthquakes, in a comprehensive program to develop mathematical models that predict the probability of radioactive releases from seismically induced events. Other SE programs include studies of the behavior of reinforced concrete structures subjected to shear and biaxial tension, of tsunamis and hurricanes, and of methods of seismic qualification.

Fuel Cycle, Environmental and Waste Management Research

NRC's Fuel Cycle, Environmental and Waste Management research is designed to develop or improve predictive models and confirm basic data related to the operation of fuel cycle facilities, the transportation of radioactive materials, the impacts of routine reactor operations, and the disposal of radioactive waste.

FUEL CYCLE RESEARCH

The decision to defer commercial recycle of uranium fuel announced by President Carter in 1977 led to a shift in NRC fuel cycle research priorities. During the year, efforts were directed principally toward the front end of the fuel cycle, i.e., the mining and milling of uranium ore and commercial fuel fabrication.

In 1978, measurements of radon gas released in mining and milling operations in New Mexico and of the transport of particulate material and the leaching of radioisotopes from tailing piles were undertaken. At the end of the year, data and models were being developed to describe the effects of wind and rain on the movement of mill-tailing residues and their impact on man. Studies of occupational exposure to workers in milling activities were done, based on measurements of ore dust levels and on deposits of uranium and radium in lungs and bones. None of the workers examined had more than sixty percent of the maximum permissible lung burden recommended by the International Committee on Radiation Protection. Other programs were aimed at testing the effectiveness of respirators designed for protection against airborne radioiodine, and of plant air filters.

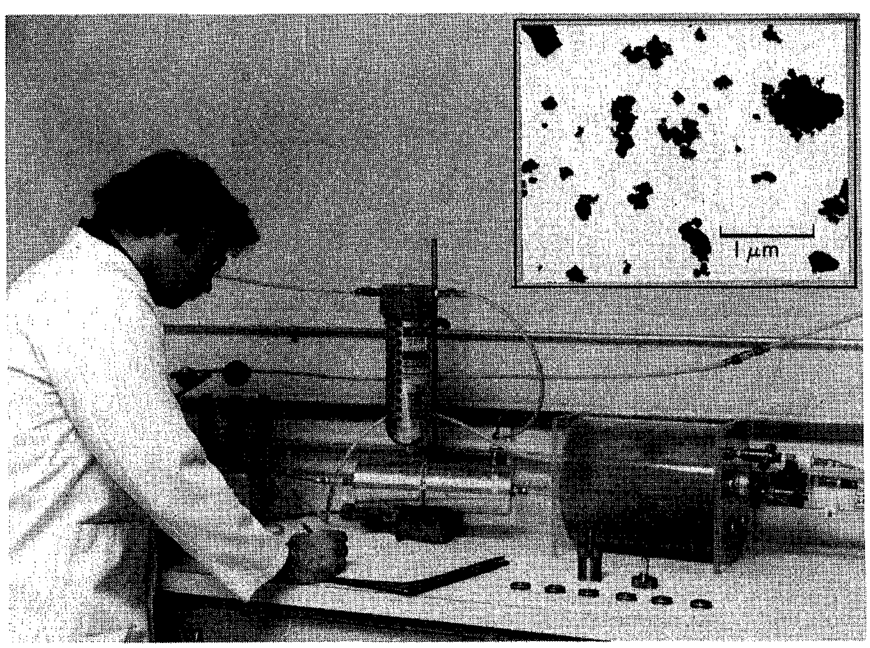
A milestone in fuel cycle research in 1978 was the Commission certification of a plutonium shipping package designed to withstand aircraft crashes. (See pp. 179 and 180, 1977 Annual Report, for background on this project.) Also during the year, development and validation continued on computer codes to predict the structural response of large shielded shipping casks to effects such as the shocks involved in railcar coupling operations, and to vibrations, punctures, etc. At year's end, experiments were being performed to quantify the potential release of radioactive materials from damaged shipping packages.

ENVIRONMENTAL RESEARCH

Radiation Dosimetry and Health Effects. Radiation dosimetry research in 1978 was directed at improving calculations of absorbed radiation doses to internal organs, accounting for the effects of age and size of exposed persons. As noted above, occupational exposures in uranium milling and fuel fabricating operations were assessed, as were radiation exposures to construction workers and power plant operating personnel.

The effects of radioiodine on people exposed to the radionuclide in childhood as part of a medical diagnostic procedure also were studied as part of a continuing investigation conducted jointly by the NRC and the Department of Health, Education, and Welfare.

Ecological Impact Studies. Chlorine toxicity and its effect on fish were studied to determine whether fish would exhibit natural avoidance mechanisms and if these might reduce the effects of chlorine on fisheries in the vicinity of nuclear reactor cooling systems. The University of Washington researchers who conducted the study for the NRC reported (UW/NRC-9) that some avoidance response had been observed. Another study of areas near reactor cooling systems, this one conducted by Oak Ridge scientists, dealt with the predation by sauger (a variety of pikeperch) on threadfin shad, and the effect of water temperature on that phenomenon. The shad are important food fish in the Southeastern United States. In studies of the impact of cooling water intake structures on striped bass and white perch in the Hudson River, emphasis was on the population dynamics of the fish species. This work has contributed to impact statements for Indian Point Power Stations.



Oxides of uranium and plutonium found in glove boxes at various fuel fabrication facilities are collected and sent to the Lovelace Inhalation Toxicology Research Institute in New Mexico. There, using the equipment shown at right, the dry powders are reconverted to aerosols and used in inhalation exposures of laboratory animals. The program includes studies of the behavior of actual fuel-material aerosols (deposition, retention, dosimetry patterns, etc.) and of their biological effects on the animals. Inset is an electron micrograph showing the nonuniformity of size and shape of such aerosol particles, features which may produce biological results different from those of more commonly used "idealized" laboratory aerosols.

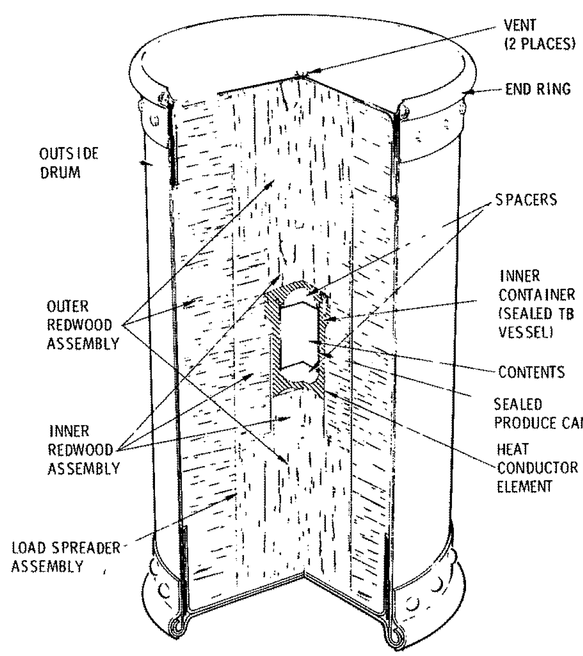
Socioeconomic Impacts and Regional Siting. NRC studies of the effects of nuclear power plant construction on labor force mobility and of the socioeconomic impacts of power plant siting, construction and operation at two power plants resulted in publication in 1978 of a report, "A Post-Licensing Study of Community Effects at Two Operating Nuclear Power Plants" (ORNL/NUREG/TM-22). Also during the year, research was undertaken on a methodology incorporating regional considerations in power plant siting, and Phase I of that research was completed for the New England region. The study provided for an integrated siting effort involving six states.

Environmental Dispersion and Effluent Monitoring. The transport of radionuclides, as sediments and in soluble forms, was studied using Cattauraugus Creek in northwestern New York. Samples at several places along the course of the stream, under all conditions of flow and in all seasons, are being collected and analyzed. By the end of fiscal year 1978, radionuclide concentrations, stream characteristics and water quality measurement parameters had been determined and reported. Plume dispersion, transport of airborne effluents and the influence of cooling towers on the generation of severe storms, as well as the important visual and aesthetic aspects of plant design also were being examined as 1978 ended. Although narrowly directed to licensing needs, this program provides independent, broadbrush assessments of environmental systems and considerations for use in evaluating power plant siting, construction and operation.

WASTE MANAGEMENT RESEARCH

Waste management research became a separate program in 1978, directed at providing the information base needed to support the licensing of high-level-waste repositories, shallow-land burial site and uranium mill tailing piles. The data base includes both empirical data and computer models for predicting the long-term behavior of radionuclides stored in deep geologic media and in shallow or open sites, as well as analyses of alternative methods of treating, packaging, and isolating wastes.

A high-level waste research program (see "Licensing Procedures for Repositories," Chapter 5) was undertaken in 1978 to provide tested analytical models and measured data for use in the establishment of standards and in licensing assessments of deep geologic waste repositories. Studies of the long-term durability of glass/ceramic waste forms, waste/container/rock interactions, and the movement of radionuclides in rock/water environments also were started.



This 500 pound Plutonium Air Transportable (PAT-1) package, shown after impact at 300 mph against an unyielding target, is the type certified by the Commission for air shipment of plutonium. The package consists of a stainless stell containment vessel inside a protective redwood over-pack, all encased within a multi-walled stainless steel drum, as shown in diagram on right.

In low-level waste management studies begun in 1978 NRC is assessing the migration of radioactivity at the shallow-land burial sites in Maxey Flats, Kentucky, and West Valley, New York, toward developing better siting criteria and standards for such sites. The characteristics of various liquid low-level wastes and of wastes which have been solidified prior to deposit also were being studied, and it is hoped that the results will lead to improved low-level waste forms and packaging. Another effort underway at year's end was aimed at finding better ways to predict the environmental impacts of waste disposal facilities and to evaluate alternatives to shallow-land burial for low-level wastes.

Efforts to develop radiological release information for assessing the environmental impact of uranium milling have produced data useful in evaluating alternative means of containing and stabilizing mill tailings.

Safeguards Research

In 1978, the Safeguards Research Program work on analytical methods and computer-codes for use in rulemaking and licensing, and on the design of safeguards system and information system designs continued. Highlights during the year included the demonstration of methods for evaluating physical protection and material control safeguards, the final design of the Integrated Safeguards Information System (ISIS), and the initiation of a study of reactor design for sabotage protection. Other research projects were concerned with improvement of nuclear material measurements and material control inspection.

Physical Protection

Fixed-Site Protection. Research by Sandia Laboratories (Albuquerque, New Mexico) into methods for evaluating fixed-site physical protection systems in 1978 resulted in a number of computer codes and a related data base, including a methodology for identifying vital reactor areas, using computerized generic fault trees, and the application of these fault trees to specific reactor sites. The Safeguards Automated Facility Evaluation (SAFE) model, which evaluates overall physical protection, including adversary paths, was exercised in a feasibility application by NRC staff. Work also was completed on the first phase of a detailed guard-adversary model and code which will be used both to calibrate the less-detailed SAFE methodology, and in connection with guard-tactics simulations, once these are developed.

Transportation Protection. Sandia Laboratories (Livermore, California) has concentrated on methods to evaluate the physical protection of nuclear materials in transit. During 1978, simulated armed attacks were staged to examine safeguards issues associated with road transportation systems. These tests involved such parameters as the adding of vehicles, guards and armor, and the employment of alternative tactics or equipment to improve convoy security. A tactical board game, called "AMBUSH," was designed to "war-game" similar parameters. AMBUSH is portable and relatively inexpensive, hence, it may be useful as a training device for transportation guards.

Material Control and Accounting

Lawrence Livermore Laboratory in 1978 demonstrated procedures for use in evaluating nuclear material control and accounting systems. In one, graphs and fault trees were used to analyze material diversion from an advanced design reprocessing plant. Another was initiated for a scrap recovery plant—a facility of less advanced design. Two important codes were designed at Livermore: one to simulate material control systems, the other to determine diversion possibilities in the complicated pipe system of a fuel facility. A concept for incorporating the dollar costs and social consequences of safeguards into such codes was also developed.

Safeguards Information System

The requirements for and progress toward achieving an Integrated Safeguards Information System (ISIS) were defined in 1977 (see p. 178, 1977 Annual Report), and in 1978 Boeing Computer Systems, Inc., defined the general design of such a system. Boeing analyzed various hardware/software design concepts to identify the alternative most appropriate for ISIS, giving consideration to such factors as cost, equipment requirements, performance features (reliability), usability features (convenience, response times, etc.), data security, training requirements, and maintenance. From this analysis a configuration was defined which included a main computer at NRC headquarters, with secure communication to terminals at NRC regional offices and possibly at licensed facilities. A cost-benefit analysis of ISIS alternatives was also completed. At the end of the fiscal year, the information was being considered by the staff for possible design implementation.

Reactor Design for Sabotage Protection

A project was initiated at Sandia Laboratories in 1978 to study design alternatives for nuclear power plants which could improve their inherent protection against sabotage. The study will characterize a baseline plant, typical of current design. Design features and damage control options representing potentially useful departures from the baseline will be identified, and these changes, along with sample physical protection systems, will be integrated in preliminary reference designs. These, in turn, will be evaluated, and the more promising options identified. A final reference design incorporating the options will then be analyzed to assess their effect on safeguards effectiveness, and on plant cost, operability and safety.

Risk Assessment Research

NRC's Probabilistic Analysis Staff (PAS) was expanded in December 1977, and given the mission of encouraging the use of quantitative risk assessment in regulatory decision-making. Risk assessment techniques are used to identify the relative importance of various contributors to potential accident risks from all elements of the nuclear fuel cycle.

In fiscal year 78, research was performed in the areas of development of quantitative risk assessment methodology, reactor risk assessment and licensing support, fuel cycle risk assessment, data analysis, emergency planning, and training programs. Work in these areas is discussed below.

A separate activity, which was discussed briefly in the 1977 Annual Report (see p. 181), and one which will affect NRC's risk assessment research program as a whole, is the work of the NRC Risk Assessment Review Group, headed by Dr. Harold W. Lewis. The final report of this group is discussed in the box on the next page.

Quantitative Risk Assessment

Programs to assess the risks to nuclear plants from fires and floods were initiated in 1978. Fires in nuclear plants are being analyzed to characterize their statistical behavior, and this information will be used in constructing models to evaluate system failure probabilities, accident probabilities, and, finally, accident risks. In the flood program, using statistical techniques which incorporate both actual flood data and scientific and engineering insights, system models will be developed to examine the effects of various levels of flooding on vital plant systems.

Risk assessment methodology research also includes work on several computer codes. The FRANTIC code (see p. 180, 1977 Annual Report) was extended to incorporate the effects of plant-to-plant variations and data uncertainties in component failure rates. The code will be used to evaluate system models in risk evaluations. In March, NUREG-0258 was published describing OCTAVIA (p. 180, 1977 Annual Report), a computer code used to compute pressure vessel failure probabilities for currently operating PWRs. The analysis permitted a more rigorous and quantitative confirmation of initial licensing decisions to reduce the frequency and maximum pressure of overpressure events.

NRC also supported development, by Sandia Laboratories and the Massachusetts Institute of Technology, of two fault tree evaluation codes, SETS and PL-MOD, respectively, for automated qualitative and quantitative evaluation of fault tree models for nuclear safety systems.

The computer code for Calculations of Reactor Accident Consequences (CRAC), developed for the Reactor Safety Study, was analyzed to determine uncertainty in the results predicted by the code. A full sensitivity study of the input data was completed; response surface techniques are being used to determine critical parameters.

RISK ASSESSMENT REVIEW GROUP

A new perspective on the capabilities and limitations of quantitative risk assessment techniques was presented late in the fiscal year with the report (NUREG-CR-0400) by NRC's Risk Assessment Review Group, appointed in 1977 to evaluate the Reactor Safety Study, (WASH-1400), known informally as the "Rasmusen Report," (See p. 181, 1977 Annual Report.) The Group completed its review the first week in September, and held public briefings on the results with the Commission and the Advisory Committee on Reactor Safeguards.

The review group was formed because of continuing public debate concerning the final report of the Reactor Safety Study, published in October 1975. Following an exchange of letters with Congressman Morris K. Udall, Chairman of the House Subcommittee on Energy and the Environment, concerning NRC's position on the study's conclusions, the Commission appointed a panel of seven distinguished scientists under the Chairmanship of Professor Harold W. Lewis of the University of California, Santa Barbara. Other members were Dr. Robert J. Budnitz (Lawrence Berkeley Laboratory, University of California), Dr. Herbert J. C. Kouts (Brookhaven National Laboratory), Dr. Walter Loewenstein (Electric Power Research Institute), Dr. William D. Rowe (Environmental Protection Agency), Dr. Frank von Hippel (Princeton University) and Dr. Fredrik Zachariasen (California Institute of Technology). Dr. Budnitz is presently on leave from the University of California and has been serving (since August 1978) as Deputy Director of the NRC's Office of Nuclear Regulatory Research.

The charter of the group had four basic elements:

- To clarify the achievements and limitations of WASH-1400.
- To assess the peer comments thereon, and the response to those comments.
- To study the present state of such risk assessment methodology.
- To recommend to the Commission how (and whether) such methodology can be used in the regulatory and licensing process.

Beginning in August 1977, the Review Group held 12 public meetings and heard the testimony of many individuals — representing the "nuclear community, the NRC, the critics of both, and others" — concerning the Reactor Safety Study and risk assessment methodology.

A summary of the group's findings, published as the introduction to its report, reads as follows:

"We find that WASH-1400 was a conscientious and honest effort to apply the methods of fault-tree/event-tree analysis to an extremely complex system, a nuclear reactor, in order to determine the overall probability and consequences of an accident. We have reviewed the methodology, the data base, the statistical procedures, and the results.

"We have found a number of sources of both conservatism and nonconservatism in the probability calculations in WASH-1400, which are very difficult to balance. Among the former are inability to quantify human adaptability during the course of an accident, and a pervasive regulatory influence in the choice of uncertain parameters, while among the latter are nagging issues about completeness, and an inadequate treatment of common cause failure. We are unable to define whether the overall probability of a core melt given in WASH-1400 is high or low, but we are certain that the error bands are understated. We cannot say by how much. Reasons for this include an inadequate data base, a poor statistical treatment, an inconsistent propagation of uncertainties throughout the calculation, etc.

"Also, both the dispersion model for radioactive material and the biological effects model should be improved and updated before they are applied in the regulatory and licensing process.

"We do find that the methodology, which was an important advance over earlier methodologies applied to reactor risks, is sound, and should be developed and used more widely under circumstances in which there is an adequate data base or sufficient technical expertise to insert credible subjective probabilities into the calculations. Even when only bounds for certain parameters can be obtained, the method is still useful if the results are properly stated. Proper application of the methodology can therefore provide a tool for the NRC to make the licensing and regulatory process more rational, in more properly matching its resources (research, quality assurance, inspection, licensing regulations) to the risks provided by the proper application of the methodology. NRC has moved somewhat in this direction, and we recommend a faster pace.

"Among our other findings are the well-known one that WASH-1400 is inscrutable, and that it is very difficult to follow the detailed thread of any calculation through the report. This has made peer review very difficult, yet peer review is the best method of assuring the technical credibility of such a complex undertaking. In particular, we find that the Executive Summary is a poor description of the contents of the report, should not be portrayed as such, and has lent itself to misuse in the discussion of reactor risks.

"In summary we find that the fault-tree/event-tree methodology is sound,* and both can and should be more widely used by NRC. The implementation of this methodology in WASH-1400 was a pioneering step, but leaves much to be desired.

*"One of us (F.v.H) is doubtful that the methodology can be implemented so as to give a high level of confidence that the probability of core melt is well below the limit set by experience."

Reactor Risk Assessment and Licensing Support

Efforts continued in 1978 to expand the application of risk methodology to a broader spectrum of LWR safety issues, and to apply the methodology and related engineering insights to issues of immediate concern to the licensing staff.

The category of safety issues included: sensitivity studies on physical phenomena associated with potential core meltdowns to aid in setting priorities for meltdown accident research; applying fault tree and event tree methodology to additional LWR design concepts to broaden engineering insights; evaluating potential risks in LWR accidents which do not lead to core melting; assessing the potential risk to the public from possible radioactive contamination of the hydrosphere as a result of core melt accidents; and assessing the impact of external events, such as transportation accidents, on nuclear plants.

The NRC also completed its evaluation of the effects of containment venting and filtering on LWR risk, and two reports were issued: NUREG-CR-0318 ("Effect of Containment Venting on the Risk of LWR Meltdown Accidents"), and NUREG-CR-0165 ("A Value/Impact Assessment of Alternate Containment Concepts"). Further efforts in this area were continuing at year's end as part of the new LWR Safety Improvement Research Program.

In the category of direct licensing support, certain technical specifications for plants employing digital computer reactor protection systems were evaluated to determine appropriate outage and testing limits; test intervals for containment spray pumps were assessed and an applicant's assessment of seismic risk to a nuclear power plant was reviewed. An evaluation of generic safety issues, from a risk perspective, also was undertaken to aid in setting priorities and allocating resources.

Fuel Cycle Risk Assessment

Fuel cycle risk assessment develops methodologies for assessing the risk (or determining significant contributors to the risk) of nuclear fuel cycle activities other than reactor operations. In 1978, a primary need was for a risk methodology applicable to the deep geologic isolation of high level waste in bedded salt, and two reports covering work at Sandia Laboratories, New Mexico, were being prepared at year's end: "Risk Methodology for Geologic Disposal of Radioactive Wastes" (NUREG-CR-0458) and "Risk Methodology for Geologic Disposal of Radioactive Waste: Sensitivity Analysis Techniques" (NUREG-CR-0394). The first report defines a "reference repository" for the purposes of analysis and the formulation of computer models to simulate the movement of radionuclides from the repository area to man. Late in the year, the models were applied to the reference site to determine their usefulness as tools for risk analysis and their possible use in the licensing process. Sensitivity analyses, described in the second report, will be aimed at determining the significant contributors to risk.

Other programs begun in 1978 were designed to develop a methodology for assessing the risk associated with spent fuel processing and to examine alternatives for the management of radioactive gases emitted by fuel cycle facilities. At the end of the report period, a program to develop a methodology to assess risks associated with "away from reactor" storage of spent fuel was in the planning stage. Efforts to assess the risk associated with shallow land burial of low level wastes, decommissioning of nuclear facilities, front end of the fuel cycle activities and ocean bed isolation of high level waste also were being considered.

Data Analysis

A significant 1978 data analysis effort to improve the basis for risk assessments involves the analysis of more than twelve thousand Licensee Event Reports (LERs) accumulated over roughly four hundred-reactor-years of LWR operating experience. The effort includes extraction of component failure rates, analysis of common cause/common mode failure statistics, and characterization of human errors, including development of gross error rate statistics. To obtain information not available from LERs (e.g., partial failures, wearout behavior, downtime distributions), a program was undertaken which uses detailed maintenance and failure information from nuclear plant log books as the basis for statistical analysis. In addition to providing an improved basis for risk assessment, these programs will provide summaries of post-failure history for licensing use, and will establish a basis for evaluating and incorporating data from the Nuclear Plant Reliability Data System (NPRDS).

Emergency Response Planning

During 1978, NRC completed a program of research on emergency response planning which used and built on the information and models developed for the Reactor Safety Study. Under NRC contract, researchers at Sandia Laboratories published an evaluation of offsite emergency protective measures (evacuation and sheltering) which might be used for postulated core meltdown accidents (SAND-78-0454), and an accident scenario book (SAND-78-0269) describing a spectrum of core meltdown accidents. The reports will be offered to emergency response planning agencies as a basis for improved realism in field exercises.

Training Programs

A critical need for training in probabilistic and risk assessment methodologies and their application was met when a total of ninety staff members completed five courses presented by the PAS staff on such subjects as Systems Reliability and Safety, Bayesian Statistical Methods, and Reliability and Data Analysis.

Improvement of Reactor Safety

As noted earlier, in November 1977 the Commission established a review group to implement an amendment (P.L. 95-209) to the Energy Reorganization Act of 1974 which directs NRC to "develop a long-term plan for projects for the development of new or improved safety systems for nuclear power plants." The Congressional intent behind this effort is "the improvement of reactor safety and not the enhancement of the economic attractiveness of nuclear power versus alternative energy sources." The Act requires that the plan be updated annually and submitted to the Congress.

In April 1978 NRC submitted to Congress a "Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants" (NUREG-0438), which presented an evaluation of concepts proposed to improve safety and recommended a three-year, $14.9 million research program. The objectives are to determine the feasibility of achieving particular improvements in safety, to evaluate the safety significance of proposed changes and to propose regulatory requirements where implementation is determined to be desirable, without preparing detailed designs.

NUREG-0438 recommended that seven research topics be pursued:

Alternate containment concepts—especially vented containments—to mitigate the consequences of postulated core meltdown accidents. This is accomplished by improving control of the release of radioactivity to the environment.

Alternate decay heat removal concepts—especially add on, bunkered systems—to reduce the probability of core meltdowns by increasing the reliability of systems designed to remove heat from the reactor core after fission ceases.

Alternate emergency core cooling concepts—to develop simpler and more clearly demonstrable systems to prevent fuel overheating in the event of pipe rupture.

Improved human performance—to reduce the risk of human error by reducing test and maintenance errors and by helping operators make correct decisions during accidents.

Advanced seismic designs—to reduce the vulnerability of plants to earthquakes by decoupling or strengthening components against seismic forces.

Scoping studies of other concepts—to determine their potential for improving safety and to assess the need for further research. The studies address protection against sabotage, better ways to monitor the condition of the plant, new siting concepts, and ways to reduce occupational exposure without increasing public risk.

Improved evaluation methodology—to assist in making more rigorous and thorough assessments of the values and impacts associated with these concepts, and in planning future safety research programs.

In 1978 NRC developed detailed work scopes, evaluated proposals and selected contractors for the three highest priority tasks; alternate containment, decay heat removal, and improved methodology. Work on the programs was initiated in December 1978, using internally reprogrammed fiscal year 1979 funds, and resources to implement the plan fully were requested in the fiscal year 1980 budget. The plan remains as originally described. Any substantive changes will be included in the annual status reports.

NRC coordinates its program with the Department of Energy to minimize duplication of effort. DOE places considerable emphasis on economic incentives for increasing the safety and availability of nuclear power plants. DOE is also willing to enhance the detail of some conceptual designs, advanced by NRC which would permit definitive engineering and safety assessments by the NRC staff.

12

Informing and Involving the Public

During fiscal year 1978, the NRC took additional steps to increase the flow to the public of information regarding nuclear regulation; to open the regulatory safety review process to public observation, and to foster meaningful participation in NRC proceedings by members of the public and State and local governments. These steps included:

- Initiation by the NRC staff of open, informal meetings with license applicants to provide the public an opportunity to observe the early stages of the NRC staff review of facility license applications.
- An overhaul of NRC's Rules of Practice governing formal proceedings to expand opportunities for participation by members of the public and State and local governments.
- Start of an automated document storage and retrieval program which should facilitate the flow of information to the public.
- A task force study regarding an expansion in the number of internal documents made available in NRC public document rooms.

In other actions on matters of public concern, the Commission (1) established formal agency-wide procedures for prompt notification of the licensing and appeal boards and the Commission of new information considered by the staff to be relevant to any licensing proceedings, and (2) began development of agency-wide procedures for the expression of, and response to, differing professional opinions from the staff which will go beyond NRC's current "open door" policy for employees.

These and other NRC activities aimed at informing and involving the public in nuclear regulatory matters, providing for a freer flow of information both internally and externally, and responding to Congressional concerns, are described below.

INFORMING THE PUBLIC

Hundreds of actions are taken each year by the NRC to inform the public directly or make information available regarding nuclear regulation. These take the form of announcements and *Federal Register* notices; publication of reports, providing access to documents in localities across the country; holding meetings and workshops with the concerned public and State and local representatives on issues of widespread interest; responding to public and Congressional inquiries; and opening of Commission, staff and advisory committee meetings to public observation as well as the many public hearings on rulemaking and licensing carried on in the normal course of the agency's business.

As the most direct means of communicating to the public, the NRC issues announcements on a wide range of topics from headquarters and the five regional offices to some 5,000 members of the news media, industry, the scientific community and the general public.

Making Documents Available

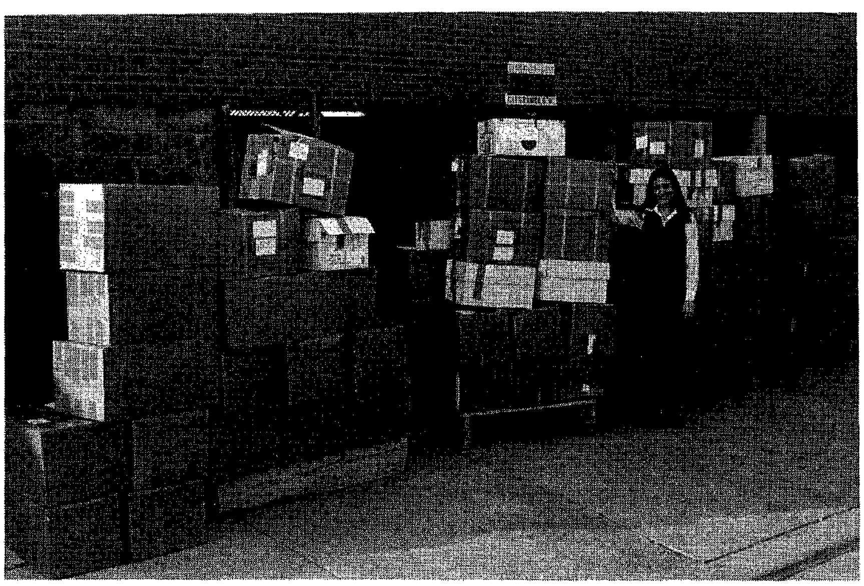
Most NRC reports and documents may be obtained by the public through the National Technical Information Service (NTIS), Springfield, Virginia, or the Government Printing Office in Washington, D.C. Available documents are listed in a monthly issuance by NRC and also in Government Research Abstracts, published by NTIS, and Atomindex, published by the International Atomic Energy Agency. These documents also are made available to Government Depository Libraries.

The NRC maintains its principal Public Document Room (PDR) at 1717 H Street, N.W., Washington, D.C., and has established more than 130 local public document rooms throughout the country. The local PDRs are typically located in libraries in cities and towns near proposed and actual nuclear plant sites, and contain detailed information specific to the nearby facilities which are either licensed or under regulatory review. (See Appendix 3 for list of all local PDRs.)

The main PDR in Washington contains about 800,000 documents — either produced by NRC or submitted to it for consideration — comprising some 13 million pages. The facilities afforded insluce reference assistance, copying services, and a microfiche reader/printer. During an average month in fiscal year 1978, visitors to the Washington PDR requested access to about 430 files. The PDR staff also responded to an average of 80 letters a month. More than 1,000,000 pages of documents and 17,000 microfiche cards were reproduced for the public during the year.

At year-end, a task force was completing a study of PDR operations with a view toward increasing the range of documents that will be made available to the general public.

During the year, the NRC took steps to apply the latest automated storage and retrieval technology to the entire range of regulatory documents. As projected, the system is expected not only to reduce the staff time required in technical studies and evaluations, but also to facilitate searches for materials in response to



License applications are submitted to NRC in 40 to 70 copies, depending on the type of permit applied for and the number of agencies involved in its review. In this photo taken at NRC's Bethesda headquarters, Carol Rossomondo of the Distribution Services Branch is shown beside a single delivery of safety analysis reports and other application documents — in 40 copies — for the Byron and Braidwood power plants in Illinois. A copy of each such document is automatically placed in NRC Public document rooms both in Washington and in the community near the plant involved.

Freedom of Information Act requests and other inquiries from the public. (See Chapter 14.)

Freedom of Information Act. Like other government agencies, the NRC is required under the Freedom of Information Act (FOIA) to make any identifiable record in its possession available on request to the public for public inspection and reproduction, unless the record requested falls within one of nine categories of exemption. Among the kinds of records exempted from FOIA requirements are information that is classified in the interest of national security or foreign policy; trade secrets and commercial or financial information; certain investigatory files; and certain interagency and intra-agency memoranda of a "pre-decisional" nature.

The NRC has, from the outset of the FOIA enactment, followed a liberal disclosure policy in releasing thousands of pages of documentation for public perusal. All material released as a result of FOIA requests is placed in the Headquarters Public Document Room, 1717 H St., N.W., Washington, D.C., where full public access is given to documents released to any individual. In addition, documents released under FOIA which pertain to a particular facility under NRC license or review are furnished to the appropriate NRC local public document rooms.

The large number of FOIA requests being received and the prompt response mandated by the Act call for a substantial commitment of staff for document searches and processing. During fiscal year 1978, the NRC received 358 FOIA requests, resulting in the release of almost 60,000 pages of material. More than 20,000 man-hours were expended by agency staff in meeting these requests, about half of which were devoted to answering requests from public interest groups.

The Privacy Act of 1974. This law, which became effective in 1975, provides that individuals have the right to determine the existence of agency records about themselves, to seek access to those records, and to have records corrected when they are not accurate, relevant, timely or complete for agency purposes. During fiscal year 1978, the NRC received 37 Privacy Act requests, most of which came from agency employees seeking access to personnel security related records about themselves.

INVOLVING THE PUBLIC

The Commission took additional steps during 1978 to facilitate more meaningful and practical involvement of the public in nuclear regulatory affairs, both informally and formally.

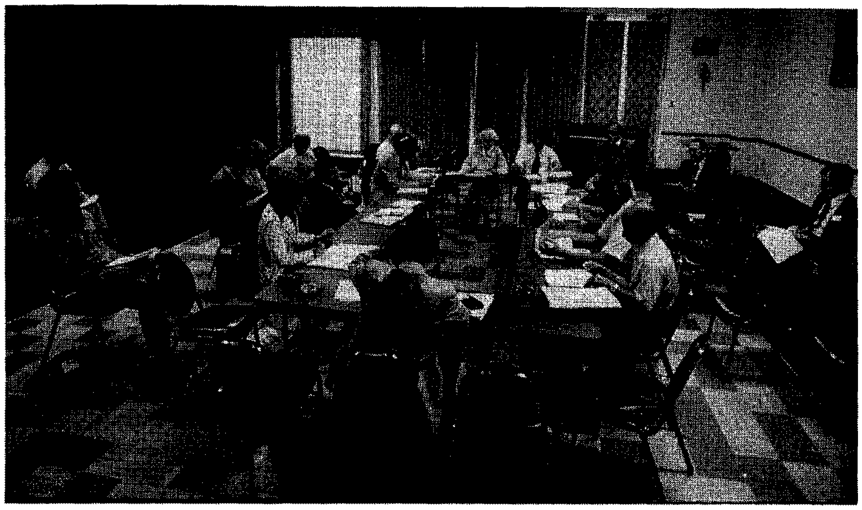
Informal Participation

While opportunities for formal public participation in nuclear regulatory proceedings have been provided from the beginning, and expanded from time to time, the NRC has sought practical ways to involve the public informally in its deliberations. For example, the NRC staff has sponsored conferences, workshops and regional meetings on public issues and other factors in its continuing development of criteria and regulations for managing nuclear wastes. The views received are taken into account in formulating regulatory policy and practices.

The NRC staff has, from time to time, conducted public meetings to assist in its consideration of standards on other issues of widespread interest. In April 1978, the Office of Standards Development sponsored a public meeting in Washington, D.C., to review and critique recent studies of the health effects of low-level radiation as a part of its development of future regulatory policy in this area. In July, NRC sponsored a public meeting and a workshop in New York City concerning findings of studies on the transportation of radioactive material through densely populated areas.

In other 1978 activities, regional workshops and meetings were conducted on decommissioning of nuclear facilities, physical security requirements for nuclear plants, and problems with steam generators; two public meetings were held on plans for developing improved safety systems for reactors; and a two-day seminar was conducted on the public hearing process. And 12 meetings to receive the public's views were conducted by the Commission-appointed Risk Assessment Review Group during its year-long study of the achievements and limitations of the Reactor Safety Study (see Chapters 1 and 11).

Staff Reviews Opened to Public. One of the recommendations of the NRC staff study of ways to improve the licensing process (see Chapter 2) was that public participation during



In 1978, NRC invited representatives of State governments, industry and the general public to participate in three regional workshops concerning policy on the decommissioning of nuclear facilities. The workshop scene left was in Philadelphia. Similar meetings occurred in September in Atlanta, Ga., and Albuquerque, N.M. Reports on the discussions have been made available in the NRC Public Document Room in Washington, D.C.

early stages of staff review be increased. In particular, it proposed arranging for NRC staff-applicant-intervenor meetings to take place in the vicinity of a proposed facility site. The formal hearing process has for some time provided for public participation, and early meetings near the site involving local officials and the public have usually been conducted during the environmental review. But, until recently, there had been little practical opportunity for members of the public near the site to become aware of, and understand, the NRC staff role in the important early stages of the safety review, before the commencement of public hearings.

Intervenors and other identified interested parties are invited to discuss safety and environmental questions with NRC staff and, as a matter of course, receive notices of all meetings between applicants and staff. Some ACRS subcommittee meetings are held in the vicinity of the site and have been well attended by the general public. Finally, the ASLB hearings are normally held in the vicinity of the proposed site and members of the public can make limited appearances to state their position and request information and explanation.

The use of informal meetings has significantly expanded the opportunities for public observance of and participation in the early non-adjudicatory stages of the licensing process. A Commission policy statement (June 28, 1978) approved and confirmed the practice of opening staff meetings with license applicants to all parties or potential parties to the proceedings.

The first opportunity for holding such informal meetings was the review of an application by Arizona Public Service Company for permits to construct two additional nuclear units at its Palo Verde site near Phoenix. The staff scheduled two meetings in the month of October 1978 to discuss the NRC licensing process, and the staff's review of environmental matters, safety matters related to the site, and safety matters related to the proposed reactors. The meetings were open to the public, with time set aside during or following the meetings to permit public comments or questions.

"Government in the Sunshine." During fiscal year 1978, the Commission opened two-thirds of its meetings to public observation in compliance with the Government in the Sunshine Act. That statute, which became effective on March 13, 1977, regulates the conduct of meetings of collegial agencies like the NRC, and makes their deliberative processes more accessible to the public.

The law requires the Commission to open all of its meetings to public attendance unless one or more of 10 exemptions applies. The exemptions are designed to permit closed discussion of matters such as classified information, proprietary commercial information, matters involving personal privacy, and issues in pending litigation. However, transcripts or recordings must be made of most closed meetings and are released to the public when appropriate.

The Commission's regulations carrying out the Act (Subpart C of 10 CFR Part 9) outline procedures for deciding whether to close a meeting, what records are to be kept, and other administrative details. They also specify that advance notices of meetings be published in the *Federal Register*, placed in the public Document

Room, and mailed to special lists and newspapers.

Since the effective date of the Sunshine Act, the Commission has expanded its conference room and installed audio and visual systems to assist the public, has made available copies of the papers to be discussed at open meetings, and has published a brochure describing how Commission business is conducted. These documents are available at each public meeting as well as in the Washington PDR. The Commission transmits an annual report to the Congress on implementation of the Act. The latest report, dated April 14, 1978, is available in the Public Document Room at 1717 H Street, N.W., Washington, D.C.

The Commission firmly supports the principles of open government enunciated in the Sunshine Act. Chairman Hendrie, in his testimony of August 4, 1978 before the Subcommittee on Federal Spending Practices and Open Government of the Senate Committee on Governmental Affairs, noted that the Commission has not only opened an increasing proportion of Commission meetings, but has also voluntarily chosen to go beyond the literal requirements of the Act to adopt policies that advance its purposes. For example, staff papers and documentation pertaining to the proposed issuance of export and import licenses are made available in the PDR; some Commissioners' correspondence is placed in the PDR; NRC regulations permit tape recordings of open meetings by attendees; radio and television coverage of licensing board hearings under certain conditions is being permitted on a trial basis; and, in several recent cases of general interest, the public has been permitted to attend Commission adjudicatory sessions that could have been closed under Exemption 10 of the Sunshine Act.

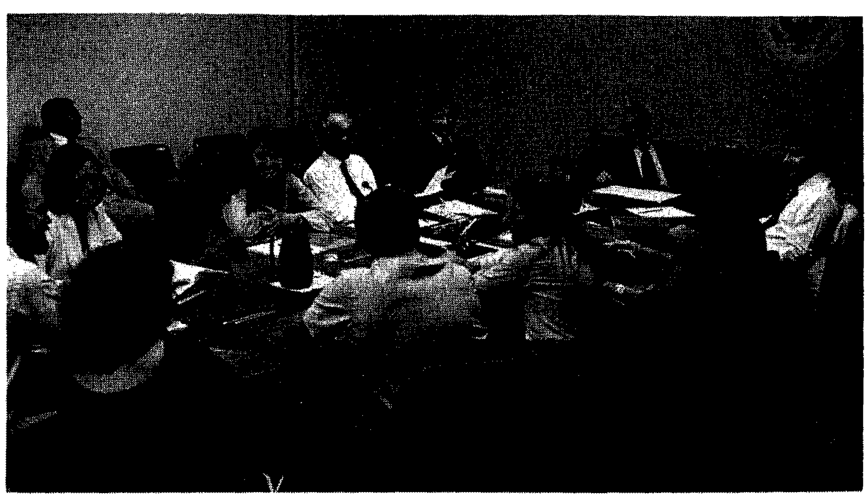
Formal Public Participation

NRC regulations provide for formal participation by members of the public as parties in rulemaking, licensing and other proceedings. Opportunities for hearings are indicated in the accompanying table.

Commission regulations require that a public hearing on each application for a major nuclear facility construction permit be conducted by an Atomic Safety and Licensing Board (see Chapter 13). Notice of such a hearing is published well in advance in the *Federal Register* and posted in a public document room near the proposed construction site, together with a copy of the application. Local newspapers also carry notice of the hearing. Interested persons or groups are invited to participate in the hearing by: (1) submitting a written statement at the hearing; (2) making an oral presentation at the hearing; or (3) petitioning the licensing board for the right to become an "intervenor" in the proceeding with full participatory rights, including cross-examination of other participants. Intervenors participate fully in prehearing conferences with other interested parties for the exchange of data and identification of issues in contention.

If the licensing board disallows a petition, the petitioner may appeal to the Atomic Safety and Licensing Appeal Board (see Chapter 13). In some instances, the Commission may rule on a

Faculty members and graduate students from eight colleges and universities met with NRC Chairman Joseph M. Hendrie on July 12, 1978, to discuss the responsibilities of the Nuclear Regulatory Commission. The group was in Washington, D.C., to participate in a symposium on energy sponsored by the Washington Center for Learning Alternatives. Included were representatives of schools in Maine, Massachusetts, New York, West Virginia, Virginia, Tennessee, Illinois and Minnesota.



petition. Ultimately a petitioner may seek a ruling in the appropriate Federal Court of Appeals and the Supreme Court of the United States.

The same rights and procedures for public participation apply to operating license hearings, with the difference that such hearings are not mandatory and need not take place unless requested by one or more interested parties.

To facilitate public participation, hearings of the licensing boards, with rare exceptions, are held in communities near each proposed facility site.

Enhancing Public Participation. On April 26, 1978, the NRC published the first major revision of its basic rules of practice (10 CFR Part 2) since they were restructured by the Atomic Energy Commission in July 1972. Key elements in the amendments, which became effective on May 26, are designed to enhance public participation in the review and hearing process for facility license applications and to improve coordination with states, counties and municipalities. Under the amended rules:

- Interested persons may make limited appearances at prehearing conferences.
- Interested cities, counties and local government agencies may participate in licensing proceedings without taking a position on the issues, a privilege previously accorded only to States.
- Interested States, counties, cities and/or agencies thereof may file proposed findings of fact and conclusions of law, exceptions to initial decisions, and petitions for Commission review.
- Procedures have been established for *amicus* participation in appeals before boards or the Commission.
- Motions for summary disposition are no longer limited to initial licensing proceedings.
- Licensing boards have authority to consolidate two or more proceedings for hearing.
- Joint hearings with States or other Federal agencies are authorized on matters of concurrent jurisdiction; however, NRC rules of practice may not be waived, and the action must be conducive to the proper dispatch of Commission business and the ends of justice. Joint hearings will be considered on a case-by-case basis.

ENHANCING INTERNAL COMMUNICATION

The Commission took important steps in 1978 to improve the flow of information within the agency, including (1) establishment of procedures to assure prompt notification of licensing boards of new information considered relevant to licensing proceedings, and (2) laying the foundation for agency-wide procedures to accommodate differing professional opinions among the staff.

New Board Notification Policy

As reported in the 1977 NRC Annual Report (pp. 187-188), the Commission became concerned—after learning that, in 1973, certain information relevant to the North Anna Power Station (Virginia) licensing action was known to the NRC staff for some time before it was made known to the licensing board involved—that potentially important data be transmitted to the appropriate board(s) as expeditiously as possible. The Commission admonished the staff (November 1976), stating that "the Licensing Board, the parties and the public have a right to be promptly informed of a discovery...before staff evaluation and regardless of whether the record is technically open." The staff was directed to ensure enforcement of the practice of reporting information to affected boards and parties routinely and promptly—even if that sometimes meant having to send the staff evaluation of the data separately and later.

In 1978, the Commission approved a policy which formalized the notification procedure and promulgated it agency-wide, and further steps were taken to assure that all staff personnel were aware of the obligation to keep the boards informed. On May 12, 1978, the NRC Executive Director for Operations directed every NRC office to set up procedures by which any professional staff member could convey information to one or more licensing boards. The guidelines prescribed that, in general, any information which the staff deemed significant enough to warrant a request that the applicant clarify or amplify the matter should be forwarded immediately to the board(s). In any case, any staff member was entitled and obliged to transmit to the board(s) any information which, to the best of the originator's knowledge, was new to the licensing record, potentially important to it, and relevant to one or more licensing proceedings then current before a licensing board, appeal board, or the Commission. The originator's supervisors would be apprised of the information to be transmitted and could request additional documentation or discussion of its importance and/or relevance, but in no case could a supervisor prevent or unreasonably delay the transmittal.

NRC offices developed and transmitted to all staff personnel formal board notification procedures, setting forth the essential policy considerations and steps to be taken to forward information to the board(s). Each office appointed a coordinator to be responsible for timely processing of information and keeping records of the transmittal (with any additions or comments by supervisory staff) and its final disposition. A panel of legal and licensing staff conducted training sessions in the procedure for NRC headquarters staff and then traveled to the five regional offices to give a similar orientation to the NRC field staff.

Differing Professional Opinions

In July 1978, the Commission released for public comment the results of a survey of policies and procedures used or considered by a number of Federal agencies, business corporations, professional societies and other private organizations for bringing differing professional views to the attention of management, and for appropriate management response (NUREG-0500).

The survey describes concepts that NRC plans to use to develop formal procedures for making known to management employees' opinions on any substantive matter within the agency's purview that differ from an existing policy or a proposed staff position on the matter. Comments were solicited from both NRC employees and the public for consideration in developing an agency-wide plan.

The survey identifies and discusses procedural steps that could provide for: making employee differences known to management, management response, alternatives if the employee is dissatisfied with the response, follow-up on resolution of the issue, and follow-up to ensure that the employee is not subjected to retaliatory actions. In addition, the survey describes criteria that could be used to judge the effectiveness — and perhaps the acceptability — of any mechanism designed to handle differing professional opinions.

The three successive Chairmen of the NRC have supported an "open door" policy for all employees extending up through the management chain to the Commissioners' offices, and to the independent Advisory Committee on Reactor Safeguards on appropriate safety matters. (See NRC Annual Report for 1976, pp. 199-201, and 1977, pp. 185-187.) However, Chairman Hendrie, in a memorandum transmitting the survey to all employees in July 1978, expressed general dissatisfaction with progress in the matter and asked for comments and suggestions in the effort "to make the 'open door' policy more of a reality both in concept and in practice."

By year-end, NRC had received a number of constructive comments from within and outside the agency.

CONGRESSIONAL OVERSIGHT

The number of hearings by the several Congressional committees exercising jurisdiction over NRC activities and hearings by other committees involving NRC continued to increase during 1978. During the fiscal year, all of the Commissioners, the Executive Director for Operations, and many of the senior staff participated one or more times in 36 days of hearings conducted by 14 Congressional committees or subcommittees. Chairman Hendrie testified at 24 hearings.

Topics requiring the most hearing days were authorization and budgetary matters (6), waste management (6), proposed licensing legislation (6), uranium mill tailings control (5), and export matters (3). The following list shows the date, committee and subject of each hearing:

10/13/77—House Committee on Public Works and Transportation, Subcommittee on Public Buildings and Grounds (NRC Building Consolidation)

10/13/77—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Staff Handling of North Anna Case)

1/25/78—Senate Committee on Governmental Affairs (An Act to Combat International Terrorism)

2/6/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (NRC Fiscal Year 1979 Budget and Fiscal Year 1978 Supplemental Requests)

2/7/78—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (Nuclear Licensing Reform Legislation)

2/8/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (NRC Fiscal Year 1979 Budget)

2/15/78—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (NRC Fiscal Year 1979 Budget and Fiscal Year 1978 Supplemental Requests)

2/27/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (Alleged Misrepresentation of Facts in Congressional Testimony of the Executive Director for Operations)

2/28/78—House Committee on Government Operations, Subcommittee on Environment, Energy and Natural Resources (Sheffield, Illinois Low-Level Radwaste Burial Site and GE Spent Fuel Storage Facility at Morris, Illinois)

3/1/78—House Committee on Appropriations, Subcommittee on Public Works (NRC Fiscal Year 1979 Appropriations Request)

3/22/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Nuclear Waste Management)

3/23/78—Senate Committee on Governmental Affairs, Subcommittee on Nuclear Proliferation, Energy and Federal Services (Nuclear Terrorism)

4/10/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Safety Research Programs)

4/11/78—Senate Committee on Environment and Public Works, Subcommittee on Public Buildings and Grounds (NRC Building Consolidation)

4/18/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (NRC Fiscal Year 1979 Authorization)

4/19/78—House Committee on Government Operations, Subcommittee on Environment, Energy and Natural Resources (Federal Radiation Protection Efforts)

4/25/78—Senate Committee on Appropriations, Subcommittee on Public Works (NRC Fiscal Year 1979 Appropriations and Supplemental for Fiscal Year 1978)

5/15/78—House Committee on Merchant Marine and Fisheries, Subcommittee on Oceanography (Ocean Disposal of Radioactive Materials)

5/22/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (Nuclear Siting and Licensing Act of 1978)

5/24/78—Senate Committee on Foreign Relations, Subcommittee on Arms Control, Oceans and International Environment (Export License Application XSNM-1060 — Tarapur Reactor)

6/6/78—House Committee on Science and Technology, Subcommittee on Environment and Atmosphere (Oversight: Federal Ionizing Radiation Research)

6/8/78—House Committee on International Relations (Tarapur Export License)

6/14/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Nuclear Waste Management)

6/20/78—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (Uranium Mill Tailings)

NRC officials appeared before 14 Congressional Committees during fiscal year 1978. Shown here in an appearance before the Subcommittee on Energy and Power of the House Committee on Interstate and Foreign Commerce, are, left to right, Commissioner Bradford, Commissioner Kennedy, Chairman Hendrie, and Commissioner Gilinsky.

6/27/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Uranium Mill Tailings)

6/28/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Nuclear Siting and Licensing Act of 1978)

6/28/78—House Committee on Science and Technology, Subcommittee on Fossil and Nuclear Energy Research, Development and Demonstration (Nuclear Waste Management)

7/10/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and Environment (Uranium Mill Tailings)

7/13/78—Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation (Nuclear Siting and Licensing Act of 1978)

7/20/78—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (Nuclear Siting and Licensing Act of 1978)

7/24/78—Senate Committee on Energy and Natural Resources, Subcommittee on Energy Production and Supply (Uranium Mill Tailings)

7/26/78—Senate Committee on Governmental Affairs, Subcommittee on Energy, Nuclear Proliferation and Federal Services (Nuclear Waste Management)

8/2/78—House Committee on Interstate and Foreign Commerce, Subcommittee on Energy and Power (Uranium Mill Tailings)

8/4/78—Senate Committee on Governmental Affairs, Subcommittee on Federal Spending Practices and Open Government (Government in Sunshine Act)

8/8/78—House Committee on Interior and Insular Affairs, Subcommittee on Energy and the Environment (Nuclear Siting and Licensing Act)

8/9/78—Senate Committee on Commerce, Science and Transportation, Subcommittee on Science, Technology and Space (Nuclear Waste Management)

Reports to Congress

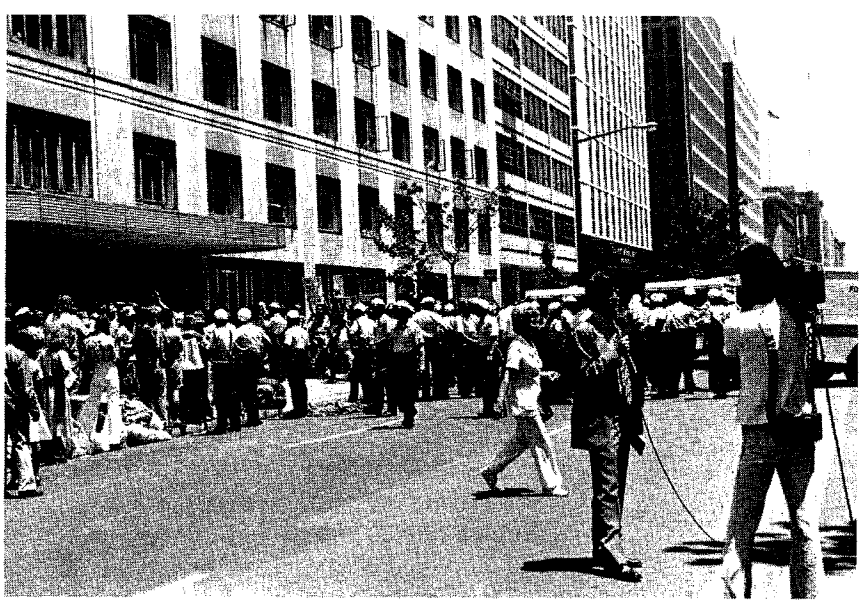
The NRC is required to keep committees having jurisdiction over its functions under rules of the Senate and the House "fully and currently informed" regarding the Commission's activities. Information on significant developments is forwarded routinely to the appropriate committees, and special reports are issued in response to frequent inquiries by committees and individual members of Congress.

Periodic reports to Congress or Congressional committees are required by law on the following matters*:

- Overall Annual Report to the President, for his transmittal to the Congress on a fiscal year basis.
- Abnormal occurrences in regulated nuclear activities (quarterly).
- Indemnity activities under the Price-Anderson Act (annual; now being incorporated in the overall Annual Report).
- Administration of the Freedom of Information Act (annual).
- Implementation of the Government in the Sunshine Act (annual).
- Printing plant report (annual).
- Annual plant inventory (annual).
- Major organizational components and numbers of employees (annual).
- Steps to meet provisions of Equal Opportunity Act (quarterly).
- Progress on resolving generic safety issues related to nuclear power plants (annual; being incorporated in overall Annual Report).
- Updating of long-term research plan for projects to develop new or improved safety systems for nuclear power plants (annual).
- Commission's views and recommendations on U.S. policies and actions to prevent proliferation (annual).
- ACRS report concerning nuclear reactor safety research program (annual).

GAO Reports. A number of other Congressional reports are generated as the result of studies by the General Accounting Office, which has broad authority to assist Congress, its committees, and individual members in carrying out their legislative and oversight responsibilities.

*Additional reporting requirements in NRC authorization legislation for fiscal year 1979, signed by the President in November 1978, are described in Chapter 1.



More than 100 demonstrators gathered in front of NRC headquarters in Washington, D.C., to await the decision of the Commission on continued construction at the Seabrook site in New Hampshire (June 1978).

Agencies that are the subject of GAO reports which recommend corrective actions are required by law to report within 60 days to the Government Operations Committees of the House and Senate on steps taken or planned to implement the recommendations. During fiscal year 1978, the GAO issued 16 reports covering various aspects of NRC activities, 13 of which required responses. These responses are available in the main NRC Public Document Room in Washington, D.C. GAO reports issued during the year were:

10/4/77—"An Evaluation of the Administration's Proposed Nuclear Nonproliferation Strategy."

10/28/77—Letter report to Congressman William J. Hughes on the NRC's environmental review process.

2/16/78—Letter report to Congressman Tom Bevill on the Tennessee Valley Authority's Hartsville and Phipps Bend licensing proceedings.

3/6/78—Letter report to Senator Gary W. Hart on the NRC's practice of submitting information to Atomic Safety and Licensing Boards. (See Chapter 12.)

3/7/78—Letter report to Senator Lloyd Bentsen regarding NRC's role in selecting fission technologies.

4/27/78—"Nuclear Power Plant Licensing: Need for Additional Improvement."

5/5/78—Letter report to Congressman John Dingell on reconciliation of special nuclear material unaccounted for (no response required.)

5/15/78—"Administrative Law Process: Better Management is Needed."

6/20/78—"The Uranium Mill Tailings Cleanup: Federal Leadership at Last?" (No response required.)

7/18/78—"New Ways of Preparing Data for Computers Could Save Money and Time and Reduce Errors."

7/20/78—"An Evaluation of Federal Support of the Barnwell Reprocessing Plant and the Department of Energy's Spent Fuel Storage Policy." (No response required.)

7/21/78—"Use of Discount Airline Fares and Teleticketing Would Help Save on Government Travel Expenses."

8/4/78—"Off-Gas Explosions at Nuclear Power Plants." (See Chapter 2.)

8/16/78—"Need for Greater Regulatory Oversight of Commercial Low-Level Radioactive Waste." (See Chapter 5.)

9/7/78—"The Nuclear Regulatory Commission Needs to Aggressively Monitor and Independently Evaluate Nuclear Power Plant Safety." (See Chapter 6.)

9/13/78—"Before Licensing Floating Nuclear Power Plants, Many Answers are Needed." (See Chapter 2.)

Proceedings and Litigation

13

The following are accounts of the adjudicatory activity of the NRC during the report period, specifically the activity of the Atomic Safety and Licensing Boards, the Atomic Safety and Licensing Appeal Boards, the Commission, and the NRC as a party to Federal court actions.

ATOMIC SAFETY AND LICENSING BOARDS

The Atomic Energy Act of 1954 requires that a public hearing be held on each application for a construction permit for a nuclear power plant and related facilities. The Act requires that a second opportunity for hearing be provided before a license may be issued to operate the facility. (A similar opportunity is provided before certain license amendments may be issued.) And the Act requires that, prior to the issuance of a construction permit for a nuclear power plant or related facility, a determination be made by the NRC as to whether the activities licensed by it would create or maintain a situation inconsistent with the antitrust laws, raising still another possibility and opportunity for public hearing. All such hearings are conducted by an independent Atomic Safety and Licensing Board (ASLB) (See 1977 NRC Annual Report, pp. 193-194, for procedural details of ASLB activities.)

Each of these licensing boards consists of three members drawn from the membership of the Atomic Safety and Licensing Board Panel—a body of legal and technical experts appointed by the Commission. As of September 30, 1978, the Panel included 16 full-time and 42 part-time members. Of these 58 members, 18 are lawyers, 17 environmental scientists, 12 engineers, 8 physicists, 2 economists and 1 chemist. (See Appendix 2 for names of members.) The Commission appoints members to the Panel based upon recognized experience, achievement, and independence in the appointee's field of endeavor. In assigning individuals to a given licensing board, consideration is given to the kinds of issues involved in the proceeding before that board. A hearing on a particular application may be divided into two phases—one concerning the health and safety, and common defense and security aspects of the application, as required by the Atomic Energy Act; and the other concerned with the environmental considerations required by the National Environmental Policy Act. Separate Initial Decisions covering these matters may be issued. (Antitrust problems in an application are heard and decided by a board of three antitrust experts.)

During the report period, the boards have been called upon to decide some novel questions of law and fact. These have included such matters as: whether a board has the authority to control the staff's review of an application by setting due dates for the filing of the results of the staff's review; whether, in certain circumstances, the staff should consider accidents whose consequences are severe but whose probability is remote (so-called "class 9" accidents); where the boundary between two States lies, in order to determine which had jurisdiction under the Federal Water Pollution Control Act; what threat would be posed to a nuclear station by tanker traffic carrying liquified natural gas on the river next to it; and whether a utility's procedures for identifying items required to be reported to the staff are adequate.

Further, during the report period, a board denied a Limited Work Authorization request because of the inadequacy of the alternative site analysis required by NEPA. Another board found that a construction permit should be suspended pursuant to a show-cause order issued by the staff to investigate the firing of a worker who had provided information to NRC. Boards were also called upon to hear and decide applications to expand the capacity of reactor spent fuel storage pools during this period, as utilities sought to ensure the continued ability of reactors to operate in the absence of off-site storage facilities.

The licensing process as it currently exists may require a number of decisions prior to construction and eventual operation of a nuclear power plant. Thus, a prospective licensee may apply for a Limited Work Authorization (LWA), by which it may gain an early start on plant construction (at its own risk, with no guarantee the construction permit will later be authorized), but the LWA will not be issued until a favorable Initial Decision on environmental and site-suitability issues is made. Two decisions were issued covering environmental and site-suitability matters leading to LWAs. These decisions involved four nuclear units.

An applicant who has received an LWA and carried out the authorized construction work may proceed to certain structural work, still at its own risk, under a second authorization (LWA-2), if such is approved by the licensing board. Two such decisions were rendered during the report period, affecting four units. In addition, one of the original LWA decisions also authorized an LWA-2 for two units.

Complete construction of the plant may be carried out only after a licensing board has made favorable findings in regard to radiological

The Atomic Safety and Licensing Board Panel held its annual meeting in 1978 at the Oak Ridge National Laboratory (ORNL). The three-day meeting included tours of ORNL's reactor safety research and environmental sciences programs. Here Robert Bryan of ORNL's Engineering Technology Division describes the Heavy Section Steel Program, which provides for the testing of nuclear reactor pressure vessels.

health and safety matters. Seven such decisions were issued during the report period, covering 15 units. Two of these decisions also dealt with environmental matters. There were two operating license decisions issued by boards during the period, covering three units.

The NRC adopted regulations in fiscal year 1977 under which applicants may obtain early site review and approval of proposed sites for nuclear generating stations, and these reviews may also entail a hearing before a licensing board. If the board approves the site, the approval would remain in effect for a period of five years, barring any substantive change in circumstances. Two proceedings under this regulation were initiated during fiscal year 1978.

Antitrust considerations were dealt with in one Initial Decision during this period. This decision involved the Farley plant and imposed license conditions after an earlier decision (1977 Annual Report p. 194) found a situation inconsistent with the antitrust laws.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

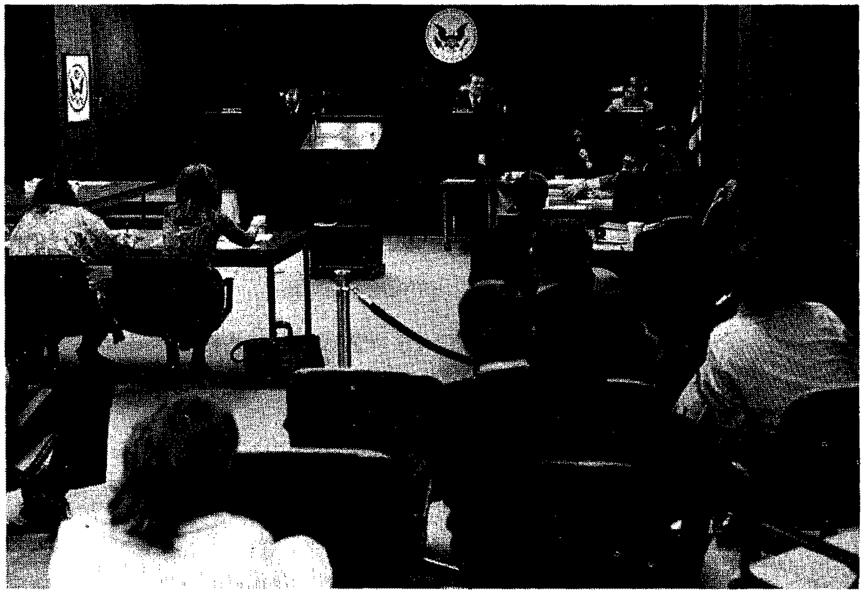
During the past year, Atomic Safety and Licensing Appeal Boards continued to perform the functions for which they have been responsible since their creation in 1969 by the Atomic Energy Commission. These three-member boards are authorized to exercise the Commission's authority and perform its review functions in facility licensing proceedings. With the establishment of the Atomic Safety and Licensing Appeal Panel in 1972, board members for particular cases have been selected from that Panel by its Chairman (or, in his absence, its Vice-Chairman). (See Appendix 2 for membership of the Panel.)

Appeal boards entertain appeals from Initial Decisions of licensing boards and certain licensing board orders respecting intervention. They also review Initial Decisions on their own initiative and, in limited circumstances, consider interlocutory questions posed or rulings referred by a licensing board. Appeal boards occasionally conduct evidentiary hearings, either as part of their appellate review functions or on direction from the Commission. The appeal board is the highest level within the Nuclear Regulatory Commission at which a party may seek administrative review as a matter of right. However, parties are permitted to seek discretionary Commission review of certain board rulings. The Commission also may decide to review an appeal board action on its own initiative. Where there is no Commission review, the decision of an appeal board represents the final order of the Nuclear Regulatory Commission; it is then subject to review in the Federal courts.

During fiscal year 1978, appeal boards completed or undertook review of 252 matters. They produced 66 published decisions (numbered ALAB-435 through ALAB-500), which appeared in the NRC's monthly publication, *Nuclear Regulatory Commission Issuances.* The monthly issues are bound into hard-backed volumes; during 1978, Volume 5, Books I and II (covering issues from January through March and April through June 1977, respectively) and Volume 6, pages 1-524 (covering issues from July through September, 1977) were released. Brief summaries of appeal board (as well as Commission and licensing board) opinions, headnotes of significant legal issues, and references to important technical questions which appear with the published opinions were prepared by the Appeal Panel staff.

The past year presented appeal boards with the challenge of numerous difficult questions bearing on both the environmental acceptability and the compatibility with the public health and safety of nuclear reactors. Although the boards held no evidentiary hearings during fiscal year 1978, several decisions set the stage for future evidentiary hearings by appeal boards.

In the environmental area, two appeal boards issued opinions which spelled out in considerable detail the scope and depth of investigations which must be undertaken with regard to alternate sites. In both the much-publicized *Seabrook* (New Hampshire) proceeding and the *Pilgrim* (Massachusetts) proceeding, the respective boards reviewed some of the evidentiary details which must be included in a record as a basis for an informed determination whether an alternate site is "obviously superior" to a proposed site and, if it is, whether the standard enunciated previously by the Commission requires that the proposed site be rejected. The *Pilgrim* Board upheld the Licensing Board's determination not to authorize a limited work authorization because of deficiencies in the alternate site



A session of the Atomic Safety and Licensing Appeal Board hearings in connection with the Seabrook, N.H., proceeding is shown. The appeal board is the highest level within the NRC at which a party may seek administrative review as a matter of right.

review, whereas the *Seabrook* Appeal Board remanded the proceeding to the Licensing Board for further hearings on this question. In declining to review the *Seabrook* ruling, the Commission directed that the further hearings which had been ordered in that proceeding be conducted by the appeal board. That hearing is to be held in fiscal year 1979. (See discussion below, under "Commission Review," and "Judicial Review," and also in Chapter 1.)

The Appeal Board in the *Three Mile Island* (Pennsylvania) proceeding will also hold an evidentiary hearing to consider health and safety questions bearing upon probabilities of airplane crashes at that facility. In its consideration of an appeal on that question, the Appeal Board determined that there were serious record deficiencies which precluded an informed decision as to the probabilities of an air crash given future predicted levels of air traffic. Because of the generic implications of some aspects of this question, as well as the extensive familiarity with the issue which it acquired during its consideration of an intervenor's appeal, the Appeal Board determined that it should conduct the hearing itself.

Several questions considered by appeal boards in the past year had both environmental and health and safety implications. One such question arose in the *Offshore Power Systems* proceeding, involving an application for a manufacturing license for floating nuclear plants. The board there determined that, although Commission policy barred the consideration of severe "class 9" accidents with respect to land-based plants (because of the improbability of their occurrence), that policy did not extend to floating plants. Hence, such accidents could be considered in the review of floating plants.

Also involving both environmental and health and safety considerations were the decisions concerning the effects of emissions of radon (Rn-222) from mill tailings produced in the mining and milling of uranium. In *Three Mile Island*, the Appeal Board treated these releases as covered by Table S-3 of 10 CFR 51.20(c) and, accordingly, as not eligible for particularized consideration in individual licensing proceedings. Thereafter the Commission determined that the radon-release values in Table S-3 were incorrect and amended the Table to delete them, directing that there be a reevaluation of the health effects of those emissions in individual licensing proceedings (see "Commission Review," below). In a joint opinion encompassing the *Peach Bottom* (Pennsylvania) and 16 other pending cases, the Appeal Boards established a procedural framework in which the issue could be resolved, with one proceeding (*Perkins*, in North Carolina) designated as "lead case" and provision for supplementing that proceeding's record and decision in the other cases.

Another ruling with both environmental and health and safety implications was the joint decision by the Appeal Boards in the *Prairie Island* (Minnesota) and *Vermont Yankee* (Vermont) proceedings. Those boards identified the safety and environmental standards relevant to proposals to expand reactors' spent fuel pools.

Several decisions reflected the burgeoning interest of States in the licensing of nuclear power plants. In the *Perry* (Ohio) proceeding, the board delineated the relationship between a State regulatory agency's required determinations and the NRC's authority to issue a construction permit. In the *Shearon-Harris* (North Carolina) proceeding, the Appeal Board treated the degree of deference which the Commission can accord to the need-for-power determinations of a State. In the *Indian Point 2* (New York) proceeding, the board discussed the extent to which the National Environmental Policy Act preempts State action in an area where NRC has performed an environmental review and imposed environmental conditions. And in the *Exxon* Nuclear Fuel Recovery and Recycling Center proceeding, the Appeal Board ruled that a State distant from a reprocessing facility site could in specified circumstances be admitted as an "interested State" in a proceeding involving that facility.

In the area of health and safety, a number of other decisions worthy of note were issued by appeal boards during the past year. The Appeal Board in the *River Bend* case (Louisiana facility) extensively treated the manner in which unresolved generic safety issues are to be taken into account by licensing boards which are considering applications for construction permits. The discussion in that case was applied to an operating license application in the *North Anna* (Virginia) proceeding. The *Three Mile Island* decision gave rise to a lengthy discussion of the requirements of evacuation plans. And the *Tyrone* (Wisconsin) and *Wolf Creek* (Kansas) decisions expanded upon existing rulings setting forth the scope and content of the Commission's inquiry into the financial qualifications of an applicant. In *Marble Hill* (Indiana), it was specifically held that all co-owners of a facility must become co-applicants for a sought license.

Other environmental matters of interest treated by appeal boards during the past year included a further delineation of work which may be performed by an applicant prior to its receipt of a limited work authorization (*Skagit* (Washington) proceeding), extensive discussions of various questions bearing upon the sufficiency of the supply of uranium for a proposed reactor (*River Bend* and *Wolf Creek* proceedings), a clarification of the respective weight which should be given to environmental impacts and financial costs in a comparative evaluation of alternatives (*Midland* (Michigan) proceeding), and the degree to which the end uses of electricity to be produced by a proposed plant are to be taken into account in the environmental review of that plant (*Midland* proceeding). In an opinion involving the *Hartsville* (Tennessee) reactors, the Appeal Board considered the relationship of the Endangered Species Act to an NRC license application.

Besides the proceedings involving radon releases (above), appeal boards issued a number of decisions dealing with procedural questions. Of particular interest were the discussions of requirements for reopening a record, contained in the *Wolf Creek* and *Perry* decisions, and of the standards relevant to motions for summary disposition appearing in the same *Perry* opinion. As in the past, appeal boards were required to determine the sufficiency of the standing of various parties to participate in NRC licensing proceedings.

With respect to the Commission's antitrust responsibilities, the *Midland* Appeal Board produced the first appellate decision on the merits of the antitrust aspects of an application. The board extensively reviewed the legislative history of the antitrust provisions of the Atomic Energy Act, the Commission's implementation of those provisions, their relationship to general antitrust principles, and the application of the relevant statutes and principles to utilities which are seeking an NRC license. The board applied its analysis to the conduct of the applicant, as reflected by the record; reversing the Licensing Board, it determined that the applicant possessed monopoly power in the relevant product and geographic markets, that the company monopolized those markets in contravention of the Sherman Act and its underlying policies, and that it was reasonably probable that licensing the Midland units without appropriate remedial conditions would maintain a situation inconsistent with the antitrust laws within the meaning of Section 105c of the Atomic Energy Act. The case was remanded to the Licensing Board for the formulation and imposition of appropriate license conditions; it was pending before that board at the close of the report period.

COMMISSION REVIEW

GESMO Decision

On December 23, 1977 the Commission terminated its proceedings related to the reprocessing of spent nuclear reactor fuel and recycling of the unused uranium and plutonium into new reactor fuel ("plutonium recycle"). A statement of the reasons behind the decision was issued on May 8, 1978. This decision terminated the extensive hearings and staff work on the Generic Environmental Statement on Mixed Oxide Fuel (GESMO) and proceedings on three recycle-related facilities: Allied-General Nuclear Services' nearly completed Barnwell, S.C., reprocessing plant; Westinghouse Electric Corporation's proposed Anderson, S.C., mixed oxide fuel fabrication plant; and Exxon Nuclear Company's proposed reprocessing plant at Oak Ridge, Tennessee. The decision followed President Carter's April 7, 1977 Statement on Nuclear Power Policy. The President recognized that if other countries which were not nuclear weapons states decided to recycle plutonium from their commercial nuclear power programs, there was a risk that some countries might divert the separated plutonium to the production of nuclear explosives. To counter this risk, the President announced an Administrative policy of indefinitely deferring domestic commercial plutonium recycle and initiated national and multinational studies of alternative nuclear fuel cycles which would reduce the risk of nuclear weapons proliferation.

The Commission reached its decision after two rounds of public comment and receipt of an October 4, 1977 letter on behalf of the President stating this view that the Commission's termination of its recycle-related activities would assist the President's non-proliferation initiatives. The Commission also decided to re-examine its decision after the completion of the alternative fuel cycle studies and to publish the staff's then nearly complete safeguards supplement to the final GESMO statement, "Safeguarding a Domestic Mixed Oxide Industry Against A Hypothetical Subnational Threat." (See also Chapter 3.)

UCS Petition

On November 4, 1977, the Union of Concerned Scientists (UCS) filed with the Commission a petition alleging safety problems in two areas: fire protection, and environmental qualification of electrical components (See "Abnormal Occurrences–1978," in Chapter 7). The UCS asked the Commission to take a number of corrective actions, and requested that all operating reactors be shut down and all reactor construction be halted pending resolution of the issues raised by the petition. In a Memorandum and Order dated April 13, 1978, the Commission declined to shut down reactors or halt construction, but did order the NRC staff to undertake a number of actions related to the matters alleged in the petition.

UCS filed a petition for reconsideration of the Commission's decision on May 3, 1978. The Commission agreed to entertain this petition, and sought further input from the NRC staff and the public on its merit.

Seabrook

During the past year the Commission issued three significant decisions involving the Seabrook Nuclear Station (N.H.).

In January 1978, the Commission approved issuance of the Seabrook construction permits. In that decision it addressed several issues, in particular the financial qualifications of the Seabrook applicants and the extent to which findings by the Environmental Protection Agency may be relied upon by the Commission in carrying out its obligations under the National Environmental Policy Act. The Commission also discussed its responsibilities for assuring a prospective licensee's financial qualifications under the Atomic Energy Act and announced its intention to begin a rulemaking investigation of current NRC regulations and policies in this area to determine whether any changes are needed.

In June 1978, after the United States Court of Appeals for the First Circuit vacated the Environmental Protection Agency's approval of a once-through cooling system for Seabrook, the Commission suspended the Seabrook construction permits by a divided vote. The majority's decision in part relied upon an appeal board decision that Seabrook with closed-cycle cooling had not received a proper environmental analysis. The suspension was based on the determination that if once-through cooling was not permitted at Seabrook, continuation of construction might foreclose the Commission's ability to choose environmentally superior alternative sites to Seabrook, if such existed. The Commission also ordered the appeal board to compare Seabrook with closed-cycle cooling against potential alternative sites, but it terminated further alternative-site comparisons of Seabrook using once-through cooling.

Subsequently, in August 1978, the Environmental Protection Agency reaffirmed its decision that once-through cooling was permissible at Seabrook. The Commission then determined that that decision removed the uncertainty about the type of cooling system that could be used at Seabrook—which was the primary factor that had required suspension of construction— and it restored the permits.

The appeal board is continuing its hearing on Seabrook as a facility with closed cycle cooling as a precautionary measure, since the EPA's approval of once-through cooling is being challenged in the courts.

Fuel Cycle Rule Amended

On April 11, 1978, the Commission amended its rule which prescribes how the environmental impacts of the uranium fuel cycle are to be set forth in environmental reports and environmental impact statements for individual light water nuclear power reactors. The fuel cycle rule specifies that, in these reports and statements, the radioactive effluents associated with fuel cycle activities shall be identified as they are set forth in a table ("Table S-3") and that "[no] further discussion of the environmental effects addressed by the table shall be required." The clarifying amendment removes the value contained in Table S-3 for releases of radon because in the staff's judgment, it significantly underestimates the total release of radon expected to result from fuel cycle activities. The Commission's action makes clear that the fuel cycle rule as amended does not preclude discussion, in individual cases, of the impacts of radon or the health effects of the other effluents described in Table S-3. A series of NRC programs now in progress is expected to provide further information useful in the updating of Table S-3, with the possible inclusion of a radon value.

Part of the overall updating program for the fuel cycle rule includes the development of a final rule treating the impacts associated with waste management and reprocessing. On March 19, 1977, the Commission promulgated an Interim Rule, effective for 18 months, which revised the waste management and reprocessing values; this action followed a decision by the Court of Appeals for the District of Columbia which invalidated the portion of the fuel cycle rule incorporating those values (see "Judicial Review," below). The Commission appointed a three-person hearing board to conduct legislative-type public hearings regarding the form and content of a final rule. Through written submission and oral testimony by 22 participants—including the NRC staff, public interest groups, and representatives of the nuclear industry—the hearing board compiled an extensive record.

On August 31, 1978, the board submitted to the Commission a summary and outline of this record, together with its view that an extension of the Interim Rule for a period sufficient to allow the Commission to consider the record carefully prior to acting on a final fuel cycle rule was warranted. On September 11, 1978 the Commission voted to extend the Interim Rule for a period of six months, until March 14, 1979. (See also Chapter 3.)

Antitrust Decision

The Commission issued one decision related to an antitrust question during the report period: *In the Matter of Florida Power & Light Company* (St. Lucie Plant, Unit 2).

At the time of application for a construction permit, (1973) for the facility cited above, the applicant and NRC agreed upon certain license conditions to resolve outstanding antitrust issues which were of concern to the Attorney General. (NRC antitrust reviews routinely take place at the time of application for permission to construct a commercial power reactor. Section 105(c) of the Atomic Energy Act requires that each such application be "promptly" transmitted to the Attorney General, who must render his advice within 180 days of receiving the application.) Because of this agreement, and in the absence of any request for an antitrust hearing, none was held at that time.

Some 31 months later, but before issuance of the construction permit for the facility, a group of Florida intervenors requested that the Commission conduct an antitrust hearing. The Commission's decision examined competing policies found in Section 105(c): the policy favoring resolution of antitrust matters early in the licensing process on the one hand, and the policy of granting smaller utilities, municipals and cooperatives access to the licensing process to pursue their interests, in a situation where a larger utility might use a government license to create or maintain an anticompetitive market position. The Commission granted a hearing, concluding that "while the statute encourages petitioners to voice their concerns early in the licensing process, we do not think that we can reasonably cut off all rights to NRC antitrust review for late requests so long as they are made concurrent with licensing." The Commission observed, however, that a very late petition must present a very strong reason for the Commission to entertain it.

JUDICIAL REVIEW

Significant Cases

Duke Power Company v. Carolina Environmental Study Group, et al. (Sup. Ct., No. 77-262).

NRC, et al. v. Carolina Environmental Study Group, et al. (Sup. Ct., No. 77-375).

Carolina Environmental Study Group, Inc. v. AEC, et al. (W.D.N.C., No. C-C-73-139).

This suit, filed by a citizen group, challenged the granting of a construction permit to the Duke Power Company for the McGuire facility in North Carolina. Plaintiffs alleged that the Commission's environmental review, required under the National Environmental Policy Act (NEPA), was inadequate. They also attacked, on constitutional grounds, the limitation of liability in the Price-Anderson Act. The district court held this case in abeyance pending the D.C. Circuit Court's decision in *C.E.S.G.* v. *AEC.* Following that decision (510 F.2d 796), the court dismissed the case except as to the Price-Anderson issue. On March 31, 1977, the district court concluded that the plaintiffs had standing, that the case was ripe for decision, and that the limitation on liability violated both the due process and equal protection clauses of the Constitution. An appeal was made to the Supreme Court.

On June 26, 1978, the Supreme Court unanimously upheld the constitutionality of the Price-Anderson Act limitation on liability as consistent with the Due Process Clause.

Vermont Yankee Nuclear Power Corporation v. Natural Resources Defense Council, Inc. (Sup. Ct., No. 76-149)

Natural Resources Defense Council, Inc., et al. v. NRC, et al. (D.C. Cir., No. 74-1385).

The Court of Appeals for the District of Columbia Circuit, by its decision of July 21, 1976 in these consolidated cases, set aside the waste management and reprocessing portions of the Commission's uranium fuel cycle rule ("Table S-3"). That rule had assigned numerical limits to the environmental effects attributable to the operation of a nuclear power plant for purposes of qualifying the plant for licensing. Without Table S-3 in place, the Commission's analysis of the environmental effects of the proposed Vermont Yankee plant was found by the court to be inadequate, and the Vermont Yankee operating license decision was remanded to the Commission for further consideration pending an adequate assessment of the fuel cycle issues. On February 22, 1977, the Supreme Court granted Vermont Yankee's certiorari petition and consolidated it with the *Aeschliman* case, discussed below.

On April 3, 1978, the Supreme Court unanimously reversed the lower court, deciding that the Administrative Procedure Act requires only notice and comment for informal rulemaking and that the Court of Appeals erred in invalidating portions of the Commission's Table S-3 rule for lack of sufficiently adjudicatory procedures. The case was remanded to the D.C. Circuit to consider the substantive adequacy of the fuel cycle rule. On June 2, 1978, NRC moved to dismiss the remanded cases urging that the interim fuel cycle rule promulgated by the Commission after the lower court decision was not arbitrary or capricious. The Court subsequently ordered these cases held in abeyance pending Commission issuance of a final fuel cycle rule.

Consumers Power Company v. Nelson Aeschliman, et al. (Sup. Ct., No. 76-528) (435 U.S. 519).

Nelson Aeschliman, et al. v. AEC, et al. (D.C. Cir., No. 73-1776).

Vermont Yankee Nuclear Power Corporation v. NRDC, (435 U. S. 519).

On review of the construction permits issued for Consumer Power Company's Midland (Michigan) facility, the Court of Appeals for the District of Columbia Circuit disapproved the Commission's treatment of energy conservation issues, ruling that the Commission had placed too stringent an evidentiary burden on groups seeking Commission consideration of energy conservation issues. The court also held that Advisory Committee on Reactor Safeguards (ACRS) reports must be sufficiently explicit to inform the public of all identified hazards of reactor operation and that licensing boards have the obligation to return cryptic reports to the ACRS for further elaboration. The court remanded the case to the Commission for the purpose of reappraising the NEPA cost/benefit balance, including an assessment of unaddressed fuel cycle issues. On February 22, 1977, the Supreme Court granted certiorari and consolidated this case with the *Vermont Yankee* fuel cycle case. These cases were argued on November 28, 1977.

On April 3, the Supreme Court reversed the Court of Appeals, holding that the sufficiency of an Environmental Impact Statement (EIS) is to be judged from the perspective of the time when it was written and that energy conservation was not an obvious alternative to a nuclear power plant in 1972. The Supreme Court also thought it proper under NEPA for the Commission to impose a threshold burden on those wishing novel alternatives covered in environmental impact statements. Finally, the Supreme Court held that ACRS reports need not be written for a layman's understanding, and remanded the case to the Court of Appeals for further consideration of the fuel cycle rule.

GESMO LITIGATION

(A) *State of New York v. NRC* (2nd Cir., No. 75-4278).
Natural Resources Defense Council, Inc., et al. v. NRC, et al. (2nd Cir., No. 75-4276).
Allied-General Nuclear Service, et al. v. NRDC, et al. (Sup. Ct., No. 76-653).
Commonwealth Edison Company, et al. v. NRDC, et al. (Sup. Ct., No. 76-762).
Baltimore Gas & Electric Company, et al. v. NRDC et al. (Sup. Ct., No. 76-774).
Westinghouse Electric Corporation v. NRDC, et al. (Sup. Ct., No. 76-769).

In these cases, New York State and certain citizen groups sought review of the Commission's November 14, 1975 *Federal Register* notice which set forth procedures for hearings on the Generic Environmental Statement on Mixed-Oxide Fuel (GESMO) and outlined agency standards for licensing activities related to the use of mixed oxide fuel prior to a Commission decision on wide-scale use of plutonium recycle. On May 26, 1976, the Court of Appeals for the Second Circuit issued its decision upholding, in full, both the GESMO hearing procedures and associated individual licensing procedures. However, interim licensing (except for "experimental and feasibility purposes") was forbidden. The Supreme Court granted petitions for certiorari by a number of utilities and a manufacturer. On December 23, 1977, however, the Commission voted to terminate the GESMO proceedings.

In January 1978, the Solicitor General filed a suggestion of mootness on behalf of NRC with the Supreme Court, submitting that the Commission's decision in December on mixed oxide fuel rendered the Second Circuit's decision moot, and that the opinion should be vacated and the case remanded with instructions to dismiss. On January 16, 1978, the Supreme Court vacated the Second Circuit's judgment and remanded the case to the Second Circuit "to consider the question of mootness." Those cases are now pending in the Second Circuit.

(B) *Westinghouse Electric Corporation v. NRC* (3rd Cir., Nos. 78-1188, 78-1198).
Exxon Nuclear Company, Inc. v. NRC (9th Cir., No. 78-1403) (3rd Cir., No. 78-1840) (dismissed by Exxon, August 11, 1978).
Allied-General Nuclear Service v. NRC (D.C. Cir., Nos. 78-1144, No. 78-1422).
Scientists and Engineers for Secure Energy, Mid-Atlantic Legal Foundation, and Capital Legal Foundation v. NRC (3rd Cir., No. 78-1204).

This is a series of cases challenging the Commission's December 23 order which terminated GESMO and related proceedings. The cases have all been consolidated in the Third Circuit for argument and decision. Petitioners argue that completion of an EIS was necessary to terminate the proceedings, that the Commission showed too great a deference to the President's foreign policy judgments, and that the Commission is obliged to pass upon all license applications under Atomic Energy Act standards. The NRC brief was filed in November 1978.

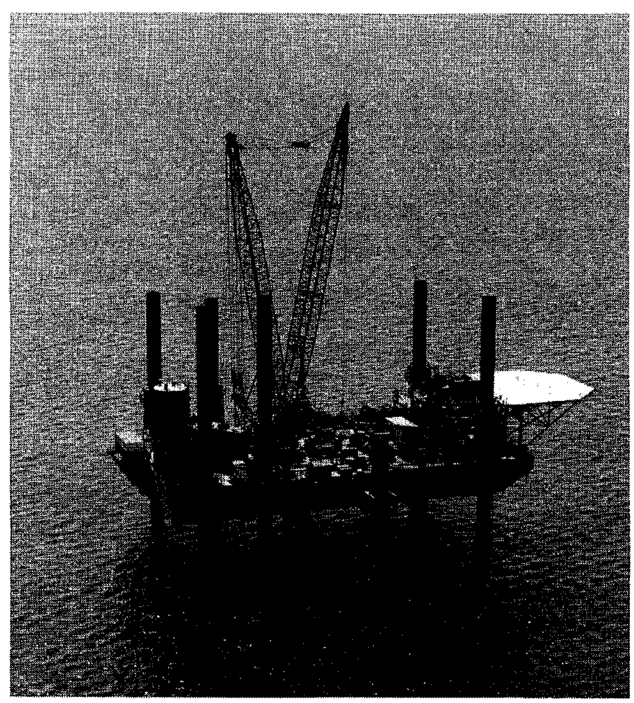
SEABROOK LITIGATION

(A) *New England Coalition on Nuclear Pollution, et al. v. NRC* (1st Cir., Nos. 77-1219, 77-1306, 77-1342, 78-1013).
Public Service Company of New Hampshire v. NRC (1st Cir., No. 77-1419).
New England Coalition v. NRC (1st Cir., Nos. 76-1469, 76-1525).
Audubon Society of new Hampshire v. United States, et al. (1st Cir., No. 76-1347).

The cases involving the Seabrook Nuclear Station in New Hampshire were among the most significant NRC cases decided during the past year.

On August 22, 1978, the Court of Appeals for the First Circuit decided four Seabrook cases brought by the New England Coalition on Nuclear Pollution, Nos. 77-1219, et al. (582 F. 2d 87). The petitioners challenged the NRC decision on many issues, including whether the Seabrook site was so close to a population center as to violate NRC regulations, whether the applicants were financially qualified to build the facility, and the legitimacy of supplementing a final environmental statement through testimony at a hearing. Also before the court was the "obvious superiority" test for comparing alternate sites, which provides that an application will be rejected on "alternate site" grounds only if the alternate site is obviously superior to the proposed site. Another issue considered by the court was the completion cost comparison ("sunk costs") rule, which provides that alternative sites will be compared on the basis of how much it would cost to complete a facility on each site. The court considered use of the "substitution theory" for need for power, which provides that use of nuclear power as a substitute for more costly fossil fuel is a legitimate factor favoring approval of an application. The use of EPA findings as part of

Litigation over the Seabrook power plant in New Hampshire focused in substantial part on site-related issues. One deals with the ecological value of a salt marsh and estuarine waters near the site. In the photo above, taken during early construction at Seabrook, the marsh and estuary are clearly visible in the background. Below, work on the Seabrook cooling water inlet and discharge tunnels was conducted offshore from "Texas Tower" platforms such as this one. The Seabrook "once-through" cooling system also was the subject of litigation during the year.



the NRC's NEPA analysis and several other NEPA and administrative law questions were also considered by the court. The First Circuit upheld the Commission's position on every point at issue.

Two months earlier, on June 21, 1978, the same court had upheld the licensing board's authority to decide that protection of an ecologically valuable marsh required that the Seabrook applicants reroute part of two transmission lines associated with the facility. The Supreme Court has declined to review that decision.

The First Circuit also dismissed as moot two Seabrook cases carried over from the previous years (Nos. 76-1347, 76-1469). Another challenge (No. 76-1525), to the Table S-3 rule in the Seabrook proceeding is still pending before the court.

(B) *Seacoast Anti-Pollution League, et al. v. NRC* (1st Cir., No. 78-1172).

This NEPA case was brought by two environmental groups challenging the consideration of alternative sites in connection with the Seabrook application. The case has been briefed and argued and is awaiting decision by the court.

(C) *New Hampshire State Building and Construction Trades Council (AFL-CIO), et al. v. NRC, et al.* (D.D.C., No. 78-1321).

Plaintiffs sued the NRC to overturn the Commission decision to suspend construction at the Seabrook site pending a decision by the EPA administrator on the environmental suitability of the Seabrook cooling system. NRC opposed the motions for stay arguing that the district court did not have jurisdiction to review the Commission's final order, since review of Commission orders is reserved exclusively to the court of appeals. The court agreed and dismissed the complaint.

Natural Resources Defense Council, Inc. v. NRC (2d Cir., No. 77-4157).

On August 25, 1977, the Natural Resources Defense Council (NRDC) filed a petition to review the Commission order denying NRDC's request that (1) a rulemaking proceeding be initiated to determine whether radioactive wastes generated in nuclear reactors can be safely disposed of and (2) that the licensing of plants be suspended pending such a determination. On July 5, 1978, the Second Circuit held unanimously that the Atomic Energy Act does not require that a finding that nuclear wastes can be disposed of safely must be made prior to licensing power plant operation.

Natural Resources Defense Council, Inc., et al. v. Robert C. Seamans, Jr., et al. (D.D.C., No. 76-1691).

Natural Resources Defense Council, Inc. v. NRC (D.C. Cir., No. 77-1489) (on appeal, D.C. Cir., No. 78-1576).

In re Robert W. Fri, Acting Administrator of ERDA (D.C. Cir., No. 77-121D).

NRDC and other environmental groups sued ERDA and NRC seeking to block construction of the waste tanks intended for the Hanford and Savannah River facilities. The complaint alleged that ERDA (now the Department of Energy) had failed to comply with NEPA by not issuing an environmental impact statement for the waste tank construction and had failed to obtain licenses from NRC under section 202(4) of the Energy Reorganization Act. NRC was named a defendant because plaintiffs seek a declaratory judgment that NRC has licensing authority in this matter.

On May 8, 1978, the District Court judge issued a 34-page opinion upholding NRC's position that it lacks licensing authority over the Hanford and Savannah River storage tanks, but found that DOE erred in not preparing project specific environmental impact statements for the waste tanks. Cross-appeals have been filed to the Court of Appeals.

People of the State of Illinois v. NRC et al. (7th Cir., No. 78-1171).

Illinois petitioned the Court of Appeals to review the denial of its request for enforcement action (under Commission regulation 10 CFR 2.206) on the General Electric facility at Morris, Ill. Petitioner alleges that the Morris facility had been "converted" to long-term storage for radioactive waste without preparation of an impact statement and without an evidentiary hearing. The briefs have been filed and the case argued; it was awaiting decision at the close of the report period.

State of Minnesota, By the Minnesota Pollution Control Agency v. NRC and the United States (D.C. Cir., No. 78-1269).

New England Coalition on Nuclear Pollution v. NRC (2d Cir., No. 78-4103).

Minnesota seeks review of the Appeal Board's decision in ALAB-455 (*Northern States Power*

Company), which authorized expanded spent fuel storage at the applicant's Prairie Island facility. The New England Coalition also sued to review ALAB-455, claiming that the decision illegally failed to give consideration to the environmental impacts of long-term on-site storage of spent fuel in connection with a spent fuel pool expansion proceeding. The cases have been consolidated in the D.C. Circuit Court of Appeals and are being briefed.

Mississippi Power and Light Company, et al. v. NRC, et al. (5th Cir., No. 78-1565).
Nuclear Engineering Company v. NRC, et al. (5th Cir., No. 78-1871).
Chem-Nuclear Systems v. NRC, et al. (5th Cir., No. 78-2200).

A number of utilities sued the NRC on its February 9, 1978 license fee rule. The utilities allege that NRC exceeded its statutory authority in setting the fees. They seek a declaration that the fee schedules are invalid, a suspension of collections in the interim, and a refund of all fees collected under the rule and its 1973 predecessor. The cases have been briefed and await argument.

ANTITRUST LITIGATION

Ft. Pierce Utilities Authority of the City of Ft. Pierce, et al. v. NRC, et al. (D.C. Cir., Nos. 77-2101, 77-1926).
Central Power & Light Company v. NRC, et al. (D.C. Cir., Nos. 77-1464, 77-1654).

In *Ft. Pierce Utilities*, petitioners have asked the Court of Appeals to review two related Commission actions denying an antitrust hearing. Petitioners argue that a Commission antitrust review may be initiated at any time, independent of licensing reviews. The cases have been briefed and argued and await a decision.

In *Central Power & Light*, the Court of Appeals, on June 12, 1978, dismissed an antitrust challenge as moot when the Commission, based on the Attorney General's recommendation, decided to hold an antitrust hearing on the competitive implications of issuing an operating license for South Texas Project, Units 1 and 2.

Concluded Cases

Martha G. Drake, et al. v. The Detroit Edison Company, et al. (E.D. Mich., No. G77-364 CA-7).

Plaintiffs challenged the sale by Detroit Edison Company to Northern Michigan Electric Cooperative, Inc., and to Wolverine Electric Cooperative, Inc., of a 20 percent interest in Detroit Edison's proposed Fermi Unit 2.

On June 26, 1978, the District Court ruled that the decision of the Director, Nuclear Reactor Regulation, not to take 10 CFR 2.206 enforcement action against Detroit Edison for its sale of a 20 percent interest in Fermi Unit 2 prior to Commission approval was wholly committed to NRC's discretion and not subject to judicial review. He also ruled that even if plaintiff's electric rates had increased by reason of the utility's unlawful action, plaintiffs lacked standing to sue the utility for damages under the Atomic Energy Act.

Virginia Electric and Power Company v. NRC (4th Cir., No. 76-2215).
North Anna Environmental Coalition v. NRC (4th Cir., No. 76-2331).

VEPCO and the North Anna Environmental Coalition petitioned the Fourth Circuit to review the Commission's *North Anna* opinion which imposed a $32,500 fine on the utility for false statements concerning geologic faulting at the site. NRC argued that the $32,500 civil penalty assessed against VEPCO was proper; that an intent to deceive is not a necessary element of an actionable false statement; that the materiality of the statement must be judged from the point of view of an NRC employee reviewing the utility's application for a power plant license, not the lay public's understanding; and that omission of information can constitute a false statement. The case was argued on December 6, 1977. On February 28, 1978, the Court of Appeals affirmed the Commission's order (571 F.2d 1289).

Culpeper League for Environmental Protection v. NRC (D.C. Cir., No. 76-1484).
Fauquier League for Environmental Protection v. NRC (D.C. Cir., No. 76-1532).

Petitioners challenged an appeal board decision, not reviewed by the Commission, which concerned the routing of high-voltage transmission lines from VEPCO's North Anna Power Station. Petitioners contended that an alternate route would have been preferable from an environmental standpoint. The appeal board, relying in large measure on evidence brought out during seven days of licensing board hearings, concluded that the route chosen was environmentally sound. The court consolidated these cases, heard oral argument on April 25, 1977, and, on March 16, 1978, affirmed the Commission in all respects (574 F.2d 633).

Natural Resources Defense Coucil, Inc. v. NRC (D.C. Cir., No. 76-1525).

Petitioners sought leave to intervene in two NRC export license proceedings involving applications to ship reactor fuel to the Tarapur Atomic Power Station in India. On May 7, 1976, the Commission denied the motions, on grounds that petitioners lacked legal standing to intervene, but afforded petitioners a legislative hearing as a matter of discretion. Petitioners sought review of the denial in the Court of Appeals.

While the case was awaiting decision, the Nuclear Non-Proliferation Act was enacted explicitly providing for discretionary hearing procedures. On July 3, 1978, the D.C. Circuit ruled that under the Nuclear Non-Proliferation Act the Commission need not afford any person an adjudicatory hearing in a nuclear export license proceeding. The court ruled that the standing issue had been rendered moot and that the new statute and regulations would govern that issue in the future. (580 F.2d 698.)

Garrett, et al. v. NRC (D. Ore., No. 78-269).

On March 28, 1978, plaintiffs sued the NRC claiming that removing spent fuel from the Trojan Nuclear Plant and placing it in the plant's spent fuel pool was illegal, absent an environmental impact statement assessing prolonged storage at the Trojan site. At a hearing held March 30, 1978, a magistrate denied plaintiffs' request for a temporary restraining order, finding no immediate irreparable injury. After a two-day evidentiary hearing, the magistrate on May 11, denied the motion for a preliminary injunction. The court's decision is based on its finding that the plaintiffs had failed to raise any substantial question that extended storage of spent fuel in the Trojan pool would have a significant adverse environmental impact. The court granted a consent motion to dismiss the complaint on May 25, 1978.

A. R. Martin-Trigona v. NRC (N.D. Ill., Civ. No. 77C-4454) (O'Hare Shipments Case).

On December 6, 1977, plaintiff sued NRC alleging that NRC must prepare an environmental impact statement for the transportation of radioactive materials through metropolitan areas, specifically Chicago's O'Hare Airport. NRC moved to dismiss, relying in part on the compliance with NEPA through publication of the NRC's "Environmental Impact Statement on the Transportation of Radioactive Materials by Air and Other Modes" (December 1978). On June 19, 1978, the District Court dismissed the complaint for plaintiff's failure to respond to the motion.

A. R. Martin-Trigona v. State of Illinois, NRC, and Nuclear Engineering Company (N.D. Ill., No. 78C-917) (Sheffield Case).

Plaintiff sought a declaratory judgment that the waste deposit site at Sheffield, Illinois, was a nuisance under Illinois law. No particular relief was sought against NRC. On April 10, 1978, the complaint was dismissed for plaintiff's failure to pursue it.

A. R. Martin-Trigona V. NRC, et al. (N.D. Ill., No. 78C-690) (University of Illinois Case).

Plaintiff sought to enjoin the University of Illinois from incinerating radioactive wastes and to order the NRC to more closely regulate the university's waste disposal. On July 7, 1978, the court granted defendants' unopposed motions for summary judgment.

People of the State of Illinois v. NRC, et al. (N.D. Ill., No. 77C-4190).

Illinois sought an injunction to require NRC to act on the license renewal application for Nuclear Engineering Company's (NECO) Sheffield site, pending since 1968. The State sought to restrain NECO from accepting or burying any additional low level waste until the NRC acts. The complaint stated that failure to act is both an abuse of discretion and a NEPA violation. The parties cross-moved for summary judgment. On June 16, 1978, the District Court dismissed the complaint as moot, because NECO can no longer bury wastes at Sheffield and an EIS is being prepared on NECO's application for renewal and expansion.

Atlantic County, et al. v. NRC, et al. (D.N.J., No. 77-2077).

Four coastal New Jersey counties sued NRC and the utilities which serve southern and central New Jersey with nuclear power, challenging the constitutionality of the Price-Anderson Act. Since the constitutional issue was pending before the Supreme Court, the parties stipulated that this District Court action should be stayed pending the Supreme Court's Price-Anderson Act decision. On June 26, 1978, the Supreme Court upheld the constitutionality of the limitation on liability, and on August 9, 1978, plaintiffs voluntarily dismissed their case.

Lewis, et al. v. NRC and TVA (N.D. Miss., No. EC-77237).
A group of University of Mississippi law students interested in making limited appearances in the Yellow Creek proceeding sued the NRC, arguing that they had not received 30-days notice of the time and place of hearing, in violation of the Atomic Energy Act and NRC regulations. On application for a temporary restraining order, the court ordered NRC to afford such notice. NRC complied and the case was dismissed on March 15, 1978.

Opened Cases

Mid-America Coalition for Energy Alternatives, Inc. v. NRC (D.C. Cir., No. 78-1294).
Petitioner sought review of the Appeal Board's March 9, 1978 decision in ALAB-452 which affirmed the Licensing Board's authorization of a contruction permit for Wolf Creek Generating Station, Unit 1. On April 24, petitioner sought a stay from the Court of Appeals which NRC opposed. On July 6, the Court of Appeals denied the stay but ordered that the case should be set for argument as soon after the filing of the NRC brief as business permits. The case has been briefed and argued.

Chauncey Kepford v. NRC, et al. (D.C. Cir., No. 78-1160).
Petitioner sued the NRC to stay operation of the Three Mile Island, Unit 2, facility, primarily because of the level of radon-222 releases from tailings produced in uranium mining and milling. On March 8, 1978, the court denied petitioner's motion for a stay. On March 22, 1978, the court, on its own motion, held further review in abeyance pending completion of the administrative appeals.

Chauncey Kepford v. NRC (D.C. Cir. No. 78-1933).
On September 21, 1978, petitioner sued the NRC for review of ALAB-480, an Appeal Board decision which established a procedure to conduct evidentiary hearings on the radon issue in cases pending before the Appeal Board. Petitioner seeks review only insofar as ALAB-480 affects the *Three Mile Island* proceeding.

Chauncey Kepford v. NRC (D.C. Cir. No. 78-2170).
Petitioner sued the NRC on November 13, 1978 for review of the Commission's affirmation of an Appeal Board decision which involved all but two of the issues associated with the *Three Mile Island* facility and which permitted its continued operation. On November 30, 1978, the NRC moved to hold the petition for review in abeyance pending the outcome of administrative hearings into one of the issues raised by the petitioner, that is, the probability that a very large aircraft will crash into the reactor.

Detroit Edison Company, et al. v. NRC (6th Cir., No. 78-3187).
National Association of Regulatory Utility Commissioners v. NRC (6th Cir., No. 78-3196).
These cases involve challenges to the Commission's denial of a rulemaking petition filed by the Detroit Edison Company. Detroit Edison had requested that the NRC amend its regulations to provide that NRC lacked authority to require rerouting of transmission lines associated with nuclear plants. The cases have been briefed and are awaiting argument.

Porter County Chapter of the Izaak Walton League of America, et al. v. NRC (D.C. Circ., No. 78-1556).
People of the State of Illinois v. NRC (D.C. Cir., No. 78-1599).
The City of Gary, Indiana v. NRC (D.C. Cir., No. 78-1560).
The Lake Michigan Federation v. NRC (D.C. Cir., No. 78-1561).
These petitions seek review of the Commission's April 20, 1978 decision affirming the denial by the Director, NRC Office of Nuclear Reactor Regulation, of a 2.206 enforcement request relating to the Bailly Generating Station. The cases were consolidated on June 23. Briefing was in progress at the close of the report period.

Commonwealth of Kentucky v. NRC (D.C. Cir., No. 78-1369).
The Commonwealth of Kentucky seeks review of the appeal board's decision of February 16, 1978, and of a Licensing Board decision of April 4, 1978, defining the Kentucky/Indiana border

for purposes of deciding from which State the utility must obtain a Section 401 water quality certificate for its Marble Hill facility.

On June 27, the D.C. Circuit dismissed the petition for review insofar as it related to the Licensing Board decision but retained jurisdiction over the petition to the extent it sought review of the appeal board's February 16 decision. Judicial review has been held in abeyance pending completion of administrative proceedings.

John Paskavitch v. NRC (D. Conn., No. H78-371).

On July 26, 1978, plaintiff sued the NRC for injunctive relief, alleging a series of harmful effects attributable to nuclear power in general and to the Millstone Nuclear Power Station in particular. The NRC moved to dismiss the complaint on jurisdictional grounds. The court granted the motion to dismiss on October 26, 1978.

Jeannine Honicker v. Joseph Hendrie, Chairman, NRC, et al. (M.D. Tenn., Civ. No. 78-3371-NA-CV) (6th Cir., No. 78-1405).

Plaintiff sued the NRC for injunctive relief alleging that the NRC had permitted nuclear power reactors to operate while underestimating the magnitude of health effects of the nuclear fuel cycle. Plaintiff seeks revocation of all licenses and dismantling of all existing fuel cycle facilities. The District Court, on September 6, denied a temporary restraining order and set a preliminary injunction hearing for October 2, 1978. Plaintiff has appealed the denial of the temporary restraining order.

Jeannine Honicker v. NRC (D.C. Cir. No. 78-2137).

On November 6, 1978, petitioner sought review of the Commission's decision to deny emergency relief to petitioner. Final disposition of the action will be reported in the next Annual Report.

Ecology Action of Oswego, New York v. NRC, et al. (D.C. Cir., No. 78-1855).

Petitioner sued the NRC to set aside the construction permit for the Sterling nuclear facility. Petitioners appeal from the denial of their application for a stay before the Appeal Board. The case had not been briefed at the close of the report period.

Akron, Canton & Youngstown Railroad Company, et al. v. Interstate commerce Commission, et al. (6th Cir., No. 78-3425).

This proceeding was brought by 22 railroads petitioning the Sixth Circuit to set aside the order of the I.C.C. in five consolidated cases. The railroads seek a declaration that they are not common carriers of highly radioactive nuclear materials. The NRC filed a limited appearance before the I.C.C. to argue that the I.C.C. lacked jurisdiction to examine the health and safety aspects of nuclear materials transportation. The NRC has moved to intervene in the case.

Pending Cases

Natural Resources Defense Council, Inc., et al. v. NRC, et al. (D. New Mexico, No. 77-240-B).

Natural Resources Defense Council, Inc., et al. v. NRC (D.C. Cir., No. 77-1570).

Natural Resources Defense Council, Inc. v. NRC, et al. (10th Cir., Nos. 77-1996, 78-1069).

These cases, brought by the Natural Resources Defense Council, challenge operations of a uranium milling facility in New Mexico. On May 3, 1977, NRDC, the Central Clearinghouse of New Mexico, and two individuals filed suit against NRC and the New Mexico Environmental Improvement Agency (NMEIA), seeking to enjoin operations of United Nuclear's Church Rock Mill, which NMEIA licensed on May 3, alleging violations of NEPA and the Atomic Energy Act. The gist of the complaint is that neither NRC nor New Mexico has prepared an environmental impact statement for the Church Rock Mill. Plaintiffs contend that New Mexico, as signatory to a section 274 State Agreement to regulate radioactive materials, is exercising Federal power and therefore must comply with NEPA, and that NRC's continuing review powers over State programs constitutes sufficient Federal involvement to call for preparation of an environmental impact statement (EIS). Second, plaintiffs argue that, in order to comply with section 274, State programs must be "compatible" with the NRC program and that compatibility requires preparation of an EIS where NRC would prepare one in a non-agreement State. NRC currently prepares an EIS for each new milling license and first renewal. A similar petition was filed June 30, 1977, in the D.C. Circuit naming only NRC as a respondent (No. 77-1570).

The D.C. Circuit on January 6, 1978, issued an order which rejected NRDC's theory that New Mexico as an Agreement State is exercising delegated federal power. The court also found that NRDC's allegations that the NRC has been "intimately involved" with the licensing of Church Rock demonstrate, if true, only State-Federal cooperation rather than a final decision-making authority retained by NRC. The court took no view on whether the New Mexico regulatory program is compatible with the Federal regulatory framework. That order brought the proceedings before the D.C. Circuit to a close.

Motions for summary judgment have been filed before the District Court for the District of New Mexico both by NRDC and by the NRC; trial is scheduled early in 1979. The Kerr-McGee Nuclear Corporation has intervened before the District Court. This intervention comes after the June 15, 1978 decision of the Tenth Circuit, on the appeal of Kerr-McGee and the American Mining Congress, which reversed the district judge's decision denying them intervention. (10th Cir., Nos. 77-1996, 78-1069).

Natural Resources Defense Council, Inc. v. NRC, et al. (D.C. Cir., No. 77-1448).

On May 13, 1977, NRDC filed a petition for review of the NRC's March 14 *Federal Register* notice promulgating an interim rule quantifying the environmental effects of the uranium fuel cycle. On July 5, NRDC requested that the D.C. Circuit hold the case in abeyance until the Supreme Court reaches a decision in the *Vermont Yankee* fuel cycle case. NRC consented to that motion.

On June 7, 1978, the D.C. Circuit requested the parties' views on how to dispose of this other fuel cycle case. On June 27, NRC advised the court that the interim rule case should be dismissed or held in abeyance pending a challenge to the final fuel cycle rule. The court is holding the case in abeyance until promulgation of the final rule.

United States of America and the Trustees of Columbia University in the city of New York v. City of New York, et al. (S.D.N.Y., 77 Civ. 3485).

The United States, on behalf of NRC and ERDA (now DOE), and Columbia University, filed a joint complaint against the City of New York asserting that the city's refusal, on radiological health and safety grounds, to permit an NRC-licensed reactor to operate violates the supremacy clause of the United States Constitution. The complaint seeks a declaration and injunction against enforcement of section 105.107(c) of the city's Health Code which purports to require a city radiological health and safety review and permit for operation of an NRC-licensed reactor.

The case is now awaiting argument on cross-motions for summary judgment. NRC contends that the Atomic Energy Act preempts local authorities from regulating the health and safety aspects of nuclear reactor operation.

John Abbotts, et al. v. NRC (D.D.C., No. 77-624).

John Abbotts, the Public Interest Research Group, and the Natural Resources Defense Council, Inc., brought a Freedom of Information Act suit challenging an NRC decision to withhold certain safeguards documents. The safeguards documents involved fall into three categories: (1) records related to the NRC program for onsite review of SSNM facilities initiated in early 1976; (2) records concerning the NRC investigation and review of conditions at the Nuclear Fuel Services facility in Erwin, Tennessee, in late 1975 and early 1976; and (3) studies done for or related to NRC's Special Safeguards Study and the Draft Safeguards Supplement. Parties have cross-moved for summary judgment and the court must now decide whether to review the documents *in camera* and whether there is a valid exemption claim by NRC.

Minnesota Environmental Control Citizen's Association, et al. v. Atomic Energy Commission, at al. (D. Minn., No. 4-72-109).

Plaintiffs, a citizens association, sought to enjoin further development and operation of Northern States Power Company's Monticello and Prairie Island facilities on the ground that the Prairie Island construction permit and the Monticello provisional operating license were issued without preparation of an environmental impact statement. On July 28, 1972, the District Court issued an opinion refusing to enjoin the construction or provisional operation, but holding that before full operating permits for these facilities could be granted a full NEPA review was required. The court retained jurisdiction over the matter to ensure that such a review was performed. During the past six years, the Commission has undertaken this environmental review, and both licensing proceedings are nearing completion. Once these are completed, NRC intends to move to dismiss the complaint on the grounds of mootness, as well as the statutory mandate that only a court of appeals shall review final orders of the Commission.

West Michigan Environmental Action Council, Inc. v. AEC, et al. (W.D. Mich., No. G-58-53).

Citizen group plaintiffs sought an injunction against increased use of mixed oxide fuel in Consumers Power Company's Big Rock Point power reactor. In June 1974, the court placed the case in abeyance pending the outcome of the GESMO proceedings and NRC review of Executive Branch comments. The utility has not pressed its application nor prepared the required environmental report, so the case may eventually be moot.

State of New York v. NRC, et al. (S.D.N.Y., No. 75 Civ. 2121) (2d Cir., Nos. 75-6115, 76-6002 and 76-6081).

New York State sought to halt air shipments of plutonium pending the preparation of an environmental impact statement. New York appealed the District Court's denial of its motion for preliminary injunction, motion for summary judgment, and dismissal of the Civil Aeronautics Board and the Customs Service as parties to the litigation. The Second Circuit in essence upheld the rulings and remanded the case to the District Court for further proceedings. The NRC environmental statement on transportation by air and other modes was issued in December 1977. The case is presently pending in the District Court. On September 6, 1978, New York filed its amended complaint in this case.

United States v. New York City (S.D.N.Y., No. 76 Civ. 273).

On January 15, 1976, the plaintiffs—the NRC, ERDA and Department of Transportation (DOT)—sought a judgment declaring a New York City Health Code provision dealing with the transportation of nuclear materials through the city to be inconsistent with the Federal statutory scheme governing the transportation of hazardous materials. The Government's request for a preliminary injunction against enforcement of the Health Code provision was denied on January 30, 1976, the court finding that no irreparable injury would occur pending a decision on the merits of the case. DOT has published regulations under the Hazardous Materials Transportation Act (which became effective January 1977) which allow interested persons to seek a ruling that a local ordinance is inconsistent with DOT regulations. On February 28, Brookhaven filed its request for such a regulation with DOT, arguing that the city's restrictions on shipping new and spent fuel were inconsistent with DOT's regulations. NRC and ERDA (now DOE) have written DOT in support of Brookhaven's position. On April 4, 1978, DOT ruled that the city ordinance was not inconsistent with DOT policy, but that a rulemaking would be held to consider whether regulations regarding the routing of nuclear materials by road are warranted.

Martin Hodder, et al. v. NRC, et al. (D.C. Cir., Nos. 76-1709, 78-1149).

Petitioners brought two petitions for review on the administrative decisions in this case challenging the NRC construction permit for St. Lucie Unit 2 on the east coast of Florida. The three issues raised by the case are whether the Commission treatment of Class 9 accidents satisfies NEPA, whether the St. Lucie Unit 2 site complies with Part 100, and whether the comparison of alternative sites was sufficient to support a decision to build the reactor at St. Lucie. The cases have been briefed by the parties and argued.

Long Island Lighting Company v. Lloyd Harbor Study Group, Inc. (Sup. Ct., No. 76-745).

Lloyd Harbor Study Group, Inc. v. NRC (D.C. Cir., No. 73-2266).

Baltimore Gas & Electric Company, et al. v. NRDC, et al. (Sup. Ct., No. 76-653).

NRDC, et al. v. AEC, et al. (D.C. Cir., No. 74-1586).

The common issue in these cases is a challenge to the Commission's Table S-3 rule prescribing the manner of accounting, in individual licensing cases, for the environmental consequences of the uranium fuel cycle activities. In both cases, petitions for writs of certiorari had been pending before the Supreme Court which, in effect, held them in abeyance pending the outcome of the *Vermont Yankee* case (see "Significant Cases," above). On April 18, 1978, the Supreme Court vacated the D.C. Circuit's orders and remanded them for reconsideration in light of the *Vermont Yankee* decision (98 Sup. Ct. 1600).

Coalition for the Environment, St. Louis Region and Utility Consumers Council of Missouri v. NRC (D.C. Cir., No. 77-1905)

On October 5, 1977, petitioners sued to suspend the construction permit for the Callaway Nuclear Plant based on a challenge to the Commission's interim fuel cycle rule. On December 1, 1977, the court held this case in abeyance until 30 days after the Supreme Court's decision in the *Vermont Yankee* case. On June 7, 1978, the court requested the parties' views as to how this and the other fuel cycle cases should be handled. NRC advised that it should be dismissed or held in abeyance pending a final fuel cycle rule. The case is in abeyance, as of the close of the report period.

A. R. Martin-Trigona v. Department of Justice, et al. (S.D. Ill., No. 78-4006).

On January 30, 1978, plaintiff sued the Justice Department, Commonwealth Edison, and the NRC concerning the withholding under the FOIA of documents pertaining to the Quad Cities power station. NRC is asserting exemption 7 as grounds for withholding the documents.

Basdekas v. NRC, et al. (D.D.C., No. 78-465).

On March 17, 1978, an NRC employee sued to compel disclosure of documents under the FOIA and Privacy Act. The documents are an investigative report of the Commission's Office of Inspector and Auditor and two memoranda from the Office of the General Counsel to the Commission. The case has proceeded through the filing of descriptive affidavits and discovery. The parties have cross-moved for summary judgment and the case awaits argument. The NRC asserts that portions of the documents are exempt from disclosure under exemptions 5 and 6

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Administration and Management

During fiscal year 1978, the NRC was operating with an authorized personnel strength of more than 2,700 and funding of $292 million. In August, the Commission attained its full five-member strength for the first time since April 1976. Headquarters activities continued to be dispersed in nine buildings in the District of Columbia and Maryland suburbs pending consideration of new consolidation studies requested by a Congressional committee. These and other management and administrative support developments, including organizational, personnel and fiscal matters, are discussed below.

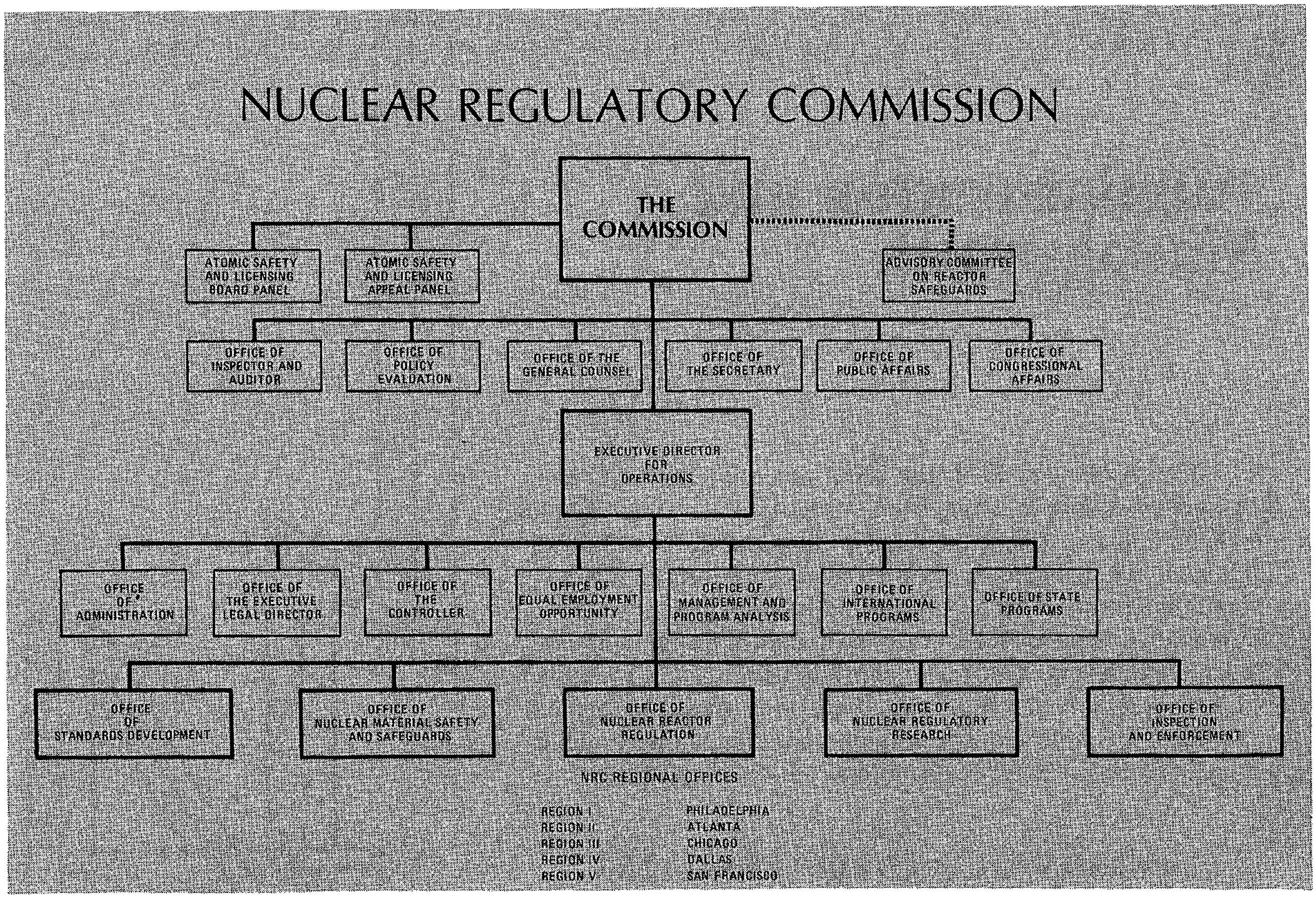
PERSONNEL AND ORGANIZATION

The NRC's authorized personnel strength at year-end rose to 2,723, almost 9 percent above the fiscal year 1977 level of 2,499. Approximately 70 percent of the agency's employees are in the major program offices, about 21 percent in program direction and coordination, and some 9 percent are employed at the Commission and Commission staff levels, including the independent advisory and adjudicatory bodies.

More than half of all NRC employees are trained as scientists or engineers in many disciplines. Nearly 70 percent hold college degrees, including almost 30 percent with masters or law degrees and almost 10 percent with doctorates.

New Commissioner and Office Directors

John F. Ahearne, appointed by the President to serve as the fifth member of the Commission, was sworn in on July 31, 1978, for a term expiring June 30, 1983. His arrival brought the Commission to full strength for the first time since April 1976. Before coming to the Commission, Mr. Ahearne was Deputy Assistant Secretary for Resource Applications in the Department of Energy, and had also served on the staff of the White House Energy Policy and Planning Office, working primarily on the nuclear and conservation portions of national energy legislation.



Changes occurred in the top management of two of the five program offices: Harold R. Denton was appointed Director of Nuclear Reactor Regulation, filling a post vacated in 1977 by Ben C. Rusche, whose duties had been handled on an acting basis by Edson G. Case; and John G. Davis was named acting director of the Office of Inspection and Enforcement following the resignation of Dr. Ernst Volgenau. In addition, Dr. Clifford V. Smith, Jr., resigned as Director of Nuclear Material Safety and Safeguards in December.

Other key management changes in staff offices included the elevation of William J. Dircks* to Deputy Executive Director for Operations from the position of Assistant Executive Director, which was abolished; the designation of Learned W. Barry as controller, moving from an acting status; and the naming of James J. Cummings as director, Office of Inspector and Auditor following the retirement of Thomas J. McTiernan.

*In December 1978, Mr. Dircks succeeded Dr. Smith as Director of Nuclear Material Safety and Safeguards.

Staff Offices Consolidated

The most significant organizational change was the establishment of the Office of Management and Program Analysis, which began operations in April 1978. The move consolidated the functions and staff of the former Office of Management Information and Program Control, former Office of Planning and Analysis, and two branches from the office of the Executive Director for Operations. Norman M. Haller was named director of the MPA, which provides management information and analyses for the NRC, Congress, other agencies and the public.

At the end of the fiscal year, the NRC was engaged in elevating its nuclear waste management functions to divisional status.

(Principal NRC officials and offices are listed, and their functions described in Appendix 1.)

Supergrade Audit

In response to recommendations by the Office of Management and Budget, the NRC arranged for an audit of all its existing and proposed supergrade positions. This audit was carried out independently by a competitively selected contractor whose report was issued in December 1977. At the end of the fiscal year, recommendations by the Executive Director for Operations were submitted to the Commission for consideration.

MANAGEMENT—EMPLOYEE RELATIONS

Union Election

In the spring of 1978, the National Treasury Employees Union (NTEU) challenged the status of the American Federation of Government Employees (AFGE) as the exclusive representative for NRC headquarters employees. Accordingly, a secret ballot election was conducted on June 6, 1978, under the supervision of the Department of Labor, to determine whether headquarters members would be represented by the AFGE, by the NTEU, or by no union.

While the tally of ballots favored NTEU as the exclusive representative, AFGE filed an objection to conduct by NTEU which allegedly affected the results of the election. The objection was subsequently dismissed, and NTEU was certified as the exclusive representatiave of all employees in the bargaining unit by the Labor Management Services Administration, Department of Labor, on November 17, 1978.

Equal Employment Opportunity

Progress at GS-11 and Up. In response to Congressional mandate, NRC began to submit quarterly reports to the Congress in 1978 on minorities and women hired and promoted, and on other actions to improve the agency's EEO posture, with main emphasis on grade levels GS-11 and above. Of all new employees hired at GS-11 and higher grades during the year, 8.0 percent were from minority groups and 12.6 percent were women. Promotions to grades GS-11 and above numbered 376, with minority personnel representing 8.2 percent and women constituting 16 percent. On September 30, 1977, there were 126 minority and 143 women employees at GS-11 and above. By September 30, 1978, these figures had changed to 152 and 171, respectively.

Recruiting Emphasis. During the year recruiters visited 28 colleges and universities for the purpose of attracting candidates for the NRC Intern, Cooperative Education and Summer Intern Programs. Visits to six of these schools were for the primary purpose of recruiting minority and women employees. Of the 143 persons hired under these programs, 51 percent were minority personnel and/or women.

NRC EMPLOYMENT PROFILE

	SEPTEMBER 30, 1977				SEPTEMBER 30, 1978			
	MEN		WOMEN		MEN		WOMEN	
	NON-MINORITY	MINORITY	NON-MINORITY	MINORITY	NON-MINORITY	MINORITY	NON-MINORITY	MINORITY
EXECUTIVE	10	1	0	0	12	1	0	0
GS-18	23	0	0	0	22	0	0	0
GS-17	53	3	0	0	54	3	1	0
GS-16	104	3	1	0	95	2	2	0
GS-15	434	15	6	0	463	17	10	0
GS-14	493	41	15	0	515	49	19	1
GS-13	244	30	18	4	236	29	24	7
GS-12	119	8	36	7	147	14	44	7
GS-11	61	6	46	10	65	9	43	13
GS-1—10	92	30	460	126	75	31	489	135
OTHER*	17	11	1	0	24	7	0	0

*Employees whose salaries are set by wage board, scientific and technical schedule, or administrative determination.

The agency feels that continued EEO emphasis in such programs will significantly increase the representation of minorities and women in the higher grades in the future.

Special EEO Programs. A number of EEO programs were held during the year highlighting Hispanic heritage, black history, awareness training, and career counseling. An *ad hoc* Committee on Age Discrimination was formed to study the special concerns of employees aged 40 to 70. The Committee sponsored programs for all NRC employees on age discrimination problems; worked with other offices to have employees' dates of birth removed from Forms 702 (Personal History Statements); surveyed employees' attitudes toward aging and age discrimination, and published the results agency-wide.

Women's Program. In February 1978 the NRC Federal Women's Program Advisory Committee initiated a survey to identify problems affecting NRC women which should be brought to management attention, including the concerns of women whose positions had been audited following promotion requests. The FWP also arranged a series of meetings between FWP members and key office directors, beginning in May, to discuss questions raised by their women employees. In October, the Program Manager and officers of the Advisory Committee briefed the Commissioners and submitted a report identifying 12 areas of potential sex discrimination.

Class Action. On June 16, 1978, an NRC women employee, as the agent of a class, filed a complaint alleging that NRC had discriminated against women and blacks in its employment practices. On October 10, 1978, following recommendations of the Civil Service Commission Complaints Examiner, NRC accepted an allegation of the class complaint charging sex discrimination in the auditing of positions. The complaint will be scheduled for a hearing on the merits before a Complaints Examiner to be appointed by the Equal Employment Opportunity Commission (the transfer of this function from the CSC to the EEOC will be effective January 1, 1979.)

Differing Professional Views

At year-end, the NRC was developing a comprehensive policy and procedures for bringing to management attention employee opinions on important matters that differ from existing policy or proposed staff positions, and for appropriate response to these concerns. (See discussion in Chapter 12.)

INSPECTION AND AUDIT

The Office of Inspector and Auditor (IA) was active throughout the year in auditing, investigating and inspecting various NRC operations

NRC's 1978 observances of Black History month in February featured a display of important artifacts and documents and special programs high-lighting the contributions and achievements of blacks in America. James Farmer, former Director of the Congress of Racial Equality (CORE), and now Associate Director, Coalition of American Public Employees, was keynote speaker at the Black History opening program. He is shown at left above, with, left rights, Edward E. Tucker, NRC Director of EEO; Howard K. Shapar, Executive Legal Director; and Lee V. Gossick, Executive Director for Operations.

Members of the NRC Federal Women's Program Advisory Committee briefed the Commission on the results of a survey of women employees' attitudes and problems and on the thrust of FWP interviews with office directors regarding issues raised by women in their organizations. Twelve areas of possible sex discrimination were outlined during the briefing. Shown facing camera, left to right, are Janet Thot-Thompson, Alicia Ong, Carol Peabody and Estelle Berdeguez of the Committee. Shown with backs to camera are Commissioner Victor Gilinsky, at left, and Chairman Joseph M. Hendrie. Executive Director for Operations Lee V. Gossick is at far end of table.

to provide the Commission with independent reviews and appraisals. Some of the more important 1978 activities are summarized below.

Standardization Policy. The draft report on NRC's proposed revision of the policy for licensing standardized nuclear power plants (see 1977 Annual Report, page 210) was completed in December 1977. It summarized industry and NRC staff views on two standardization options—the reference system, most widely used, and replication, with which the staff and the industry have experienced the greatest number of problems. IA recommendations to improve effectiveness of the program are generally reflected in the Revised Standardization Policy approved by the Commission on July 25, 1978 (see Chapter 2).

Executive Director for Operations' Testimony Concerning Question of Nuclear Material Diversion. In response to a Commission request, IA and the Office of General Counsel conducted an inquiry focusing on testimony by the Executive Director for Operations (EDO) before two Congressional committees in July and August 1977 on the question of whether special nuclear material (SNM) had been stolen from the Apollo, Pa., fuel plant formerly operated by the Nuclear Materials and Equipment Corporation. The principal matter addressed was a Congressional allegation that the EDO failed to present "an accurate description of the current understanding of the Apollo matter" in his July 1977 testimony to the effect that the Commission had no evidence that significant amounts of SNM had been stolen. A 594-page report was issued to the Commission on February 17, 1978, which generally concluded that, although the EDO's testimony was incorrect in some instances, he did not intentionably misrepresent the facts. Hearings were held on the matter on February 27 by the Subcommittee on Energy and Environment of the House Committee on Interior and Insular Affairs.

Review of License Renewal Process. A report to the EDO in January 1978 reviewed the efficiency of the process for renewing licenses for fuel fabrication facilities, the adequacy of documentation of NRC decisions in such renewals, and the impact of delays on NRC inspection efforts as well as on licensees' operations. A number of recommendations were made for improvement. The EDO noted that corrective actions were either underway or under study.

Inquiry on North Anna Fault Matter. Commission and Congressional interest in further probing AEC/NRC employee conduct in connection with discovery of a geologic fault at the North Anna nuclear plant site in Virginia in 1973 resulted in an inquiry and report by IA. (See 1977 Annual Report, pp. 187-189 for background.) The report, issued to the Commission in August 1978, answered 14 questions of Chairman Hendrie and Commissioner Bradford. The inquiry found no evidence indicating any intention on the part of AEC employees to deliberately withhold information from the Atomic Safety and Licensing Board; however, it concluded that the geologic problems at the North Anna site should have been brought to the board's attention by both the AEC and the licensee (Virginia Electric and Power Company) at a much earlier date. The inquiry could not conclusively determine whether the AEC was informed of a fault, a possible fault, or a geologic anomaly on May 17, 1973; however, because the AEC staff visited the site on June 18, 1973 and concluded that a fault existed, the report concluded that the NRC should support the date of June 18, 1973, as the date the AEC first learned that a fault existed. The IA informed the Commissioners that it believed there was no need for further investigation into the matter.

Chairman Hendrie sent a copy of the report to Senator Gary Hart (D-Colo.), Chairman of the Nuclear Regulation Subcommittee of the Senate Committee on Environment and Public Works, who had requested it, and who subsequently made it public.

Allegation of Employee Misconduct. An inquiry addressed allegations that AEC employees in 1974 failed to provide information to an Atomic Safety and Licensing Board about the Florida Power & Light Company's (FP&L) power grid disturbances and that AEC staff excluded the St. Lucie Nuclear Power Plant site from an investigation concerning the grid. IA's inquiry did not disclose any misconduct on the part of AEC employees in their handling of the grid instability issue during the licensing process for the St. Lucie plant and the report noted that the grid disturbance did not affect the safe operation of the nuclear plants on the FP&L grid. IA's report, issued in June 1978, was sent to the Commission and released to the public.

FUNDING AND BUDGET MATTERS

At the end of this chapter are NRC resource charts and NRC financial statements.

The charts show allocations of authorized personnel and funds to the various NRC activities which were carried out in fiscal year 1978 and which are projected for fiscal year 1979.

The increase in personnel for fiscal year 1979 is mainly for reducing the backlog of licensing actions and for increased efforts in reactor safety research and international and State programs.

The increase in funds for fiscal year 1979 results primarily from the transfer of LOFT operating costs from DOE to the NRC, inflation, increase in personnel strength, and some increases in research programs such as "3D," which is a project jointly sponsored by the United States, the Federal Republic of Germany, and Japan.

The financial statements following the charts are self-explanatory.

Contracting and Reimbursable Work

A large proportion of NRC's operating funds is expended in reimbursable arrangements with other agencies and contracts for confirmatory research and technical assistance in support of phases of virtually every major area of the agency's activity.

Approximately $128 million was allocated to NRC's confirmatory research program during fiscal year 1978, of which about $111 million went for reimbursable work performed for NRC by other Federal agencies. The Department of Energy's share of this was approximately $107 million for work performed in DOE's national laboratories and other facilities. Work done through DOE during the year included major research projects such as experiments at the Loss-of-Fluid Test (LOFT) facility ($15 million), the Power Burst Facility ($14 million), and the Semiscale facility ($6 million). (Specific research programs are described in Chapter 11.)

Technical and administrative assistance contracts (except work performed through DOE), as well as general purchases of all kinds, are administered through the Division of Contracts in the Office of Administration. Such contracts totaled more than $40 million during fiscal year 1978. Major activities regarding these contracts have focused on implementing socio-economic programs in Federal procurement, principally those relating to contracting in labor surplus areas and with small and minority-owned businesses.

Efforts were begun to develop an automated contract information system to provide promptly to the public and other Government agencies data on NRC procurement practices.

AUTOMATED CONTROL OF DOCUMENTS

During the year, the NRC initiated a program to apply the latest automated storage and retrieval technology to the entire range of NRC documents.

A $10 million contract was awarded in June to TERA Advanced Services Corporation to install and operate the document storage, indexing, abstracting and remote retrieval system. It is designed to search the indexed base by computer and to produce page images on video screens in response to user queries. Such capability is expected to reduce substantially the time required by NRC staff to evaluate license applications, prepare decisions, and study such matters as generic safety issues. The contractor will also prepare periodic abstract, index, and title list reports which will replace the high-cost, time-sharing contracts that have been used for this purpose.

A facility near NRC headquarters in Bethesda, Md. will house the contractor's staff of engineers, computer specialists, indexers, technical coders, technicians, and computer and microfiche equipment.

A monthly catalog will contain the abstracts and subject indexing of all documents made available to the public by any means, including those placed in the Public Document Room, released in response to Freedom of Information Act requests, sold through the National Technical Information Service or the Government Printing Office, and announced in the *Federal Register.*

PHYSICAL FACILITIES

During 1978 the NRC continued to house approximately 2,400 headquarters employees in nine buildings—one located in the District of Columbia, and eight in Maryland suburbs of the Capital.

In October 1977, the House Committee on Public Works and Transportation approved a General Services Administration (GSA) report which proposed consolidation of NRC headquarters in a single facility to be constructed on an urban renewal site in Washington, D.C. (See 1977 NRC Annual Report, page 208.) However, as a result of differing views presented by employee representatives and Maryland officials and legislators at an April 1978 hearing of the Subcommittee on Public Buildings and Grounds of the Senate Committee on Environment and Public Works, that committee requested GSA to analyze several additional factors. One of the main factors to be considered was the feasibility of consolidating in Montgomery County, Maryland.

GSA's expanded study, submitted to the Senate committee on July 25, 1978, recommended that two locations in the District of Columbia and three sites in Montgomery County be considered. The committee was evaluating the study at year-end.

NRC LICENSE FEES REVISED

On March 23, 1978, the NRC revised its license fee schedule which had been in effect since August 1973. The practice of charging fees was first adopted by the AEC in October 1968, in accordance with provisions of the Independent Offices Appropriations Act of 1952 and established Administration policy on recovery of user charges.

The NRC expects the revised schedule to recover less than 10 percent of its budget. All costs associated with generic licensing issues, research activities, standards development, State and international programs, and export licensing have been excluded from recovery. Also, costs of contested licensing hearings have been excluded.

Fees collected in fiscal year 1978 amounted to $13 million, of which $5.3 million is held by the Department of the Treasury in a suspense account awaiting calculation of actual costs after action on the permit or license involved has been completed. The total collected since fees were first imposed in 1968, through September 1978, was $88.9 million, of which $6.3 million has been refunded.

Basis for Charges

In the revised schedule, charges for facility and fuel cycle licenses, permits and approvals are based on actual cost to the NRC of processing the licenses. Fees for most materials licenses are based on the average cost of processing the application for a particular category of license.

The revised schedule includes, for the first time, fees for (1) review of applications from vendors and architect-engineers for standard design approvals; (2) utility applications referencing standard designs; (3) license amendments; (4) routine inspections; (5) special projects and reviews; (6) requests for approval of spent fuel casks and shipping containers; (7) requests for approval of sealed sources and devices containing or utilizing byproduct, source or special nuclear material; and (8) licenses for the receipt and storage of spent nuclear fuel.

Fuel Cycle Fee Increases. Fees were increased substantially in some cases due to the increased review effort, particularly environmental, that has occurred since 1973. For example, the maximum total fee for a license to use 5 or more kilograms of enriched uranium for fuel processing and fabrication has increased from $85,000 to $136,000 (including application filing fee), and for a similar license to use 2 or more kilograms of plutonium, from $135,250 to $771,900. The fee for a license for uranium milling operations (except *in situ* leaching) increased from $10,050 to a maximum of $107,700. And the fee for a license authorizing commercial disposal of nuclear waste by burial increased from $3,000 to a maximum of $323,100 (including application filing fee). All of the foregoing charges will be based on actual costs incurred in conducting each review and will not exceed the maximum published in the regulation (10 CFR Part 170).

Power Reactor Fees. The revised schedule takes into account the NRC's standardization plan and establishes fees for each approach (see accompanying table). Fees for nuclear power plants have not changed substantially, except that charges are reduced considerably for a second unit of similar design to be located at the same site and which is reviewed concurrently with the first unit. A unit built on another site using a previously approved design also would be subject to reduced fees.

Litigation Concerning Fees

In March 1974 the U.S. Supreme Court decided two cases which challenged the validity of annual fees for licenses assessed by the Federal Communications Commission and the Federal Power Commission under authority of the Independent Offices Appropriations Act of 1952. The Court ruled that the Act permitted an agency to assess fees only for special benefits rendered to identifiable persons as measured by the "value to the recipient" of the agency's service. As a result, the NRC (then AEC) promptly discontinued its annual license fees and announced procedures for the refund of all annual fees collected.

On November 11, 1974, the AEC published for comment a proposed revision of the remainder of its fee schedule. However, in December 1976, the Court of Appeals for the District of Columbia Circuit issued four opinions which invalidated a revised FCC license fee schedule. These cases provided additional guidance to the NRC for updating its license fee schedule.

Several electric utilities and two waste disposal licensees have petitioned the Fifth Circuit Court of Appeals to review the revised NRC license fee schedule. The petitioners argue that the schedule is invalid because fees are being assessed from applicants and licensees in circumstances where both the licensee and the public benefit from the services rendered by NRC.

NATIONAL EMERGENCY PREPAREDNESS

The NRC has taken several steps to assure continuation of essential agency functions during a national emergency, including an attack on the United States.

The Commission has identified essential uninterruptible functions and approved members and alternates for three executive teams which would provide dispersal of selected personnel and continuity of such functions if national security were threatened. A telephone "cascade"

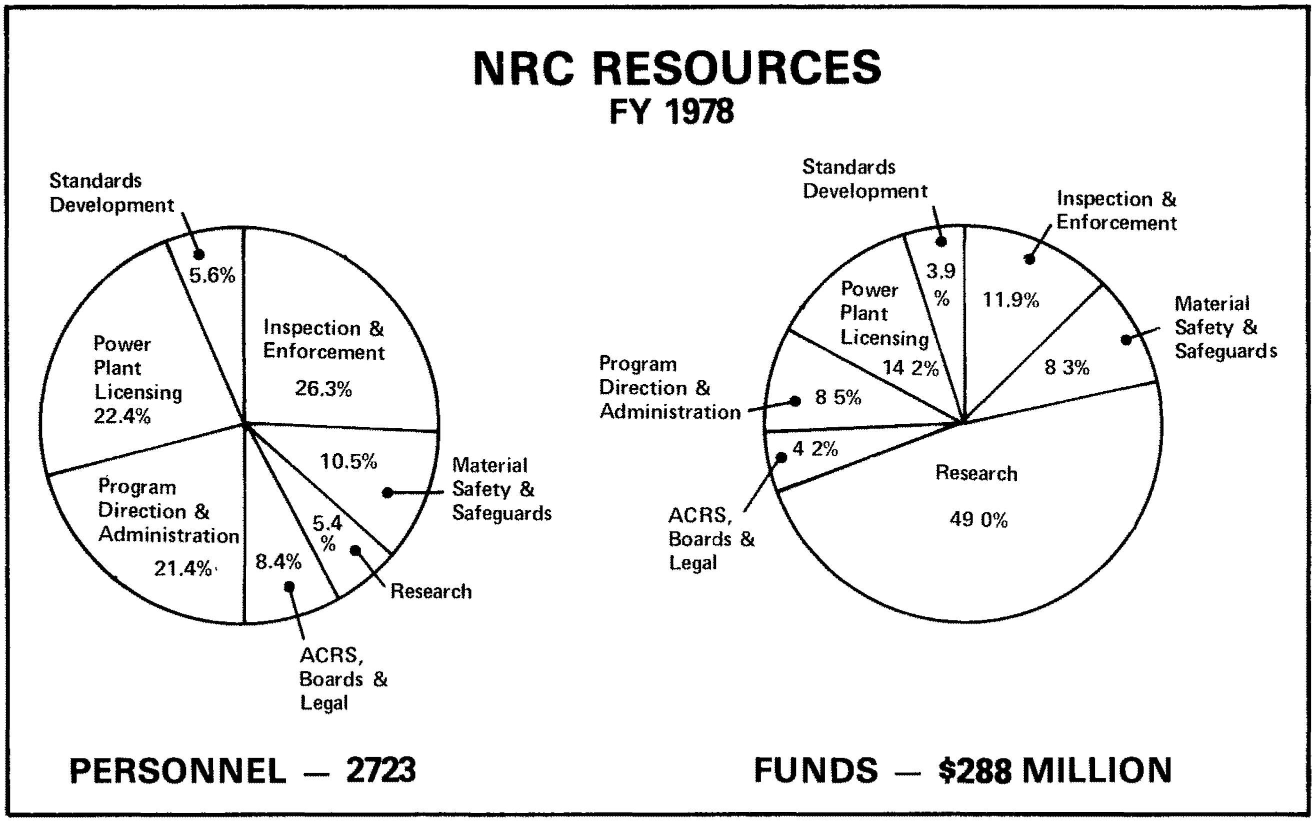
Maximum Fees for Nuclear Power Plants

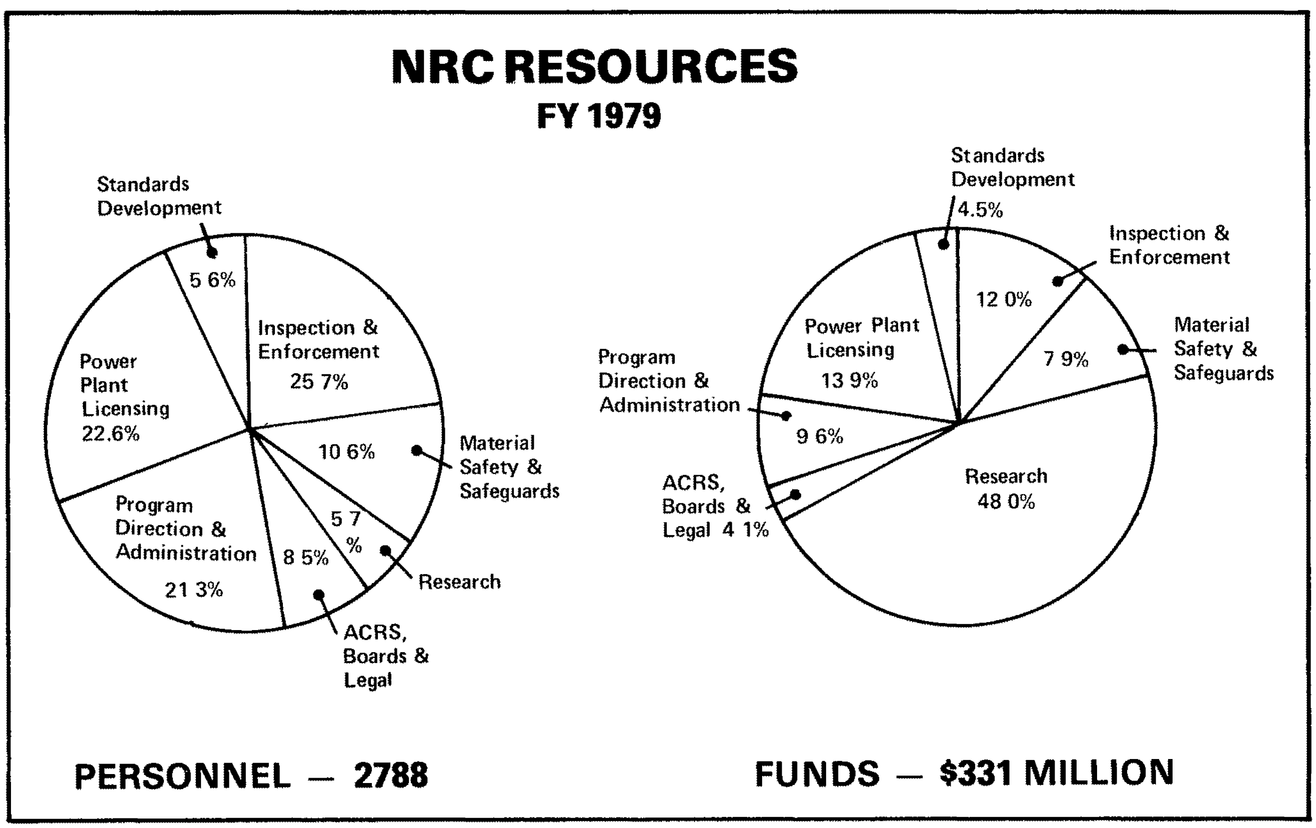
Facility categories	*Types of fees*	*Fee*
A. Power reactors:		
1. Custom	Application—Construction permit	$ 125,000
	Construction permit—First unit	944,000
	Construction permit—Concurrent unit	174,000
	Operating license—First unit	1,024,500
	Operating license—Concurrent unit	302,800
2. Standardized design—duplicate unit	Application—Construction permit	125,000
	Construction permit—First unit	944,000
	Construction permit—Concurrent unit	174,000
	Construction permit—First identical unit additional site(s)	757,100
	Operating license—First unit	1,024,500
	Operating license—Concurrent unit	300,200
	Operating license—First identical unit additional site(s)	712,000
3. Standardized design—replicate unit	Application—Construction permit	125,000
	Construction permit—First unit	811,600
	Construction permit—Concurrent unit	164,200
	Construction permit—First identical unit additional site(s)	725,900
	Operating license—First unit	914,400
	Operating license—Concurrent unit	293,900
	Operating license—First identical unit additional site(s)	691,500
4. Standardized design—Reference systems concept:		
a. Utility referencing a standardized nuclear steam supply system and custom balance of plant for both CP and OL stages.	Application—Construction permit	125,000
	Construction permit—First unit	853,600
	Construction permit—Concurrent unit	162,500
	Construction permit—First identical unit additional site(s)	725,900
	Operating license—First unit	934,100
	Operating license—Concurrent unit	292,100
	Operating license—First identical unit additional site(s)	669,200
b. Utility referencing a standardized nuclear steam supply system and standardized balance of plant for both the CP and OL stages.	Application—Construction permit	125,000
	Construction permit—First unit	721,800
	Construction permit—Concurrent unit	162,500
	Construction permit—First identical unit additional site(s)	725,900
	Operating license—First unit	829,100
	Operating license—Concurrent unit	292,100
	Operating license—First identical unit additional site(s)	669,200
5. Manufacturing license concept:		
a. Vendor—review of preliminary design.	Application	125,000
	Manufacturing license	1,477,500
b. Vendor—review of final design.	Final design amendment	448,100
c. Utility referencing a manufacturing license.	Application—Construction permit	125,000
	Construction permit—First unit	730,000
	Construction permit—Concurrent unit	61,500
	Operating license—First unit	1,001,200
	Operating license—Concurrent unit	221,000
6. Advanced reactors	Application—Construction permit	125,000
	Construction permit	1,781,000
	Operating license	1,954,900

provides for contacting team members. Emergency operating sites were identified and team orientation visits were conducted.

During the year, lines of succession to head NRC headquarters and regional offices were established, and the Commission issued appropriate delegations of authority. Regional office roles also were established. NRC staff worked with the staff of the Department of Energy toward a memorandum of understanding concerning joint efforts in the electric power aspects of the emergency preparedness program.

The Federal Preparedness Agency of the General Services Administration, responsible for coordinating the emergency preparedness activities of Federal agencies, reviewed the NRC program and found that it was progressing satisfactorily.


NRC RESOURCES
FY 1978
Standards Development
5.6%
Inspection & Enforcement
26.3%
Power Plant Licensing
22.4%
10.5%
Material Safety & Safeguards
Program Direction & Administration
21.4%
5.4 %
8.4%
Research
ACRS, Boards & Legal
Standards Development
3.9 %
Inspection & Enforcement
11.9%
Power Plant Licensing
14 2%
Material Safety & Safeguards
8 3%
Program Direction & Administration
8 5%
4 2%
Research
49 0%
ACRS, Boards & Legal
PERSONNEL — 2723
FUNDS — $288 MILLION


NRC RESOURCES
FY 1979
Standards Development
5 6%
Inspection & Enforcement
25 7%
Power Plant Licensing
22.6%
10 6%
Material Safety & Safeguards
Program Direction & Administration
21 3%
5 7 %
8 5%
Research
ACRS, Boards & Legal
Standards Development
4.5%
Inspection & Enforcement
12 0%
Power Plant Licensing
13 9%
7 9%
Material Safety & Safeguards
Program Direction & Administration
9 6%
ACRS, Boards & Legal 4 1%
Research
48 0%
PERSONNEL — 2788
FUNDS — $331 MILLION

Fiscal Year 1978—NRC Financial Statements

Balance Sheet (in thousands)

Assets	September 30, 1978	September 30, 1977
Cash:		
Appropriated Funds in U.S. Treasury	$ 129,149	$ 110,003
Other (See Note 1 below, and Note 5, next page)	10,841	6,204
	139,990	116,207
Accounts Receivable:		
Federal Agencies	142	39
Miscellaneous Receipts (See Note 2, below)	3,259	367
Other	85	16
	3,486	422
Plant:		
Completed Plant and Equipment (See Note 3, below)	5,716	7,022
Less - Accumulated Depreciation	1,428	1,663
	4,288	5,359
Advances and Prepayments:		
Federal Agencies	129	230
Other	984	634
	1,113	864
Total Assets	$ 148,877	$ 122,852

Liabilities and NRC Equity	September 30, 1978	September 30, 1977
Liabilities:		
Funds held for Other (See Note 1, below, and Note 5, next page)	$ 10,841	$ 6,204
Accounts Payable and Accrued Expenses:		
Federal Agencies	39,176	31,638
Other	15,318	11,253
Accrued annual leave of NRC Employees	5,552	4,843
Deferred revenue (See Note 5, next page)	2,067	
Total Liabilities	72,954	53,938
NRC Equity: October 1, 1977, Balance	68,914	64,855
Additions:		
Funds Appropriated-net	290,023	248,780
Non-reimbursable transfers from DOE		1,233
	358,937	314,868
Deductions:		
Net Cost of Operations (See Note 3, below)	273,153	233,220
Funds returned to U.S. Treasury (See Note 2, below)	9,861	12,734
	283,014	245,954
Total NRC Equity	75,923	68,914
Total Liabilities and NRC Equity	$ 148,877	$ 122,852

Note 1. As of September 30, 1978, includes $4,832,195.18 of funds received under cooperative researach agreements involving NRC, DOE, Federal Republic of German, Japan, Austria, and the Netherlands. Included also is $5,293,610.00 of funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processing the applicable application has been finalized and accordingly, are not available for NRC use. (See Note 5, next page.)

Note 2. These funds are not available for NRC use.

Note 3. On March 11, 1977, NRC and DOE signed a policy agreement which stated that for all equipment purchased by NRC for use at DOE facilities on NRC requested projects, title would rest with DOE. This agreement was considered to be retroactive to January 19, 1975. Therefore, $1,478,736.12 in plant and equipment cost at September 30, 1977, associatd with NRC projects being performed at DOE facilities was transferred from the Completed Plant & Equipment accounts and charged to Net Cost of Operations during Fiscal Year 1978.

Fiscal Year 1977/1978 Statement of Operations (in thousands)

	Fiscal Year 1978 (October 1, 1977, thru September 30, 1978)	Fiscal Year 1977 (October 1, 1976, thru September 30, 1977)
Personnel Compensation	$ 77,144	$ 68,430
Personnel Benefits	7,172	6,375
Program Support	161,817	133,808
Administrative Support	19,120	14,045
Travel of Persons	5,378	4,854
Training (Technical)	759	724
Equipment (Technical) (see Note 4. below)	7,687	6,016
Construction (See Note 4, below)	1,672	2,205
Taxes and Indemnities	5	7
Refunds to Licensees	189	473
Representational Funds	9	10
Reimbursable work	273	170
Increase in Annual Leave Accrual	709	838
Depreciation Expense	469	521
Equipment Write-offs and Adjustments	229	200
Total Cost of Operations	$ 282,632	$ 238,676
Less Revenues:		
Reimbursable work for Other Federal Agencies	273	170
Fees (to be deposited in U.S. Treasury as Miscellaneous Receipts (See Note 2, preceding page))		
Indemnity	1,793	2,805
Material Licenses	321	159
Facility Licenses	7,383	9,321
Other	1,188	215
Total Revenue	10,958	12,670
Net Cost of Operations before prior Year Adjustment	271,674	226,006
Prior Year Adjustment (See Note 3, preceding page)	1,479	7,214
Net Cost of Operations	$ 273,153	$ 233,220

U.S. Government Investment In The Nuclear Regulatory Commission (From January 19, 1975, Through September 30, 1978—in thousands)

Appropriation Expenditures:	
Fiscal Year 1975 (January 19, 1975, through June 30, 1975)	$ 52,792
Fiscal Year 1976 (July 1, 1975, through September 30, 1976)	226,248
Fiscal Year 1977 (October 1, 1976, through September 30, 1977)	230,559
Fiscal Year 1978 (October 1, 1977, through September 30, 1978)	$ 270,877
	$ 780,476
Unexpended Balance of Appropriated Funds in U.S. Treasury, September 30, 1978	129,149
Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975	429
Total Funds Appropriated	910,054
Less:	
Funds returned to U.S. Treasury (See Note 2, preceding page)	43,620
Assets and Liabilities transferred from Other Federal Agencies without Reimbursement	2,018
Net Cost of Operations from January 19, 1975, through September 30, 1978	788,493
Total Deductions	834,131
NRC Equity at September 30, 1978, as shown on Balance Sheet	$ 75,923

Note 4. Represents current year cost of plant and equipment acquisitions for use at DOE facilities.

Note 5. 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 (generally after license has been issued), 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, preceding page.)

Appendix 1

NRC ORGANIZATION

(As of September 30, 1978)

COMMISSIONERS

Joseph M. Hendrie, Chairman
Victor Gilinsky
Richard T. Kennedy
Peter A. Bradford
John F. Ahearne

The Commission Staff

General Counsel, James L. Kelley, Acting*
Office of Policy Evaluation, Kenneth S. Pedersen, Director
Office of Public Affairs, Joseph J. Fouchard, Director
Office of Congressional Affairs, Carlton C. Kammerer, Director
Office of Inspector and Auditor, O. Gene Abston, Acting Director**
Secretary of the Commission, Samuel J. Chilk

Other Offices

Advisory Committee on Reactor Safeguards, Stephen Lawroski, Chairman
Atomic Safety & Licensing Board Panel, James R. Yore, Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, Lee V. Gossick
Deputy Executive Director for Operations, William J. Dircks***
Technical Advisor, Stephen H. Hanauer

Program Offices

Office of Nuclear Reactor Regulation, Harold R. Denton, Director
Office of Nuclear Material Safety and Safeguards, Clifford V. Smith, Jr., Director
Office of Nuclear Regulatory Research, Saul Levine, Director
Office of Standards Development, Robert B. Minogue, Director
Office of Inspection and Enforcement, John G. Davis, Acting Director

Staff Offices

Office of Administration, Daniel J. Donoghue, Director
Executive Legal Director, Howard K. Shapar
Controller, Learned W. Barry
Office of Equal Employment Opportunity, Edward E. Tucker, Director
Office of Management and Program Analysis, Norman M. Haller, Director
Office of International Programs, James R. Shea, Director
Office of State Programs, Robert G. Ryan, Director

Regional Offices

Region I Philadelphia, Pa., Boyce H. Grier, Director
Region II Atlanta, Ga., James P. O'Reilly, Director
Region III Chicago, Ill., James G. Keppler, Director
Region IV Dallas, Texas, Karl V. Seyfrit, Director
Region V San Francisco, Calif., Robert H. Engelken, Director

*Leonard Bickwit was named General Counsel in December 1978.

**James J. Cummings was named director of OIA in November 1978.

***Mr. Dircks was named Director of Nuclear Material Safety and Safeguards in December, succeeding Dr. Smith, who resigned.

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 Non-proliferation 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 confirmatory 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 and the development of policy options for Commission consideration.

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 the 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. NRR also reviews the financial responsibility of each applicant for a construction permit, confirms that each applicant is properly indemnified against accidents, and verifies that the applicant(s) is not in violation of antitrust laws.

The Office of Nuclear Material Safety and Safeguards is responsible for ensuring public health and safety, and protection of national security and environmental values in the licensing and regulation of facilities and materials associated with the processing, transport, and handling of nuclear materials. NMSS reviews and assesses safeguards against potential threats, thefts, and sabotage, and works closely with other NRC organizations 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 implements research programs of nuclear regulatory research which are deemed necessary for the performance of the Commission's licensing and regulatory functions. Research programs cover reactor safety areas such as materials behavior, site safety, systems engineering, and computer code development and assessment. Research is also performed on safeguards, health effects associated with the nuclear fuel cycle, environmental impact of nuclear power, waste treatment and disposal, and transportation of radioactive materials.

The Office of Standards Development develops regulations, guides, and other standards needed for regulation of facilities and materials with respect to radiological health and safety and environmental protection, for materials safeguards and plant protection, and for antitrust review. The Office also coordinates NRC participation in national and international standards activities.

The Office of Inspection and Enforcement inspects nuclear facilities and materials licensees to determine whether facilities are constructed and operations are conducted in compliance with license provisions and Commission regulations, and to identify conditions that may adversely affect the protection of nuclear materials and facilities, the environment, or the health and safety of the public; inspects applicants and their facilities to provide a basis for recommending issuance or denial of licenses; investigates accidents, incidents, and allegations of improper actions that involve nuclear material and facilities; and enforces NRC regulations and license provisions. IE, on behalf of NRC, manages and directs the Commission's five regional offices, located in Philadelphia, Pa., Atlanta, Ga., Chicago, Ill., Dallas, Texas, and San Francisco, Calif.

The Commission Staff

The Office of the Secretary provides secretariat services for the conduct of Commission business and implementation of decisions, including planning meetings and recording deliberations, manages the staff paper system, monitors the status of actions, and maintains the Commission's official records. The office also processes institutional correspondence, controls the service of documents in adjudicatory and public proceedings, supervises the Washington, D.C. Public Document Room, administers the NRC historical program, and provides administrative support for the Commission.

The Office of General Counsel serves the Commission in a variety of legal capacities. The Office assists the Commission in the review of Appeal Board decisions, petitions seeking direct Commission relief, and rulemaking proceedings, and drafts 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 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 the 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. 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.

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.

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; data processing; management development and training; and other administrative housekeeping and special services.

The Office of the Controller develops and maintains the Commission's financial management program, including accounting, budgeting, pricing, contract finance, automatic data processing equipment acquisition, and accounting for capitalized property. Prepares reports necessary to the management of NRC funds. Maintains liaison with the General Accounting Office, Office of Management and Budget, Congressional Committees, other agencies, and industry. The Controller also prepares the NRC Five-Year Plan and performance resource evaluation studies.

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 Equal Employment Opportunity develops and recommends overall policy providing for equal employment opportunity, recommends improvements or corrections to achieve this goal, and monitors the agency's affirmative action program.

The Office of International Programs plans and implements programs of international cooperation; coordinates NRC export-import policies, issuing licenses as directed by the Commission; and establishes regulatory relationships with foreign nations and international organizations.

The Office of Management and Program Analysis provides NRC staff with management information and program analyses; identifies and analyzes major NRC policy, program and management issues and conducts long- and short-range planning to assist NRC operating officials; develops and implements management information and control systems and recommends policy on use of such systems for agency-wide applications; develops and implements application of sound statistical practices within NRC; and coordinates special information projects on overall NRC policies and programs.

The Office of State Programs directs programs relating to regulatory relationships with State governments and organizations and interstate bodies; manages the NRC State Agreements program; and provides Federal agency leadership in assisting State and local governments in radiological emergency response planning.

Other Offices

Advisory Committee on Reactor Safeguards. A statutory committee of 15 scientists and engineers, advises the Commission on the safety aspects of proposed and existing nuclear facilities and the adequacy of proposed reactor safety standards, and performs such other duties as the Commission may request.

Atomic Safety and Licensing Board Panel. Three-member licensing boards drawn from the Panel—made up of lawyers and others with expertise in various technical fields—conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, suspend, revoke, or amend NRC licenses.

Atomic Safety and Licensing Appeal Panel. Three-member appeal boards selected from the Panel exercise the authority and perform the review functions which would otherwise be carried out by the Commission in licensing proceedings. ASLB decisions are reviewable by an appeal board, either in response to an appeal or on its own initiative. The appeal board's decision also is subject to review by the Commission on its initiative or in response to a petition for discretionary review.

Appendix 2

NRC Committees and Boards

Advisory Committee on Reactor Safeguards

The ACRS was made a statutory committee in 1957 by Section 29 of the Atomic Energy Act of 1954, as amended. The committee reviews safety studies and facility license applications referred to it in accordance with the Atomic Energy Act and the Energy Reorganization Act and makes reports thereon which are made part of the public record of the proceeding. The committee provides advice with respect to the hazards of new or existing nuclear facilities and the adequacy of related safety standards. The committee also performs such other additional duties as the Commission may request. The members are appointed for four-year terms by the Commission. The committee annually elects its own chairman and vice chairman. As of September 30, 1978 the members were:

DR. STEPHEN LAWROSKI, *Chairman,* Senior Engineer, Chemical Engineering Division, Argonne National Laboratory, Argonne, Ill.

DR. MAX W. CARBON, *Vice Chairman,* Professor and Chairman of Nuclear Engineering Department, University of Wisconsin, Madison, Wis.

MYER BENDER, Director, Engineering Division, Oak Ridge National Laboratory, Oak Ridge. Tenn.

JESSE EBERSOLE, Retired Head Nuclear Engineer, Division of Engineering Design, Tennessee Valley Authority, Knoxville, Tenn.

HAROLD ETHERINGTON, Consulting Engineer (Mechanical Reactor Engineering), Jupiter, Fla.

DR. HERBERT S. ISBIN, Professor, Chemical Engineering, University of Minnesota, Minneapolis, Minn.

PROF. WILLIAM KERR, Professor of Nuclear Engineering, Director of Michigan Memorial-Phoenix Project, University of Michigan, Ann Arbor, Mich.

DR. J. CARSON MARK, Retired Division Leader, Los Alamos Scientific Laboratory, Los Alamos, N.M.

WILLIAM M. MATHIS, Retired Director, Planning, United Nuclear Industries, Inc., Richland, Wash.

DR. DADE W. MOELLER, Chairman, Department of Environmental Health Sciences, School of Public Health, Harvard University, Boston, Mass.

DR. DAVID OKRENT, Professor, School of Engineering and Applied Science, University of California, Los Angeles, Calif.

DR. MILTON S. PLESSET, Professor, Department of Engineering Science - Emeritus, California Institute of Technology, Pasadena, Calif.

JEREMIAH J. RAY, Retired Chief Electrical Engineer, Philadelphia Electric Company, Philadelphia, Pa.

DR. PAUL G. SHEWMON, Professor, Chairman of Metallurgical Engineering Department, Ohio State University, Columbus, Ohio

DR. CHESTER P. SIESS, Professor, Head of Civil Engineering Department, University of Illinois, Urbana, Ill.

Atomic Safety and Licensing Board Panel

Section 191 of the Atomic Energy Act of 1954 authorizes the Commission to establish one or more atomic safety and licensing boards, each comprised of three members, one of whom is to be qualified in the conduct of administrative proceedings and two of whom will have such technical or other qualifications as the Commission deems appropriate to the issues to be decided. The boards conduct such hearings as the Commission may direct and make such intermediate or final decisions as it may authorize in proceedings with respect to granting, suspending, revoking or amending licenses or authorizations. The Atomic Safety and Licensing Board Panel (ASLBP)—Office with a permanent chairman who coordinates and supervises the ASLBP activities—serves as spokesman for the panel, and makes policy recommendations to the Commission concerning conduct of hearings and hearing procedures. Pursuant to subsection 201 (g)(1) of the Energy Reorganization Act of 1974, the functions performed by the licensing boards were specifically transferred to the Nuclear Regulatory Commission. As of September 30, 1978 the ASLBP was composed of the following members and professional staff ("*" denotes full-time ASLBP members and staff):

JAMES R. YORE, *Chairman,* ASLBP, Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.*

ROBERT M. LAZO, Executive Secretary, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.*

DR. GEORGE C. ANDERSON, Department of Oceanography, University of Washington, Seattle, Wash.

ELIZABETH S. BOWERS, ASLBP Attorney, Bethesda, Md.*
JOHN H. BREBBIA, Attorney with law firm of Alston, Miller & Gaines, Washington, D.C.
R. BEECHER BRIGGS, Retired Senior Research Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.
GLENN O. BRIGHT, ASLBP Engineer, Bethesda, Md.*
DR. A. DIXON CALLIHAN, Retired Physicist, Union Carbide Corporation, Oak Ridge, Tenn.
DR. E. LEONARD CHEATUM, Retired Director of Institute of Natural Resources, University of Georgia, Watkinsville, Ga.
HUGH K. CLARK, Retired Attorney, E. I. duPont de Nemours & Company, Kennedyville, Md.
DR. RICHARD F. COLE, ASLBP Environmental Scientist, Bethesda, Md.*
DR. FREDERICK P. COWAN, Retired Physicist, Brookhaven National Laboratory, Stuart, Fla.
DR. FRANKLIN C. DAIBER, College of Marine Studies, University of Delaware, Newark, Del.
VALENTINE B. DEALE, Attorney at Law, Washington, D.C.
RALPH S. DECKER, Retired Engineer, U.S. Atomic Energy Commission, Cambridge, Md.
DR. DONALD P. DE SYLVA, Professor, Biology and Living Resources, School of Marine and Atmospheric Science, University of Miami, Miami, Fla.
MICHAEL A. DUGGAN, College of Business Administration, University of Texas, Austin, Tex.
DR. KENNETH G. ELZINGA, Department of Economics, University of Virginia, Charlottesville, Va.
DR. GEORGE A. FERGUSON, Professor of Nuclear Engineering, Howard University, Washington, D.C.
DR. HARRY FOREMEN, Director, Center for Population Studies, University of Minnesota, Minneapolis, Minn.
JOHN H. FRYE, III, ASLBP Legal Counsel, Bethesda, Md.*
MICHAEL GLASER, Partner, law firm of Glaser and Fletcher, Washington, D.C.
ANDREW C. GOODHOPE, Retired Administrative Law Judge, Federal Trade Commission, Wheaton, Md.
DR. DAVID B. HALL, Los Alamos Scientific Laboratory, Los Alamos, N.M.
DR. CADET H. HAND, JR., Director, Bodega Marine Laboratory, University of California, Bodega Bay, Calif.
DR. DAVID L. HETRICK, Professor, Nuclear Engineering Department, University of Arizona, Tucson, Ariz.
ERNEST E. HILL, Engineer, Lawrence Livermore Laboratory, University of California, Livermore, Calif.
DR. ROBERT L. HOLTON, School of Oceanography, Oregon State University, Corvallis, Ore.
DR. FRANK F. HOOPER, Chairman, Resource Ecology Program, School of Natural Resources University of Michigan, Ann Arbor, Mich.
SAMUEL W. JENSCH, Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.*
ELIZABETH B. JOHNSON, Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.
DR. WALTER H. JORDAN, Retired Senior Research Advisor & Physicist, Oak Ridge National Laboratory, Oak Ridge, Tenn.
LESTER KORNBLITH, JR., ASLBP Engineer, Bethesda, Md.*
DR. JAMES C. LAMB, III, Department of Environmental Sciences & Engineering, University of North Carolina, Chapel Hill, N.C.
MARGARET M. LAURENCE, Partner, law firm of Laurence, Stokes and Neilan, Arlington, Va.
DR. J. V. LEEDS, JR., Professor, Environmental and Electrical Engineering, Rice University, Houston, Tex.
GUSTAVE A. LINENBERGER, ASLBP Physicist, Bethesda, Md.*
DR. LINDA W. LITTLE, Research Triangle Institute, Research Triangle Park, N.C.Department of Environmental Sciences & Engineering, University of North Carolina, Chapel Hill, N.C.
DR. M. STANLEY LIVINGSTON, Retired Associate Director, Atomic Energy Commission National Accelerator Laboratory, Santa Fe, N.M.
DR. EMMETH A. LUEBKE, ASLBP Physicist, Bethesda Md.*
EDWARD LUTON, ASLBP Attorney, Bethesda, Md.*
DR. MARVIN M. MANN, ASLBP Technical Advisor, Bethesda, Md.*
DR. WILLIAM E. MARTIN, Senior Ecologist, Battelle Memorial Institute, Columbus, Ohio
DR. KENNETH A. McCOLLOM, Dean, Division of Engineering, Technology and Architecture, Oklahoma State University, Stillwater, Okla.
GARY L. MILHOLLIN, University of Wisconsin Law School, Madison, Wis.
MARSHALL E. MILLER, ASLBP Attorney, Bethesda, Md.*
DR. OSCAR H. PARIS, ASLBP Environmental Scientist, Bethesda, Md.*
DR. HUGH PAXTON, Los Alamos Scientific Laboratory, Los Alamos, N.M.
DR. PAUL W. PURDOM, Director, Environmental Studies Institute, Drexel University, Philadelphia, Pa.
DR. FORREST J. REMICK, Director, Institute of Science and Engineering, Pennsylvania State University, University Park, Pa.
DR. DAVID R. SCHINK, Department of Oceanography, Texas A&M University, College Station, Tex.
CARL W. SCHWARZ, Partner, law firm of Metzger, Noble, Schwarz & Kempler, Washington, D.C.
FREDERICK J. SHON, ASLBP Physicist, Bethesda, Md.*
IVAN W. SMITH, ASLBP Attorney, Bethesda, Md.*
DR. MARTIN J. STEINDLER, Chemist, Argonne National Laboratory, Argonne, Ill.

DR. QUENTIN J. STOBER, Research Associate Professor, Fisheries Research Institute, University of Washington, Seattle, Wash.
JOSEPH F. TUBRIDY, Attorney at Law, Washington, D.C.
JOHN F. WOLF, Attorney, law firm of Lamensdorf, Leonard & Moore, Washington, D.C.
SHELDON J. WOLFE, ASLBP Attorney, Bethesda, Md.*

Atomic Safety and Licensing Appeal Panel

An Atomic Safety and Licensing Appeal Board, established effective 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 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 (or, in his absence, the Vice Chairman) 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 September 30, 1978, was composed of the following full-time members and professional staff:

ALAN S. ROSENTHAL, Appeal Panel *Chairman,* U.S. Nuclear Regulatory Commission, Bethesda, Md.
DR. JOHN H. BUCK, Appeal Panel *Vice Chairman,* U.S. Nuclear Regulatory Commission, Bethesda, Md.
MICHAEL C. FARRAR, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
RICHARD S. SALZMAN, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JEROME E. SHARFMAN, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
CHARLES BECHHOEFER, Counsel, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
CARDIS L. ALLEN, Technical Advisor, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
S. LORRAINE CROSS, Legal Intern, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
LENORE R. MAGIDA, Legal Intern, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.

During fiscal year 1978, the Appeal Panel also included the following part-time members:

DR. LAWRENCE R. QUARLES, Dean Emeritus, School of Engineering and Applied Science, University of Virginia, Charlottesville, Va.
DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

Advisory Committee on Medical Uses of Isotopes

The Advisory Committee on Medical Uses of Isotopes was established in July 1958. The ACMI, composed of qualified physicians and scientists, considers medical questions referred to it by the NRC staff, and renders expert opinion regarding medical use of radioisotopes. The ACMI also advises the NRC staff, as requested, on matters of policy. Members are employed under yearly personal services contracts. The Deputy Director, Division of Fuel Cycle and Material Safety, serves as Committee Chairman. As of September 30, 1978 the members were:

RICHARD E. CUNNINGHAM, *Chairman*, ACMI, Deputy Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Silver Spring, Md.
DR. FRANK H. DE LAND, Chief, Nuclear Medicine Department, Veterans' Administration Hospital, Lexington, Ky.
DR. DAVID E. KUHL, Chief, Division of Nuclear Medicine, University of California School of Medicine, Los Angeles, Calif.
DR. JAMES L. QUINN, III, Director, Nuclear Medicine Department, Northwestern Memorial Hospital, Chicago, Ill.
DR. HENRY N. WAGNER, JR., Professor, Radiology and Radiological Sciences, Johns Hopkins Medical Institution, Baltimore, Md.
DR. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.
DR. JOSEPH B. WORKMAN, Associate Professor of Radiology, Duke University Medical Center, Durham, N.C.
DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.
DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.

Appendix 3

Public Document Rooms

Most documents originated by NRC, or submitted to it for consideration, are placed in the Commission's Public Document Room at 1717 H Street, N.W., Washington, DC, for public inspection. In addition, documents relating to licensing proceedings or licensed operation of specific facilities are made available in local public document rooms established in the vicinity of each proposed or existing nuclear facility. The locations of these local PDRs as of December 1978, and the name of the facility for which documents are retained, are listed below. (NOTE: Due to changes in the location of local PDRs, an updated listing may be obtained by writing to the Local Public Document Room Branch, Division of Rules and Records, U.S. Nuclear Regulatory Commission, Washington, DC 20555.)

ALABAMA

- Mrs. Maude S. Miller
  Athens Public Library
  South and Forrest
  Athens, Ala. 35611
  Browns Ferry Nuclear Plant
- Mr. Wayne Love
  G. S. Houston Memorial Library
  212 W. Verdeshaw Street
  Dothan, Ala. 36301
  Farley Nuclear Plant
- Mrs. Joanne Wyatt
  Clanton Public Library
  100 First Street
  Clanton, Ala. 35045
  Barton Nuclear Plant
- Mrs. Peggy McCutchen
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

ARIZONA

- Mrs. Mary Carlson
  Phoenix Public Library
  Science and Industry Section
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde Nuclear Plant

ARKANSAS

- Mr. Vaughn
  Arkansas Polytechnic College
  Russellville, Ark. 72801
  Arkansas Nuclear One

CALIFORNIA

- Mr. C. Combs
  Kern County Library
  1315 Truxtun Avenue
  Bakersfield, Calif. 93301
  San Joaquin Nuclear Plant
- Mrs. Alice Rosenberger
  Palo Verde Valley District Library
  125 West Chanslorway
  Blythe, Calif. 92255
- Mr. William B. Rohan
  San Diego County Law Library
  1105 Front Street
  San Diego, Calif. 92101
  Sundesert Nuclear Plant
- Mrs. Lucille A. Martel
  Mission Viejo Branch Library
  24851 Chrisanta Drive
  Mission Viejo, Calif. 92676
  San Onofre Nuclear Plant
- Mrs. Patricia Clark
  San Luis Obispo County Free Library
  888 Morro Street
  San Luis Obispo, Calif. 93406
  Diablo Canyon Nuclear Plant
- Mrs. Judy Klapprott
  Humboldt County Library
  636 F Street
  Eureka, Calif. 95501
  Humboldt Bay Nuclear Plant
- Mrs. Dorothy Harvey
  Business & Municipal Department
  Sacramento City-County Library
  828 I Street
  Sacramento, Calif. 95814
  Rancho Seco Nuclear Plant
- Mr. Andrew LaMance
  Stanislaus County Free Library
  1500 I Street
  Modesto, Calif. 95345
  Stanislaus Nuclear Plant

COLORADO

- Miss Ester Fromm
  Greeley Public Library
  City Complex Building
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Plant
- Mrs. Elizabeth Morrissett
  Acquisitions Department
  Auraria Library
  University of Colorado at Denver
  Lawrence and 11th
  Denver, Colo. 80204
  Atlas Corp. Uranium Mill

CONNECTICUT

- Mrs. Judy Liskov
  Waterford Public Library
  Rope Ferry Road—Route 156
  Waterford, Conn. 06385
  Millstone Nuclear Plant
- Mr. William Van Beynum
  Russell Library
  119 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Nuclear Plant

DELAWARE

- Mrs. Yvonne Puffer
  Newark Free Library
  750 East Delaware Avenue
  Newark, Del. 19711
  Summit Nuclear Plant

FLORIDA

- Ms. Sally Litton
  Jacksonville Public Library
  122 North Ocean Street
  Jacksonville, Fla. 32204
  Offshore Power Systems Manufacturing Facility
- Mrs. R. Scott
  Indian River Community College Library
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Nuclear Plant
- Mrs. Rene' Daily
  Environmental and Urban Affairs Library
  Florida International University
  Miami, Fla. 33199
  Turkey Point Nuclear Plant
- Mrs. Bonsall
  Crystal River Public Library
  668 N.W. First
  Crystal River, Fla. 32639
  Crystal River Nuclear Plant

GEORGIA

- Mrs. J. W. Borom
  Burke County Library
  Fourth Street
  Waynesboro, Ga. 30830
  Vogtle Nuclear Plant
- Ms. Annette Osborne
  Appling County Public Library
  Parker Street
  Baxley, Ga. 31513
  Hatch Nuclear Plant

ILLINOIS

- Mr. Ed Anderson
  Illinois Valley Community College
  Rural Route #1
  Oglesby, Ill. 16348
  LaSalle Nuclear Plant
- Mrs. Pam Wilson
  Morris Public Library
  604 Liberty Street
  Morris, Ill. 60451
  Dresden Nuclear Plant
  Midwest Fuel Recovery Plant
- Mrs. Marie Hoschied
  Moline Public Library
  504 17th Street
  Moline, Ill. 61255
  Quad Cities Nuclear Plant
- Ms. Jo Ann Ellingson
  Zion-Benton Public Library
  2600 Emmaus Avenue
  Zion, Ill. 60099
  Zion Nuclear Plant
- Mrs. M. Evans
  Vespasian Warner Public Library
  120 West Johnson Street
  Clinton, Ill. 61727
  Clinton Nuclear Plant
- Mrs. Penny O'Roarke
  Byron Public Library
  Third and Washington Streets
  Byron, Ill. 61010
  Byron Nuclear Plant
- Mr. Thomas Carter
  Wilmington Township Public Library
  201 S. Kankakee Street
  Wilmington, Ill. 60481
  Braidwood Nuclear Plant
- Savanna Township Public Library
  326 Third Street
  Savanna, Ill. 61074
  Carroll Nuclear Plant

INDIANA

- Mr. David Palmer
  West Chester Township Public Library
  125 South Second Street
  Chesterton, Ind. 46304
  Bailly Nuclear Plant
- Mr. Don C. Johnson
  Madison-Jefferson County Public Library
  420 West Main Street
  Madison, Ind. 47250
  Marble Hill Nuclear Plant

IOWA

- Miss Kay Burke
  Reference Service
  Cedar Rapids Public Library
  428 Third Avenue, S.E.
  Cedar Rapids, Ia. 52401
  Duane Arnold Nuclear Plant

KANSAS

- Mr. Jack Scott
  Coffey County Courthouse
  Burlington, Kans. 66839
  Wolf Creek Nuclear Plant

KENTUCKY

- Mr. Clarence R. Graham
  Louisville Free Public Library
  4th and York Streets
  Louisville, Ky. 40203
  Marble Hill Nuclear Plant

LOUISIANA

- Business & Science Division
  New Orleans Public Library
  219 Loyola Avenue
  New Orleans, La. 70140
  Offshore Power Systems Manufacturing Facility
- Mr. Ken Owen
  University of New Orleans Library
  Lousiana Collection, Lakefront
  New Orleans, La. 70122
  Waterford Nuclear Plant
- Miss Janie Videtto
  Audubon Library, West Feliciana Branch
  Ferdinand Street
  St. Francisville, La. 70775
- Mr. Jimmie H. Hoover
  Government Documents Department
  Lousiana State University
  Baton Rouge, La. 70803
  River Bend Nuclear Plant

MAINE

- Mrs. Barbara Shelton
  Wiscasset Public Library
  High Street
  Wiscasset, Me. 04578
  Maine Yankee Nuclear Plant

MARYLAND

- Mrs. Elizabeth Hart
  Charles County Library
  Garrett and Charles Streets
  La Plata, Md. 20646
  Douglas Point Nuclear Plant
- Mrs. Marie Barrett
  Calvert County Library
  Prince Frederick, Md. 20678
  Calvert Cliffs Nuclear Plant
- Ms. Pamela R. Schott
  Harford Community College
  401 Thomas Run Road
  Bel Air, Md. 21014
  Perryman Nuclear Plant

MASSACHUSETTS

- Mrs. Margaret Howland
  Greenfield Community College
  One College Drive
  Greenfield, Mass. 01301
  Yankee Rowe Nuclear Plant
- Mr. Mark Titus
  Plymouth Public Library
  North Street
  Plymouth, Mass. 02360
  Pilgrim Nuclear Plant
- Ms. Sue SanSoucie
  The Carnegie Library
  Avenue A
  Turner Falls, Mass. 01376
  Montague Nuclear Plant

MICHIGAN

- Mrs. Diana Shamp
  Reference Department
  Kalamazoo Public Library
  315 South Rose Street
  Kalamazoo, Mich. 49006
  Palisades Nuclear Plant
- Mrs. Katherine Thomson
  St. Clair County Library
  210 McMorran Boulevard
  Port Huron, Mich. 48060
  Greenwood Nuclear Plant
- Mrs. M. B. Wallick
  Charlevoix Public Library
  107 Clinton Street
  Charlevoix, Mich. 49720
  Big Rock Point
- Mrs. Joan Somerville
  Grace Dow Memorial Library
  1710 West St. Andrews Road
  Midland, Mich. 48640
  Midland Nuclear Plant
- Ms. Ann Stobbe
  Maude Preston Palenske Memorial Library
  500 Market Street
  St. Joseph, Mich. 49085
  D. C. Cook Nuclear Plant
- Mrs. Marcia Learned
  Reference Department
  Monroe County Library System
  3700 South Custer Road
  Monroe, Mich. 48161
  Fermi Nuclear Plant

MINNESOTA

- Mrs. Copeland
  Environmental Conservation Library
  Minneapolis Public Library
  300 Nicollet Mall
  Minneapolis, Minn. 55401
  Monticello Nuclear Plant
  Prairie Island Nuclear Plant

MISSOURI

- Mrs. Freida Mittwede
  Fulton City Library
  709 Market Street
  Fulton, Mo. 65251
- Mrs. Ranata Rotkowicz
  Olin Library of Washington University
  Skinker & Lindell Boulevard
  St. Louis, Mo. 63130
  Callaway Nuclear Plant

MISSISSIPPI

- Mrs. Stella Jennings
  Clairborne County Chancery Clerk
  Clairborne County Courthouse
  Port Gibson, Miss. 39150
  Grand Gulf Nuclear Plant
- Mr. William McMullin
  Corinth Public Library
  1023 Fillmore Street
  Corinth, Miss. 38834
  Yellow Creek Nuclear Plant

NEBRASKA

- Mrs. Leona Hansen
  Blair Public Library
  1665 Lincoln Street
  Blair, Neb. 68008
  Ft. Calhoun Unit 1 Nuclear Plant
- Mr. Frank Gibson
  W. Dale Clark Library
  215 South 15th Street
  Omaha, Neb. 68102
  Ft. Calhoun Unit 2 Nuclear Plant
- Mrs. Loy Mowery
  Auburn Public Library
  118 15th Street
  Auburn, Neb. 68305
  Cooper Nuclear Plant

NEW HAMPSHIRE

- Miss Pamela Gjettum
  Exeter Public Library
  Front Street
  Exeter, N.H. 03883
  Seabrook Nuclear Plant

NEW JERSEY

- Mr. Arthur Flandreu
  Stockton State College Library
  Pomona, N.J. 08240
  Offshore Power Systems Manufacturing Facility
  Atlantic Nuclear Plant
- Miss Elizabeth Fogg
  Salem Free Public Library
  112 West Broadway
  Salem, N.J. 08079
  Salem Nuclear Plant
  Hope Creek Nuclear Plant
- Mrs. Gail Colure
  Ocean County Library
  Brick Township Branch
  401 Chambers Bridge Road
  Brick Town, N.J. 08723
  Oyster Creek Nuclear Plant
  Forked River Nuclear Plant

NEW MEXICO

- Ms. Sandra Coleman
  General Library, Reference Department
  University of New Mexico
  Albuquerque, N.M. 87131
- Ms. Ingrid Vollnhofer
  New Mexico State Library
  Box 1629
  Santa Fe, N.M. 87503
  Waste Isolation Pilot Plant

NEW YORK

- Mr. Ralph W. Schmidt
  Oswego County Office Building
  46 East Bridge Street
  Oswego, N.Y. 13126
  Nine Mile Point Nuclear Plant
  Sterling Nuclear Plant
  FitzPatrick Nuclear Plant
- Mrs. June Rogoff
  Rochester Public Library
  Business & Social Science Division
  115 South Avenue
  Rochester, N.Y. 14604
  Ginna Nuclear Plant
- Mr. Oliver Swift
  White Plains Public Library
  100 Martine Avenue
  White Plains, N.Y. 10601
  Indian Point Nuclear Plant
- Mr. Richard Lusak
  Comsewogue Public Library
  170 Terryville Road
  Port Jefferson, N.Y. 11776
  Shoreham Nuclear Plant
- Mrs. E. Overton
  Riverhead Free Library
  330 Court Street
  Riverhead, N.Y. 11901
  Jamesport Nuclear Plant
- Mrs. Dorothy Augustine
  Catskill Public Library

One Franklin Street
Catskill, N.Y. 12414
Greene County Nuclear Plant

- Mr. Stanley Zukowzki
Buffalo & Erie County Public Library
Lafayette Square
Buffalo, N.Y. 14203
- Ms. Marsha Russell
Town of Concord Public Library
23 North Buffalo Street
Springville, N.Y. 14141
NFS Fuel Reprocessing Plant and UF_6 Facility
- Mr. Sol Becker
Public Health Library
New York City Department of Health
125 Worth Street
New York, N.Y. 10013
Columbia University Research Reactor
- Mr. Harold Ettelt
Columbia-Greene Community College
P.O. Box 100
Hudson, N.Y. 12534
Greene County Nuclear Plant

NORTH CAROLINA

- Mrs. Ruth Osborne
Public Library of Charlotte & Mecklenburg County
310 North Tryon Street
Charlotte, N.C. 28202
McGuire Nuclear Plant
- Mr. Roy Dicks
Wake County Public Library
104 Feyetteville Street
Raleigh, N.C. 27601
Shearon Harris Nuclear Plant
- Mr. David G. Ferguson
Davie County Public Library
416 North Main Street
P.O. Box 158
Mocksville, N.C. 27028
Perkins Nuclear Plant
- Mr. Phillip Barton
Southport-Brunswick County Library
109 West Moore Street
Southport, N.C. 28461
Brunswick Nuclear Plant
- Mrs. Charlotte Ellis
Franklin County Library
1026 Justice Street
Louisburg, N.C. 27549
Gulf Youngsville Fuel Fabrication Facility

OHIO

- Mrs. Betty Waltman
Perry Public Library
3753 Main Street
Perry, Ohio 44081
Perry Nuclear Plant
- Ms. Edith Holman
Clermont County Library
Third and Broadway Streets
Batavia, Ohio 45103
Zimmer Nuclear Plant
- Mr. Donald Fought
Ida Rupp Public Library
310 Madison Street
Port Clinton, Ohio 43452
Davis-Besse Nuclear Plant
- Mrs. Esther Schedley
Berlin Township Public Library
Four East Main Street
Berlin Heights, Ohio 44814
Erie Nuclear Plant

OKLAHOMA

- Mrs. Linda Hill
Tulsa City-County Library
400 Civic Center
Tulsa, Okla. 74102
Black Fox Nuclear Plant
- Mrs. O. J. Grosclaude
Sallisaw City Library
111 North Elm
Sallisaw, Okla. 74955
Sequoyah UF_6 Facility
- Ms. Hazel Nicholson
Guthrie Public Library
402 East Oklahoma Street
Guthrie, Okla. 73044
Cimarron Pu Fabrication Plant and Uranium Fuel Facility

OREGON

- Mr. H. B. Allen
City Hall, Records Office
Arlington, Ore. 97812
Pebble Springs Nuclear Plant
- Mr. Zimmer
Columbia County Courthouse
Law Library Circuit Court Room
St. Helens, Ore. 97501
Trojan Nuclear Plant

PENNSYLVANIA

- Reference Department
Osterhout Free Library
71 South Franklin Street
Wilkes-Barre, Pa. 18701
Susquehanna Nuclear Plant
- Mr. John Geschwindt
Government Publications Section
State Library of Pennsylvania
Education Building
Commonwealth and Walnut Street
Harrisburg, Pa. 17126
Peach Bottom Nuclear Plant
Three Mile Island Nuclear Plant
Fulton Nuclear Plant
- Mrs. Gordon Bauerle
Pottstown Public Library
500 High Street
Pottstown, Pa. 19464
Limerick Nuclear Plant
- Apollo Memorial Library
219 North Pennsylvania Avenue
Apollo, Pa. 15613
Apollo UF_6 and Pu Facilities
- Mr. Anthony Martin
Carnegie Library of Pittsburgh
4400 Forbes Avenue
Pittsburgh, Pa. 15213
Cheswick Fuel Development Laboratories
- Mr. F. E. Virostek
B. F. Jones Memorial Library
663 Franklin Avenue
Aliquippa, Pa. 15001
Beaver Valley Nuclear Plant
Shippingport Light Water Breeder Reactor

PUERTO RICO

- Mrs. Rosario Cabrera
Public Library, City Hall
Jose de Diego Avenue
P.O. Box 1086
Arecibo, P.R. 00612
- Mrs. Amalia Ruiz De Porras
Etien Totti Public Library
College of Engineers, Architects & Surveyors
Urb Roosevelt Development
Hato Rey, P.R. 00918
North Coast Nuclear Plant

RHODE ISLAND

- Mrs. Ann Crawford
Cross Mill Public Library
Old Post Road
Charlestown, R.I. 02831
- Mrs. Ann Shaw
University of Rhode Island
University Library

Government Publications
Office
Kingston, R.I. 02881
New England Nuclear Plant

SOUTH CAROLINA

- Joe E. Garcia
York County Library
325 South Oakland Avenue
Rock Hill, S.C. 29730
Catawba Nuclear Plant
- Reference Department
Richland County Public
Library
1400 Sumter Street
Columbia, S.C. 29201
Summer Nuclear Plant
- Miss Louise Marcum
Oconee County Library
201 South Spring Street
Walhalla, S.C. 29691
Oconee Nuclear Plant
- Mrs. Allene Reep
Hartsville Memorial Library
Home and Fifth Avenues
Hartsville, S.C. 29550
H.B. Robinson Nuclear
Plant
- Mr. David Lyon
Cherokee County Library
300 East Rutledge Avenue
Gaffney S.C. 29340
Cherokee Nuclear Plant
- Mr. Fred Bodiford
County Office Building
Room 105
P.O.Box 443
Barnwell, S.C. 29812
Barnwell Fuel Plant
UF_6 Facility
Barnwell Fuel Storage
Station
- Mr. Carl Stone
Anderson County Library
202 East Greenville Street
Anderson, S.C. 29621
Recycle Fuel Plant
- Mrs. Ellen Jenkins
Barnwell County Library
Hagood Avenue
Barnwell, S.C. 29812
Chem-Nuclear Plant

TENNESSEE

- Miss Kendall J. Cram,
Director
Tennessee State Library
and Archives
403 Seventh Avenue, North
Nashville, Tenn. 37219
Hartsville Nuclear Plant
- Ms. Dorothy Dismuke
Oak Ridge Public Library
Civic Center
Oak Ridge, Tenn. 37830
- Mrs. Patricia Rugg
Lawson McGhee Public
Library
500 West Church Street
Knoxville, Tenn. 37902
Clinch River Breeder Plant
Exxon Nuclear Fuel
Recovery Center
Fuel Fabrication Facility
- Mr. Wally Keasler
Chattanooga-Hamilton County
Bicentennial Library
1001 Broad Street
Chattanooga, Tenn. 37402
Sequoyah Nuclear Plant
Watts Bar Nuclear Plant
- Mr. T. Cal Hendrix
Kingsport Public Library
Broad and New Streets
Kingsport, Tenn. 37660
Phipps Bend Nuclear Plant
- Mr. H. E. Zittel
Oak Ridge National
Laboratory
P.O. Box X
Oak Ridge, Tenn. 37830
Tyrone Nuclear Plant

TEXAS

- Mrs. Tim Whitworth
Somervell County Public
Library
On The Square
P.O. Box 417
Glen Rose, Tex. 76043
Comanche Peak Nuclear
Plant
- Newton County Library
P.O. Box 657
Newton, Tex. 77034
Blue Hills Nuclear Plant
- Matagorda County Courthouse
Matagorda County Law
Library
P.O. Box 487
Bay City, Tex. 77414
South Texas Nuclear Plant
- Mrs. Kroesche
Sealy Public Library
P.O. Box 836
Sealy, Tex. 77474
Allens Creek Nuclear Plant

VERMONT

- Mrs. June Bryant
Brooks Memorial Library
224 Main Street
Brattleboro, Vt. 05301
Vermont Yankee Nuclear
Plant

VIRGINIA

- Ms. Sandra Peterson
Swem Library
College of William & Mary
Williamsburg, Va. 23185
Surry Nuclear Plant
- Mr. Edward Kube
Board of Supervisors
Louisa County Courthouse
P.O. Box 27
Louisa, Va. 23093
- Mr. Gregory Johnson
Alderman Library
Manuscripts Department
University of Virginia
Charlottesville, Va. 22901
North Anna Nuclear Plant

WASHINGTON

- Ms. D. E. Roberts
Richland Public Library
Swift and Northgate Streets
Richland, Wash. 99352
WPPSS 1, 2 and 4 Nuclear
Plants
Exxon Fuel Plant
- Mrs. D. Stendal
Sedro Wooley Library
802 Ball Avenue
Sedro Wooley, Wash. 98294
Skagit Nuclear Plant
- Ms. Selma Nielsen
W. H. Abel Memorial Library
125 Main Street South
Montesano, Wash. 98563
WPPSS 3 and 5 Nuclear
Plants

WISCONSIN

- Mrs. Jane Radloff
LaCrosse Public Library
800 Main Street
LaCrosse, Wis. 54601
LaCrosse BWR Nuclear
Plant
- Mr. Arthur M. Fish
Document Department,
Library
University of Wisconsin
Stevens Point
Stevens Point, Wis. 54481
Point Beach Nuclear Plant
Wood Nuclear Plant
- Mrs. Frances Wendtland
Mead Public Library
710 North Eighth Street
Sheboygan, Wis. 53081
Haven Nuclear Plant

- Madison Public Library
  Business and Science Division
  201 West Mifflen Street
  Madison, Wis. 53703
  Koshkonong Nuclear Plant
- Ms. Sue Grossheuch
  Kewaunee Public Library
  833 Juneau Street
  Kewaunee, Wis. 54216
  Kewaunee Nuclear Plant
- Mr. John Jax
  University of Wisconsin
  Stout Library
  Menomonie, Wis. 54751
- Mr. Robert Fetvedt
  University Library
  University of Wisconsin—
  Eau Claire
  Park and Garfield Avenues
  Eau Claire, Wis. 54710
- Mrs. Robert Goodrich
  Durand Free Library
  315 Second Avenue, West
  Durand, Wis. 54736
  Tyrone Nuclear Plant

WYOMING

- Mrs. Carroll Highfill
  Converse County Library
  Douglas, Wyo. 82633
  Highland Uranium Mill
- Mrs. Margaret Baker
  Carbon County Public Library
  Courthouse
  Rawlins, Wyo. 82301
  Shirley Basin Uranium Mill

Appendix 4

Regulations and Amendments—Fiscal Year 1978

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 thereto, which were published in the Federal Register during fiscal year 1978, are set forth below.

Clarifying and Corrective Amendments—Part 10

On October 6, 1977, amendments to Part 10 were published, effective immediately, which substitute certain office and officer designations so as to reflect the Commission's present staff functions and organization, or to make the designations consistent throughout, and which correct language inaccuracies regarding various statutory provisions of the Atomic Energy Act of 1954, as amended, and delete certain words which are superfluous.

Petitions for Rulemaking—Part 2

On October 31, 1977, an amendment to Part 2 was published, effective immediately, to state that the Director, Division of Rules and Records, Office of Administration, or his designee, will prepare on a quarterly basis a summary of petitions for rulemaking pending before the Commission, including status thereof, and a copy of the report will be available for public inspection and copying in the Commissions's Public Document Room, 1717 H Street, N.W., Washington, D.C.

General License for Government Agencies' Operational Use of Small Quantities of Source Material—Part 40

On December 7, 1977, an amendment to Part 40 was published, effective January 6, 1978, to include Federal, State, and local governmental agencies' research, development, education, or operational use of small quantities of source material in the general license which authorizes certain persons to use small quantities of source material.

Requests for Classification Review—Part 9

On December 9, 1977, an amendment to Part 9 was published, effective immediately, to provide guidance to members of the public desiring a classification review of a classified document of NRC.

Release of Transcripts of Closed Commission Meetings—Part 9

On December 13, 1977, an amendment to Part 9 was published, effective immediately, to remove the requirement that counsel attend all closed Commission meetings to advise the Commission on possible withholding of transcripts of those meetings.

Guard Force Response to an Alarm—Part 73

On December 22, 1977, an amendment to Part 73 was published, effective January 23, 1978, to clarify the alarm response requirements for onsite guards to protect special nuclear material from theft and licensed plants from industrial sabotage.

Caution Signs, Labels, Signals, and Controls—Part 20

On December 27, 1977, an amendment to Part 20 was published, effective March 14, 1978, which establishes additional requirements to improve safety in the use of and reduce the probability of accidental exposure of workers to sealed radioactive sources that produce very high levels of radiation.

Mixed Oxide Fuel—Order

On December 30, 1977, the Commission published an order announcing its decision to (1) terminate the GESMO proceeding; (2) terminate the proceedings on pending or future plutonium recycle-related license applications with certain exceptions; (3) reexamine the above matters after the completion of the ongoing alternative fuel cycle studies; (4) publish the draft safeguards supplement to the GESMO document as a staff technical report; (5) withdraw the November 1975 policy statement; and (6) reserve for decision the question of whether a facility may be licensed for experimental and feasibility purposes on a noncommercial basis to investigate processes which support the nation's nonproliferation objectives.

Foreign Gifts and Decorations—Part 0

On January 13, 1978, an amendment to Part 0 was published, effective January 1, 1978, which incorporates recently enacted amendments to the Foreign Gifts and Decorations Act of 1966.

Group Licensing for Certain Medical Uses—Part 35

On January 16, 1978, an amendment to Part 35 was published, effective immediately, to add a new reagent kit to its lists of authorized radioactive drugs, reagent kits, and procedures.

Exemption of Persons Using Spark Gap Irradiators Containing Cobalt-60—Part 30

On January 17, 1978, an amendment to Part 30 was published, effective February 16, 1978, to exempt from licensing and regulatory requirements persons using nuclear material near the spark gap of oil furnaces to prevent ignition problems.

Group Licensing for Certain Medical Uses—Part 35

On February 7, 1978, an amendment to Part 35 was published, effective immediately, to add a new reagent kit to the list of authorized radioactive drugs, reagent kits, and procedures.

Export and Import of Nuclear Facilities and Materials—Parts 2, 30, 31, 32, 33, 36, 40, 50, 51, 70, 73, and 110

On February 17, 1978, amendments to Parts 2, 30, 31, 32, 33, 36, 40, 50, 51, 70, 73, and 110 were published, effective May 3, 1978, to add a new part providing for standards, procedures, and rules of practice for licensing the export and import of utilization facilities, source, byproduct, and special nuclear materials. Conforming changes were also made to other parts of the Commission's regulations relating to export and import matters.

Distribution of Environmental Impact Statements—Parts 2 and 51

On February 21, 1978, amendments to Parts 2 and 51 were published, effective immediately, to reflect the transfer to the Environmental Protection Agency from the Council on Environmental Quality of certain responsibilities for the receipt and filing of Environmental Impact Statements and to change certain statutory citations to make them conform to the citations provided for by present law.

Revision of Fee Schedule—Part 170

On February 21, 1978, amendments to Part 170 were published, effective March 23, 1978, which revise the Commission's schedule of fees for applications, permits, and licenses.

Licensee Safeguards Contingency Plans—Parts 50, 70, and 73

On March 23, 1978, amendments to Parts 50, 70 and 73 were published, effective June 6, 1978, which require that licensees authorized to operate a nuclear reactor and those authorized to possess strategic quantities of plutonium, uranium-233, or uranium-235 develop and implement acceptable plans for responding to threats, thefts, and industrial sabotage of licensed nuclear facilities and materials.

Public Records—Part 9

On March 29, 1978, an amendment to Part 9 was published, effective immediately, to allow members of the public to make electronic sound recordings of open Commission meetings.

Uranium Fuel Cycle Impacts From Spent Fuel Reprocessing and Radioactive Waste Management—Part 51

On April 14, 1978, amendments to Part 51 were published, effective immediately, which remove the value contained in Table S-3 for releases of radon and to clarify that Table S-3 does not include health effects from the effluents described.

Orders to Show Cause—Part 2

On April 19, 1978, an amendment to Part 2 was published, effective immediately, to provide that the Director, Office of Administration, may institute a proceeding to modify, suspend, or revoke a license or for such other actions as may be proper by serving on the licensee an order to show cause.

Codes and Standards for Nuclear Power Plants—Part 50

On April 24, 1978, an amendment to Part 50 was published, effective May 24, 1978, to incorporate by reference a new edition and addenda of a national code that provides rules for the construction of nuclear power plant components.

Miscellaneous Amendments—Parts 2 and 8

On April 26, 1978, amendments to Parts 2 and 8 were published, effective May 26, 1978, to facilitate public participation in NRC facility license application review and hearing process, to improve coordination with States, counties, and municipalities, and to make certain other improvements.

Minor and Clarifying Amendments—Part 50

On May 1, 1978, an amendment to Part 50 was published, effective immediately, which relates to the service of copies of an updated application for a license to construct a production or utilization facility so as to reflect the current practice regarding the NRC officers who are to receive copies of such updated applications, and which provides clarifying language as to when the application should be updated, and which substitutes clarifying language regarding the service of copies of any subsequent amendments to the application.

Export and Import of Nuclear Equipment and Material—Part 110

On May 19, 1978, amendments to Part 110 were published, effective immediately and also requesting public comment by July 8, 1978. The amended regula tions incorporate the new export criteria mandated by the Nuclear Non-Proliferation Act of 1978 to govern exports of nuclear facilities, source material, and special nuclear material for peaceful nuclear uses.

Removal or Defacing of Radioactive Materials Labels on Empty Containers—Part 20

On May 24, 1978, an amendment to Part 20 was published, effective June 23, 1978, which requires the removal or the defacing of radioactive labels on empty, uncontaminated containers prior to disposal.

Authority of Secretary to Rule on Procedural Matters— Part 2

On May 25, 1978, an amendment to Part 2 was published, effective immediately, which expands the authority of the Secretary of the Commission to grant extensions of time for Commission review of certain decisions and actions.

Group Licensing for Certain Medical Uses—Part 35

On June 16, 1978, an amendment to Part 35 was published, effective immediately, which adds a new reagent kit to the lists of authorized radioactive drugs, reagents kits, and procedures.

Extension of the Implementation Period for Quality Assurance Program Requirements—Part 71

On June 23, 1978, an amendment to Part 71 was published, effective immediately, which extends until January 1, 1979, the date for filing a description of a quality assurance program for transport packages.

Domestic License Applications; Open Meetings and Statement of NRC Staff Policy

On June 28, 1978, the Commission announced its policy to allow concerned citizens to attend meetings conducted by the NRC technical staff as part of its review of a particular domestic license or permit application (including an application for an amendment to a license or permit).

Additional Functions of Executive Director for Operations—Part 1

On July 3, 1978, an amendment to Part 1 was published, effective immediately, to delegate to the Executive Director for Operations additional functions for dealing with petitions for rulemaking.

Clarifying Amendments—Part 20

On July 7, 1978, amendments to Part 20 were published, effective immediately, to clarify in 20.103 that exposure to radon-222 and its daughters may be averaged over 1 year and make minor editorial changes.

Advisory Committee on Reactor Safeguards—Part 1

On July 14, 1978, an amendment to Part 1 was published, effective immediately, to clarify that, upon requests from the Department of Energy, the Advisory Committee on Reactor Safeguards performs reviews, provides reports, and advises DOE with regard to the hazards of DOE nuclear activities and facilities.

Notice of Hearing—Part 2

On July 18, 1978, an amendment to Part 2 was published, effective immediately, to provide specifically for 30 days' Federal Register notice of the time and place of the initial hearings to be held on applications seeking construction permits for production or utilization facilities. Once this notice has been given, the presiding officer may reschedule the commencement of the hearing for a later date or reconvene a recessed hearing without again providing 30 days' notice.

Change of Address of Region II Office—Parts 1, 20, and 73

On July 28, 1978, amendments to Parts 1, 20, and 73 were published, effective immediately, to show the new address for the NRC Inspection and Enforcement Regional Office II.

Maintaining Integrity of Structures, Systems, and Components Important to Safety During Construction at Multiunit Sites—Part 50

On August 7, 1978, amendments to Part 50 were published, effective September 6, 1978, to require that, for multiunit sites, applicants for construction permits and operating licenses take proper precautions to assure the integrity of structures, systems, and components important to the safety of the operating unit or units during all construction activities.

Physical Protection of Plants and Materials—Part 73

On August 7, 1978, an amendment to Part 73 was published, effective immediately, to grant a one-time extension to delay full implementation of the physical protection requirements of 10 CFR 73.55 from August 24, 1978, to February 23, 1979.

Open Meetings—Part 9

On August 23, 1978, an amendment to Part 9 was published, effective immediately, to make explicit the present practice of not permitting parties to cite statements made at open meetings by Commissioners or Commission employees in support of argument presented to its adjucatory boards.

Application Form for Materials License—Medical—Part 35

On August 23, 1978, an amendment to Part 35 was published, effective November 6, 1978, to require use of a new form NRC-313M, "Applications for Materials License—Medical."

Security Personnel Qualification Training and Equipment Requirements—Part 73

On August 23, 1978, amendments to Part 73 were published, effective October 23, 1978, to impose upgraded guard qualification, training, and equipping requirements for security personnel protecting against theft of special nuclear materials and industrial sabotage of nuclear facilities or nuclear shipments.

Group Licensing for Certain Model Uses—Part 35

On September 7, 1978, an amendment to Part 35 was published, effective immediately, to add a new procedure to NRC's lists of authorized radioactive drugs, reagent kits, and procedures.

Personnel Monitoring Reports—Part 20

On September 29, 1978, amendments to Part 20 were published, effective December 13, 1978, to extend to all NRC specific licensees the requirement for submission of an annual statistical summary report on radiation exposure of workers.

Seismic and Geologic Siting Criteria for Nuclear Power Plants—Part 100

On January 19, 1978, a notice was published inviting public comments, suggestions, and information on the planned reassessment of current seismic criteria.

Notices, Instructions and Reports to Workers, Inspections; Standards for Protection Against Radiation—Parts 19 and 20

On February 6, 1978, proposed amendments to Parts 19 and 20 were published for comment which would require licensees to control the total occupational radiation dose of individuals.

Decommissioning Criteria for Nuclear Facilities—Parts 30, 40, 50, and 70

On March 13, 1978, an advance notice of proposed rulemaking to Parts 30, 40, 50, and 70 was published for comment which would provide more specific guidance on decommissioning criteria for production and utilization facility licensees and byproduct, source, and special nuclear material licensees.

Change in License Conditions for Certain Medical Licenses—Part 35

On March 17, 1978, proposed amendments to Part 35 were published for comment which would (1) permit physicians greater latitude, when they use certain low risk diagnostic radiopharmaceuticals, by no longer designating authorized clinical procedures and (2) delete from several licensing groups certain chemical forms not approved by FDA.

Regulation of the Medical Uses of Radioisotopes—Part 35

On March 17, 1978, a proposed policy statement to Part 35 was published for comment which is intended to inform NRC licensees, other Federal and State agencies, and the public of the Commission's general intention regarding the regulation of the medical uses of radioisotopes.

Amendments of Radiography Regulations—Part 34

On March 27, 1978, proposed amendments to Part 34 were published for comment which would reduce the radiation overexposure rate of radiographers and would formalize as regulations current licensing practices.

Licenses for Radiography and Radiation Safety Requirements for Radiographic Operations—Part 34

On March 27, 1978, an advance notice of proposed rulemaking was published for comment which would reduce radiation overexposures caused by equipment failure.

Antitrust Review Procedures—Parts 2 and 50

On April 26, 1978, proposed amendments to Parts 2 and 50 were published for comment to reduce or eliminate the requirements for submission of antitrust information in certain "de minimis" instances and to clarify requirements for antitrust review of applications for licenses for Class 103 facilities (commercial facilities) other than power reactors.

Petitions for Rulemaking—Part 2

On April 28, 1978, a proposed amendment to Part 2 was published for comment which would require the petitioner to include a statement in support of the petition setting forth the specific issues involved, the petitioner's views regarding those issues, and relevant technical, scientific, or other data involved which is reasonably available to the petitioner.

Facilities and Access for Resident Inspection—Parts 50 and 70

On May 9, 1978, proposed amendments to Parts 50 and 70 were published for comment which would require power reactor licensees and construction permit holders and selected fuel facilities licensees to provide (1) onsite, rent-free, exclusive use office space and (2) prompt licensee facility access to Commission inspection personnel.

Export of Certain Minor Quantities of Nuclear Material—Part 110

On May 9, 1978, proposed amendments to Part 110 were published for comment to establish or expand general and specific licensing provisions for the export of (1) gram quantities of special nuclear material, (2) certain classes of source material, and (3) certain classes of byproduct material.

General License Requirements for Any Person Who Possesses Special Nuclear Material in Transit—Part 70

On May 24, 1978, proposed amendments to Part 70 were published for comment which would require any person who possesses or who exercises control over formula quantities of special nuclear material in transit to be responsible for assuring that the special nuclear material is protected against theft and sabotage by a security system which is implemented in accordance with a Transportation Security Plan that has received prior NRC approval.

Special Nuclear Material of Moderate and Low Strategic Significance—Parts 70, 73, and 150

On May 24, 1978, proposed amendments to Parts 70, 73, and 150 were published for comment which would require physical protection measures against theft of special nuclear material of moderate and low strategic significance.

Safeguards on Nuclear Material—Implementation of US/IAEA Agreement—Parts 40, 50, 70, 75, and 150

On May 25, 1978, proposed amendments to Parts 40, 50, 70, 75, and 150 were published for comment which would require licensees (1) to submit information concerning their installations, for the use of IAEA; (2) to establish, maintain, and follow prescribed material accounting and control procedures; (3) to provide specific reports; and (4) to permit inspection by IAEA representatives.

Generic Issue of Financial Qualification—Part 50

On May 25, 1978, the NRC published a notice of its intention to initiate a rulemaking proceeding with respect to the requirement for demonstrating financial qualifications to obtain Part 50 licenses for production and utilization facilities.

Changes in License Conditions for Certain Medical Therapy Licensees—Part 35

On June 28, 1978, a proposed amendment to Part 35 was published for comment which would require licensees authorized to treat patients with implants incorporating radioactive material to confirm the removal of the implants at the end of the treatment by (1) a source count and (2) a radiation survey of the patient.

Misadministration Reporting Requirements—Part 35

On July 7, 1978, a proposed amendment to Part 35 was published for comment to require licensees to (1) keep records of all misadministrations of radioactive material or radiation from radioactive material and (2) promptly report potentially dangerous misadministrations to the NRC, to the patient's referring physician, and to the patient or the patient's responsible relative.

Change in License Conditions for Byproduct, Source, and Special Nuclear Material—Parts 30, 40, and 70

On July 27, 1978, proposed amendments to Parts 30, 40, and 70 were published for comment to require each licensee to notify the Commission when the licensee decides to permanently discontinue all activities involving materials authorized under a license.

Physical Protection of Plants and Materials—Parts 70 and 73

On August 9, 1978, proposed amendments to Parts 70 and 73 were published for comment for strengthened physical protection for strategic special nuclear material and for certain fuel cycle facilities, associated transportation, and other activities involving significant quantities of strategic special nuclear material.

Appendix E—Emergency Plans for Production and Utilization Facilities—Part 50

On August 23, 1978, a proposed amendment to Part 50 was published for comment to address emergency planning considerations that may extend to areas outside the low population zone.

Domestic Licensing of Special Nuclear Material—Parts 30 and 70

On September 28, 1978, proposed amendments to Parts 30 and 70 were published for comment which would establish requirements to be accomplished in the event of an irretrievable well-logging source (any sealed source containing licensed material which is pulled off or not connected to the wireline down-well, and for which all reasonable effort at recovery, as determined by the Commission, has been expended).

Appendix 5

Regulatory Guides — Fiscal Year 1978

Regulatory guides describe and make available to the public methods acceptable to the NRC staff 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 provide guidance to applicants concerning information needed by the staff in its review of applications for permits and licenses.

Comments and suggestions for improvements in guides are encouraged at all times, and guides will be revised, as appropriate, to accommodate comments and to reflect new information or experience. Regulatory guides may also be withdrawn when they are superseded by the Commission's regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes in methods and techniques have made them obsolete.

When guides are issued, revised, or withdrawn, notices are placed in the *Federal Register* and public announcements made. Single copies of guides may be obtained by writing to the Director, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. The following guides were issued or revised (or withdrawn as noted) during the period October 1, 1977, to September 30, 1978.

Division 1—Power Reactor Guides

1.28 Quality Assurance Program Requirements (Design and Construction) (Revision 1)
1.29 Seismic Design Classification (Revision 3)
1.31 Control of Ferrite Content in Stainless Steel Weld Metal (Revision 3)
1.33 Quality Assurance Program Requirements (Operation) (Revision 2)
1.52 Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Revision 2)
1.56 Maintenance of Water Purity in Boiling Water Reactors (Revision 1)
1.63 Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants (Revision 2)
1.68 Initial Test Programs for Water-Cooled Nuclear Power Plants (Revision 2)
1.68.2 Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Revision 1)
1.72 Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Revision 1)
1.75 Physical Independence of Electric Systems (Revision 2)
1.84 Code Case Acceptability—ASME Section III Design and Fabrication (Revisions 11, 12, and 13)
1.85 Code Case Acceptability—ASME Section III Materials (Revisions 11, 12, and 13)
1.91 Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (Revision 1)
1.109 Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Revision 1)
1.117 Tornado Design Classification (Revision 1)
1.118 Periodic Testing of Electric Power and Protection Systems (Revisions 1 and 2)
1.120 Fire Protection Guidelines for Nuclear Power Plants (Revision 1)
1.122 Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Revision 1)
1.124 Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Revision 1)
1.126 An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Revision 1)
1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants (Revision 1)
1.129 Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Revision 1)
1.136 Material for Concrete Containments
1.137 Fuel-Oil Systems for Standby Diesel Generators
1.138 Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants

1.139 Guidance for Residual Heat Removal
1.140 Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants
1.141 Containment Isolation Provisions for Fluid Systems
1.142 Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)
1.143 Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

Division 2—Research and Test Reactor Guides

2.5 Quality Assurance Program Requirements for Research Reactors (Revision O-R)

Division 3—Fuels and Materials Facilities Guides

3.4 Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (Revision 1-R)
3.5 Standard Format and Content of License Applications for Uranium Mills (Revision 1)
3.8 Preparation of Environmental Reports for Uranium Mills (Revision 1)
3.11 Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills (Revision 2)
3.40 Design Basis Floods for Fuel Reprocessing Plants and for Plutonium Processing and Fuel Fabrication Plants (Revision 1)
3.43 Nuclear Criticality Safety in the Storage of Fissile Materials

Division 4—Environmental and Siting Guides

4.10 WITHDRAWN Irreversible and Irretrievable Commitments of Material Resources
4.15 Quality Assurance for Radiological Monitoring Programs (Normal Operations)—Effluent Streams and the Environment
4.16 Measuring, Evaluating, and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Airborne Effluents from Nuclear Fuel and Fabrication Plants

Division 5—Materials and Plant Protection Guides

5.54 Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants
5.55 Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities
5.56 Standard Format and Content of Safeguards Contingency Plans for Transportation

Division 6—Product Guides

None

Division 7—Transportation Guides

7.6 Design Criteria for the Structural Analysis of Shipping Cask Containment (Revision 1)

Division 8—Occupational Health Guides

8.8 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable (Revision 3)
8.18 Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions Will Be As Low As Reasonably Achievable
8.19 Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants — Design Stage Man-Rem Estimates
8.20 Applications of Bioassay for I-125 and I-131
8.21 Health Physics Surveys for Byproduct Material at NRC-Licensed Processing and Manufacturing Plants
8.22 Bioassay at Uranium Mills

Division 9—Antitrust and Financial Review Guides

9.4 Suggested Format for Cash Flow Statements Submitted As Guarantees of Payment of Retrospective Premiums

Division 10—General Guides

None

Appendix 6

Nuclear Electric Generating Units in Operation, Under Construction or Planned

(As of September 30, 1978)

The following listing includes 212 nuclear power reactor electrical generating units which were in operation, under NRC review for construction permits, and ordered or announced by utilities in the United States at the end of September 1978, representing a total capacity of approximately 209,000 MWe. TYPE is indicated by: BWR—boiling water reactor, PWR—pressurized water reactor, HTGR—high temperature gas-cooled reactor, and LMFBR—liquid metal cooled fast breeder reactor. STATUS is indicated by: OL—has operating license, CP—has construction permit, UR—under review for construction permit, A/O—announced or ordered by the utility but application for construction not yet docketed by the NRC for review. The dates for operation are either actual or those scheduled by the utilities (N/S—not yet scheduled).

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ALABAMA						
Decatur	Browns Ferry Nuclear Power Plant Unit 1	1,065	BWR	OL	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant Unit 2	1,065	BWR	OL	Tennessee Valley Authority	1975
Decatur	Browns Ferry Nuclear Power Plant Unit 3	1,065	BWR	OL	Tennessee Valley Authority	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 1	829	BWR	OL	Alabama Power Co.	1978
Dothan	Joseph M. Farley Nuclear Plant Unit 2	829	PWR	CP	Alabama Power Co.	1980
Scottsboro	Bellefonte Nuclear Plant Unit 1	1,235	PWR	CP	Tennessee Valley Authority	1981
Scottsboro	Bellefonte Nuclear Plant Unit 2	1,235	PWR	CP	Tennessee Valley Authority	1981

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ARIZONA						
Winterburg	Palo Verde Nuclear Generating Station Unit 1	1,270	PWR	CP	Arizona Public Service Co.	1982
Winterburg	Palo Verde Nuclear Generating Station Unit 2	1,270	PWR	CP	Arizona Public Service Co.	1984
Winterburg	Palo Verde Nuclear Generating Station Unit 3	1,270	PWR	CP	Arizona Public Service Co.	1986
Winterburg	Palo Verde Nuclear Generating Station Unit 4	1,270	PWR	UR	Arizona Public Service Co.	1988
Winterburg	Palo Verde Nuclear Generating Station Unit 5	1,270	PWR	UR	Arizona Public Service Co.	1990
ARKANSAS						
Russelville	Arkansas Nuclear One Unit 1	850	PWR	OL	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One Unit 2	912	PWR	OL	Arkansas Power & Light Co.	1978
CALIFORNIA						
Eureka	Humboldt Bay Power Plant Unit 3	65	BWR	OL	Pacific Gas & Electric Co.	1963
San Clemente	San Onofre Nuclear Generating Station Unit 1	436	PWR	OL	So. Calif. Ed. & San Diego Gas & Electric Co.	1968
San Clemente	San Onofre Nuclear Generating Station Unit 2	1,140	PWR	CP	So. Calif. Ed. & San Diego Gas & Electric Co.	1980
San Clemente	San Onofre Nuclear Generating Station Unit 3	1,140	PWR	CP	So. Calif. Ed. & San Diego Gas & Electric Co.	1981
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 1	1,084	PWR	CP	Pacific Gas & Elec. Co.	1979
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 2	1,106	PWR	CP	Pacific Gas & Elec. Co.	1979
Clay Station	Rancho Seco Nuclear Generating Station Unit 1	917	PWR	OL	Sacramento Municipal Utility District	1975

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
*	Stanislaus Unit 1	1,200	BWR	A/O	Pacific Gas & Elec. Co.	Indef.
*	Stanislaus Unit 2	1,200	BWR	A/O	Pacific Gas & Elec. Co.	Indef.
Clay Station	Rancho Seco Nuclear Generating Station Unit 2	1,100		A/O	Sacramento Municipal Utility District	Indef.
COLORADO						
Platteville	Fort St. Vrain Nuclear Generating Station	330	HTGR	OL	Public Service Co. of of Colorado	1978
CONNECTICUT						
Haddam Neck	Haddam Neck Generating Station	575	PWR	OL	Conn. Yankee Atomic Power Co.	1968
Waterford	Millstone Nuclear Power Station Unit 1	660	BWR	OL	Northeast Nuclear Energy Co.	1971
Waterford	Millstone Nuclear Power Station Unit 2	830	PWR	OL	Northeast Nuclear Energy Co.	1975
Waterford	Millstone Nuclear Power Station Unit 3	1,159	PWR	CP	Northeast Nuclear Energy Co.	1986
DELAWARE						
Summit	Summit Power Station Unit 1	1,200		A/O**	Delmarva Power & Light Co.	N/S
FLORIDA						
Florida City	Turkey Point Station Unit 3	693	PWR	OL	Florida Power & Light Co.	1972
Florida City	Turkey Point Station Unit 4	693	PWR	OL	Florida Power & Light Co.	1973
Red Level	Crystal River Plant Unit 3	825	PWR	OL	Florida Power Corp. Light Co.	1977
Ft. Pierce	St. Lucie Plant Unit 1	802	PWR	OL	Florida Power Corp. Light Co.	1976
Ft. Pierce	St. Lucie Plant Unit 2	842	PWR	CP	Florida Power Corp. Light Co.	1983

*Site not selected.

**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
GEORGIA						
Baxley	Edwin I. Hatch Plant Unit 1	786	BWR	OL	Georgia Power Co.	1975
Baxley	Edwin I. Hatch Plant Unit 2	795	BWR	OL	Georgia Power Co.	1978
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 1	1,100	PWR	CP	Georgia Power Co.	1984
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 2	1,100	PWR	CP	Georgia Power Co.	1985
ILLINOIS						
Morris	Dresden Nuclear Power Station Unit 1	200	BWR	OL	Commonwealth Edison Co.	1960
Morris	Dresden Nuclear Power Station Unit 2	794	BWR	OL	Commonwealth Edison Co.	1970
Morris	Dresden Nuclear Power Station Unit 3	794	BWR	OL	Commonwealth Edison Co.	1971
Zion	Zion Nuclear Plant Unit 1	1,040	PWR	OL	Commonwealth Edison Co.	1973
Zion	Zion Nuclear Plant Unit 2	1,040	PWR	OL	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Station Unit 1	789	BWR	OL	Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co.	1973
Cordova	Quad-Cities Station Unit 2	789	BWR	OL	Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co.	1973
Seneca	LaSalle County Nuclear Station Unit 1	1,078	BWR	CP	Commonwealth Edison Co.	1979
Seneca	LaSalle County Nuclear Station Unit 2	1,078	BWR	CP	Commonwealth Edison Co.	1980
Byron	Byron Station Unit 1	1,120	PWR	CP	Commonwealth Edison Co.	1981
Byron	Byron Station Unit 2	1,120	PWR	CP	Commonwealth Edison Co.	1982
Braidwood	Braidwood Unit 1	1,120	PWR	CP	Commonwealth Edison Co.	1981
Braidwood	Braidwood Unit 2	1,120	PWR	CP	Commonwealth Edison Co.	1982
Clinton	Clinton Nuclear Power Plant Unit 1	950	BWR	CP	Illinois Power Co.	1982
Clinton	Clinton Nuclear Power Plant Unit 2	950	BWR	CP	Illinois Power Co.	1988
Savannah	Carroll County Station Unit 1	1,120		A/O	Commonwealth Edison Co.	1984

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Savannah	Carroll County Station Unit 2	1,120		A/O	Commonwealth Edison Co.	1985
INDIANA						
Westchester Town	Bailly Generating Station	660	BWR	CP	Northern Indiana Public Service Co.	1984
Madison	Marble Hill Unit 1	1,130	PWR	CP	Public Service of Indiana	1982
Madison	Marble Hill Unit 2	1,130	PWR	CP	Public Service of Indiana	1984
IOWA						
Pala	Duane Arnold Energy Center Unit 1	538	BWR	OL	Iowa Elec. Light & Power Co.	1975
Vandalia	Iowa Power Unit 1	1,270	BWR	A/O	Iowa Po. & Lt. Co.	N/S
KANSAS						
Burlington	Wolf Creek	1,150	PWR	CP	Kansas Gas & Elec. Co.	1983
LOUISIANA						
Taft	Waterford Steam Electric Station Unit 3	1,165	PWR	CP	Louisiana Power & Light Co.	1981
St. Francisville	River Bend Station Unit 1	934	BWR	CP	Gulf States Utilities Co.	1984
St. Francisville	River Bend Station Unit 2	934	BWR	CP	Gulf States Utilities Co.	N/S
MAINE						
Wiscasset	Maine Yankee Atomic Power Plant	790	PWR	OL	Maine Yankee Atomic Power Co.	1972
MARYLAND						
Lusby	Calvert Cliffs Nuclear Power Plant Unit 1	845	PWR	OL	Baltimore Gas & Elec. Co.	1975
Lusby	Calvert Cliffs Nuclear Power Plant Unit 2	845	PWR	OL	Baltimore Gas & Elec. Co.	1977

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Douglas Point	Douglas Point Generating Station Unit 1	1,146	BWR	UR	Potomac Electric Power Co.	Indef.
MASSACHUSETTS						
Rowe	Yankee Nuclear Power Station	175	PWR	OL	Yankee Atomic Elec. Co.	1961
Plymouth	Pilgrim Station Unit 1	655	BWR	OL	Boston Edison Co.	1972
Plymouth	Pilgrim Station Unit 2	1,180	PWR	UR	Boston Edison Co.	1985
Turners Falls	Montague Unit 1	1,150	BWR	UR	Northeast Nuclear Energy Co.	N/S
Turners Falls	Montague Unit 2	1,150	BWR	UR	Northeast Nuclear Energy Co.	N/S
MICHIGAN						
Big Rock Point	Big Rock Point Nuclear Plant	72	BWR	OL	Consumers Power Co.	1963
South Haven	Palisades Nuclear Power Station	805	PWR	OL	Consumers Power Co.	1971
Lagoona Beach	Enrico Fermi Atomic Power Plant Unit 2	1,123	BWR	CP	Detroit Power Co.	1980
Bridgman	Donald C. Cook Plant Unit 1	1,054	PWR	OL	Indiana & Michigan Elec. Co.	1975
Bridgman	Donald C. Cook Plant Unit 2	1,100	PWR	OL	Indiana & Michigan Elec. Co.	1978
Midland	Midland Nuclear Power Plant Unit 1	492	PWR	CP	Consumers Power Co.	1982
Midland	Midland Nuclear Power Plant Unit 2	818	PWR	CP	Consumers Power Co.	1981
St. Clair County	Greenwood Energy Center Unit 2	1,200	PWR	UR	Detroit Edison Co.	N/S
St. Clair County	Greenwood Energy Center Unit 3	1,200	PWR	UR	Detroit Edison Co.	N/S
MINNESOTA						
Monticello	Monticello Nuclear Generating Plant	545	BWR	OL	Northern States Power Co.	1971
Red Wing	Prairie Island Nuclear Generating Plant Unit 1	530	PWR	OL	Northern States Power Co.	1973

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Red Wing	Prairie Island Nuclear Generating Plant Unit 2	530	PWR	OL	Northern States Power Co.	1974
MISSOURI						
Fulton	Callaway Plant Unit 1	1,150	PWR	CP	Union Elec. Co.	1982
Fulton	Callaway Plant Unit 2	1,150	PWR	CP	Union Elec. Co.	1987
MISSISSIPPI						
Port Gibson	Grand Gulf Nuclear Station Unit 1	1,250	BWR	CP	Mississippi Power & Light Co.	1981
Port Gibson	Grand Gulf Nuclear Station Unit 2	1,250	BWR	CP	Mississippi Power & Light Co.	1984
Yellow Creek	Yellow Creek Unit 1	1,285	PWR	UR**	Tennessee Valley Authority	1985
Yellow Creek	Yellow Creek Unit 2	1,285	PWR	UR**	Tennessee Valley Authority	1985
NEBRASKA						
Fort Calhoun	Fort Calhoun Station Unit 1	457	PWR	OL	Omaha Public Power District	1973
Brownville	Cooper Nuclear Station	778	BWR	OL	Nebraska Public Power District	1974
NEW HAMPSHIRE						
Seabrook	Seabrook Nuclear Station Unit 1	1,194	PWR	CP	Public Service of N.H.	1983
Seabrook	Seabrook Nuclear Station Unit 2	1,194	PWR	CP	Public Service of N.H.	1985
NEW JERSEY						
Toms River	Oyster Creek Nuclear Power Plant Unit 1	650	BWR	OL	Jersey Central Power & Light Co.	1969
Forked River	Forked River Generating Station Unit 1	1,070	PWR	CP	Jersey Central Power & Light Co.	1984

**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Salem	Salem Nuclear Generating Station Unit 1	1,090	PWR	OL	Public Service Elec. & Gas Co.	1977
Salem	Salem Nuclear Generating Station Unit 2	1,115	PWR	CP	Public Service Elec. & Gas Co.	1979
Salem	Hope Creek Generating Station Unit 1	1,067	BWR	CP	Public Service Elec. & Gas Co.	1984
Salem	Hope Creek Generating Station Unit 2	1,067	BWR	CP	Public Service Elec. & Gas Co.	1986
Little Egg Inlet	Atlantic Generating Station Unit 1	1,150	PWR	UR	Public Service Elec. & Gas Co.	N/S
Little Egg Inlet	Atlantic Generating Station Unit 2	1,150	PWR	UR	Public Service Elec. & Gas Co.	N/S
*	Atlantic Generating Station Unit 3	1,150	PWR	A/O	Public Service Elec. & Gas Co.	N/S
*	Atlantic Generating Station Unit 4	1,150	PWR	A/O	Public Service Elec. & Gas Co.	N/S
NEW YORK						
Indian Point	Indian Point Station Unit 1	265	PWR	OL	Consolidated Edison Co.	1962
Indian Point	Indian Point Station Unit 2	873	PWR	OL	Consolidated Edison Co.	1973
Indian Point	Indian Point Station Unit 3	965	PWR	OL	Consolidated Edison Co.	1976
Scriba	Nine Mile Point Nuclear Station Unit 1	610	BWR	OL	Niagara Mohawk Power Co.	1969
Scriba	Nine Mile Point Nuclear Station Unit 2	1,080	BWR	CP	Niagara Mohawk Power Co.	1983
Ontario	R. E. Ginna Nuclear Power Plant Unit 1	490	PWR	OL	Rochester Gas & Elec. Co.	1970
Brookhaven	Shoreham Nuclear Power Station	854	BWR	CP	Long Island Lighting Co.	1980
Scriba	James A. FitzPatrick Nuclear Power Plant	821	BWR	OL	Power Authority of State of N.Y.	1975

*Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Long Island	Jamesport Unit 1	1,150	PWR	UR	Long Island Lighting Co.	1988
Long Island	Jamesport Unit 2	1,150	PWR	UR	Long Island Lighting Co.	1990
*	New Haven 1	1,250	PWR	A/O	N.Y. State Elec. & Gas. Co.	Indef.
*	New Haven 2	1,250	PWR	A/O	N.Y. State Elec. & Gas Co.	Indef.
Sterling	Sterling Power Project Unit 1	1,150	PWR	CP	Rochester Gas & Elec. Co.	1988
Cementon	Greene County Nuclear Power Plant	1,270	PWR	UR	Power Authority of State of N.Y.	1986
*	Mid-Hudson East 1	1,300		A/O	Empire State Power Resources	N/S
*	Nine Mile Point 3	1,300		A/O	Empire State Power Resources	N/S

NORTH CAROLINA

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Southport	Brunswick Steam Electric Plant Unit 2	821	BWR	OL	Carolina Power & Light Co.	1975
Southport	Brunswick Steam Electric Plant Unit 1	821	BWR	OL	Carolina Power & Light Co.	1977
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 1	1,180	PWR	CP	Duke Power Co.	1979
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 2	1,180	PWR	CP	Duke Power Co.	1981
Bonsal	Shearon Harris Plant Unit 1	915	PWR	CP	Carolina Power & Light Co.	1983
Bonsal	Shearon Harris Plant Unit 2	915	PWR	CP	Carolina Power & Light Co.	1985
Bonsal	Shearon Harris Plant Unit 3	915	PWR	CP	Carolina Power & Light Co.	1989
Bonsal	Shearon Harris Plant Unit 4	915	PWR	CP	Carolina Power & Light Co.	1987
Davie Co.	Perkins Nuclear Station Unit 1	1,280	PWR	UR	Duke Power Co.	1988
Davie Co.	Perkins Nuclear Station Unit 2	1,280	PWR	UR	Duke Power Co.	1991
Davie Co.	Perkins Nuclear Station Unit 3	1,280	PWR	UR	Duke Power Co.	1993

*Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
*	Carolina P&L Unit 8	1,150	PWR	A/O	Carolina Power & Light Co.	—
*	Carolina P&L Unit 9	1,150	PWR	A/O	Carolina Power & Light Co.	—
OHIO						
Oak Harbor	Davis-Besse Nuclear Power Station Unit 1	906	PWR	OL	Toledo Edison-Cleveland Elec. Illum. Co.	1977
Oak Harbor	Davis-Besse Nuclear Power Station Unit 2	906	PWR	UR**	Toledo Edison-Cleveland Elec. Illum. Co.	1986
Oak Harbor	Davis-Besse Nuclear Power Station Unit 3	906	PWR	UR**	Toledo Edison-Cleveland Elec. Illum. Co.	1988
Perry	Perry Nuclear Power Plant Unit 1	1,205	BWR	CP	Cleveland Elec. Illum. Co.	1981
Perry	Perry Nuclear Power Plant Unit 2	1,205	BWR	CP	Cleveland Elec. Illum. Co.	1983
Moscow	Wm. H. Zimmer Nuclear Power Station Unit 1	810	BWR	CP	Cincinnati Gas & Elec. Co.	1979
Berlin Hgts.	Erie Unit 1	1,260	PWR	UR	Ohio Edison Co.	1986
Berlin Hgts.	Erie Unit 2	1,260	PWR	UR	Ohio Edison Co.	1988
OKLAHOMA						
Inola	Black Fox Unit 1	1,150	BWR	UR**	Public Service Co. of Oklahoma	1983
Inola	Black Fox Unit 2	1,150	BWR	UR**	Public Service Co. of Oklahoma	1985
OREGON						
Prescott	Trojan Nuclear Plant Unit 1	1,130	PWR	OL	Portland General Elec. Co.	1976
Arlington	Pebble Springs Unit 1	1,260	PWR	UR	Portland General Elec. Co.	1986
Arlington	Pebble Springs Unit 2	1,260	PWR	UR	Portland General Elec. Co.	1989

*Site not selected.
**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
PENNSYLVANIA						
Peach Bottom	Peach Bottom Atomic Power Station Unit 2	1,065	BWR	OL	Philadelphia Elec. Co.	1974
Peach Bottom	Peach Bottom Atomic Power Station Unit 3	1,065	BWR	OL	Philadelphia Elec. Co.	1974
Pottstown	Limerick Generating Station Unit 1	1,065	BWR	CP	Philadelphia Elec. Co.	1983
Pottstown	Limerick Generating Station Unit 2	1,065	BWR	CP	Philadelphia Elec. Co.	1985
Shippingport	Shippingport Atomic Power Unit 1	90	PWR	—[1]	Duquesne Light Co. & ERDA	NA
Shippingport	Beaver Valley Power Station Unit 1	852	PWR	OL	Duquesne Light Co. Ohio Edison Co.	1976
Shippingport	Beaver Valley Power Station Unit 2	852	PWR	CP	Duquesne Light Co. Ohio Edison Co.	1982
Goldsboro	Three Mile Island Nuclear Station Unit 1	819	PWR	OL	Metropolitan Edison Co.	1974
Goldsboro	Three Mile Island Nuclear Station Unit 2	906	PWR	OL	Metropolitan Edison Co.	1978
Berwick	Susquehanna Steam Electric Station Unit 1	1,052	BWR	CP	Pennsylvania Power & Light Co.	1980
Berwick	Susquehanna Steam Electric Station Unit 2	1,052	BWR	CP	Pennsylvania Power & Light Co.	1982
Fulton	Fulton Generating Station Unit 1	1,160		UR	Philadelphia Elec. Co.	N/S
Fulton	Fulton Generating Station Unit 2	1,160		UR	Philadelphia Elec. Co.	N/S
RHODE ISLAND						
No. Kingston	New England Unit 1	1,194	PWR	UR	New England Power Co.	1987
No. Kingston	New England Unit 2	1,194	PWR	UR	New England Power Co.	1989
SOUTH CAROLINA						
Hartsville	H. B. Robinson S. E. Plant Unit 2	700	PWR	OL	Carolina Power & Light Co.	1971
Seneca	Oconee Nuclear Station Unit 1	887	PWR	OL	Duke Power Co.	1973

[1]Operable but OL not required.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Seneca	Oconee Nuclear Station Unit 2	887	PWR	OL	Duke Power Co.	1974
Seneca	Oconee Nuclear Station Unit 3	887	PWR	OL	Duke Power Co.	1974
Broad River	Virgil C. Summer Nuclear Station Unit 1	900	PWR	CP	So. Carolina Elec. & Gas Co.	1980
Lake Wylie	Catawba Nuclear Station Unit 1	1,145	PWR	CP	Duke Power Co.	1981
Lake Wylie	Catawba Nuclear Station Unit 2	1,145	PWR	CP	Duke Power Co.	1983
Cherokee County	Cherokee Nuclear Station Unit 1	1,280	PWR	CP	Duke Power Co.	1984
Cherokee County	Cherokee Nuclear Station Unit 2	1,280	PWR	CP	Duke Power Co.	1986
Cherokee County	Cherokee Nuclear Station Unit 3	1,280	PWR	CP	Duke Power Co.	1988

TENNESSEE

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Daisy	Sequoyah Nuclear Power Plant Unit 1	1,140	PWR	CP	Tennessee Valley Authority	1979
Daisy	Sequoyah Nuclear Power Plant Unit 2	1,140	PWR	CP	Tennessee Valley Authority	1980
Spring City	Watts Bar Nuclear Plant Unit 1	1,165	PWR	CP	Tennessee Valley Authority	1979
Spring City	Watts Bar Nuclear Plant Unit 2	1,165	PWR	CP	Tennessee Valley Authority	1980
Oak Ridge	Clinch River Breeder Reactor Plant	350	LMFBR	UR	U.S. Government	Indef.
Hartsville	TVA Plant 1 Unit 1	1,205	BWR	CP	Tennessee Valley Authority	1982
Hartsville	TVA Plant 1 Unit 2	1,205	BWR	CP	Tennessee Valley Authority	1983
Hartsville	TVA Plant 2 Unit 1	1,205	BWR	CP	Tennessee Valley Authority	1983
Hartsville	TVA Plant 2 Unit 2	1,205	BWR	CP	Tennessee Valley Authority	1984
Phipps Bend	Phipps Bend Unit 1	1,220	BWR	CP	Tennessee Valley Authority	1983
Phipps Bend	Phipps Bend Unit 2	1,220	BWR	CP	Tennessee Valley Authority	1984

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station Unit 1	1,150	PWR	CP	Texas P&L, Dallas P&L, Texas Elec. Service	1981
Glen Rose	Comanche Peak Steam Electric Station Unit 2	1,150	PWR	CP	Texas P&L, Dallas P&L, Texas Elec. Service	1983
Wallis	Allens Creek Unit 1	1,213	BWR	UR	Houston Lighting & Power Co.	1985
Bay City	South Texas Nuclear Project Unit 1	1,250	PWR	CP	Houston Lighting & Power Co.	1980
Bay City	South Texas Nuclear Project Unit 2	1,250	PWR	CP	Houston Lighting & Power Co.	1982
VERMONT						
Vernon	Vermont Yankee Generating Station	514	BWR	OL	Vermont Yankee Nuclear Power Corp.	1972
VIRGINIA						
Gravel Neck	Surry Power Station Unit 1	822	PWR	OL	Va. Electric & Power Co.	1972
Gravel Neck	Surry Power Station Unit 2	822	PWR	OL	Va. Electric & Power Co.	1973
Mineral	North Anna Power Station Unit 1	907	PWR	OL	Va. Electric & Power Co.	1978
Mineral	North Anna Power Station Unit 2	907	PWR	CP	Va. Electric & Power Co.	1979
Mineral	North Anna Power Station Unit 3	907	PWR	CP	Va. Electric & Power Co.	1982
Mineral	North Anna Power Station Unit 4	907	PWR	CP	Va. Electric & Power Co.	1983
*	Central Virginia 1	1,150		A/O	American Electric Power Co.	1990
*	Central Virginia 2	1,150		A/O	American Electric Power Co.	1990
WASHINGTON						
Richland	N-Reactor/WPPSS Steam	850	GR	—[1]	Wash. Public Power Supply System	

*Site not selected.
[1] Operable but OL not required.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
Richland	WPPSS No. 1 (Hanford)	1,267	PWR	CP	Wash. Public Power Supply System	1982
Richland	WPPSS No. 2 (Hanford)	1,103	BWR	CP	Wash. Public Power Supply System	1980
Satsop	WPPSS No. 3	1,242	PWR	CP	Wash. Public Power Supply System	1984
Richland	WPPSS No. 4	1,267	PWR	CP	Wash. Public Power Supply System	1984
Satsop	WPPSS No. 5	1,242	PWR	CP	Wash. Public Power Supply System	1985
Sedro Wooley	Skagit Nuclear Power Project Unit 1	1,277	BWR	UR	Puget Sound Power & Light Co.	1985
Sedro Wooley	Skagit Nuclear Power Project Unit 2	1,277	BWR	UR	Puget Sound Power & Light Co.	1987
WISCONSIN						
Genoa	Genoa Nuclear Generating Station (LaCrosse)	50	BWR	OL	Dairyland Power Coop.	1969
Two Creeks	Point Beach Nuclear Plant Unit 1	497	PWR	OL	Wisconsin Michigan Power Co.	1970
Two Creeks	Point Beach Nuclear Plant Unit 2	497	PWR	OL	Wisconsin Michigan Power Co	1972
Carlton	Kewaunee Nuclear Power Plant Unit 1	535	PWR	OL	Wisconsin Elec. Power Co.	1974
Durand	Tyrone Energy Park Unit 1	1,150	PWR	CP	Northern States Power Co.	1985
Ft. Atkinson	Haven Nuclear Plant Unit 1	900	PWR	UR	Wisconsin Elec. Power Co.	1987
Ft. Atkinson	Haven Nuclear Plant Unit 2	900	PWR	UR	Wisconsin Elec. Power Co.	1989
PUERTO RICO						
Arecibo	North Coast Nuclear Plant Unit 1	583	PWR	UR	Puerto Rico Water Resources Authority	Indef.

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