U. S. Nuclear Regulatory Commission

Annual Report 1975

The President

The White House

Dear Mr. President:

We have the honor to submit herewith the first 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 January 19, 1975, through December 31, 1975.

Respectfully,

William A. Anders

Chairman

U. S. Nuclear Regulatory Commission

Annual Report 1975

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Annual Report Statutory Requirements

Section 307(c) of the Energy Reorganization Act of 1974 directs the Commission to include in its annual report statements and descriptions covering a number of specific areas. These requirements and the chapters in which they are addressed are as follows:

"... a clear statement of the short-range and long-range goals, priorities, and plans of the Commission as they relate to the benefits, costs, and risks of commercial nuclear power." (See Chapter 1.)

"... a clear description of the Commission's activities and findings in the following areas —

"(1) insuring the safe design of nuclear power plants and other licensed facilities ..." (See discussions, "Ensuring Safe Design" and "Advanced Reactor Reviews" in Chapter 2.)

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

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle ..." (See Chapter 5.)

"(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 Chapter 5.)

"(5) insuring the safe, permanent disposal of high-level radioactive wastes through the licensing of nuclear activities and facilities ..." (See discussion, "Nuclear Waste Management," in Chapter 4.)

"(6) protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities." (See discussion, "Control of Radioactive Emissions," in Chapter 3.)

The President met with the NRC Commissioners-Designate at the White House on January 17 to support the launching of the new and independent regulatory agency. From left, Commissioner Richard T. Kennedy, Commissioner Marcus A. Rowden, Chairman William A. Anders, President Gerald R. Ford, Commissioner Victor Gilinsky, and Commissioner Edward A. Mason.

Introduction and Overview

THE NRC MISSION

The Nuclear Regulatory Commission began official operation on January 19, 1975, the effective date of the Energy Reorganization Act of 1974, which was signed into law on October 11, 1974. That Act dissolved the Atomic Energy Commission and created both NRC and the Energy Research and Development Administration, the latter to carry out research and development on all forms of energy. When he signed the legislation, President Ford said of the tasks facing NRC:

> "The highly technical nature of our nuclear facilities and the special potential hazards which are involved in the use of nuclear fuels fully warrant the creation of an independent and technically competent regulatory agency to assure adequate protection of public health and safety. NRC will be responsible for the licensing and regulation of the nuclear industry under the provisions of the Atomic Energy Act. This means that NRC will be fully empowered to see to it that reactors using nuclear materials will be properly and safely designed, constructed and operated to guarantee against hazards to the public from leakage or accident. NRC will also exercise strengthened authority to assure that the public is fully safeguarded from hazards arising from the storage, handling and transportation of nuclear materials being used in power reactors, hospitals, research laboratories or for any other purpose."

NRC Authorities

The Energy Reorganization Act, among other things, transferred to NRC the responsibility for carrying out the regulatory provisions of the Atomic Energy Act of 1954, as amended. These provisions established national policy and a framework for regulating civilian nuclear energy activities to ensure that they are conducted in a manner which will protect public health and safety, maintain national security, and comply with the antitrust laws.

Chairman Anders takes his oath of office from Supreme Court Justice Harry Blackmun at a Capitol ceremony presided over by Vice President Nelson Rockefeller, as Mrs. Anders holds the Bible. Justice Blackmun also administered the oath to Commissioners Kennedy, Gilinsky, Mason, and Rowden.

Under the National Environmental Policy Act of 1969, the Commission also assumed responsibility for evaluating the nonradiological as well as radiological impact on the environment of major nuclear facilities proposed for licensing, and for balancing the benefits of such facilities against their environmental and social costs.

The Energy Reorganization Act expanded nuclear regulation activities to be conducted by NRC and charged the agency with several special tasks. For example, it authorized the NRC to administer a strong regulatory research program, to license and regulate certain nuclear waste management activities, and to conduct extensive studies on safeguards needs and the feasibility of nuclear energy centers. The Act also took note of the expanding civilian use of nuclear materials by directing the NRC to increase significantly the emphasis placed on safeguarding strategic nuclear materials and nuclear facilities against theft, diversion or sabotage.

NRC Programs

To accomplish its missions, NRC maintains broad programs of standards-setting and rulemaking, technical reviews and studies, licensing actions, inspection and enforcement, evaluation of operating experience, and regulatory research. The objectives and first-year accomplishments of these programs are described in detail in later chapters of this report.

The major portion of NRC effort is directed to the regulation of nuclear power reactors. As of December 31, 1975, there were 236 nuclear power generating units in operation, being built, or planned in the U.S., representing a total capacity of some

236,000 electrical megawatts (MWe). Of these, 198 reactors had advanced far enough into the planning stage to be under direct surveillance of the NRC regulatory process; 56 of the 198, with a total capacity of about 39,000 MWe, were already licensed for operation.

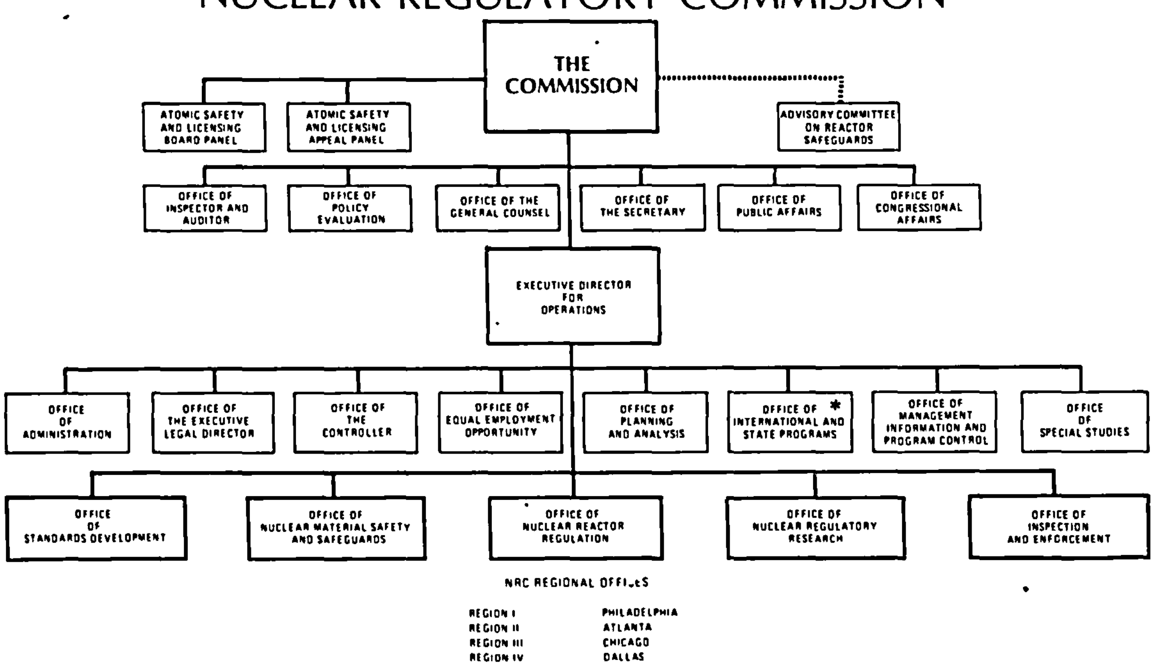
NRC is charged with regulating most of the activities associated with the nuclear fuel cycle, including the milling of uranium ore, its conversion into nuclear fuel materials, and the fabrication of these materials into fuel elements. The Commission also regulates the utilization of finished fuel cycle materials, the reprocessing and transportation of "spent" fuel elements, and their interim and final disposition. The mining of uranium, and its enrichment in government plants are under the purview of other government agencies.

Outside the fuel cycle, NRC regulates the production and use of a wide varierty of nuclear materials in industry, medicine, and research involving quantities ranging from a few millionths of a curie of radioactivity to millions of curies. In 1975, NRC was directly administering a total of about 11,000 facility, materials and individual operator licenses held by more than 9,000 licensees throughout the country. The NRC also was monitoring nuclear regulatory programs in 25 States which, under the terms of agreements with the Commission, exercise regulatory control over an additional 10,500 materials licenses.

NRC ORGANIZATION

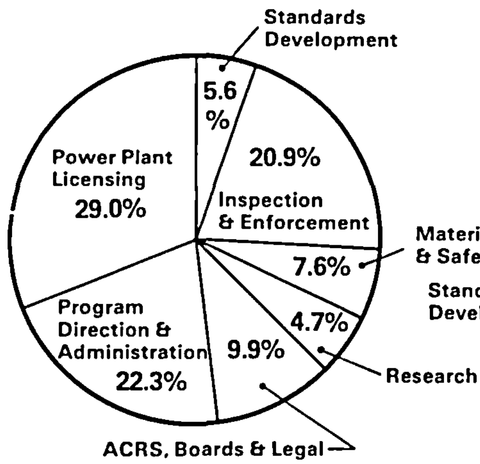
The Energy Reorganization Act assigns responsibility for carrying out NRC's obligations to five Commissioners, each appointed by the President with the advice and consent of the Senate. One member, designated by the President as Chairman, acts as Chief Executive Officer of the Commission and is its official spokesman. Members of the first Nuclear Regulatory Commission were formally sworn into office on

NUCLEAR REGULATORY COMMISSION

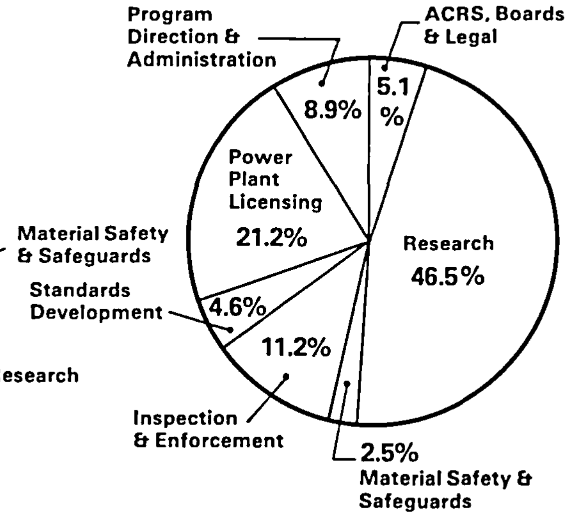


*At year-end, NRC was in the process of establishing separate offices for International Programs and State Programs.

NRC RESOURCES FY 1975



PERSONNEL — 2006 (June 30, 1975)



FUNDS — $148 MILLION

January 23, 1975, during a ceremony presided over by Vice President Nelson A. Rockefeller, with the oath of office administered by U.S. Supreme Court Justice Harry M. Blackmun.

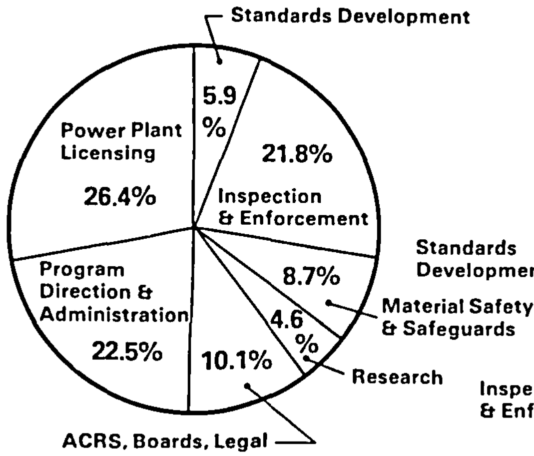
The Energy Reorganization Act also specified several organizational units below the Commission level, including an Executive Director for Operations and three principal "line" offices—Nuclear Reactor Regulation, Nuclear Material Safety and Safeguards, and Nuclear Regulatory Research. During the transition phase of organizational development, the Commission created two additional line offices—Inspection and Enforcement, and Standards Development—and completed the supporting framework of staff offices. The Commission and its top executives were responsible for managing fiscal year 1975 and 1976 budgets of $148 million and $222 million respectively, and a staff that numbered 2,199 as of December 31, 1975. In keeping with the diversity and complexity of its mission, the NRC comprises one of the broadest spectrums of technical and scientific disciplines in government. More than 90 percent of the professional staff hold college degrees, and about half also have advanced degrees.

GOALS AND OBJECTIVES

The primary reasons for nuclear regulation as expressed in the statutes governing NRC are protection of public health and safety, national security and the environment. These aims are constants which demand continuous vigilance to achieve and maintain levels of protection that are acceptable to society. Maintaining and enhancing these protective levels is the overall goal of the Commission. They require the establishment and attainment of both short-term objectives—for example, the early resolution of technical problems arising in the operation of today's light-water-cooled power reactors, and the assurance of safe interim storage for spent nuclear fuel elements—and longer-term objectives such as closing the nuclear fuel cycle, regulating the safe disposition of high-level radioactive wastes, and improving capability to evaluate the safety of advanced reactor proposals.

Balancing the benefits of nuclear activities against safety, security and environmental costs and risks in the public interest is the heart of NRC decisionmaking. The Commission believes, however, that consideration of the administrative and economic burdens of regulatory actions can

NRC RESOURCES FY 1976



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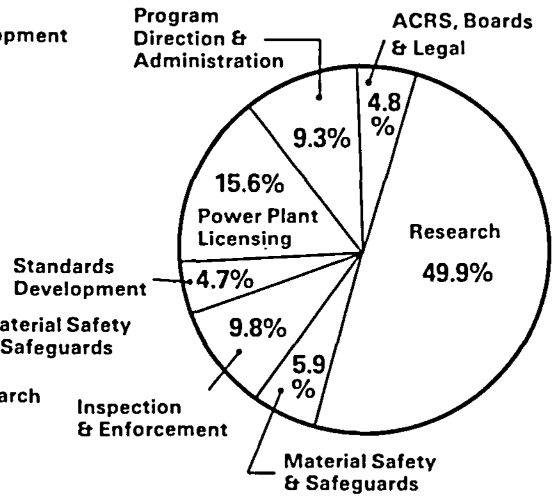

FUNDS — $222 MILLION

and should be part of the decisionmaking process. In some instances efficiency calls for regulatory change; in others, it demands regulatory stability. During its first year, the Commission established several operating principles and took steps to apply them in all NRC activity. These initiatives sought to enhance the ability of NRC to perform its safety, safeguards and environmental missions by improving the effectiveness and efficiency of its regulatory processes, assuring the objectivity and independence of its decisions, and keeping the agency open and responsive to the public which it serves.

In this regard, high priority was assigned to actions in support of the following objectives:

- Improving regulatory effectiveness and efficiency;
- Stabilizing and clarifying NRC practices to increase predictability and remove regulatory uncertainty;
- Shortening the time required to conduct licensing reviews to the fullest extent possible consistent with adequate protection of public health and safety, the common defense and security, and the environment;
- Improving coordination with, and reducing unnecessary overlaps among government agencies at all levels;
- Addressing a number of important unresolved policy issues, especially in the nuclear fuel cycle area.

Actions in each of these areas are described in the various chapters of this report. Summarized here are some of the more noteworthy activities and accomplishments of the year:

FIRST YEAR HIGHLIGHTS

The year 1975 was a period of transition for the new Commission—a time of planning, assessing issues and establishing priorities while at the same time maintaining the momentum of ongoing regulatory work. It was a year of vigorous recruitment of outstanding managerial and technical talent to fill key positions, and of careful shaping of a new organizational structure to meet the demanding mandates of Congress. It also was a year of achievement toward resolving long standing issues, and progress in establishing and moving toward new goals.

This period of intense transitional activity was punctuated by several challenging and unanticipated reactor operational events of potential safety significance which required swift response by

NRC RESPONSIBILITIES — NUCLEAR FACILITIES AND MATERIALS

- PROTECTING PUBLIC HEALTH AND SAFETY
  - Consequences of accidents
  - Routine releases of radioactivity
- PROTECTING THE ENVIRONMENT
  - Total impact on environment
  - Balance between economic and environmental values
- MATERIALS AND PLANT PROTECTION (SAFEGUARDS)
  - Theft or diversion of materials to unauthorized uses
  - Sabotage of plants
- ASSURING CONFORMITY WITH ANTITRUST LAWS

the Commission, and extensive NRC investigations. For example:

- The NRC was less than two weeks old when hairline cracks in safety-system pipes were discovered at several plants. Because of the lack of sufficient data concerning the integrity of stainless steel piping of the type involved, the Commission issued an order that 23 reactors with similar piping be inspected within 20 days. While these inspections turned up no additional cracking at the time, the problem reoccurred at the first plants involved and elsewhere during the year. More frequent inspections and more stringent leak-detection standards imposed by NRC kept the situation under control while an intensive study of possible causes was undertaken. Corrosion from a combination of factors was identified by the NRC study group in late 1975 as the basic cause of cracking. Research into permanent remedies is continuing. The NRC concluded that, with the close surveillance of the piping now required, there is no threat to public health and safety.
- Another unanticipated problem involved unusual vibrational wear in some metal channel boxes which hold the fuel assemblies in boiling water reactors. In this case, existing understanding and available data permitted a finely-tuned regulatory approach. Using information on the operating history of each affected plant and vibration magnitude data, individualized solutions—sometimes involving power reductions—were implemented by each licensee under close NRC supervision. This procedure gave the public full protection while allowing the utilities to avoid power supply problems which would have proved costly and inconvenient to the consuming public.
- A major fire at the Browns Ferry reactor site in Alabama in March prompted the Commission to launch independent investigations into the events and implications associated with the fire, and to order follow-up inspections at every operating power reactor in the U.S. Results of investigations and inspections relating to the fire and provisions to assure safety during the restoration period were made available to the public. Assessment of NRC and industry procedures in light of this incident was continuing by a special review group at year-end.

Toward Effectiveness and Efficiency

A major Commission objective during the first year was careful examination of NRC programs, procedures and requirements to determine the extent of their contributions to effective and efficient regulation. On the one hand, this meant assuring that NRC practices protect the public health and safety, security and the environment in light of experience and current needs, and identifying areas in which additional effort was required. On the other, it meant taking steps to eliminate or reduce requirements that were shown to be unnecessary. While the paramount NRC goal is assurance of full protection for the public and the environment, the Commission has taken the position that neither the public nor the regulated industry should be required to bear costs and inconveniences resulting from unjustifiable regulatory burdens.

Among actions taken during 1975 to improve regulatory performance were:

- A Standing Committee on Regulatory Effectiveness, composed of senior NRC managers, was established to oversee the overall program of regulatory improvement. In addition to monitoring implementation of a number of immediate actions, the committee is preparing an "Action Plan" for specific steps and schedules to further improve agency performance in 1976 and beyond.
- Impact/value analysis was made an integral part of NRC decisionmaking—to be utilized in policy proposals as well as in assessing other contemplated regulatory actions. This involves a systematic assessment of the values and adverse impacts, including added costs to the public, which can be expected to result from the various alternatives.
- Planning was initiated to determine how advanced probabilistic approaches in assessing reactor accident risks can be used in licensing considerations. A concerted staff effort is being undertaken to determine whether, and to what extent, NRC regulation could benefit from using the methodologies employed by Dr. Norman Rasmussen and his associates in their three-year study on reactor safety. The study, released by the NRC in 1975, made extensive use of highly sophisticated probabilistic risk assessment techniques. These techniques may enable NRC both to focus its licensing, inspection and research activities more accurately on those areas where the greatest safety benefits are likely to be realized, and to identify requirements or practices which are unproductive or overly strict, and thus could be eliminated or relaxed without increasing public risk.
- Development was begun on a research plan aimed at expanding NRC's ability to apply impact/value assessment usefully, with emphasis on quantifying health and safety-related benefits.
- Programs were initiated for direct NRC inspections of the quality assurance programs of major nuclear manufacturers, suppliers and architect-engineers and for the use of third-party inspections of reactor systems and components. These programs are designed to give NRC better control over vendor quality assurance and to eliminate duplication of inspection effort. In this area, more direct government involvement in inspection could produce savings to the public and industry of more than $12 million a year.

Increasing Predictability

Stable and understandable requirements are essential to conduct of the NRC regulatory process in an orderly and responsible manner. Some in the regulated industry have contended that uncertainty about regulatory requirements and practices is a major concern that can hamper sound management planning.

Steps taken in 1975 by the Commission to reduce uncertainty in the regulatory process included actions designed to introduce improved management controls over regulatory changes, stabilize the review process, and assure that NRC's requirements and procedures are clearly communicated and consistently applied. Examples are:

- More than 70 new and revised regulatory guides were developed and published to provide industry and the public with clear guidance on NRC requirements, and to suggest methods of satisfactorily complying. The use of regulatory guides aids in understanding the requirements for safe reactor design and operation, encourages uniform approaches to regulation, promotes understanding among all parties to the regulatory process, and helps eliminate delays.

- All proposed changes to existing NRC requirements and practices for nuclear power plant licensing not requiring direct Commission action must now be reviewed and approved by a standing Regulatory Requirements Review Committee (RRRC) before becoming effective. The RRRC, composed of senior NRC officials, is authorized to reject or defer changes that appear unnecessary, or need further clarification and analysis.
- Standard Review Plans for the licensing safety review stage were completed and put in force. These plans provide NRC reviewers with detailed guidance in carrying out their reviews of applications to construct and operate nuclear power plants. Broad-scale use of the plans will benefit the public and industry by improving the overall quality, uniformity and predictability of staff reviews. The plans will also serve as a well-defined basis for evaluating proposed changes.
- A revised Standard Format and Content document was issued to provide definitive guidance to industry on information required of applicants for licenses for nuclear power plants. Keyed to the Standard Review Plans, the document defines applicant requirements more completely and clearly than heretofore, and should reduce uncertainty and delay created when the NRC staff must return to an applicant for additional information.
- Export licensing procedures for certain special nuclear materials were clarified and streamlined. At year-end, the staff was considering further simplification by eliminating the need for a specific license for each export of a small quantity of certain special nuclear materials.
- Regulatory document management improvement was begun which, when completed, is expected to make important regulatory information available to the public and industry up to a month sooner than in the past. The reduction in regulatory-related cost imposed on industry and the public that should result is estimated at more than $1 million a year.
- The staff issued the first three Preliminary Design Approvals for standardized nuclear power plant designs. By the end of 1975, all five U.S. reactor manufacturers had submitted at least one standard reactor design to NRC for review, and three architect-engineering firms had submitted standard designs for the nonnuclear balance of the plant. Increased use of NRC-approved standard plant designs offers the safety benefits of a stabilized technology and prospects for a more consistent review process which should also help government and industry to predict the time and capital needed for plant approval and construction.

Eliminating Licensing Delays

To be fully effective and efficient, regulation must also be timely. Neither the safety of the public nor its economic interest is well-served by a regulatory process that needlessly prolongs the time required to design, construct and place a nuclear power plant in safe operation. Consequently, another Commission objective during 1975 was to continue efforts to shorten the licensing process and remove it from the critical path of plant development to the fullest possible extent while maintaining high standards of regulatory quality and thoroughness, and otherwise meeting statutory requirements.

Among actions to assure prompt reviews of, and timely regulatory decisions on nuclear power plant license applications were:

- Limited work authorizations (LWAs) were granted during the year for 19 proposed power reactors. These authorizations permitted applicants to begin limited site work, at their own risk, following successful completion of the environmental and site suitability reviews and public hearings but prior to completion of the safety review and issuance of a construction permit. Where safety-related work was involved, appropriate safety reviews and hearings were conducted. As a result, facility construction was authorized an average of seven months sooner than would have been possible previously.

- The Commission announced that, owing to improved management and scheduling procedures, it proposed to complete staff reviews and begin the required public hearing process leading to an LWA decision some 11 months after docketing of an application to construct a nuclear power plant. Under normal circumstances this would, on the average, result in an LWA decision about 16 months after docketing of the application—a 20 percent saving over the recent average of about 20 months.
- At year-end the NRC was considering development of early site review regulations and procedures to permit reviews and decisions on applications for nuclear power plant sites in advance of actual need to begin construction. This step, coupled with NRC-approved standardized plant designs, could, when fully utilized, reduce the overall time required to place a nuclear plant in operation by as much as a year while still providing for full consideration of safety and environmental matters.
- In May, the Commission forwarded proposed legislation to Congress to further improve the licensing process. The legislation would give the NRC new or expanded authority to: (1) provide for separate and early site approvals; (2) encourage greater reactor standardization; (3) provide for combined construction permits and operating licenses and avoid duplicate hearings; and (4) enhance public participation by providing for earlier hearings on site suitability and plant design questions. These improvements could allow the licensing process to be removed altogether from the critical path for most plants.

Intergovernmental Coordination

Sound, efficient regulation of nuclear energy requires that the Commission coordinate its activities with actions of other government agencies at home and abroad with expertise and authority related to the NRC mission. During NRC's first year, emphasis was given to developing, implementing and strengthening procedures for dealing with other Federal agencies, State and local officials and foreign regulatory authorities. The major thrust was toward improved regulatory effectiveness through constructive cooperation and sharing of information while reducing the confusion, inefficiency, and excessive regulatory burdens created by overlapping authorities and duplicative requirements. Efforts during 1975 included the following actions:

- Negotiated formal Memoranda of Understanding to clarify regulatory responsibilities and jurisdictional areas with the Environmental Protection Agency regarding regulation of water quality for nuclear facilities, and with the Army Corps of Engineers regarding the regulation of floating nuclear power plants on navigable waters of the U.S.
- Worked toward a new agreement with the U.S. Department of Transportation to clarify respective roles regarding the transport of radioactive materials.
- Reached interagency agreement on procedures for licensing nuclear exports and imports to assure that, in carrying out its responsibilities, the NRC will receive needed information promptly from other Federal agencies.
- Sponsored a major Federal-State Power Plant Siting Conference of national,

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State and local regulatory officials. It produced a number of recommendations for improving intergovernmental coordination in regulating the siting of nuclear power reactors—a particularly complex and delay-prone area. During the year, NRC began to implement many of these recommendations and initiated planning for a second conference in 1976.

- Concluded bilateral agreements for exchange of regulatory information with the United Kingdom, Italy, Denmark and West Germany. In addition, agreements to exchange the results of specific reactor safety research projects were concluded with West Germany, Japan and France. These agreements will further NRC's ability to assure safe reactor operation through early identification of potential problems, and to make effective, fully-informed regulatory decisions that take account of the experiences of others.

Confronting the Issues

The Commission, during its first year, has resolved a number of specific problems and major policy questions, and has taken significant steps toward resolution of issues requiring long-term consideration. Some of these were foreseen or inherited, while others came as unexpected challenges. In particular, substantial effort has been directed to the resolution of nuclear fuel cycle issues which—due in large part to insufficient attention being paid to this area in the past by both government and industry—have become pacing items for the continued development of nuclear power.

NRC's responses to some of the year's more challenging problems included:

- A major new regulation and revised guides were issued providing definitive instruction to licensees on how to comply with the NRC requirement that routine releases of radioactive materials in effluents from light-water power reactors and any resultant public exposure be kept "as low as reasonably achievable." This was a complex and vital issue inherited from the AEC. The new provisions, resulting from intensive impact/value analysis and a public rulemaking proceeding lasting more than three years, provide licensees a firm quantitative basis for determining the relative costs and benefits of seeking further reductions in the already low radioactive levels of effluents from these plants.
- Interim procedures for licensing of short-term storage facilities or pools for spent reactor fuel were clarified and staff work begun on an environmental impact statement addressing the generic aspects of spent fuel storage. While the impact statement is being prepared, the NRC staff will continue to review applications to augment storage capacity on an individual basis. Without the provisions for interim reviews, as many as 10 reactors would have to shut down in the next few years due to lack of authorized storage space.
- Procedures were established for reaching a decision on health and safety, environmental, and safeguards issues involved in wide-scale use of mixed uranium and plutonium oxides as fuel in nuclear power reactors. Criteria were also prepared to govern interim licensing actions on applications for limited use of mixed-oxide fuel. All NRC staff work is to be completed during 1976, to be followed by public hearings and possibly a decision on wide-scale use of mixed-oxide fuel as early as the first quarter of 1977.
- NRC tightened existing requirements for protecting nuclear materials from theft or diversion and nuclear plants from sabotage, carried out major studies, and conducted contingency planning aimed at assuring the continuing adequacy of safeguard measures.
- A major effort was initiated to develop a

comprehensive program for regulating the management of all types of radioactive wastes, including long-term storage and disposal of high-level wastes. The Commission created a special organizational unit for waste management regulations and began building capability to review and license long-term storage proposals. Several research projects were initiated to provide sound, independent bases for establishing and applying regulatory requirements governing various methods of long-term storage, such as deep burial in stable geological formations.

LOOKING AHEAD

A viable nuclear power option is a key element in the nation's energy future, and the Commission is convinced that its continued viability as an energy choice for the American public will depend in part on the effectiveness and efficiency with which the NRC discharges its responsibilities for protecting public health and safety and the environment in the years ahead. Thus, while NRC's first year saw many accomplishments, it was also a time of preparation and planning. Solid regulatory foundations were laid for the realization of increasing benefits in the future.

For example, in 1976 the Standing Committee on Regulatory Effectiveness will continue to develop plans for further improvement of the effectiveness and efficiency of NRC performance. Priority attention will be given to upgrading and stabilizing regulatory procedures and requirements. Members of the NRC staff who propose modifications in current policies, practices or rules must demonstrate both the rationale for proposed changes and how—and how much—such changes will contribute to furthering the NRC mission, and at what impact. Particularly important to adding further quality and predictability to NRC regulation will be the development during 1976 of Standard Review Plans for the environmental phase of

At one of the scores of working sessions of NRC Commissioners during 1975: Commissioners Kennedy and Mason, Chairman Anders, and Commissioners Rowden and Gilinsky.

the reactor licensing cycle. These plans will complement those implemented during 1975 for the safety review phase.

In the coming months the Commission will also press its efforts to improve and shorten the licensing process. These will include, for example:

- Making decisions on issuing Preliminary Design Approvals for additional standard power plant designs;
- Implementing a full-scale early-site review program for nuclear plants, and strengthening efforts with the States in this area; and
- Expediting completion of staff reviews for nuclear plant construction authorizations.

Other high priority areas will include:

- Completion and transmittal to Congress of the two special reports on Nuclear Energy Centers and a Federal Security Agency which were mandated by the Energy Reorganization Act.
- Reaching a decision on wide-scale commercial use of mixed-oxide reactor fuel.
- Development of an integrated program to set priorities for and carry out the research, standards development and licensing actions necessary for regulating management of all types of nuclear wastes.
- Development of programs and staff capabilities for sound, timely reviews of new industry initiatives, such as a decision to construct and operate a uranium enrichment facility.
- Development and implementation of updated and strengthened policies, procedures and contingency plans for safeguarding of nuclear materials and facilities.
- Implementation of improved mechanisms for setting priorities to guide future research and further development of standards, and to assure that these actions are effectively phased with the planning for advanced nuclear concepts and engineering applications by ERDA and industry.

In addition to the activities discussed here, there will continue to be the thousands of routine licensing, standards-setting and inspection and enforcement actions that comprise day-to-day nuclear regulation, and which are vital to ensuring protection of the public health and safety and the environment. If the past year is any guide, there will be unforeseen challenges as well. The Commission is confident of NRC's ability to face up to and deal with tomorrow's regulatory demands with effective, independent decisions, efficiently and openly made.

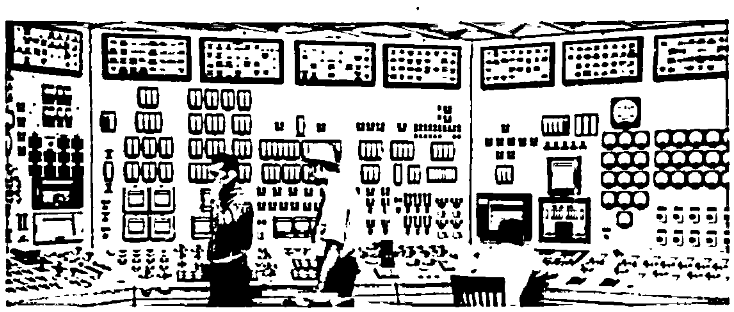
Regulating Nuclear Reactors

Safety—Bedrock of Regulation

A primary goal in the regulation of nuclear power has been and continues to be preservation of the health and safety of the public. The regulatory staff of the former AEC was formed for this purpose, and safety remains paramount in NRC attention.

NRC's safety review of nuclear reactors proposed for construction and operation—as well as the environmental protection, antitrust and indemnification aspects of each case—is centered in the Office of Nuclear Reactor Regulation. Its evaluations are conducted by staff drawn from across the spectrum of the technical disciplines and organized into 38 functional branches and an antitrust and indemnification group. The special expertise of other Federal agencies and consultants is used wherever appropriate.

This chapter discusses all major aspects of the licensing process and develops the relationship between licensing actions and concepts and the primary objective: the safe operation of nuclear power plants. It covers the manifold reviews and decisions of the process; the "defense-in-depth" concept applied to plant design; specific licensing actions during 1975; the prospects for and purposes of standardization of plant design; NRC administrative actions on standardization and other measures to expedite the



process without compromise in thoroughness; NRC recommendations for new legislation to improve and facilitate licensing procedures; and other matters related to this most basic operation of the agency.

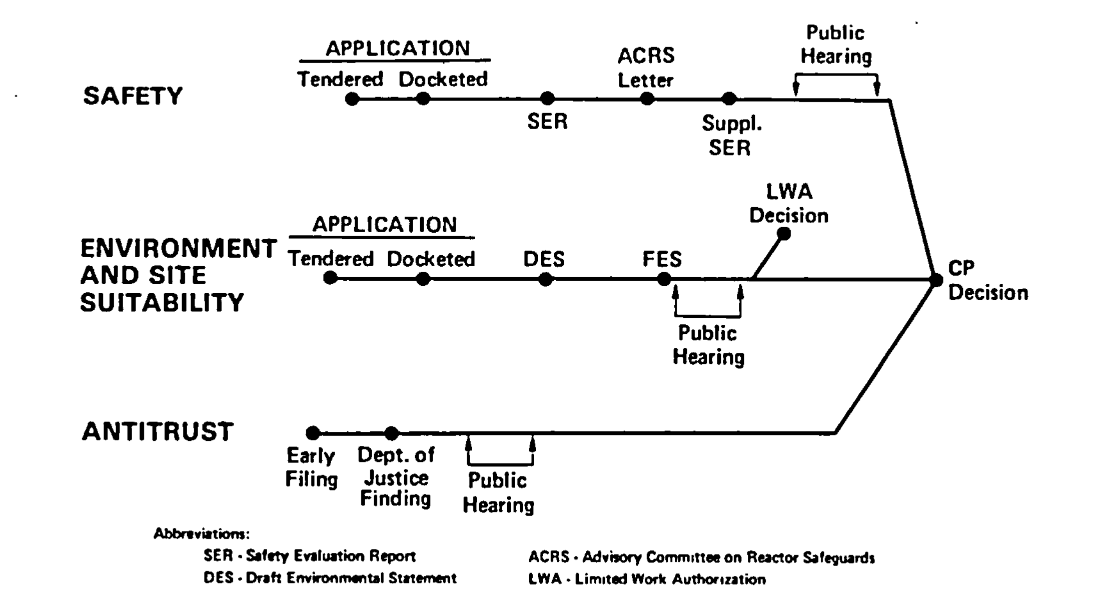
THE LICENSING PROCESS

Obtaining a construction permit for a power reactor or other major nuclear facility involves: First, the filing and acceptance of the application, generally consisting of eight or nine large volumes of material covering both safety and environmental factors; second, safety environmental, safeguards and antitrust reviews by the NRC staff; third, a safety review by the independent Advisory Committee on Reactor Safeguards (ACRS), which is required by law; and fourth, a mandatory public hearing by a three-man 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, and could ultimately go to the Commissioners for final decision.

A notice of receipt of application is published in the *Federal Register*. Copies of the application are furnished to appropriate State and local authorities and to a public document room established in the vicinity of the proposed site. At the same time, a notice of hearing is published in the *Federal Register* (and local newspapers) which provides 30 days for interested members of the public to petition to intervene in the proceeding. Such petitions are considered by the ASLB appointed to the case.

The NRC staff's safety, environmental, and antitrust reviews proceed in parallel. During the course of the environmental review (described in Chapter 3), the staff prepares a draft environmental statement which is reviewed by Federal, State, and local agencies and other interested persons. Their comments are taken into account in the preparation of a final environmental statement. Both documents are made available to the public. The results of the staff's

PARALLEL TRACKS IN CONSTRUCTION PERMIT REVIEW PROCESS



safety review (described in this chapter) are embodied in a Safety Evaluation Report.

When the ACRS has completed its safety review, the NRC staff issues a supplement to the Safety Evaluation Report which discusses any action taken as a result of ACRS recommendations. A public hearing can then be held, generally taking place in a community near the proposed site. A single hearing may cover both radiological health and safety and environmental matters, or separate hearings may be held on these factors.

In appropriate cases, NRC may grant a limited work authorization to an applicant in advance of the final decision on the construction permit to allow certain work to begin at the site some seven months earlier than would otherwise be possible. This can be done only when the environmental portion of the application is submitted early, and after environmental impact and site suitability reviews have been completed by the NRC staff and the ASLB presiding in the case has made favorable findings following a public hearing.

When a plant is nearing completion, the applicant must go through virtually the same process for an operating license. Most of the same steps pertain—filing an application, reviews by the NRC staff and the ACRS, and issuances of a Safety Evaluation Report and an updated environmental statement. While a public hearing is not mandatory at this stage, one may be held if requested by affected members of the public or at the initiative of the Commission itself.

Antitrust reviews of license applications (described in this chapter) are conducted by the NRC and the Attorney General in advance of, or concurrently with, other licensing reviews. Any antitrust hearing that may be required is conducted separately from those on environmental and safety matters.

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 of its operation that are to be met. Once licensed, a nuclear facility remains under NRC surveillance and undergoes periodic inspections throughout its operating life.

ENSURING SAFE DESIGN

Section 307(c) of the Energy Reorganization Act of 1974 directs the Commission to include in its annual report to Congress, among other things, a description of activities and findings in the area of assuring the safe design of nuclear power plants. These activities are addressed in the following discussion.

NRC RESOURCES
PERSONNEL (Technical Disciplines)

Biology & Radiobiology	Metallurgy	Engineering:
Chemistry	Meteorology	Chemical
Ecology & Radioecology	Oceanography	Civil
Geology	Physics	Electrical
Hydrology	Rad. Health Physics	Instrumentation
Limnology	Seismology	Mechanical
Mathematics		Nuclear
		Systems

Defense-in-Depth Concept

Underlying the regulatory requirements imposed to achieve safety objectives is the defense-in-depth concept employed in the design of each nuclear plant: three successive and mutually reinforcing echelons of defense are required to prevent a serious accident affecting the public.

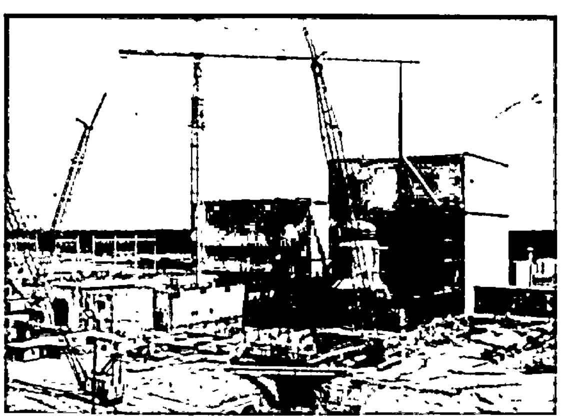
The first echelon of defense emphasizes accident prevention. It requires that the plant be soundly and conservatively designed, so that it can be built, tested, operated, and maintained in accordance with stringent quality standards and engineering practices with a high degree of freedom from faults and errors. The basic design selected must be inherently stable and have a high tolerance for the possibility of system malfunctions or off-normal conditions.

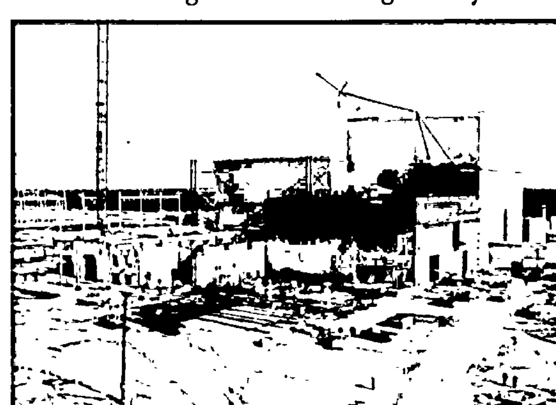
The second echelon of defense is based on the assumption that failures or operating errors that potentially could lead to safety problems will occur during the service life of a nuclear power plant, despite the care taken to prevent them. Accordingly, safety systems are required to prevent or minimize damage from such failures. The aim here is to detect off-normal conditions and prevent minor incidents from escalating into major ones. Conservative design practices, adequate safety margins, inspectability, and redundant detecting and actuating equipment must be incorporated into protection systems to assure both the effectiveness and the reliability of this second echelon of defense.

The third echelon of defense supplements the first two through features that provide additional margins to protect the public against unlikely accidents. These margins are assessed primarily by evaluating the response of the plant to a number of arbitrarily assumed events, involving in some instances the assumption of an independent failure of a redundant protective system simultaneously with the occurrence of the accident it is intended to control. From analyses of these postulated events, a number of accident sequences called "design basis accidents" are selected as a basis for the incorporation of additional features required for the extra margin of protection.

Conduct of Safety Review

The safety portion of the application for a nuclear power plant construction permit is organized in accordance with the NRC guide, "Standard Format and Content of Safety Analysis Reports," which describes the informational needs of the NRC staff for review. The conduct of the safety review is in accordance with the recently published NRC Standard Review Plans which describe in some detail how the safety review of light-water-cooled reactor applications is accomplished, and states the criteria applied in assessing systems, components and structures important to safety and in determining site parameters as a basis for design. The criteria used in the review process include NRC regulations and regulatory





Construction of a nuclear power plant must be in accordance with plans approved by the NRC and the construction permit issued by the NRC. NRC inspectors examine the construction of the plant periodically at the site. This series of photos follows the recent progress of construction of Georgia Power Company's Edwin I. Hatch nuclear plant Unit 2, shown in front of Unit 1 turbine and reactor buildings. Featured in the series from left to right are the Unit 2 drywell, the reactor building enclosing the drywell, the reactor pressure vessel being lifted by crane, and the reactor pressure vessel being lowered into the drywell. Unit 1 was licensed to operate in August 1974.

guides, and consensus standards developed by technical societies in conjunction with NRC.

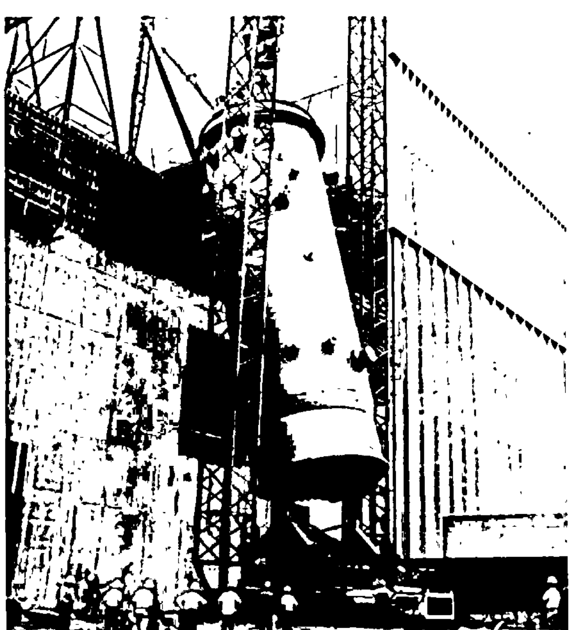
When a construction permit application is submitted, it is first subjected to a preliminary review to determine whether it contains sufficient information for a detailed review. If it does not, the staff makes specific requests for the required information, and formally dockets the application only after it meets acceptance criteria. In addition, before the Preliminary Safety Analysis Report is submitted, the staff conducts a substantive review and inspection of the applicant's quality assurance program covering design and procurement.

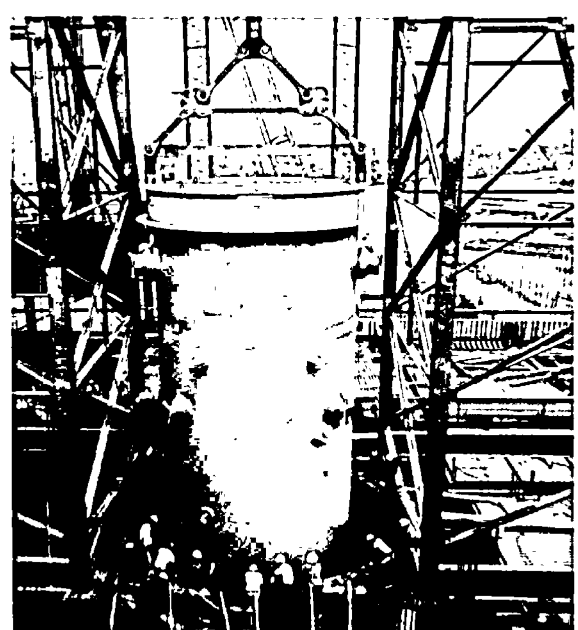
The staff must determine that the plant design is safe and consistent with NRC rules and regulations. Design methods and procedures of calculations are examined to establish their validity and audits of actual calculations and other procedures of design and analysis are made by the staff to establish the validity of the applicant's design and to determine that the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support required findings in respect to safety.

Design Changes Required

As the review proceeds, if any portion of the application is found to be inadequate, the staff requires the applicant to make modifications so that it will be acceptable. No application has been received to date that has not undergone some required changes. Necessary modifications in design features have been required by the staff in its review before the NRC is satisfied that safety objectives will be met; often these are dictated by the unique nature of the site. Some recent design changes required in nuclear power reactor applications include:

- Modification of cooling lake dam design to include an auxiliary spillway to provide adequate freeboard on the dam in the event of a maximum postulated flood.
- Modification of earthquake intensity factor to be considered, affecting design to accommodate safe shutdown after a seismic event.
- Modification of roof design of structures to properly account for vertical





velocities of tornado-generated missiles.

- Modification of design criteria to protect against postulated high-energy pipe breaks outside the containment.
- Modification of the filter system design for the auxiliary building to include safety-grade filters for effective removal of iodine.
- Modification of spent fuel pool cooling system to meet seismic design requirements.
- Modification of spent fuel transfer tube access design to minimize radiation exposure to plant personnel.
- Modification of cooling lake dams to withstand postulated earthquake.

Reviews of Technical Problems

In conjunction with the safety review of nuclear power plant applications, the NRC technical staff conducts evaluations of potential safety problems that may apply to many reactors of a given design type. The detailed reviews and independent analyses of emergency core cooling system performance, the reliability of automatic shutdown (scram) systems, and containment pressure during accidents are examples of this type of study. The staff also conducts engineering audits of reactor vendors' and architect-engineers' design calculations and procedures to assure conformance with safety design practice. The safety review of operating problems described in Chapter 7 of this report is another means of insuring safe design by applying the findings reached in these reviews to the licensing process.

Verification and Research

The review process includes consideration of programs proposed by an applicant for the post-construction permit period, to verify plant design features and to confirm design margins. It also includes consideration of programs of research and development to be conducted to assure the resolution of safety questions associated with safety features or components requiring such research. The review must identify any research and development work that will be conducted to confirm the adequacy of, or to resolve any safety questions associated with, the design of a particular facility. Further, a schedule for completion of the research and development work must show that such safety questions will be resolved in a timely manner.

Data obtained from research and development programs on particular facilities and from the Commission's confirmatory research program are factored into the licensing reviews. These programs are discussed in some detail in Chapter 8.

When the review and evaluation of the construction permit application progresses to the point that the staff concludes that acceptable criteria, preliminary design information and financial information are documented in the application, a Safety Evaluation Report (SER) will be prepared. This report represents a summary of the review and evaluation of the application by the staff relative to the anticipated effect of the proposed facility on the public health and safety.

During the ongoing NRC staff technical review, the ACRS conducts its independent review of design safety features of the reactor. The Committee's recommendations are taken into account in a supplement to the SER.

Operating License Review

When the construction of the nuclear facility has progressed to the point where substantial final design information and plans for operation are available, the applicant submits the Final Safety Analysis Report (FSAR) in support of an application for an operating license. This report sets forth the pertinent details of the final design of the facility, including final containment design, design of the nuclear core, the waste handling system, and the results of any research and development programs. The FSAR also supplies plans for operating and procedures for coping with emergencies. Again, the staff and the ACRS make detailed reviews of the information. A Safety Evaluation Report concerning the operating license is prepared, as during the construction permit stage.

Through its inspection and enforcement program, NRC maintains surveillance over construction and operation of each plant to assure compliance with Commission regulations for the protection of public health and safety and the environment.

PLANT LICENSING ACTIONS

During fiscal year 1975, nine operating licenses, 14 construction permits and 10 limited work authorizations were issued for nuclear power plant units. All of these actions occurred during the first half of the year (July-December, 1974). Decisions on construction permit applications involving seven planned units that had been expected during the year had to be deferred, principally because of the need to comply with the Commission's final acceptance criteria for emergency core cooling systems, which became effective on December 28, 1974. However, limited work was authorized to begin on all seven in advance of the construction permit decisions. These units were Duke Power Co.'s Catawba Units 1 and 2 in South Carolina; Commonwealth Edison Co.'s Byron Units 1 and 2, and Braidwood Units 1 and 2, in Illinois; and Florida Power and Light Co.'s St. Lucie Unit 2. Limited work authorizations were also issued in the latter half of calendar year 1975 for the Gulf States Utilities Co.'s Riverbend Units 1 and 2 in Louisiana; the Washington Public Power Supply System's Units 1 and 4; the Illinois Power Co.'s Clinton Units 1 and 2; the Union Electric Co.'s Callaway Units 1 and 2 in Missouri; the Houston Lighting and Power Co.'s South Texas Units 1 and 2; and the Toledo Edison Co.'s Davis-Besse Units 2 and 3.

On August 7, 1975, NRC issued construction permits to Duke Power Co. for the two Catawba units, located on the shore of Lake Wylie in York County, S.C. Each unit has a rated capacity of 1,153 electrical megawatts (MWe). By year's end construction permits were also issued for the four Commonwealth Edison Co.'s units cited above, the two units of the Houston Lighting and Power Co., and Unit 1 of the Washington Public Power Supply System.

Actions that had been scheduled for the last half of the fiscal year on three operating licenses were not completed because of construction delays experienced by the applicants.

On August 1, 1975, NRC issued an operating license for Northeast Nuclear Energy Co.'s Millstone Unit 2 in Connecticut, a pressurized water reactor plant with a capacity of 828 MWe. Full power was authorized in September 1975. An operating license was issued in November 1975 to the Portland General Electric Co. and two other utilities for the Trojan Nuclear Plant in Columbia County, Oregon; the plant uses a pressurized water reactor and, when operating at full power, will have a net output of 1,130 MWe. And in December 1975, an operating license was issued to the Consolidated Edison Co. authorizing the loading of nuclear fuel into the reactor of Unit 3 of the Indian Point Nuclear Generating Station at Buchanan, New York, and testing prior to start-up. The unit employs a pressurized water reactor and at full power will have an output of about 965 MWe.

Status of Nuclear Power

As of December 31, 1975, there were 236 nuclear power units either in operation, being built or planned, representing a total capacity of 236,000 MWe, as compared with the totals on December 31, 1974: 233 units with a capacity of 232,000 MWe. (See Appendix 1.)

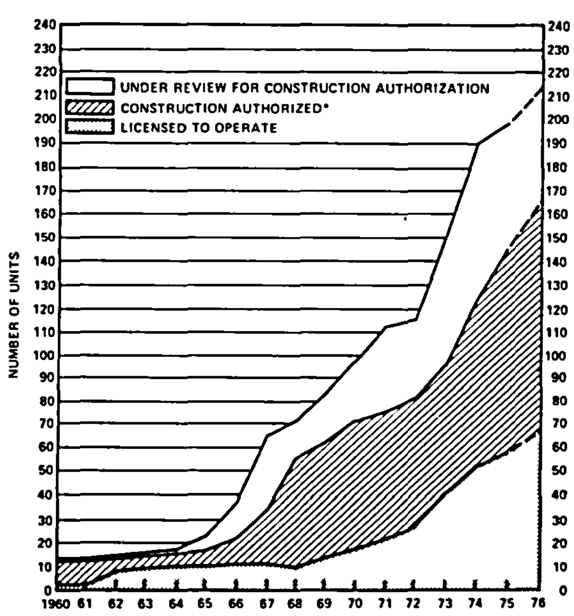
Of the 236 units operating, being built or planned, 198 were in the regulatory process as follows:

- 56 licensed to operate, with total capacity of 39,000 MWe.
- 69 with construction permits, representing 71,000 MWe of capacity.
- 73 with an aggregate capacity of 80,000 MWe under review for construction permits. Work was proceeding on 19 of these under limited work authorizations.

Of the remaining 38 units, 19 had been ordered and 19 announced.

Under the impact of difficult financial conditions and revised assessments of energy demands in the nation, a pattern of deferral and some cancellations of planned nuclear power units has persisted through and beyond fiscal year 1975. Units under review on June 30, 1974 plus those coming into the regulatory process between then and the end of 1975 total 204. For 95 of these, representing 38 utilities, postponement of the date of expected readiness for decision on a construction permit or operating license was made necessary by economic constraints or revised estimates of regional power needs. Twenty-three units were cancelled in this period. Most of these deferrals and cancellations, however, occurred prior to April 1975.

NUCLEAR POWER PLANTS UNDER NRC SURVEILLANCE



*Includes Limited Work Authorizations in 1974–1976

STANDARDIZATION STEPS

During 1975, further significant process was made toward the goal of nuclear power plant standardization, first enunciated by the Atomic Energy Commission in April 1972. NRC regards standardization of plant designs, complemented by the early review of sites planned for the location of nuclear plants, as one of the most important means for improving the efficiency and effectiveness of the licensing process.

The procedural options available to applicants for standardization of nuclear power plants are:

Reference System – a design of an entire facility or major portion thereof can be reviewed once and utilized repeatedly by reference without further staff review in individual applications for licenses.

Duplicate Plants – the design for several identical plants that would be constructed within a limited time by one or more utilities at one or more sites can be reviewed once.

License to Manufacture – the design of an entire facility can be reviewed once for manufacture at a central location. The preapproved facilities can then be moved to specific utility sites for construction and operation.

As an expansion of the duplicate plant option, a policy for "replication" was established in 1974. Replication provides for the reuse of a recently approved design for

STANDARDIZATION
MAIN FEATURES OF NRC'S APPROACH

- MAXIMUM CAPACITY—3800 MW THERMAL
- NRC WILL REVIEW:
  - Entire Facility
  - Nuclear Steam Supply System
  - Balance of Plant
- OPTIONS:
  - Reference Systems
  - Duplicate Plants (Including Replication)
  - Licenses to Manufacture

a custom plant. NRC regards replication as an interim approach to standardization until a sufficient number of reference system designs is accumulated, estimated to occur in two to four years. Each of these standardization approaches is based on the reuse of approved plant designs.

In April 1974, the AEC proposed rules for implementation of the Duplicate Plants and Reference Systems options. These rules, reissued in April 1975 as part of NRC regulations, are designated as Appendices N and O of 10 CFR Part 50. Regulations for the License to Manufacture concept were issued in 1973 as Appendix M to Part 50.

Additional guidance was made available to the public in 1974 for the implementation of the Reference System and Duplicate Plant options and replication. WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," published in August 1974, provides guidance to those involved with the Reference System and Duplicate Plant options regarding the preparation, staff review, and utilization of these designs. WASH-1340, "Policy and Procedures for the Replication of Custom Plant Designs," dated July 1974, provides guidance to those interested in the replication approach to standardization.

As one approach to standardization, in mid-1973, Offshore Power Systems submitted an application for a license to manufacture eight identical floating nuclear power plants. The plants would be fabricated in a shipyard-like facility in Jacksonville, Florida and towed to their planned location for operation. The licensing process for the floating nuclear plant concept, as for the Reference System option, involves separate applications and reviews for the plant design and for the proposed sites of operation. The first site undergoing review is for the Atlantic Generating Station proposed by the Public Service Electric and Gas Co. of New Jersey. The first stage in the review of the design, manufacture, and operating features of the floating nuclear plant was completed with the issuance in October of the NRC staff's Safety Evaluation Report and a portion of the Final Environmental Statement for the design and manufacturing of the plant. The floating and sea-going aspects of a floating nuclear plant require interagency coordination to delineate respective responsibilities for the regulation of safety and protection of the environment (see Chapter 3).

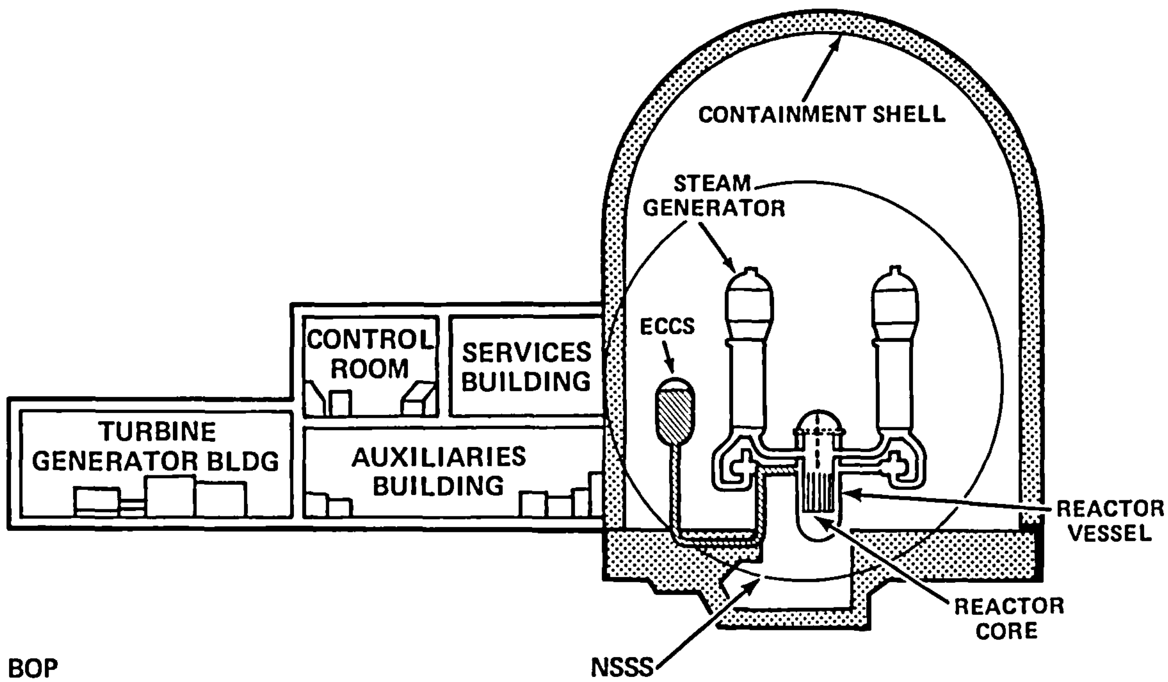
The industry's response to the Commission's standardization program has been gratifying, particularly with respect to reactor manufacturers. By the end of 1975, all five reactor vendors had submitted at least one standard reactor design and three architect-engineering firms had submitted balance-of-plant designs. Several additional architect-engineering firms were either contemplating or preparing the submission of balance-of-plant designs. A total of 21 utilities had applied for permits to build "standard" plants. Table 1 indicates standardization applications under review as of December 31, 1975. Preliminary Design Approvals were issued during December to the General Electric Co. for its GESSAR-238 nuclear island design (the first for a standardized nuclear power plant design); to Combustion Engineering for its

CESSAR nuclear steam supply system design; and to Westinghouse Electric Corp. for its RESAR-41 nuclear steam supply system design.

Table 1. STANDARDIZATION APPLICATIONS UNDER REVIEW
(As of December 31, 1975)

PROJECT	*APPLICANT*	*DOCKET DATE*	*COMMENTS*
Reference Designs			
GESSAR-238	General Electric	7-30-73	Nuclear island. Preliminary Design Approval (PDA) issued 12-31-75
CESSAR	Combustion Engineering	12-19-73	NSSS. PDA issued 12-31-75
RESAR-41	Westinghouse	3-11-74	NSSS. PDA issued 12-31-75
B-SAR-241	Babcock & Wilcox	5-14-74	NSSS (Withdrawn)
SWESSAR	Stone & Webster		
RESAR-41		6-28-74	Standard balance-of-plant (BOP) design matched to RESAR-41
CESSAR		10-21-74	BOP matched to CESSAR
RESAR-3S		10-2-75	BOP matched to RESAR-3S
B-SAR-205		12-19-75	BOP matched to B-SAR-205
C.F. Braun SSAR	C.F. Braun	12-21-74	Turbine Island matched to GESSAR-228
GASSAR	General Atomic	2-5-75	NSSS
GESSAR-251	General Electric	2-14-75	NSSS
RESAR-3S	Westinghouse	8-1-75	NSSS
GESSAR-238 (NSSS)	General Electric	9-24-75	NSSS
B-SAR-205	Babcock & Wilcox	10-24-75	NSSS (replaces B-SAR-241)
F-P SSAR	Fluor Pioneer	11-17-75	BOP matched to RESAR-41
Utility Application Using Reference Systems			
Cherokee 1-3	Duke Power	5-24-74	References CESSAR
Perkins 1-3	Duke Power	5-24-74	References CESSAR
South Texas 1 & 2	Houston Light & Power	7-5-74	References RESAR-41
WNP-3 & 5	Washington Public Power Supply System	8-2-74	References CESSAR
Palo Verde 1-3	Arizona Public Service	10-7-74	References CESSAR
Hartsville 1-4	Tennessee Valley Authority	11-11-74	References GESSAR-238

NUCLEAR STEAM SUPPLY SYSTEM (NSSS) AND BALANCE OF PLANT (BOP)



BOP

- FUEL HANDLING & STORAGE
- RADWASTE SYSTEMS
- EMERGENCY POWER
- CONTAINMENT
- SERVICE BUILDINGS
- CONTROL ROOM
- TURBINE GENERATOR

NSSS

- REACTOR
- COOLING SYSTEM
- RESIDUAL HEAT REMOVAL SYSTEM
- NUCLEAR INSTRUMENTATION
- CHEMICAL & VOLUME CONTROL
- ENGINEERED SAFETY FEATURES

PROJECT	*APPLICANT*	*DOCKET DATE*	*COMMENTS*
Black Fox 1 & 2	Public Service of Oklahoma	12-23-75	References GESSAR-238 (NSSS)
Phipps Bend 1 & 2	Tennessee Valley Authority	11-7-75 10-1-75	 References GESSAR-238
Duplicate Plants			
Byron/Braidwood	Commonwealth Edison	9-20-73	Two units at each of two sites
SNUPPS	Kansas Gas & Electric		Five units at four sites
(Wolf Creek)	Kansas City Power & Light	5-17-74	
(Callaway 1 & 2)	Union Electric	6-21-74	
(Tyrone 1)	Northern States Power	6-21-74	
(Sterling)	Rochester Gas & Electric	6-21-74	
WUPS	Wisconsin Electric Power		As many as six units on three sites
(Koshkonong 1 & 2)	Madison Gas & Electric	8-09-74	
	Wisconsin Power & Light		
	Wisconsin Public Service		
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	Reference FNP
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/ Braidwood
UEA 1 & 2	Alabama Power	*	Replicates Vogtle
NEP 1 & 2	New England Power	*	Replicates Seabrook

*Future Application
**Applicant changes necessitate rescheduling
***Schedule presently unavailable

ADVANCED REACTOR REVIEWS

LMFBR Demonstration Plant

The application for a permit to construct the nation's first large-scale demonstration liquid metal fast breeder reactor (LMFBR) at the Clinch River site near Oak Ridge, Tenn., was tendered to the AEC Regulatory staff in October 1974 for an acceptance review. It is designated the Clinch River Breeder Reactor Project (CRBRP). The application was subsequently determined to be insufficiently complete to docket.

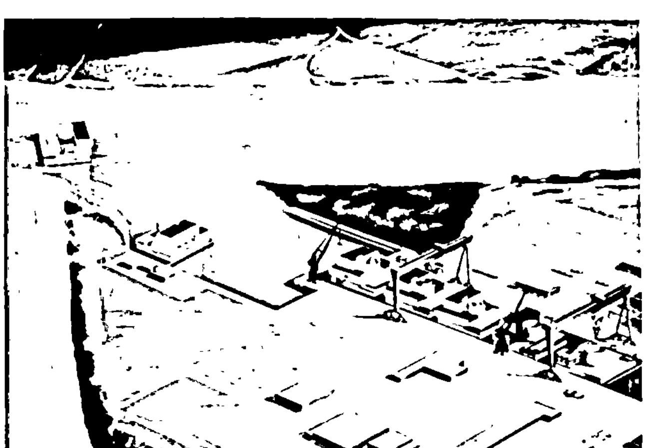
The environmental report and site-related section of the Preliminary Safety Analysis Report (PSAR) were resubmitted to the NRC in March 1975 and docketed for formal review in April. The balance of the application, consisting of the remaining PSAR sections, and other material relating to the radiological safety aspects of the proposed demonstration plant, was submitted in April for an acceptance review. The PSAR was docketed for formal review in June.

The proposed Clinch River facility is to be a 975 megawatt thermal sodium-cooled, fast neutron reactor fueled with a mixture of plutonium and uranium oxides. It will be designed to generate 380 megawatts of electrical power.

Major items of concern have been identified, discussed with the applicant, and reviewed with the project's management. The facility is a joint government-industry project. To streamline the organization that is developing the project, the Energy Research and Development Administration has assumed direct management.

In response to the NRC acceptance review, a major amendment to the PSAR describing the features designed to accommodate severe accidents, including core disruptive accidents, was submitted in October 1975. Principal items of concern for a plant of this size and type include the development of applicable principal design criteria, consideration of the need to design for the accommodation of major pipe rupture, the reliability of the shutdown heat removal systems, establishment of radiological site suitability, and assessment of the need for and adequacy of the design to accommodate core disruptive accidents.

Substantial additional effort is required to provide the special attention and arrangements necessary to review the varied and novel aspects of a plant of this type. The NRC review is being conducted in an aggressive manner: requests for additional information have brought partial response and a schedule for other responses; the ACRS has appointed a special subcommittee and has held meetings with the applicant and NRC staff; an Atomic Safety and Licensing Board has been appointed, and the



Artist's drawing of proposed facility for manufacturing floating nuclear power plants, to be located on an island near Jacksonville, Fla. The barge-mounted plants would be built in the slipway at the center of the drawing, moving through 6 positions from right to left as construction progresses. A completed plant, measuring 400' x 378' and displacing 150,000 tons, is moored outside the slipway for final (nonnuclear) testing. At upper left, the completed plant is being towed to an approved site. Other buildings on the island provide specific parts to the plant, such as steel, concrete, and electrical components.

first prehearing conference was held in September 1975. Various public organizations have indicated a strong interest in the project and are actively participating in the licensing process.

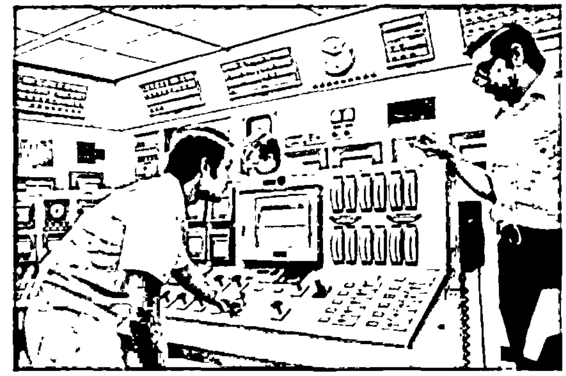
THE HUMAN FACTOR

In order to carry out an effective program of reactor regulation, the NRC examines not only the safety aspects of the design, fabrication and functioning of plants and components, but also the qualifications and organization of personnel associated with quality assurance and safety. Of obvious importance in the "human element" of reactor operation are the qualifications of those who handle the controls, all of whom must be licensed individually by the NRC.

The Commission presently issues two types of operator licenses. In general, anyone who manipulates reactor controls must be licensed as a reactor operator, while those who direct the activities of licensed operators must be licensed as senior reactor operators. Practically speaking, the reactor operator would be the control room operator, and his immediate supervisor would normally be the senior reactor operator.

Each applicant for an operator or senior operator license must submit an application to the Commission together with a certification by an authorized representative of the facility licensee that he or she has need for the license, has completed an acceptable training program, and has learned to operate the reactor controls in a competent and safe manner. A medical report must also be submitted.

To test the knowledge of applicants for each type of license, NRC examiners conduct both written examinations and onsite operating tests. The Commission administers over 600 operator examinations annually; the scope of the examinations is based, in part, on the complexity of the facility, the facility's administrative controls and the responsibilities assigned to the individuals.



An NRC examiner (at right) monitors the performance of an applicant for a reactor operator license at the San Onofre Nuclear Station. The applicant is performing a reactor startup as part of the operating portion of the license exam. He has previously completed an 8-hour written exam, and a 4-hour oral exam on his knowledge of the operation of the plant during normal and abnormal conditions. The examination covers both theory and detailed procedures.

An individual's license is effective for two years, at the end of which time he must be requalified by being actively engaged as an operator or senior operator and successfully participating in a facility-administered requalification program.

As of December 31, 1975, a total of 832 operator and 1,208 senior operator licenses were in effect.

ANTITRUST ACTIVITIES

The NRC is required by the 1970 amendment to the Atomic Energy Act to conduct prelicensing reviews of all applications for nuclear reactors or other production or utilization facilities for commercial use in order to assure that the issuance of any license will not create or maintain a situation inconsistent with the antitrust laws. NRC holds a hearing when recommended by the Attorney General and may also consider whether antitrust issues raised by other persons should be the subject of a hearing. Antitrust hearings are held separately from those on environment, health and radiological safety matters.

To assure that the antitrust review does not delay the licensing procedure, Commission regulations require applicants to submit to the NRC the information requested by the Attorney General at least nine months, and as early as 36 months, before other parts of the construction permit application are filed for acceptance review. The antitrust information, when accepted, is considered "the application" for antitrust purposes and the statutory Attorney General review can proceed from his receipt of the accepted information. This early filing permits the completion of the antitrust review process, including hearings where necessary, in advance of or concurrently with other reviews.

In the period January through June 1975, the Attorney General reviewed 16 applications for nuclear power reactor licenses, and recommended hearings in three cases. (Two of these involved applicants already engaged in antitrust proceedings, and the third was later settled by the staff and the applicant so as to eliminate the need for a hearing.) In four cases, "no hearings" were recommended if the NRC imposed license conditions to reflect certain commitments the applicants had made to the Department of Justice. In one case (the *Wolf Creek* application of the Kansas City Gas and Electric Co. and Kansas City Power and Light Co.) where the Attorney General had furnished advice in December 1974 recommending no hearing if NRC imposed such license conditions, an intervenor's request for a hearing has been granted.

During the past year, an Atomic Safety and Licensing Board completed the first antitrust evidentiary hearing (Consumer Power Co.'s application to construct Midland Units 1 and 2). The Board's initial decision, approving a construction permit without license conditions, has been appealed by the Department of Justice, the NRC staff, and the intervenors.

The antitrust hearing concerned with Alabama Power Co.'s Farley Units 1 and 2 application was conducted during fiscal year 1975 with completion expected in fiscal year 1976. Prehearing proceedings with respect to several applications of the five-member Central Area Power Coordination Group were carried out during the year, and the evidentiary hearing is expected to be conducted during fiscal year 1976.

In a significant decision, the Appeal Board in the *Wolf Creek* application upheld the views previously expressed by the Department of Justice and urged by the NRC staff, and by intervenors in this proceeding, regarding the scope of NRC's prelicensing antitrust review responsibilities.

INDEMNITY AND INSURANCE

The Price-Anderson Act provides for a system of private insurance and government indemnity totaling $560 million to pay public liability claims for personal injury and property damage in the unlikely event of a major nuclear accident. In anticipation of the expiration of the Price-Anderson Act on July 31, 1977, the Joint Committee on Atomic Energy (JCAE) held hearings in 1974 on a bill (H.R. 15323) to modify and extend the provisions of the Act. The bill, as finally passed by Congress, provided for, among other things, a five-year extension of the Act to August 1982. Because of a provision in the bill which would have allowed Congress to rescind the legislation within 30 days after the JCAE submitted its evaluation of the Rasmussen Reactor Safety Study, President Ford vetoed the bill on October 12, 1974. There was no effort made to override the President's veto during the 93rd Congress.

In July 1975, the Federal Energy Administration submitted a legislative proposal almost identical to the vetoed bill (but without the questionable provision) which would modify and extend the Price-Anderson Act for a 10-year period. Hearings on the bill (H.R. 8631) were conducted by the JCAE in September. It was passed by both

Houses of Congress in December and signed on December 31, 1975 as Public Law 94-197.

Coverage for Sabotage or Theft

In June 1975, the Commission submitted a staff study to the JCAE focusing on the question of whether financial protection should be extended to potential harm caused by the sabotage or theft of nuclear materials. The study examined the present Price-Anderson Act coverage with regard to sabotage and theft of nuclear materials, and analyzed an amendment sponsored by Senator Ribicoff. The amendment would have extended the Act to cover all nuclear incidents arising out of the theft or diversion of nuclear material. The study discussed several possible alternatives for providing financial protection for injury or property damage related to such sabotage or theft.

Indemnity Operations

As of June 30, 1975, 117 indemnity agreements with NRC licensees were in effect. Indemnity fees earned by the NRC during the January through June, 1975 period totaled $1,631,521, bringing the total fees collected since inception of the program to $10,477,737. No claims had been made under the NRC's indemnity agreements with licensees during the 18 years of the program's existence.

Insurance Premium Refunds

During the year, the two private nuclear energy liability insurance pools, the Nuclear Energy Liability and Property Insurance Association, and Mutual Atomic Energy Liability Underwriters, paid to policyholders the ninth annual refund of premium reserves under their industry credit rating plan. The refunds totaled $1,468,002, which is 69.9 percent of all premiums paid by the affected policyholders in 1965, and approximately 97.8 percent of the reserve established from these premiums.

Under the rating plan, a portion of the annual premiums is set aside as a reserve for either payment of losses or ultimate return to policyholders. The amount of the reserve available for refund is determined on the basis of loss experience of all policyholders over the preceding 10-year period.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards is a group established by law to review and report to the Commission on safety studies and license applications for nuclear power reactors and other major nuclear facilities such as spent fuel processing plants. 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.

During the last half of fiscal year 1975, the ACRS held a total of six regular meetings, and conducted 47 meetings of ACRS subcommittees and *ad hoc* working groups. In conjunction with subcommittee meetings, 10 site/facility visits were made.

The committee provided between January 1 and November 30, 1975, reports to the Nuclear Regulatory Commission and its predecessor (the Atomic Energy Commission) concerning construction permits for 14 nuclear power plants and operating licenses for three others. Reports were also provided regarding; (a) need for attention to protection against industrial sabotage; (b) reactor safety research; (c) evaluation models for NRC criteria for the emergency core cooling systems (ECCS) for light-water-cooled nuclear power reactors; (d) a power increase for the Rancho Seco Nuclear Station; (e) siting of the Douglas Point reactor; (f) a partial construction permit review for the Diablo Canyon Nuclear Plant; (g) cracking in boiling water reactor piping; (h) status of generic items relating to light-water reactors; (i) the General Electric Standard Safety Analysis Report (GESSAR 238); (j) the Reactor Safety Study, WASH-1400 (the Rasmussen Report); (k) emergency planning; (l) a permanent dewatering system for the Perry Nuclear Plant; (m) Fast Flux Test Facility interim review; (n) Combustion Engineering Standard Safety Analysis Report (CESSAR-80); (o) Westinghouse Standard Safety Analysis Report (RESAR-41); and (p) operation of the loss of Fluid Test Facility.

The committee also provided comments and recommendations regarding its future role and scope of activities in providing advice to the Nuclear Regulatory Commission, and a proposed change in the legislative requirement that the ACRS review all construction permits and operating licenses for nuclear power plants.

The ACRS provided advice to the Executive Director for Operations, NRC, on the following subjects:

(1) Locking out of ECCS power-operated valves;
(2) Safe-end welds on the Beaver Valley plant;
(3) Documentations of seismic design bases to be applied east of the Rockies;
(4) Anticipated transients without scram;
(5) Qualification for radiation protection personnel;
(6) Loss-of-coolant-accident best estimate calculational methods;
(7) Containment spray additives.

The ACRS approved and/or provided comments on five regulatory guides. The committee also approved a revision to the NRC's General Design Criterion 17—"Electrical Power Systems." The ACRS also completed a study regarding the status of the Loss-of-Fluid - Test Facility Safety

The Advisory Committee on Reactor Safeguards, a statutory body of scientists, engineers, and other experts in fields related to nuclear safety, conducts independent reviews of nuclear power plant applications and other matters referred to it by the NRC.

Research Program and other aspects of light-water reactor safety research, and the application of foreign fast-reactor safety information to U.S. fast reactors, and provided reports to the General Accounting Office.

The committee continued to make available to the public a large number of documents relating to the activities of the ACRS since it became a statutory committee in 1957, and this process was almost completed during the year. These documents include all minutes of meetings of the committee and its subcommittees, reports from consultants, and all draft documents that had been discussed with groups outside the ACRS except those precluded by law. In addition to this effort to provide the public with a more complete understanding of the nature and scope of its activities, the ACRS and its subcommittees held 45 meetings during the last half of fiscal year 1975 with sessions open to the public. Comments from the public were received and considered at several of these meetings.

The committee provided recommendations to the Commission regarding its future role as an advisory committee and is continuing to review the manner in which the ACRS may most effectively contribute to nuclear safety in light of the establishment of the Nuclear Regulatory Commission as a separate agency.

IMPROVING THE PROCESS

Many management actions have been taken during the past several years to improve the efficiency of nuclear power plant licensing reviews, while at the same time strengthening their quality.

Looking at the total time involved in nuclear power schedules, it is noted that, in recent years, an average of some 10 years has been required from the time a utility makes a decision to build a nuclear power plant until the completed facility is ready to operate under NRC license. The major portion of this time—six to seven years—is taken in construction. A strong effort is being made to reduce, or eliminate, the time (approximately two years) that licensing reviews and proceedings are on this critical path.

The Commission, during 1975, not only emphasized finding new management initiatives under existing law to develop optimum efficiency in the process consistent with its mission of effective regulation, but also supported proposed legislation to provide further opportunity to restructure the process.

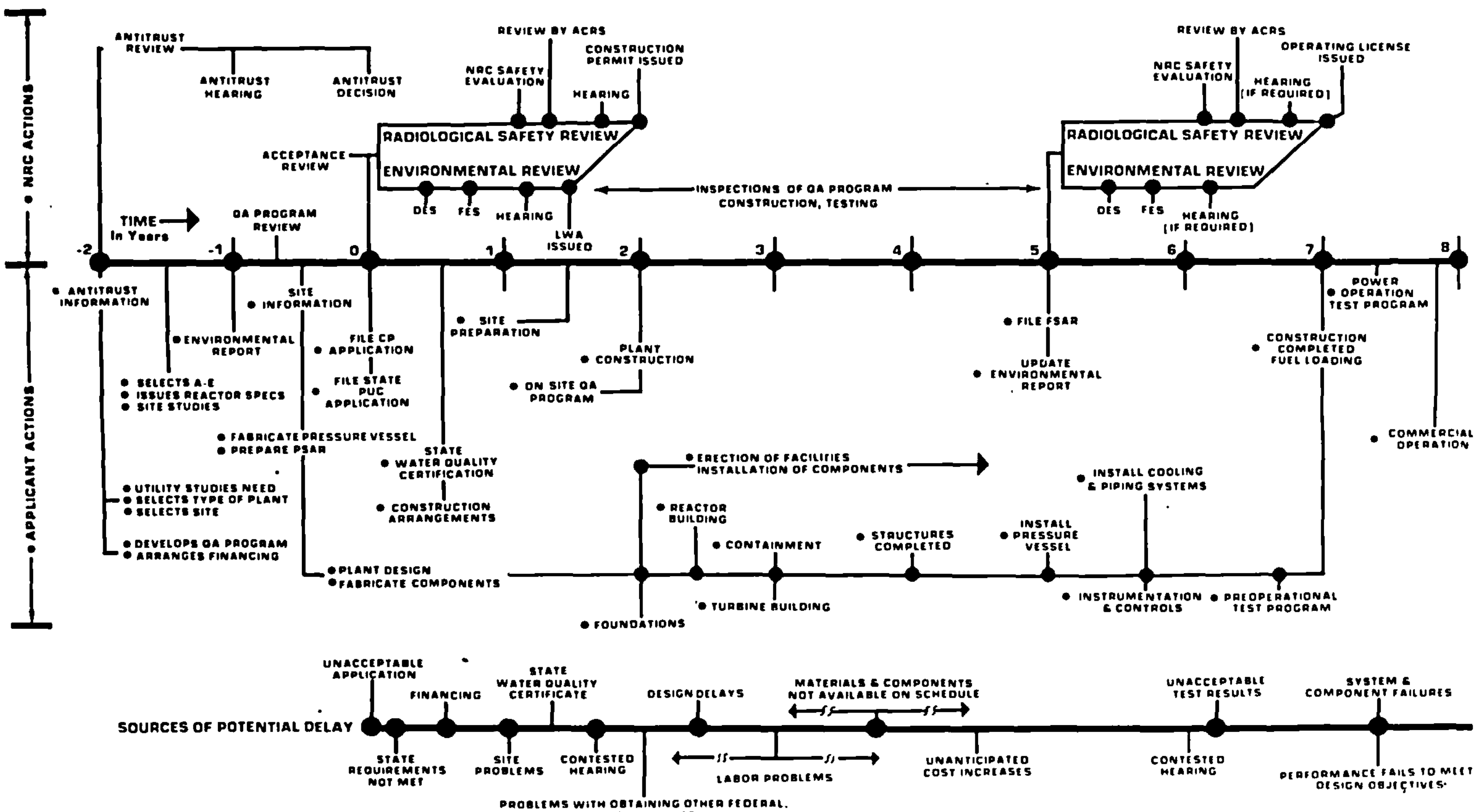
Administrative Actions

Standardization offers great promise of saving in time and money to applicants, particularly in reducing design and construction costs. The review and approval process for several reference designs is nearing completion, and a number of utilities are referencing these standard designs in their applications for construction permits. (See earlier discussion in this chapter.)

Early site reviews. Policies and procedures are being developed for the early review of sites planned for nuclear power stations, independent of the specific design and construction of the station itself. These are designed to remove site related matters from the critical path and to provide advanced assurance of site acceptability either for certain considerations or for all pertinent aspects. Two applications submitted in September 1975 which contain complete safety and environmental site information, will be reviewed under these policies and procedures.

Limited work authorizations. The LWA procedures allows certain onsite construction activities to be undertaken prior to issuance of a construction permit, but

NUCLEAR POWER PLANT LICENSING AND CONSTRUCTION PROCESS



Abbreviations A E = architect engineer; QA = quality assurance, CP = construction permit, PUC = Public Utility Commission, DES = draft environmental statement, FES = final environmental statement, LWA = limited work authorization

Table 2. LIMITED WORK AUTHORIZATION EXPERIENCE

Project	LWA Issue Date	Actual and Estimated Construction Permit Decision	Initiation of Construction Improvement (Mos.)
Grand Gulf	05/01/74	09/04/75	4.0
Waterford 3	05/14/74	11/14/74	6.0
Catawba 1 & 2	05/16/74	08/07/74	15.0
Vogtle 1-4*	05/28/74	06/28/74	1.0
Millstone 3	06/07/74	08/06/74	2.0
Bellefonte 1 & 2	09/17/74	12/24/74	3.0
Surry 3 & 4	10/04/74	12/20/74	2.5
Comanche Peak 1 & 2	10/17/74	12/19/74	2.0
Perry 1 & 2	10/21/75 (LWA-1) 12/31/75 (LWA-2)	03/15/76E**	17.0
Byron 1 & 2	12/13/74 (LWA-1) 10/30/75 (LWA-2)	12/31/75	13.0
Braidwood 1 & 2	01/14/75 (LWA-1) 10/30/75 (LWA-2)	12/31/75	12.0
St. Lucie 2	03/17/75	05/04/76E	14.0
WPPSS 1	08/01/75	12/24/75	4.5
WPPSS 4	08/01/75	***	–
Summit 1 & 2	08/07/75	***	–
South Texas 1 & 2	08/12/75	12/22/75	4.5
Callaway 1 & 2	08/14/75	04/05/76E	8.0
River Bend 1 & 2	09/05/75	02/27/76E	5.5
Clinton 1 & 2	10/01/75	02/23/76E	4.7
Davis-Besse 2 & 3	12/31/75	12/27/76E	12.0

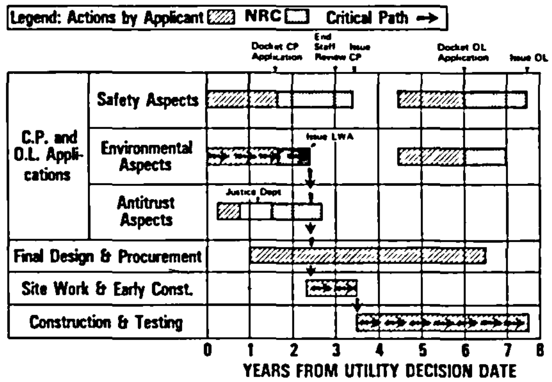
RANGE OF IMPROVEMENT: 1-17 months
AVERAGE IMPROVEMENT: 7 months

*Vogtle Units 3 and 4 have been cancelled.
**Denotes schedule estimate is not firm.
***Project deferred after issuance of LWA.

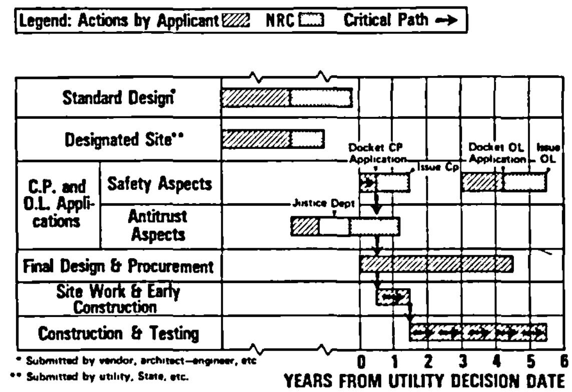
only after issuance of the final environmental impact statement and consideration of that statement in the licensing review process, including formal public hearings and issuance of a decision by the presiding Atomic Safety and Licensing Board. In addition, it must also be determined that the site is generally suitable and that there are no unanswered safety questions with respect to any construction work that may be safety related. The scope of work that may be undertaken is carefully defined and limited, and the work is undertaken at the applicant's risk. However, the LWA procedure enables applicants to start construction

TIME REQUIRED FROM CONCEPTION TO OPERATION OF NUCLEAR PLANTS

(With Limited Work Authorization Procedure)



(Assuming Standardized Plants and Designated Sites)



as much as six or more months earlier than would be the case if construction could not be started until the entire safety review process and related public hearings were completed. The head start in construction is reflected, at least on a month-for-month basis, in the total time to bring the plant on line. Since institution of the LWA procedure in April 1974, it has resulted in an average improvement of seven months in initiation of construction for 19 projects representing 35 nuclear units (see Table 2).

Standard review plans. The Standard Review Plans for safety reviews were completed and implemented during the year. The plans describe the process by which the staff determines that a given design provides adequate protection of the public health and safety. The primary purposes of the plans are to improve the quality and uniformity of staff reviews, to stabilize the safety review process, and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. The goal in adoption of these plans is to assure that only essential requirements are imposed on license applicants. Standard plans are also being developed for the environmental reviews.

Standard formats. Standard format and content guides have been published for Safety Analysis Reports and Environmental Reports that are required with reactor license applications. They identify the principal information needed in the staff reviews and define a format acceptable to the NRC staff. While conformance with the Standard Format is not required, its use minimizes the staff's need to request additional information during the course of the review and thus eliminates unnecessary delays. The second major revision of the Safety Standard Format for light-water reactors—keyed to the Standard Review Plans—was issued in September 1975 (Regulatory Guide 1.70).

Scheduling. An important responsibility of the NRC personnel who manage the review process is the establishment and maintenance of detailed schedules for the review and its milestones. The scheduling process is based on the critical path method and uses a computerized system.

Pre-docketing activities. The NRC staff has implemented a number of procedures which are carried out prior to the acceptance of an application for formal review. These procedures identify problems as early as possible in the applicant's planning process and help assure that the application is adequate before the review starts.

Meetings are held with potential applicants up to a year in advance of application submittal to inform them of review procedures and requirements. Emphasis is placed on quality assurance during this phase. Applicants may request an informal site review prior to preparation and submittal of the application to determine the general suitability of the site. The staff requests, on a routine basis, that potential applicants submit certain site related information six months prior to application tendering to enable an early start on any site-related critical-path items which might be identified. The staff performs a 30-day acceptance review of tendered applications to determine completeness and degree of conformance with the Standard Format. If the application is not acceptable, it is rejected and returned to the applicant for the necessary improvements. This is a desirable alternative to accepting an incomplete application and incurring later delays in the review process.

Regulatory Guides. A series of regulatory guides is under continuing development. The purposes are to make available to applicants methods of implementing specific parts of the NRC regulations which are acceptable to the staff, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, and provide guidance to applicants concerning certain of the information needed for review of applications. (See Chapter 9.)

Topical reports/generic reviews. The major nuclear steam supply system manufacturers, architect-engineering firms and major component manufacturers are encouraged to prepare and submit topical reports which describe proposed solutions to safety problems, results of research and development programs, and current analytical techniques. The staff evaluates these reports, which have general applicability to several plants or designs. If found acceptable, they can be simply referenced in applications, thus reducing repetitious review and accelerating the process. A related step is the staff's policy of identifying issues and problems which have applicability to a number of plants or review cases and resolving them generically rather than on a case-by-case basis.

Change requirements reviewed. Changes in NRC licensing requirements have been frequently blamed as the cause for significant delays and additional costs in the licensing process. While many of these changes involve significant safety matters, and are viewed as a justifiable part of the licensing review process, the NRC staff has taken steps to carefully manage significant changes to preclude unwarranted imposition of new requirements. All new regulatory guides, which inform the industry of acceptable licensing positions, are critically reviewed by the Regulatory Requirements Review Committee, representing top NRC management, before approval. Additionally, guidance on staff review considerations and positions is written into the staff's Standard Review Plans. Finally, NRC management will meet with applicants to resolve any disagreements with staff positions that relate to their applications. These procedures are clearly established, and information regarding them has been made public.

Legislation Urged

In May, the Commission forwarded to the Congress a legislative proposal to improve the licensing process for major nuclear facilities, which was introduced as S. 1717 and H.R. 7002. In hearings conducted in June by the Joint Committee on Atomic Energy, the Commission strongly supported the proposed licensing reform legislation as a measure that could lead to reduction of the time now required to bring a nuclear power plant on line from eight or more years to about six years.

The basic concepts of early site approvals and standard plant designs are at the heart of the proposed legislation. The

Commission noted it would provide a more efficient framework for siting and licensing without impairing the quality or thoroughness of the NRC's safety, antitrust or environmental reviews, or depriving the Commission or the public of the benefits of full public participation in the process. It would make a major contribution to attainment of more efficient, effective regulation which is essential if nuclear power is to be a viable option in meeting the country's need for electric energy.

Highlights of Legislation

Main features of the proposed legislation are:

(1) Provision for approvals of sites for nuclear plants independent of construction permit applications. Site permit applications could be filed by interested States as well as by utilities proposing to construct plants. An inventory of approved sites could be developed. There would be a complete environmental review and opportunity for formal hearing before issuance of any site permit.

(2) Encouragement of standardization of nuclear plants by providing for combined construction permits and operating licenses, by encouraging early public participation in the resolution of plant design questions, and by avoiding duplicate hearings.

MAJOR FEATURES OF NRC'S PROPOSED LICENSING LEGISLATION

- Separate and Early Site Reviews
  - Utilities, States, or Others Could File Site Application
  - Inventory of Sites Could be Developed
  - Full Environmental Review of Site with Opportunity for Public Hearing
- Plant Standardization Encouraged
  - Combined Construction Permit and Operating License Possible
  - Duplicate Hearings Avoided
- Public Participation Enhanced
  - Hearings on Site Suitability and Plant Design Issues Held Early
  - NRC Assistance to Hearing Participants Provided by Documents, Meetings, & Studies

(3) Enhancement of public participation by providing for hearings on site suitability and design questions at early points in time when they can be most effective, and by providing for certain assistance to hearing participants.

Other improvements that would be effected in the licensing process include:

- Hearings only when requested.
- ACRS review at option of Commission or ACRS.
- Issuance of interim operating license or operating license amendments for power reactors pending completion of hearing.
- Provisions to expedite commencement of construction.
- Strengthened Federal-State cooperation.

Nuclear Energy Center Survey

Throughout 1975 the NRC was engaged in conducting a study to identify resource requirements for nuclear energy centers and a national survey of large land areas to identify possible nuclear energy center sites, as directed by Section 207 of the Energy Reorganization Act of 1974. Such sites, as defined in the Act, would be large enough to support utility operations or other elements of the total fuel cycle or both, including, if appropriate, nuclear fuel reprocessing facilities, fuel fabrication plants, nuclear waste storage facilities, and uranium enrichment facilities.

The Act further stated that the survey shall include (a) a regional evaluation of natural resources, estimates of future electric power requirements that could be served by each site, assessment of economic impact, and consideration of other relevant factors; (b) evaluation of the environmental impact of such centers; and (c) consideration of federally-owned land except national parks, forests, wilderness areas and monuments.

NRC's report, including evaluation of results of the survey and conclusions and recommendations regarding the feasibility and practicality of locating nuclear power reactors and/or elements of the fuel cycle on nuclear energy centers, is scheduled for publication and transmittal to the Congress and the Council on Environmental Quality in January 1976. As mandated by the Act, the Nuclear Energy Center Site Survey (NECSS) is being conducted in cooperation with other interested Federal, State and local agencies and is considering the views of utilities, citizens' groups, and others.

Should NECs prove to be feasible and practical, one of the important additional aspects of the study will be an evaluation of siting characteristics for NECs. While siting guides will not be developed, judgments will be made regarding those siting characteristics that might affect licensability.

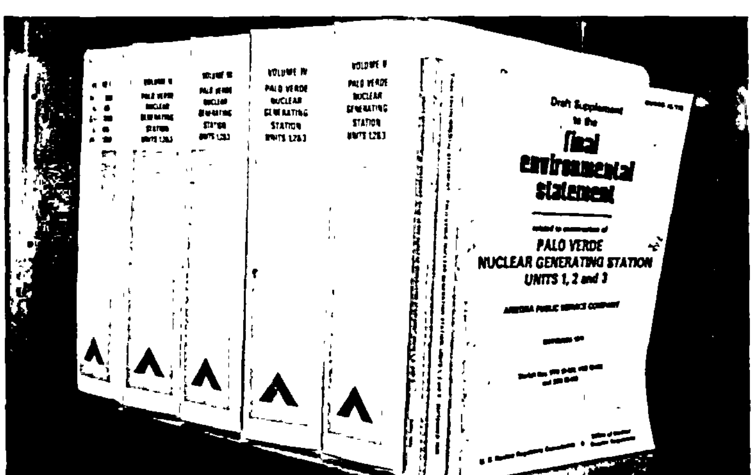
Also, should NECs prove to be feasible and practical, follow-on activities will be identified that are necessary for the further development of the concept. These activities could involve research and development programs that would need to be performed by others, the development of better information to guide the successful development of an NEC, and the performance of other studies (such as offshore siting and industrial collocation) that could affect aspects of development.

Protecting the Environment

Balancing Benefits Against Risks

All Federal agencies are required by the National Environmental Policy Act to factor into their decision-making process the consideration of the impact of major actions that could significantly affect the quality of the human environment. NRC decisions falling under this requirement include licensing of nuclear facilities; amendments, renewals and terminations of licenses; adoption of new significant regulations (rulemaking); and authorization of new methods or processes (for example, the widespread use of nuclear-powered cardiac pacemakers (Chapter 4), the recycling of plutonium as reactor fuel (Chapter 4), or the centrifuge process for the enrichment of uranium).

Each proposed major action requires comprehensive evaluations and assessments of the full range of anticipated environmental effects, both radiological and nonradiological. The results, plus a study of the available alternatives, are used to arrive at a balancing of environmental, economic, technical and other benefits against



environmental and other costs, in the public interest. NRC has a specialized staff to carry out its environmental impact reviews for both nuclear reactor and fuel cycle facilities, with appropriate support from expert consultants and personnel of ERDA's national laboratories.

If no significant environmental impact—one that would involve an amendment or affect renewal of an existing license—is indicated, a brief environmental impact appraisal report will be written and a negative declaration published in the *Federal Register* announcing that NRC has decided not to prepare an environmental statement. Minor license amendments covering changes which do not affect the environment can be made with the inclusion in the official record of the staff's assessment that the change will have no measurable environmental effect.

During 1975, the Commission underscored the importance of environmental impact knowledge by establishing within its Office of Nuclear Regulatory Research a division with specific responsibilities to plan and administer environmental research programs in support of nuclear regulation. (See Chapter 8.)

Discussed in this chapter are (1) the Commission's procedures and actions in reviewing the impact of nuclear power plants, including actions and findings concerning control of low-level radioactivity in effluents; (2) environmental activities in the nuclear fuel cycle, and (3) efforts to achieve optimum coordination with other Federal agencies having overlapping environmental responsibilities.

REACTOR REVIEWS

Staff Assessment Process

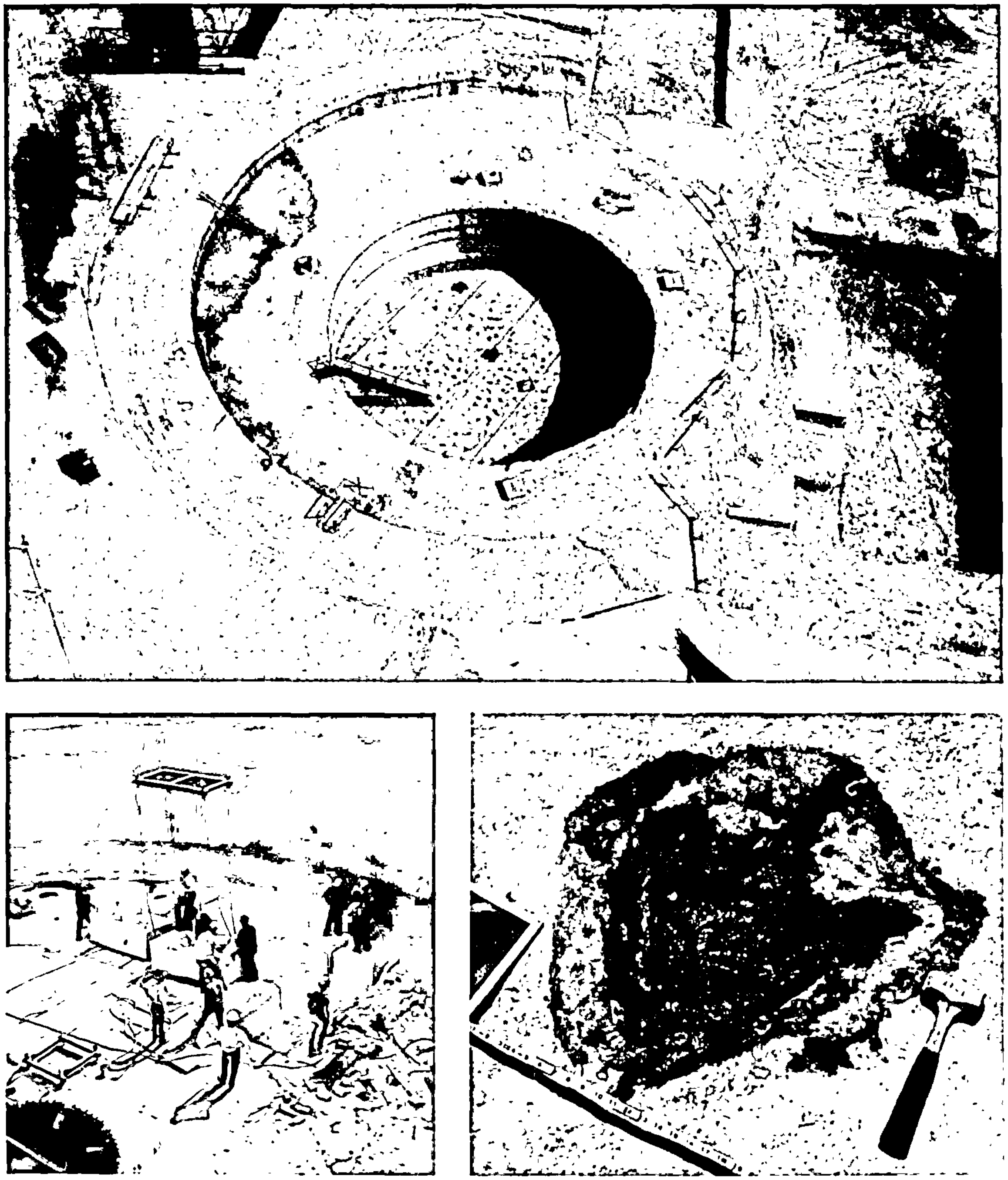
NRC procedures require an applicant for a nuclear power plant construction permit or operating license to submit an environmental report which discusses in detail:

- The site and reactor characteristics;
- Power needs in the area;
- The environmental effects of site preparation, and plant and transmission facilities construction;
- The environmental effects of plant operation;
- Effluent and environmental measurements and monitoring;
- The environmental effects of accidents;
- The economic and social effects of plant construction and operation;
- Alternative energy sources and sites; and
- Plant design alternatives.

The applicant's environmental report also must demonstrate through a cost-benefit analysis of the proposed plant why, in the applicant's judgment, the aggregate benefits outweigh the aggregate costs.

Upon receipt of an application for a construction permit or an operating license, the NRC staff reviews the applicant's environmental report to determine whether it is sufficiently complete to permit the NRC to perform its own independent and detailed evaluation of the potential environmental impact of the proposed nuclear facility, assuring that the benefit of the plant to the public outweighs possible risk to the environment.

This review for acceptability is completed in about 30 days. If the acceptance review shows that the applicant's report is adequate as a starting point for the NRC environmental review, the staff undertakes its review, which culminates in a draft environmental statement on which comments are solicited from appropriate Federal, State and local agencies and from the public. All comments received are considered in detail and appropriate revisions are reflected in a final environmental statement, which is sent to all persons who commented on the draft and is available to members of the public through the Na-



Archaeological studies of the proposed site of the Comanche Peak nuclear power plant near Glen Rose, Texas, were begun several years before construction of the plant was authorized. As a condition of the construction permit issued by the NRC, the utility was required to notify the Texas Historical Commission of any significant archaeological findings during construction. In the spring of 1975, dinosaur tracks were discovered in the excavation for Unit 1 of the plant. Although it was determined that the tracks were not unique to this geologic formation, the Texas Utilities Generating Co. removed large blocks containing five well defined "acrocanthosaurus" tracks for preservation.

Table 1. Environmental Impact Statements Issued From January 1 through November 30, 1975

Plant	Date Issued
Draft Statements	
1. Skagit	1/28/75
2. Jamesport Units 1 & 2	2/12/75
3. Davis Besse Units 2 & 3	2/14/75
4. Cherokee	3/31/75
5. Palo Verde Units 1-3	4/14/75
6. Alan R. Barton Units 1-4	4/28/75
7. Davis Besse Unit 1	4/30/75
8. Perkins Nuclear Station Units 1-3	5/9/75
9. Wolf Creek Generating Station	7/3/75
10. William B. McGuire Units 1 & 2	10/29/75
11. Montague Nuclear Power Station Units 1 & 2	11/5/75
Final Statements	
1. Indian Point Unit 3	2/21/75
2. Washington Public Power Supply System (WPPSS) Units 1 & 4	3/11/75
3. Callaway	3/26/75
4. South Texas Units 1 & 2	3/28/75
5. Pebble Springs Units 1 & 2	4/16/75
6. H. B. Robinson Unit 2	4/18/75
7. Fulton Generating Station Units 1 & 2	5/5/75
8. Skagit Units 1 & 2	6/2/75
9. WPPSS Units 3 & 5	6/3/75
10. Hartsville Nuclear Plants	6/25/75
11. Palo Verde Nuclear Generating Station Units 1-3	9/25/75
12. Davis Besse Units 2 & 3	9/30/75
13. Cherokee	10/1/75
14. Floating Nuclear Power Plant, Part 1	10/6/75
15. Jamesport Nuclear Power Station Units 1 & 2	10/7/75
16. Perkins Nuclear Station Units 1-3	10/31/75
17. Davis Besse Unit 1	10/31/75
18. Wolf Creek Unit 1	10/31/75

tional Technical Information Service of the U.S. Department of Commerce. This procedure is followed for each construction permit and operating license application, but, in the latter case, the review is limited to significant changes in design, operation or impact that might have occurred since issuance of the construction permit.

At the construction permit review stage, the staff's final environmental statement is considered at a public hearing by the Atomic Safety and Licensing Board. A similar hearing may occur at the operating license review stage if warranted by public interest.

Environmental impact statements were prepared by NRC staff from January through November 1975 on the nuclear power facilities listed in Table 1. During the same period, some 30 negative declarations were published, principally concerning changes in technical specifications of operating reactors.

Environmental Protection Actions

NRC environmental statements normally recommend that construction permits and operating licenses be conditioned to require the licensee to take specified environmental protection measures. These may range from the relocation of proposed transmission line corridors and upgrading of construction practices to major plant design modifications.

All nuclear power plant operating licenses that have been issued since January 1972 contain detailed environmental technical specifications which establish operating limitations and procedures and require detailed environmental monitoring programs, both radiological and nonradiological, to verify the anticipated impact of the plant. If unacceptable environmental effects are detected, the licensees are required to provide to the staff an acceptable analysis of the problem and a plan of action to eliminate or significantly reduce the harmful effects or damage.

Some of the changes that have been required of licensees as a result of NRC environmental reviews are as follows:

- Redesign of intake structure to limit water velocity to one foot-per-second (many plants)
- Major cooling system redesign to use closed cycle cooling (Indian Point, Brunswick, Peach Bottom and Catawba plants).
- Reduction in discharge temperature limits (McGuire plant).
- Augmented dispersion of thermal plume in receiving waters (Crystal River, Waterford, Millstone, LaSalle, North Anna plants).
- Augmented radwaste systems (Cooper, San Onofre, Arkansas, and Waterford plants).
- Expanded environmental surveillance (San Onofre, Arkansas, Fort St. Vrain, Midland and Zimmer plants).
- Modified chemical waste systems (Midland, Waterford and Point Beach plants).
- Causeway redesign (Maine Yankee plant).
- Transmission line rerouted (Midland and Seabrook plants).
- Fish screens added at intake structure (Surry plant).
- Alternative site selected for construction of plant (Newbold Island plant).

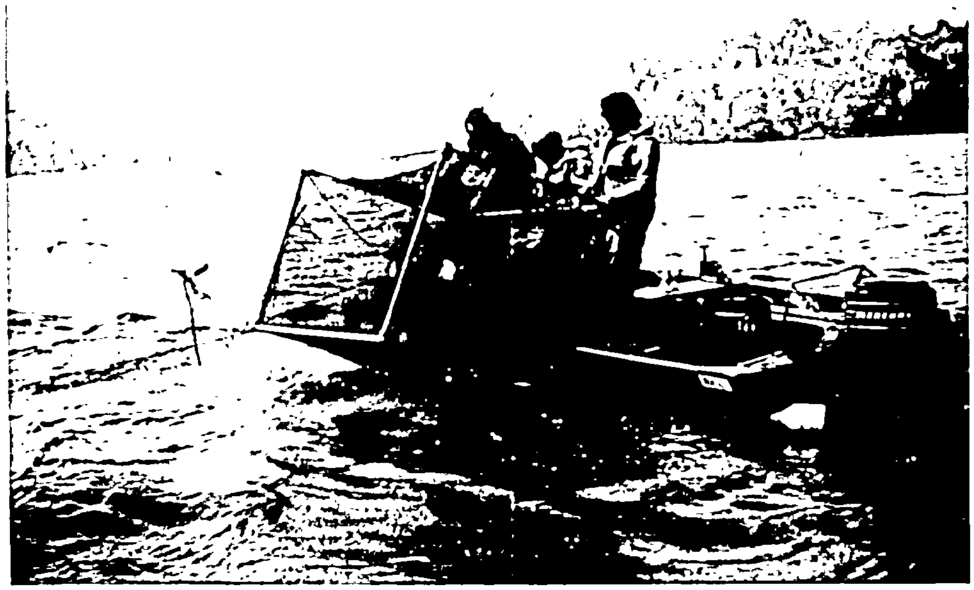
SOME PLANT CHANGES REQUIRED AS RESULT OF NRC ENVIRONMENTAL REVIEW

- Water Intake Structure Redesign
- Major Cooling System Redesign
- Modification of Thermal Plume from Water Discharge
- Augmentation of Radioactive Waste Systems
- Modification of Chemical Waste Systems
- Rerouting of Transmission Lines
- Installation of Fish Screens
- Modification of Environmental Monitoring Plans
- Recommendation for Change of Site

Industry Guidance

All of NRC's policies and procedures for implementing NEPA in licensing and rule-making proceedings are consolidated, insofar as possible, in Part 51 of NRC regulations.

The NRC staff is continuing its efforts to provide guidance to license applicants on



A box trap is about to be lowered into the water near an operating nuclear power plant to take fish samples from the river. These samples will be studied as part of the required environmental monitoring program to determine the effects of the plant's operation on the marine biota near plant discharge structures.

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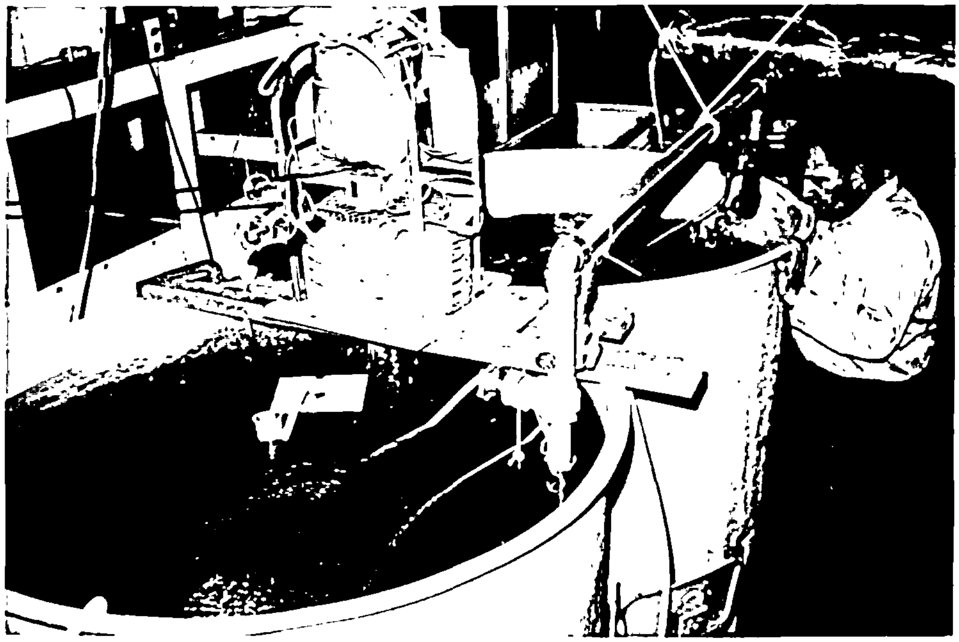


A "cold shock" experiment is under way at Consolidated Edison Company's Environmental Laboratory near the Indian Point nuclear power plant. Fish in the tanks are exposed to temperature drops to determine the maximum safe rate for lowering the temperature at the plant discharge in the event of a plant shutdown. The Laboratory is the keystone of the New York utility's ongoing ecological survey of the power plant's effects on fishlife in the Hudson River.

the siting and environmental policy requirements for nuclear power plants and other nuclear facilities.

Revisions of two important regulatory guides were issued in 1975. The first, "Preparation of Environmental Reports for Nuclear Power Stations" (Regulatory Guide 4.2, Rev. 1), provides up-to-date guidance on information which should be provided in environmental reports that accompany license applications. The second, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants" (Regulatory Guide 4.1, Rev. 1), provides the latest guidance on acceptable programs for monitoring radioactivity around nuclear power plants. Such programs are essential for providing additional assurance that the public health and safety and the environment are adequately protected; they also provide NRC with data by which to determine if the plants are operating within the conditions of their licenses.

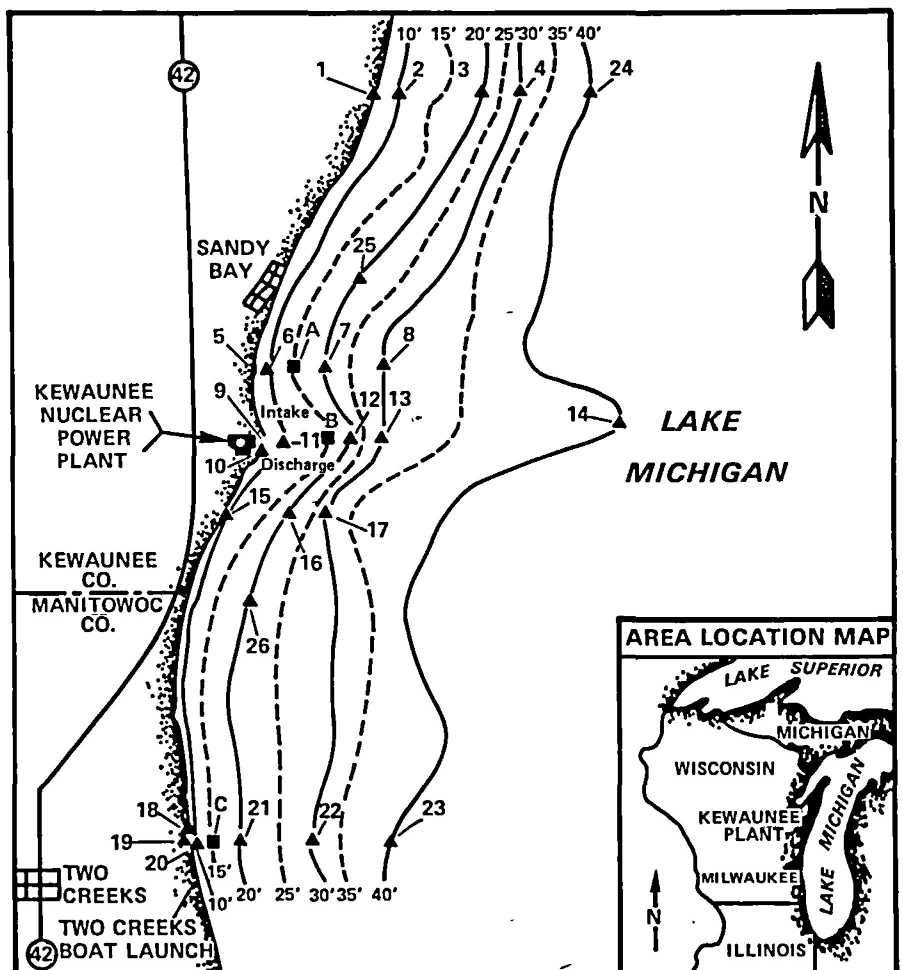
Amendments to Part 51 that became effective February 5, 1975, allow applicants in their environmental reports, and the Commission in its detailed environmental statements, to account for the environmental effects of transportation of fuel and waste to and from light-water-cooled reactors by using specified numerical values in Summary Table S-4 of Part 51, "Environmental Impact of Transportation of Fuel and Waste to and from One Light-Water-Cooled Nuclear Power Reactor."

In April 1975, the Commission published (as NUREG-75/038) Supplement I to WASH-1238, "Environmental Survey of Transportation of Radioactive Materials to and from Nuclear Power Plants." The supplement presents data and identifies methods used to derive the environmental impact and risk values in Summary Table S-4, Part 51.

New Rule on Effluent Control

On April 30, the Commission announced a significant new regulation, effective on June 4, 1975, which provided definitive numerical guidance to NRC licensees on how to comply with the previously enunciated requirement that levels of radioactive material in effluents from light-water-cooled power reactors and resultant doses to the public be kept "as low as reasonably achievable."

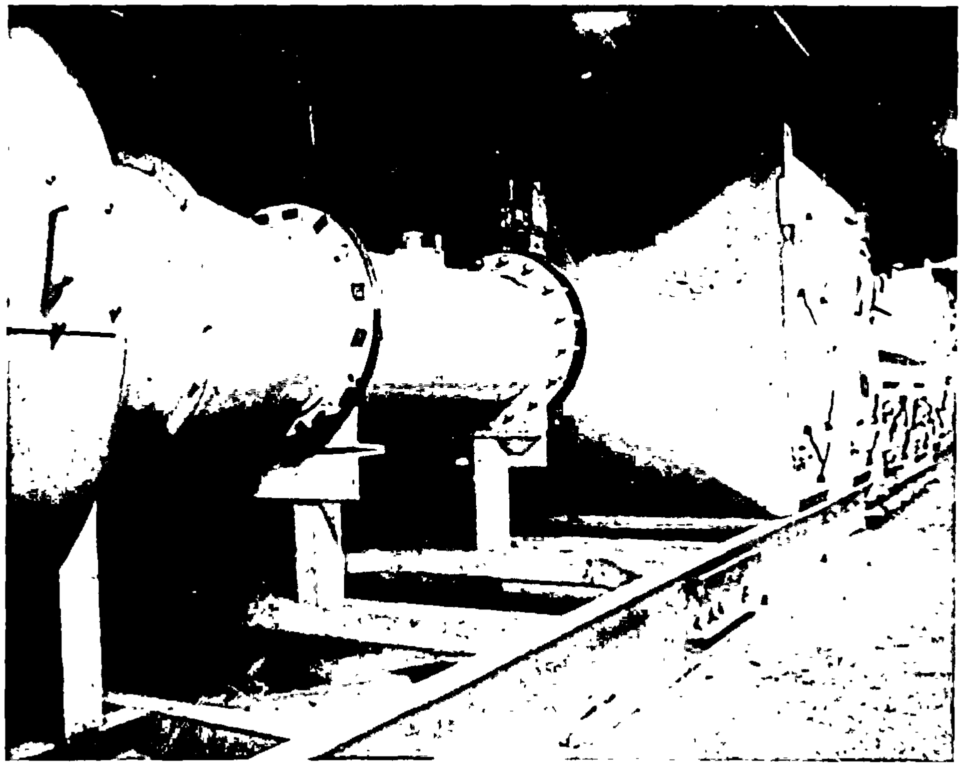
The new rule, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," (Appendix I to 10 CFR Part 50) concluded a rulemaking proceeding lasting more



SAMPLING LOCATION

Current	12,14,25,26	Zooplankton	2-4, 6-8, 11-13, 15-17, 20-22
Temperature	14,25,26	Periphyton	1, 9, 18
Profile location	2-4, 6-8, 10-14, 15-17, 20-24	Benthos	3, 6-8, 11-13, 16, 17
Water Quality & Bacteriology	2, 7, 11, 12, 14, 16, 20, 23, 24	Minnow seining	5, 9, 19
		Gill nets	A, B, C
Phytoplankton	2, 7, 11, 12, 14, 16, 20, 23, 24	Fish larvae	6, 7, 11, 12, 20, 21

Wisconsin Public Service Co.'s proposed program for environmental monitoring offshore of its Kewaunee Nuclear Power Plant was reviewed and approved by the NRC. The details of the approved program are included in Environmental Technical Specifications, which are a part of the operating license issued by the NRC.



Above is a typical gas cleanup system inside a nuclear power plant. This filter system is designed to remove radioactive iodines and particulate matter from the atmosphere inside the plant before release to the environment. Any radioactivity released from the plant must be within strict limits set by the NRC to protect the public health and safety.

than three years which involved an evidentiary public hearing, an environmental statement, more than 4,000 pages of transcript, and thousands of pages of prepared testimony and exhibits.

Implementation of the regulation will keep radiation doses to persons living near nuclear power plants at levels that are small fractions of doses occurring from natural background radiation levels[1] and of existing radiation protection standards[2] followed by all Federal agencies.

The new regulation involves both design objectives and a cost-benefit analysis requirement to determine the point at which the costs of further reductions of emissions of radioactive material are not justified.

The design objectives are to:

(1) Restrict the amount of radioactive material released in liquid effluents from any light-water-cooled power reactor to levels that would keep the annual doses to any individual in an unrestricted area to not more than three millirems for the total body and not more than 10 milli-

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[1] Average annual doses from natural background radiation in the U.S. are in the range of 100 to 125 millirems, but vary from about 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.

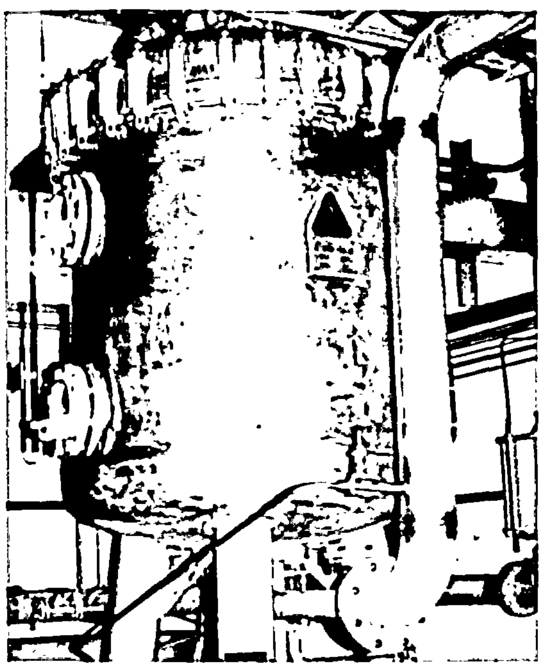
[2] Existing radiation protection guides—formerly developed and recommended for Presidential approval by the Federal Radiation Council (FRC), which was abolished in 1970—are the responsibility of the Environmental Protection Agency, to which the FRC functions were transferred. These guides are designed to establish permissible dose levels for humans rather than the lowest practicable emission levels. They limit to 500 millirems annual total-body dose for an individual in the general population from all sources of radiation except natural background and medical sources.

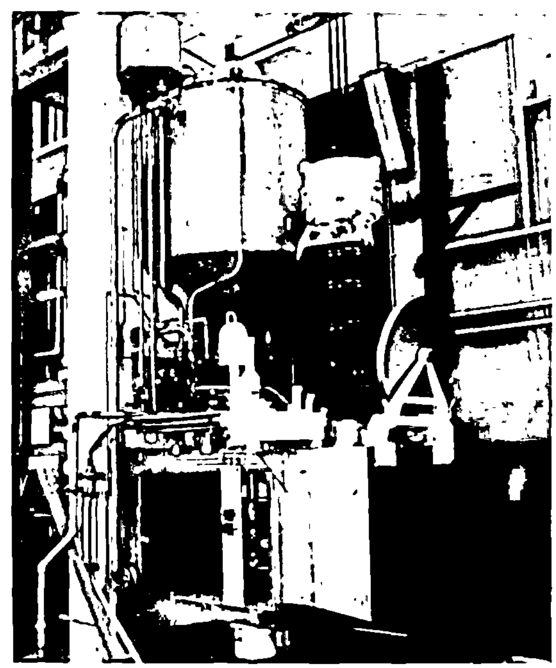
rems for any single organ;

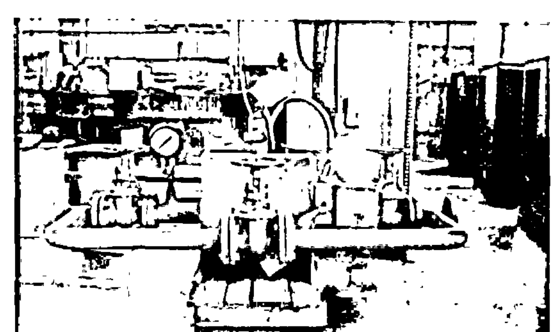
(2) Restrict the amount of radioactive material in gaseous effluents from any light-water-cooled power reactor to keep annual doses to any individual in an unrestricted area to not more than five millirems to the total body and not more than 15 millirems to the skin; and

(3) Restrict the amount of radioactive iodine and other radioactive material in gaseous effluents from any light-water-cooled power reactor to keep annual dose to any organ of any individual in an unrestricted area to no more than 15 millirems.

In its decision, the Commission adopted a landmark technique—a quantitative approach—for assessing the cost-benefit of achieving further reductions in emissions of radioactive material. This approach seeks, for the first time, to express both the costs and benefits of a reduction in radiation exposure levels in comparable units in order to achieve an objective cost-benefit balance. Contributions to the record in the rulemaking proceeding comparing the costs of further measures to the benefits of any reduction achieved ranged from $10 to $980 for total-body exposure to a standard unit of radiation dose equivalent, the "man-rem." (The man-rem is a







Liquid radioactive wastes are processed through several systems at a nuclear power plant. Radioactive solid material suspended in the liquid waste is separated out in the filter equipment shown above. These solids are compacted and packaged in drums (above right) for shipment to one of the six waste burial grounds throughout the United States. The remaining liquid waste—which has a low level of radioactivity—is measured (below right) and released outside the plant in controlled amounts in order to assure that NRC limits for radioactivity in effluents are not exceeded.

measure of collective dose equivalent for large groups of people—for example, 100 people, each receiving 0.01 rem; or 1,000 people each receiving 0.001 rem.)

The Commission indicated a further rulemaking hearing would be necessary to establish appropriate monetary values for the worth of reduction of radiation doses to the population. However, it adopted, as an interim measure, "the conservative value of $1,000 per total-body man-rem [and $1,000 per man-thyroid-rem] for these cost-benefit evaluations. Since we realize that the ultimately accepted value may well prove to be less than this, we should leave it open to demonstration in individual cases that a lower figure should be used if the applicant chooses to and can make that demonstration."

Optional Method for Compliance

In September 1975, the Commission amended the regulation to provide an optional method for certain reactors to come in compliance with the guidelines. Under this option, applicants filing for construction permits on or after January 2, 1971, and before June 4, 1976, may comply with the "as low as reasonably achiev-

Control of Radioactive Emissions

Section 307(c) of the Energy Reorganization Act directs the Commission to include in its annual report a description of the NRC activities and findings in a number of areas, including "protecting the public against the hazards of low-level radioactive emissions from licensed nuclear activities and facilities."

During 1975 the Commission—in addition to conducting its day-to-day functions of regulating radioactive material in effluents through ongoing licensing reviews, investigations, inspections, and enforcement actions—concluded a significant rulemaking action designed to keep radiation doses to the public from light-water-cooled nuclear power plants to levels far below values in existing Federal radiation protection guides. (See discussion in this Chapter).

The NRC staff and the statutory Advisory Committee on Reactor Safeguards methodically review the design for the control of radioactive waste of each major nuclear facility proposed for licensing. Changes in the radwaste system design, as well as in other systems important to health and safety that are proposed by the applicant, are required as appropriate.

Rigorous radiological monitoring requirements are imposed on reactor licensees covering all major and potentially significant paths of gaseous and liquid effluents during normal operation. NRC inspectors check the radiological monitoring and waste systems to ensure that they are built as designed and operated to keep releases within regulatory limits, and the Commission is developing a program to corroborate effluent measurements performed by laboratories for facility licensees. Cooperating in this program are the National Bureau of Standards, Energy Research and Development Administration, Environmental Protection Agency, and State agencies. A number of States are collaborating with the NRC in monitoring low-level emissions at the point of release within plants and in the plant environs.

Nuclear power plant licensees are required to inform NRC when any regulatory limit on releases in effluents is exceeded and to take appropriate action. NRC follows up with an investigation, and enforcement if appropriate. During 1975, an instance occurred in which the radioactive material in gaseous effluents from a nuclear power plant exceeded NRC limits on two occasions. Actions were taken to prevent reoccurrence of the operational problems involved, and, while the releases exceeded NRC limits, they did not pose a hazard to the public. On the whole, radioactive material in effluents from nuclear power reactors continued during the year to be small fractions of the limits set forth in NRC regulations. (See discussion under "Environmental Monitoring" in Chapter 6.)

able" regulation without making the newly required cost-benefit analysis, provided their radioactive waste systems and equipment meet the guides on design objectives that were proposed by the regulatory staff of the AEC in February 1974.

The guidelines proposed by the staff in February 1974, and followed in the design of reactors since that time, are based on a different procedure than that provided in the newly approved rule. Rather than a cost-benefit analysis to indicate the need for additional equipment to restrict population exposure, those proposed guidelines featured design objectives that included the amount of radioactive material in reactor effluents per reactor on a site. NRC staff analyses indicate it is unlikely that the cost-benefit analysis would show a need for additions to waste equipment already proposed or installed on the basis of the 1974 guidelines. At the same time, the optional method for those reactors will avoid unproductive expenditures of manpower and increased facility costs that would otherwise result from schedule delays.

The NRC staff is devoting a major effort toward producing several regulatory guides concerning aspects of the new regulation.

FUEL CYCLE ACTIONS

Essentially the same type of environmental review is performed by NRC for major fuel cycle plants as for nuclear power plants. Specialists in the Office of Nuclear Material Safety and Safeguards perform this function.

Major Actions in 1975

During 1975, the staff took the following environmental actions on fuel cycle facilities:

- Draft environmental statement on the Allied-General Nuclear Services Fuel Receiving and Storage Station, Barnwell, S. C.
- Final environmental statement on the Kerr-McGee Co.'s Sequoyah Uranium Hexafluoride Plant near Cresent, Okla.
- Environmental impact appraisal and negative declaration on modifications at the Union Carbide Corp.'s uranium mill at East Gas Hills, Wyo.
- Environmental impact appraisal and negative declaration on license renewal of the Petrotomics Co.'s uranium mill at Shirley Basin, Wyo.
- Environmental impact appraisal and negative declaration on license renewal of Combustion Engineering, Inc.'s Nuclear Fuel Fabrication Facility and Nuclear Laboratories, Windsor, Conn.
- Environmental impact appraisal and negative declaration on license renewal of General Electric Co.'s Uranium Oxide Fuel Fabrication Facility, Wilmington, N.C.

Guidance Furnished to Industry

"Interim Guidelines for Preparing Environmental Information for Nuclear Fuel Cycle Facilities," have been prepared to assist applicants in providing complete information for a thorough appraisal of actual and potential environmental impacts from a nuclear fuel cycle facility and its operations. The guide will be available to industry in 1976.

The NRC staff completed a draft regulation in early 1975 that would require operators of fuel cycle facilities to submit semiannual reports providing data on radioactive effluents and other information needed to estimate resulting radiation doses to the public. Final action on the rule and issuance of guidance for preparation of the reports is expected in 1976. Such reports are already required from nuclear power plant licensees.

Control of Uranium Mill Tailings

An environmental problem in the fuel cycle that is receiving increasing NRC attention is the stabilization of residue left from uranium milling. In the production of uranium from ore at mills, the bulk of the rock is crushed and reduced, after processing, to a fine sandy or silty residue called tailings. The tailings contain the radioactive "daughter" products of the "parent" uranium which is recovered from the ore for subsequent use as nuclear fuel.

In May 1975, the General Accounting Office (GAO) recommended that the Chairman of the NRC "(1) assess the capability and willingness of public health authorities or other State agencies to assume responsibility for and to adequately carry out programs for the long-term monitoring of tailings piles and for correcting any problems in tailings' stabilization and control programs and (2) determine whether additional Federal authority is needed to improve such programs."

In August, NRC reported to the Committees on Government Operations of the U.S. House of Representatives and the Senate the actions taken and planned to implement the GAO recommendations.

NRC and State Responsibilities

NRC has direct regulatory jurisdiction over uranium mills in the States of Utah, Wyoming and South Dakota, currently numbering 10 active facilities. Under the "Agreement State" program (see Chapter 10) whereby qualified States assume part of NRC's regulatory authority, eight other active mills are regulated–including the control of tailings–by the States of Colorado, New Mexico, Texas and Washington.

For a mill licensed by the NRC, an environmental monitoring program must be carried out, and the licensee must establish a program to prevent releases of tailings material and to restrict use of the area. NRC also will require a guarantee such as a surety bond to assure that the commitments will be fulfilled, and is discussing arrangements for the affected States to hold such surety bonds posted by the licensee.

The Agreement States maintain programs for uranium mill tailings stabilization and control that are reviewed for comparability with the NRC programs. In addition, the Conference of Radiation Control Program Directors, representing all 50 States, has formed a task force to develop guidance for States on bonding and perpetual care arrangements.

Federal Authority Question

NRC is reviewing the matter of whether additional authority is needed, as recommended by the GAO. Information used in this review will include licensing reviews of milling operations, including technical feasibility of stabilization programs, and from a Federally-funded study by ERDA, EPA and the affected States of the stabilization and control programs at 21 inactive uranium mill sites. As a part of the latter project, field studies have begun at an abandoned mill site in Salt Lake City. These will include engineering evaluations of the tailings with examination of alternative solutions, cost estimates for remedial action, and extensive radiation measurement to determine the potential for exposure to man.

Petition for Action

In March 1975, NRC received a petition requesting action to (a) prepare a generic environmental statement on uranium mill tailings, (b) cease all licensing actions connected with generating tailings until results of the environmental review are available, and (c) require mill owners to post performance bonds to assure that tailings will be stabilized prior to abandonment. The petition was published for comment and NRC will review the response and take appropriate action in 1976.

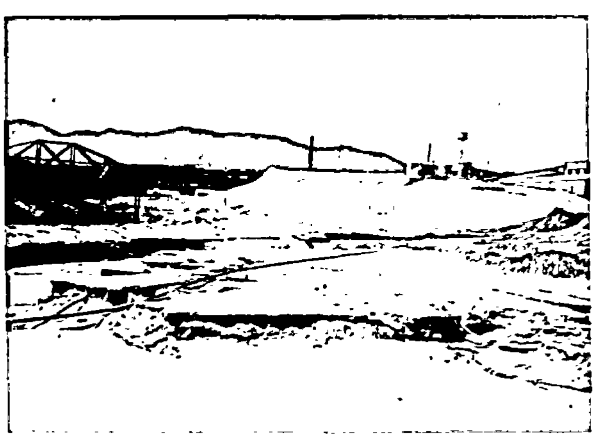
INTERAGENCY COORDINATION

Environmental regulation of nuclear activities requires close interfacing of policies and procedures with State and other Federal agencies, such as the Environmental Protection Agency, the Department of the Interior, and the Council on Environmental Quality. An aspect of growing importance during 1975 was the need to establish better coordination between the NRC and other agencies having overlapping environmental responsibilities, in order to avoid duplication and increase efficiency in both Federal and State actions.

Agreement with EPA

A memorandum of understanding was concluded by the former AEC and the EPA in January 1973, by which AEC would accept EPA's decisions under specified sections of the Federal Water Pollution Control Act (FWPCA) and, where no EPA decisions were involved, give due regard to EPA's views as expressed in comments on the AEC's draft environmental statements. As part of this agreement, AEC had reserved the right to establish certain effluent release standards including limits for byproduct, source and special nuclear materials. Based on these provisions, EPA issued a discharge permit under Section 402 of the FWPCA for the Fort St. Vrain Nuclear Generating Station which did not include limitations or standards for radioactive releases.

Subsequently, environmental groups brought suit against EPA, and in December 1974, the U.S. Court of Appeals for the 10th Circuit ruled, in *Colorado Public Interest Research Group v. EPA,* that EPA is required by the provisions of the FWPCA amendments of 1972 to issue radioactive effluent discharge permits for individual nuclear power plants. While the judgment does not diminish or dilute the NRC's current authority under the Atomic Energy Act of 1954, as amended, it results in concurrent jurisdiction by the EPA and the NRC in regulating and controlling releases



At left is an unstabilized mill tailings pile photographed in 1961, and at right a stabilized pile at the same site in 1970.
The uranium and vanadium mill near Monticello, Utah, was operated from 1942 to 1960, and accumulated approximately 900,000 tons of tailings covering about 40 acres. After shutdown of the mill, the low-level radioactive solid residues dried out and were subject to wind and water erosion, stream silting and radioactive contamination. A stabilization project was begun in August 1961 and included leveling, grading, improving drainage and, finally, fertilizing and seeding for a vegetation cover.

of liquid radioactive effluents from nuclear power reactors and fuel cycle facilities.

Supreme Court review. In June 1975, the U.S. Supreme Court decided to review the Court of Appeals' decision. Oral argument was heard by the Court in early December 1975.

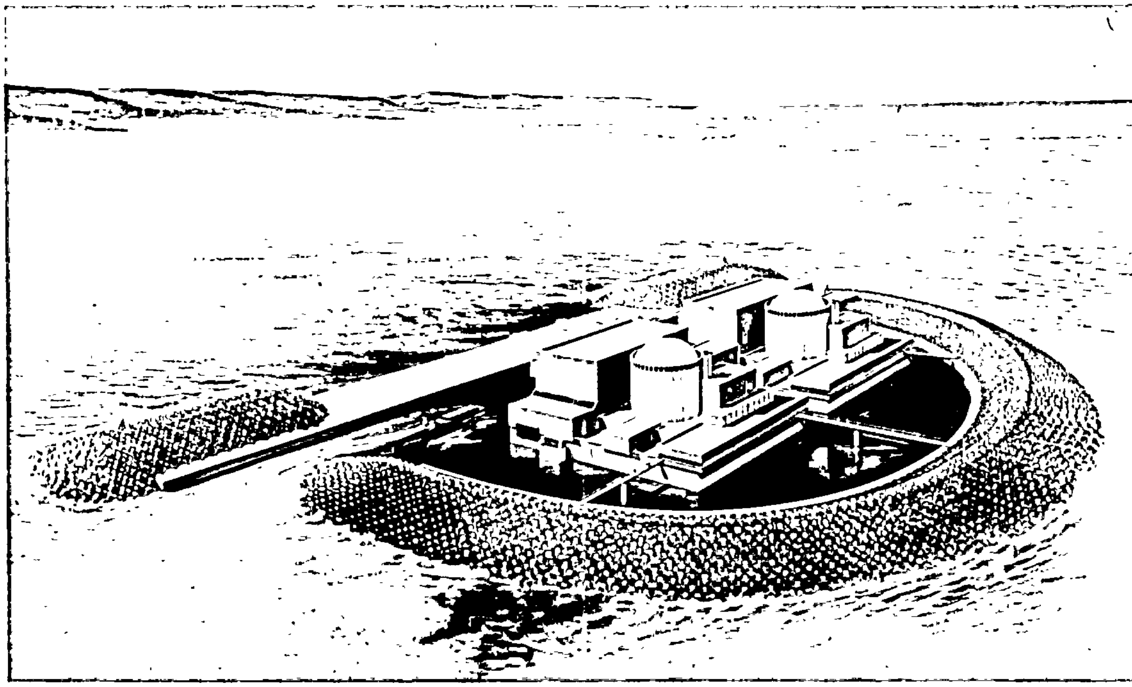
Second Memo of Understanding

Meanwhile, at the suggestion of the Chairman of the Council on Environmental Quality in late 1973 that steps be taken "to make the analysis of the water quality impact of nuclear power plants more effective and more meaningful and, at the same time, reduce the demands for data being placed upon applicants for licenses," the two agencies developed a second memorandum of understanding which was published for comment in November 1974. The proposed agreement was revised substantially to incorporate the majority of comments received from 15 sources, including organizations representing more than 200 electric utilities, the States and Federal agencies. On December 17, 1975, the second memorandum was signed by representatives of NRC and EPA, with the concurrence of the Council on Environmental Quality. It was published in the *Federal Register* and became effective in January 1976.

The main features of the second memorandum of understanding provide that:

- NRC will serve as the "lead agency" for preparation of environmental statements for nuclear power plants and other activities identified in the memorandum, as provided for in the Council on Environmental Quality's "Guidelines for Preparation of Environmental Impact Statements."
- NRC and EPA will work together to identify needed environmental information for early evaluations related to impact from the identified activities on water quality and biota.
- EPA will exercise its best efforts to evaluate impacts on water quality and biota as far as possible in advance of the issurance of NRC's final environmental impact statement for any covered activity.
- EPA will issue to the applicant, where appropriate, a complete Section 402 permit far as possible in advance of authorization by the NRC of any commencement of construction or issuance by NRC of a license or early site approval, whichever is applicable. Permits will be reissued, as appropriate, and any reissued permit, effective at the time the facility commences actual discharge, may require additional limitations and controls based on data gathered during the initial permit or may require additional studies for the purpose of confirming conclusions reached from previous predictive studies. It is expected that the early issuance of Section 402 discharge permits will assure to the maximum extent possible that considerations regarding impacts on water quality and biota will not result in the need for significant changes in plant design or in the costs and benefits of the operation of the facility subsequent to the completion of NRC's environmental review.
- EPA and NRC will consider the feasibility of holding combined or concurrent hearings on EPA's proposed Section 402 permits and NRC's proposed issuance of construction permits or other activities where appropriate.
- The first memorandum of understanding regarding implementation of certain complementary responsibilities under the FWPCA is rescinded.

EPA and NRC have been working together to identify any additional water quality and biological information that may be needed in an applicant's environmental report so that EPA and the States



Artist's drawing of the proposed Atlantic Generating Station which is under active NRC licensing review. The two-unit floating nuclear station would be located approximately 3 miles off the New Jersey coast, north of Atlantic City, where the water is 60 feet deep. The units would be moored within a breakwater for protection from ship collisions and from high waves and winds. Transmission lines from the plant would be placed in a large pipe on the ocean floor between the plant and the shore and, to protect the onshore wetlands, in an underground pipe for approximately 7 miles from the shoreline to a switchyard. The lines would be cooled by oil pumped through the pipe under pressure. Due to the unusual safety and other considerations involved in establishing and operating nuclear plants in an ocean environment, the NRC has mounted an interagency effort in evaluating floating plants.

may arrive at early determinations regarding the impacts on water quality and biota.

Coordination on Floating Plants

A number of government agencies and organizations have statutory responsibility and regulatory authority over activities taking place in the ocean. In recognition of this multi-agency involvement in the licensing of floating nuclear power plants, an Interagency Regulatory Steering Committee was formed to minimize duplicative efforts, clarify policy and provide an efficient regulatory process for industry and the public.

Departments and agencies of the Federal government represented on the Committee are: Coast Guard, Corps of Engineers, Council on Environmental Quality (observer), Department of the Interior, Environmental Protection Agency, Federal Aviation Administration, Federal Energy Administration (observer), Federal Power Commission, National Oceanic and Atmospheric Administration, and the Nuclear Regulatory Commission. The NRC representative serves as chairman of the Committee, which has existed since 1973.

For each of the major phases in the licensing process for a floating nuclear power plant, agencies represented on the Committee have identified the key points at which a required regulatory or coordination function must be undertaken by one or more of them. The principal phases are: construction of the manufacturing facility; manufacture of the floating nuclear power plants; construction of offshore and onshore facilities at the proposed site of the electric generating station; towing of the floating nuclear plant to the operating site and its installation in the breakwater; operation of the nuclear power plant. There is currently an application under review for license to manufacture and install eight floating nuclear power plants.

In July 1975, a report of the Committee was published. Single copies of this report are available without cost upon request to the Director of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. The report is entitled: "Report of the Interagency Regulatory Steering Committee for the Coordination of Federal Regulatory Activities Relative to the Licensing of Floating Nuclear Power Plants."

Work with Corps of Engineers

The NRC and the U.S. Army Corps of Engineers executed a memorandum of understanding in July 1975 regarding the regulation of nuclear power plants situated on navigable waters. Both NRC and the Corps of Engineers have responsibilities for assuring that nuclear power plants on coastal and inland navigable waters and at offshore sites are built and operated safely and with minimum impact on the environment. The new agreement is designed to allow each agency to take full advantage of the capabilities of both agencies and to avoid duplication of regulatory functions.

Covered by the agreement are floating nuclear power plants and associated structures as well as inland plants on navigable waters. In the case of floating plants, associated structures include transmission lines from the plant to a landbased substation, protective breakwater and mooring systems and other supporting facilities.

The principal provision of the agreement emphasizes the "lead agency" concept by providing for a single coordinated NEPA environmental review to be accepted by each agency as a part of its actions leading to the NRC construction permit and the Department of Army permit—both required by an applicant to commence construction of a nuclear power station. The Corps will provide information for those elements of the review falling within its area of expertise such as shoreline changes, siltation and sedimentation, dredging and filling effects and location of structures in navigable waters. The agreement calls for simultaneous issuance of the permits by each agency, unless otherwise denied by either agency. A major duplicative effort has been eliminated and, in doing so, a more closely coordinated review process is expected.

Both agencies will conduct public hearings as required. There is a mandatory NRC public hearing on radiological safety and environmental matters which provides for public participation in connection with each construction permit application. The Corps of Engineers also conducts public hearings to solicit public comments and to keep the public fully informed.

The agreement designates the Director of the NRC's Office of Nuclear Reactor Regulation as the point of contact for correspondence relating to environmental matters between licensees and both agencies. Actions requiring partial or complete shutdown of a nuclear power plant or changes from the design and operating limitations and conditions approved by either agency will be transmitted through the Director of that office. Otherwise, each agency will enforce its regulations or orders and conditions of permits and licenses separately.

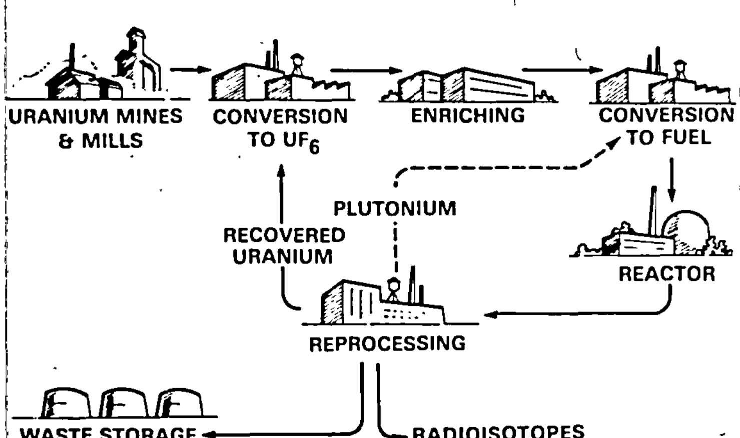
Controlling Nuclear Materials

The Nuclear Fuel Cycle

Except for uranium mining and Government-owned enrichment plants, NRC regulates all steps involved in supplying fuel to nuclear reactors. These steps include the milling of the uranium ore; refinement and conversion of the uranium to a chemical form suitable for enrichment; fabrication of the enriched uranium into fuel elements for nuclear reactors; reprocessing the "spent" fuel elements from the reactors to recover unused fissionable materials for refabrication into new fuel elements; and storing or otherwise disposing of the radioactive wastes.

These fuel-cycle activities and associated facilities are regulated by the Office of Nuclear Material Safety and Safeguards through procedures involving safety, environmental, and safeguards evaluations and licensing actions.

THE NUCLEAR FUEL CYCLE



*The Recycling of Plutonium as a Reactor Fuel (Represented by a Dotted Line) has Not been Approved for Widespread Use in Commercial Light Water Power Reactors

The Office of Inspection and Enforcement conducts periodic plant inspections during construction, startup, and operation. In all of these areas, including transportation of radioactive materials, NRC requires conformance to standards established to protect public health and safety, security, and the environment.

CLOSING THE FUEL CYCLE

Issues related to closing the nuclear fuel cycle after the initial use of nuclear material in the reactor drew increasing regulatory attention during 1975. The most salient concerns were: (1) the need to reach a timely decision on whether plutonium should be recovered from spent uranium fuel elements and be recycled as fuel in light-water-cooled nuclear power reactors, (2) the need for expanded storage facilities for spent fuel which continues to be discharged from operating nuclear power plants, and (3) the need for a comprehensive plan for the management of high-level radioactive wastes.

NUCLEAR FUEL CYCLE FACILITIES

TYPE	OPERATING	PLANNED	ESTIMATED BY YEAR 2000
Uranium Mills*	18	6	69
UF_6 Production Plants	2	1	9
Enriched Uranium Fuel Processing and Fabrication Plants	17	2	53 (combined)
Plutonium Fuel Processing and Fabrication Plants	5	1**	
Fuel Reprocessing Plants	0	3	17 (combined)
Irradiated Fuel Storage Facilities	2	3	
Waste Burial Grounds*	6	0	10
Enrichment Plants	0	4	18
TOTAL	50	20	176

*Some of the Mills and Waste Burial Grounds are Under Agreement State Licenses
**Firm Plans Affected by Pu Recycle Question

Plutonium Recycle Weighed

One of the most critical economic and technological questions faced by the nuclear power industry in planning for future operations is that of under what conditions "plutonium recycle" will be permitted.

During the uranium fissioning process in today's light-water-cooled nuclear power plants, the fissionable element plutonium (as well as highly radioactive fission products which are treated as wastes) is formed within the fuel elements. Much of this plutonium is consumed promptly in the fissioning process and contributes to the generation of useful power. About one-third of the power produced in a uranium-fueled light water reactor comes from this *in situ* use of plutonium. The unconsumed plutonium and uranium remaining in the spent fuel can be separated from the fission products and recovered at reprocessing plants. While the plutonium thus recovered has the potential to be used with uranium to manufacture mixed oxide fuel for these same light-water-cooled nuclear power plants, widescale recycling of plutonium has not yet been approved. Resolution of this question was initiated by the former AEC and remains to be concluded by the Nuclear Regulatory Commission. It involves important health and safety, environmental and safeguards issues.

Shortly after formation of the NRC, the Commissioners accorded high priority to consideration of all the factors involved and, during 1975, adopted a definitive plan by which a final decision could be made, possibly by early 1977, on the widescale use of recycled plutonium in mixed oxide fuel for light water reactors (LWRs).

Toward a Final Decision

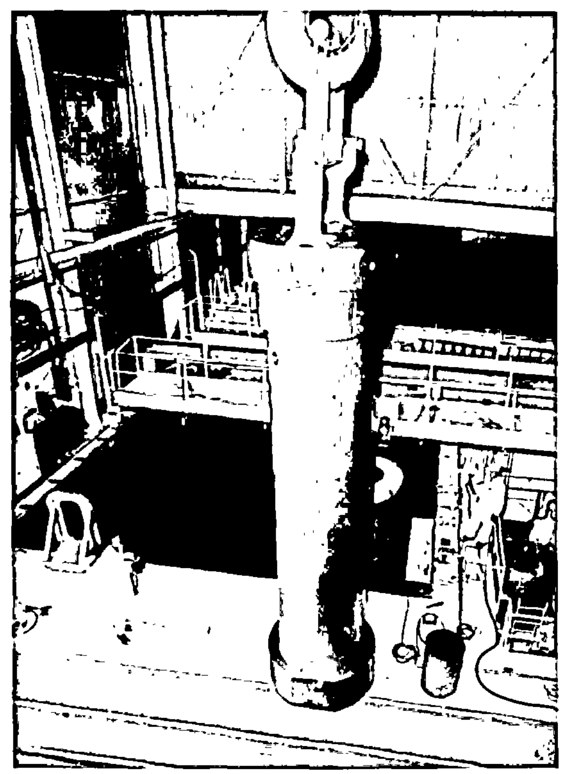
On November 12, 1975, NRC announced in definitive form the procedures and steps to be taken toward reaching a final decision on plutonium recycle using mixed oxide fuel in LWRs. In addition, the Commission outlined a policy for licensing the interim use of mixed oxide fuel in LWRs and the interim licensing of certain supporting fuel cycle facilities. The process leading up to the November announcement included the

August 1974 publication for public comment of a draft generic environmental statement, prepared by the regulatory staff of the former AEC in accordance with provisions of the National Environmental Policy Act. The technical report (WASH-1327, four volumes totaling 1,100 pages) was entitled "Generic Environmental Statement on Use of Mixed Oxide Fuel in Light Water Reactors," and became known throughout the industry by the acronym, GESMO. The detailed analyses in the report led to the following AEC staff conclusions: (1) the recycling of plutonium in LWR nuclear fuel could result in a slight decrease in the environmental effects of the total fuel cycle; (2) there would be no significant change in factors affecting the safety of nuclear plants and operations; (3) safeguards considerations need not delay the approval of plutonium recycle since there would be little change in plutonium production or utilization for several years and, therefore, the present safeguards could continue to serve adequately until upgrading plans were implemented, which would be accomplished before the widescale use of plutonium recycle; and (4) plutonium recycle would result in decreased natural resource requirements for meeting U.S. energy demands.

On May 8, 1975, NRC announced its provisional view that a decision on plutonium recycle should await preparation of a cost-benefit evaluation of alternative safeguards programs to protect the public from the consequences of a possible diversion of special nuclear materials and sabotage of facilities, in accord with a recommendation of the President's Council on Environmental Quality. More than 200 organizations and individuals responded to requests for comment, and their judgements were carefully considered by the Commission and NRC senior staff in developing the procedures announced on November 12.

There are six basic steps set forth in the definitive plan, designed to make a final decision possible as early as 1977, some 18 months earlier than was projected in the provisional plan. The six steps are:

- A cost-benefit analysis of alternative safeguards programs for the widescale use of mixed oxide will be prepared on an expedited schedule. A draft of the analysis is expected to be completed early in 1976, as is a partial final statement on health and safety and environmental matters. The final portion of the impact statement, expected in mid-1976, will include the cost-benefit analyses on safeguards and the overall cost-benefit balance on the widescale use of mixed oxide fuel for light water nuclear power plants.
- Proposed rules for the use of mixed oxide fuel will be published and public



A shipping cask, containing about 2,000 lbs. of spent uranium fuel is about to be lowered by crane into the storage pool at the Nuclear Fuel Services fuel reprocessing plant at West Valley, N.Y. Once under water, the shipping cask will be removed, and the highly radioactive spent fuel assemblies will remain covered with cooling water, until removed for reprocessing or other disposal.

comment solicited as final portions of the environmental statement are issued. The Commission expects to issue the final rules at the time of its decision on the widescale use of mixed oxide fuel in LWRs.

- The public will have the opportunity to participate in the decision-making process not only by submitting written comments on the draft supplement environmental impact statement on safeguards issues and proposed rules, but also by taking part in the public hearings which will be held on the final impact statement and on any rules to be implemented. These essentially legislative-type hearings will begin as soon as practicable after issuance of the non-safeguards portion of the final statement early in 1976 and will continue after the issuance of the final safeguards supplement in mid-1976. These may be followed by adjudicatory-type hearings on particular issues, if the need for such is demonstrated to the Commission. If no such need arises, a final decision could be reached by early 1977.

- The NRC staff will continue reviewing applications already submitted for mixed oxide fuel related activities and will commence review of any new applications received.

- Eligibility criteria have been established for considering the interim licensing of such plutonium recycle related operations as fuel reprocessing and mixed oxide fuel fabrication. The Commission noted that very little mixed oxide fuel is being made in the United States and that, at present, there is no operating reprocessing facility for commercial fuel in the country; thus, it is not likely that there could be any substantial use of mixed oxide fuel before the early to mid-1980s, regardless of licensing and construction activities. The Commission stated its confidence that the current safeguards framework is adequate for existing plants and for interim licensing of facilities to chemically separate spent fuel and to convert recovered uranium to a usable form. Upgraded safeguards requirements may be imposed in any interim licensing of facilities which convert recovered plutonium to a usable form and which fabricate mixed oxide fuel. Interim safeguard rules for these activities are expected to be published when the Commission publishes the draft safeguards supplement in early 1976; additional rules for use pending a final decision on widescale use of mixed oxides may be published when the final portion of the statement appears in mid-1976, after consideration of all public comment on the draft statement supplement on safeguards and on the proposed rules and safeguards.

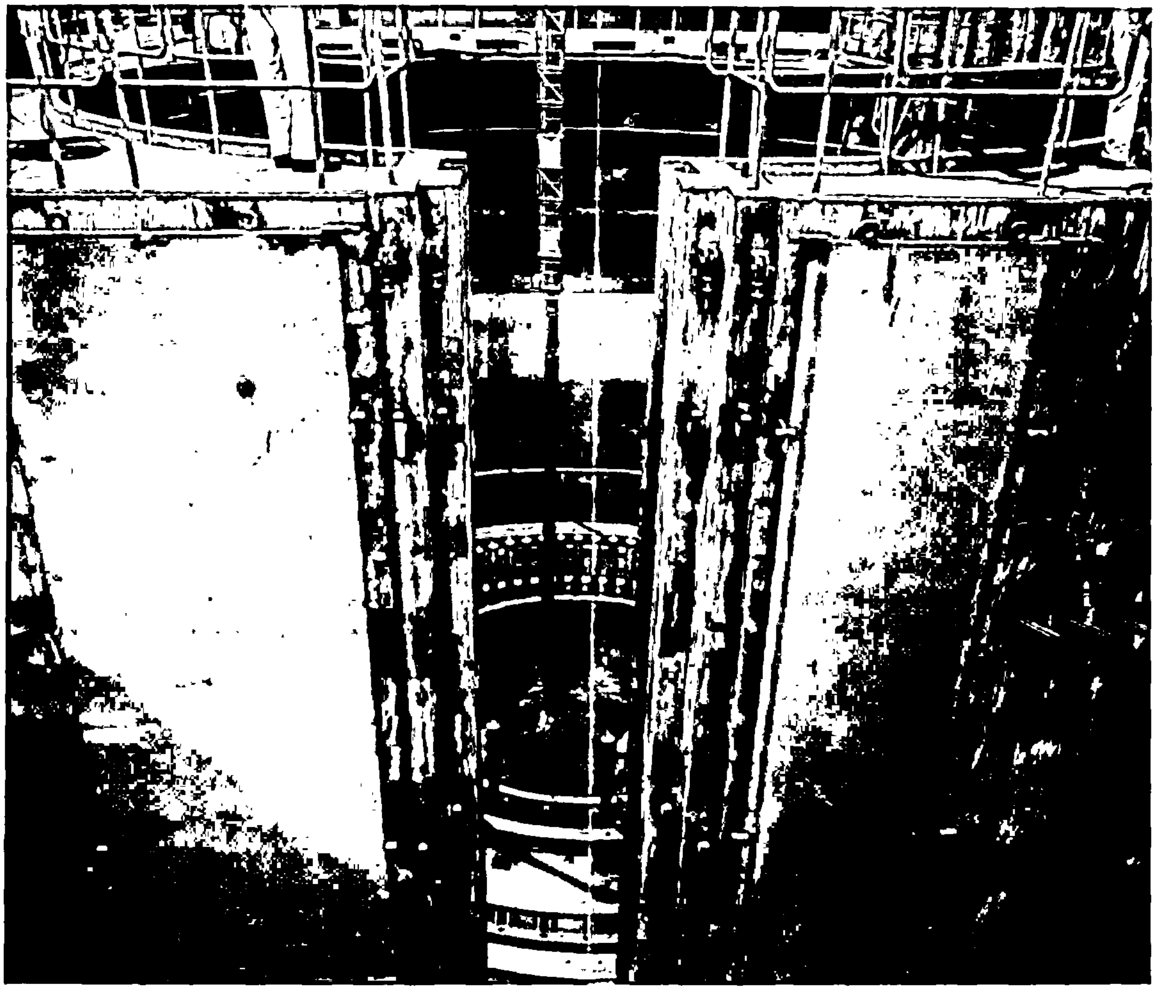
- Operating licenses and amendments to operating licenses may be issued which authorize the interim use of mixed oxide fuel in light water reactors; such use would fall short of "widescale" use because of the limited mixed oxide fuel fabrication capacity available. The Commission believes that such use would produce useful additional technical and economic data regarding this fuel.

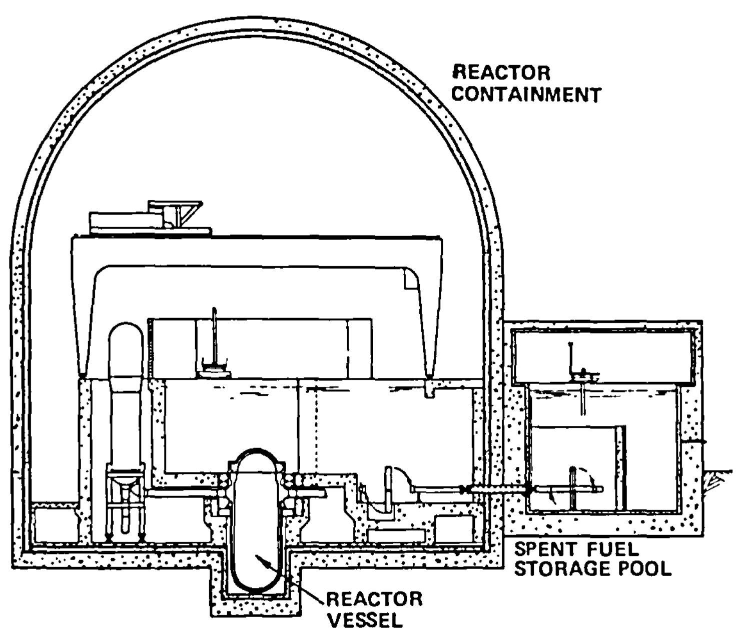
The Commission, in reaching its decision on safeguards, will have the benefit of the results of a special study on safeguards needs for recycled plutonium as well as two additional one-year studies related to safeguards which were mandated by the Energy Reorganization Act (see Chapter 5).

Action on Spent Fuel Storage

In the meantime, no commercial irradiated fuel reprocessing facility has been in operation since early 1972, and the spent fuel removed periodically from power reactors has filled many of the available storage facilities. In September, the Commission took steps concerning this situation.

Normally, when the highly radioactive used-fuel elements are removed from a nuclear power reactor, they are stored temporarily in a pool at the plant site, to permit some radioactive decay before the





Fuel Loading and Unloading. Above, the first new fuel bundle is seen through the "keyhole" about to be lowered into the reactor core. During refueling, the area over the reactor and the 40-foot-deep fuel pool (illustrated in drawing at right) are filled with water to cool and shield the highly radioactive fuel as it is withdrawn from the reactor core. The irradiated fuel is moved through the "keyhole" under water to spent fuel storage pool area. Much of the radioactivity in the fuel will decay during this temporary storage. When the fuel is shipped offsite, it will be placed inside a massive shipping cask with heavy shielding and self-contained cooling.

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elements are transported in shielded casks to a reprocessing plant. The reprocessing plant maintains storage capacity to receive spent fuel from a number of power reactor customers while working off the inventory.

Because of the continuing buildup of spent fuel in storage pools, owners of the three fuel reprocessing plants (none operating, except for storage)—located in New York, Illinois and South Carolina—have applied to NRC for licenses either to expand storage facilities already in operation or, in one case, to operate the storage basin at the site before completion of the entire plant. In addition, during 1975, four utilities applied to NRC for approval of storage capacity increases at their individual nuclear power plants—Point Beach in Wisconsin, Indian Point Unit 2 in New York, LaCrosse in Wisconsin, and Maine Yankee.

In September the Commission announced NRC would prepare a generic environmental impact statement on storage of spent fuel from light-water reactors, and would continue to act on storage license applications while the statement is being prepared. In deciding to continue to act on such applications, the Commission noted that:

(1) By mid-1978, upwards of 10 reactors with an aggregate capacity of some 6,000 MWe are expected to fill their storage pools to present capacity and, without additional spent fuel storage space, be forced to shut down with an adverse impact on the utilities' ability to meet energy requirements and possible increased electricity costs to consumers.

(2) It is likely that an individual licensing action would be justified independently and would not constitute a commitment of resources that would foreclose alternative action.

(3) It is likely that possible environmental impact from such licensing action would be addressed adequately during the individual proceeding without overlooking any cumulative impacts; and any technical issues that may arise during licensing review of an individual application could be resolved as part of that review.

Storage of spent fuel is regarded as an interim solution. On a long-term basis, spent fuel must either be reprocessed to recover the valuable uranium and plutonium for recycling in new reactor fuel, or managed as radioactive waste.

Fuel Reprocessing Plants

Currently, there are three commercial spent fuel reprocessing facilities in various stages of development in the U.S.:

- The West Valley, N.Y., reprocessing plant owned by Nuclear Fuel Services, Inc., the first and only commercial facility of its type to be placed in operation, has been shut down since early 1972 for major modifications, including an increase in capacity from 300 to 750 metric tons of irradiated uranium per year (capable of supporting 20 1,000-MWe power reactors). The NRC safety and environmental reviews for these changes continued during the year, with public hearings projected for 1977. In a separate licensing action, NFS requested authority to increase the capacity of its fuel storage pool.
- Licensing reviews continued on Allied-General Nuclear Services' reprocessing plant at Barnwell, S.C., on which construction was over 90 percent complete at the end of the fiscal year. A public hearing on environmental impact, being conducted by an Atomic Safety and Licensing Board, was expected to continue into calendar year 1976. A separate hearing will be held on Allied-General's request for approval to use

its spent fuel receipt and storage station prior to final action on the facility operating license application. The Barnwell plant is designed to reprocess 1,500 metric tons of irradiated uranium per year (capable of supporting 50 1,000-MWe power reactors).

- General Electric Co., which completed a reprocessing plant in 1974 at Morris, Ill., designed to handle 300 metric tons of spent fuel annually, decided not to place the facility in operation due to technical difficulties; the receiving and storage portion has been utilized, however, and GE requested authorization to substantially expand storage capacity. Licensing action on this request was completed in December 1975.

These heavily shielded facilities were designed to separate the highly radioactive fission products from spent fuel, purify the recovered uranium and plutonium for reuse in reactors, and concentrate the radioactive wastes for storage onsite pending solidification and transfer to the government for disposition. As with nuclear power reactors, the NRC conducts exhaustive safety and environmental reviews and will maintain surveillance over operations to assure radiological protection of workers, the public, and the environment.

Nuclear Waste Management

Wastes generated in civilian uses of nuclear fission energy range from slightly

Main buildings of Barnwell Nuclear Fuel Plant under construction at Barnwell, S.C. In the fuel receiving and storage station at upper left, spent reactor fuel would be stored under water prior to reprocessing. The storage pool is adjacent to the main reprocessing building (center) so that the fuel could be transferred under water to the processing area where it would be chopped into small pieces and then dissolved. By radiochemical processing inside heavily shielded "hot cells," uranium and plutonium can be separated from the highly radioactive fission products. In building at lower right, purified uranium would be converted to uranium hexafluoride, then shipped to another site for enrichment and later fabrication into new fuel pellets. The roofless building at upper right is the high-level liquid waste storage equipment building. In stainless steel tanks under this building, the highly radioactive fission product wastes from the reprocessing would be stored temporarily until converted to the proper form and sealed into special containers for permanent storage at another location.

contaminated trash to wastes with very high-level radioactive content produced in the reprocessing of fuel elements which have had a long period of use in nuclear power reactors.

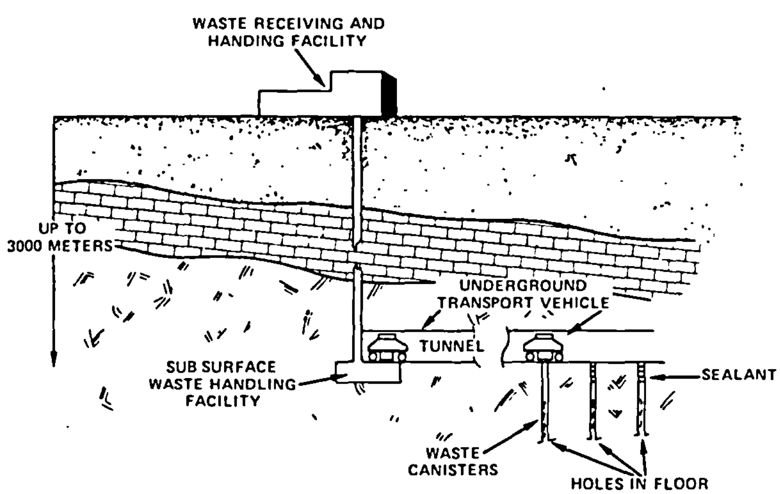
Where and how to store and dispose of highly radioactive wastes produced from nuclear reactor spent fuel reprocessing operations has been a matter of national concern for some time. A research and development program to resolve this problem, formerly carried out by the AEC, is now the responsibility of ERDA.

The Energy Reorganization Act of 1974 assigned NRC the responsibility to license the safe storage and/or disposal of high-level radioactive wastes from the commercial reprocessing industry, whether stored at the reprocessor's facilities or at a Federal repository. To meet this, as well as other nuclear waste responsibilities conferred by the Atomic Energy Act, the Commission established a Waste Management Branch in the Office of Nuclear Material Safety and Safeguards. This Branch is responsible for the necessary safety analyses and licensing activities as well as for the development of a comprehensive waste management policy for the Commission.

During the year, NRC undertook preparation of a broad program plan for nuclear waste regulation and management concerning all types of wastes ranging from tailings at uranium mills to the decontamination of nuclear facilities upon decommissioning at the end of their useful lives. Scope of the program will include standards development, backup research, and the licensing actions required to protect public health and safety in all aspects of the handling, treatment, shipping, storage and disposal of nuclear wastes.

High-level wastes. Under current NRC regulations, high-level liquid radioactive wastes resulting from fuel reprocessing operations must be solidified within five years of generation and shipped to ERDA within another five years for storage or disposal. Of the many proposals for disposition of these wastes, ERDA has concentrated studies on an interim storage technique (called a Retrievable Surface Storage Facility) and a geological disposal system in bedded salt, while studying other promising methods.

NRC is charged by law with making licensing decisions on all storage and disposal facilities. The Commission is developing detailed standards and performance criteria for high-level waste disposal to help guide ERDA's waste management research and development program, while providing flexibility to include any additional options that may be developed as the program progresses.



One concept for permanent disposal of high level radioactive waste is emplacement in a mined cavity after the waste is solidified. All handling of the material below the earth's surface would be by remote control. Initially, emplacement would be in a retrievable fashion. Under the Energy Reorganization Act, proposals for permanent disposal will be developed by the Energy Research and Development Administration, and reviewed for safety and licensing by the NRC.

Waste burial facilities. Low-level radioactive wastes are generally disposed of at commercial shallow burial grounds located in six states. One of these is licensed by NRC, two by Agreement States, and three by both NRC and Agreement States.

In September 1974, AEC proposed a new rule to prohibit commercial shallow land burial of transuranium elements (those with atomic numbers above that of uranium, such as plutonium). Because of the increased quantities of such wastes expected to be generated, the long half-life of transuranics and their high radiotoxicity, it was considered that such wastes in the future should be stored and disposed of at Government-owned facilities. The regulation would require that these wastes be solidified (if liquid), packaged and transferred to ERDA as soon as practicable, but no later than five years after their generation. At the end of 1975, NRC was evaluating public comments received on the proposed rule. Some delay in this rulemaking action has resulted from the withdrawal by ERDA of an environmental impact statement covering management of these wastes and the preparation of the broader NRC program plan.

PROCESSING FACILITIES

At the "front end" of the nuclear fuel cycle, a number of processing steps are naturally required from the time the uranium ore is mined to the point where it is ready for use in the reactor. With the exception of mining and enrichment (see below), each phase of the process is under the regulatory authority of the NRC in its overall responsibility to protect the public and preserve the environment in activities related to the non-military uses of nuclear energy.

Uranium Mining and Milling

The uranium in the ore produced by mining is extracted and concentrated in milling operations. NRC has regulatory jurisdiction over these operations in three States (Utah, Wyoming and South Dakota) which, at the end of the fiscal year, included 10 mills. ("Agreement States"—those which have assumed NRC's regulatory authority over certain nuclear materials by agreement with the Commission—maintained jurisdiction over an additional eight uranium mills. (See "Agreement State Programs" discussion in Chapter 10.)

A principal concern in regulating milling operations is stabilization of large piles of slightly radioactive tailings (crushed rock) left after the uranium has been removed. NRC is working with mill operators and the States to assure long-term control of this problem. (See Chapter 3, "Protecting the Environment.")

During 1975, an application was received from the Rocky Mountain Energy Co., for a license to construct and operate a new 1,000 ton-per-day uranium mining-milling complex in Converse County, Wyoming. Existing milling licenses held by Exxon Co. and Utah International Co. were amended to authorize increased capacities.

Continuing interest is being shown in "solution mining" of uranium—dissolving the ore in place and pumping the mineral solution to the surface for extraction of the uranium. Two licenses were issued by NRC during the year authorizing research and development activities by leaching at sites in Wyoming. The Wyoming Mineral Corp. (a subsidiary of Westinghouse Electric Corp.), which explored this method in 1974, has started operation of a plant-scale solution mining facility in Texas and has announced plans to apply for a license to build a full-scale facility in Wyoming.

Uranium Hexafluoride Facilities

From the milling operation, the uranium ore concentrates move to a facility for refinement and conversion to uranium hexafluoride (UF_6), a volatile compound of uranium and fluorine which is the chemical form used for enrichment in the gaseous diffusion process (see below). There are two such commercial facilities in the U.S.—Allied Chemical in Illinois, and Kerr-McGee in Oklahoma.

During the year, Kerr-McGee Nuclear Corp. requested licensing authority to increase the capacity of its UF_6 plant at Sequoyah, Oklahoma, from 5,000 to 10,000 tons of uranium per year. After a thorough review, approval for preliminary construction work on the facilities was granted, and the licensee anticipates full operation at the increased capacity in 1977.

Uranium Enrichment Activities

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 by industry as a commercial enterprise. Three ERDA-owned gaseous diffusion plants, originally constructed for national defense purposes, constitute the entire U.S. enriching capacity and are not regulated by NRC. However, in view of the Government's program of encouraging commercial undertaking of enrichment plants, the Commission has stepped up preparations for evaluating any license applications that may be received.

Guides on the format and content of safety analysis and environmental reports and on special nuclear material accountability requirements have been published to assist potential applicants for enrichment ment facility licenses. In addition, a description of the information required for antitrust review has been developed and made available to all interested parties. A task force has been established under the direction of a Management Overview Committee to prepare for licensing of enrichment facilities, to identify and propose solutions to licensing problems in this area area, and to meet as appropriate with applicants to discuss licensing application requirements.

Fuel Fabrication Plants

The final steps in producing fuel for nuclear power reactors are the conversion of the enriched uranium hexafluoride to uranium dioxide (UO_2) and the processing of the UO_2 into pellets which are enclosed in long, pencil-like tubular rods manufactured of zirconium. These steps are generally performed in the same facilities that fabricate the finished fuel assemblies. Currently, there are five firms actively engaged in the processing and fabrication of UO_2 fuel for nuclear power reactors.

Significant fuel fabrication licensing actions in fiscal year 1975 included:

- Full-term operating license (five years) issued in July to Exxon Nuclear Co. for its uranium fuel fabrication plant and plutonium mixed oxide laboratory at Richland, Wash., and subsequent amendment authorizing doubling the capacity of uranium fuel fabrication. Separate environmental impact statements were issued in connection with the uranium and mixed oxide fuel operations.
- Babcock & Wilcox Co. was authorized, in May, to use a new high-capacity fuel pellet line at its Commercial Nuclear Fuels facility at Lynchburg, Va.
- General Atomic Co. was permitted to terminate its license covering light-water reactor fuel fabrication at New Haven, Conn., and plutonium mixed oxide fabrication research at Pawling, N.Y. The license was terminated after decontamination of the facilities in accordance with NRC guidelines.

Licensing and Transport

Outside the nuclear fuel cycle, there were, at the end of 1975, approximately 19,000 nuclear materials licenses in effect in the United States, principally for the use of radioisotopes (reactor byproducts) in the medical, industrial and academic fields. More than half of these licenses are now administered by 25 States under regulatory agreements with NRC (see Chapter 10). NRC still holds jurisdiction over some 8,500 materials licenses, and the Office of Nuclear Material Safety and Safeguards processes new applications, amendments, and other materials licensing actions totaling over 8,000 annually.

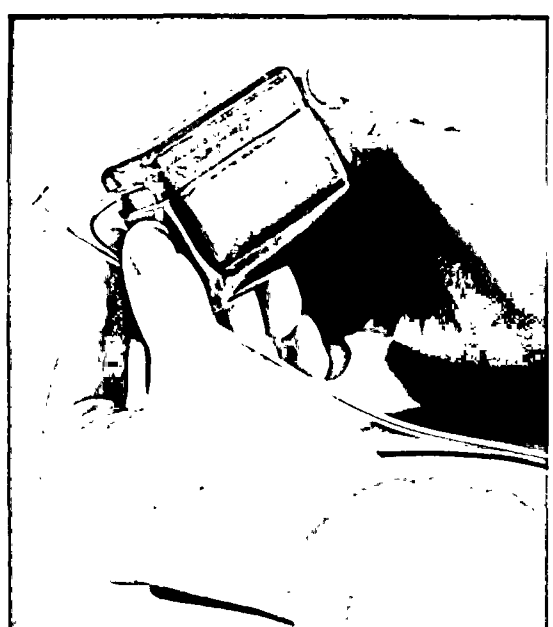
In accord with action initiated last year by the AEC, the NRC discontinued one phase of regulation by which it evaluated the safety and efficacy of radioactive drugs with respect to patients. This responsibility was assumed by the Food and Drug Administration; however, NRC continues to license such uses as they pertain to the radiation safety of the users and the public. NRC also licenses medical devices containing sealed sources of radioisotopes which are not drugs.

Nuclear-Powered Pacemakers

For several years, hospitals have been authorized to implant nuclear-powered cardiac pacemakers for investigational use to determine if they are safe and reliable, and if the benefits to the patients outweigh the risks to the public. Interest has grown, since the nuclear "batteries" using plutonium-238 in pacemakers are expected to last over 10 years before requiring surgical replacement, as compared to useful lives of two to six years for conventional chemical batteries commonly used in these devices. More than 2,000 nuclear pacemakers, produced by four manufacturers, have been licensed by NRC and the Agreement States for investigational implantations. Guides have been issued to set forth requirements for a standard protocol to be followed by all clinical investigators and for licensing hospitals to participate.

A draft generic environmental statement on the wide-scale use of plutonium-powered cardiac pacemakers was published for comment in January 1975, by the Regulatory staff of the AEC, just prior to creation of NRC. The draft statement concluded that the benefits to be derived are substantially greater than the risks to the environment, and that such usage should be authorized, subject to the following conditions:

- Requirements for accountability, recovery and disposal of plutonium sources contained in pacemakers would be imposed by license conditions or by regulations.
- Investigational implantations would continue to be followed and reported pursuant to the protocols established in order that the Commission and the manufacturer would be promptly informed if an unexpected mode of wearout or failure should occur.



Nuclear-powered cardiac pacemaker.

At yearend, NRC was preparing a final environmental statement taking into account the comments received, both favorable and unfavorable. Meanwhile, nuclear-powered pacemakers continued to be licensed in limited numbers for use under closely controlled conditions.

SAFETY REVIEWS OF SHIPMENTS

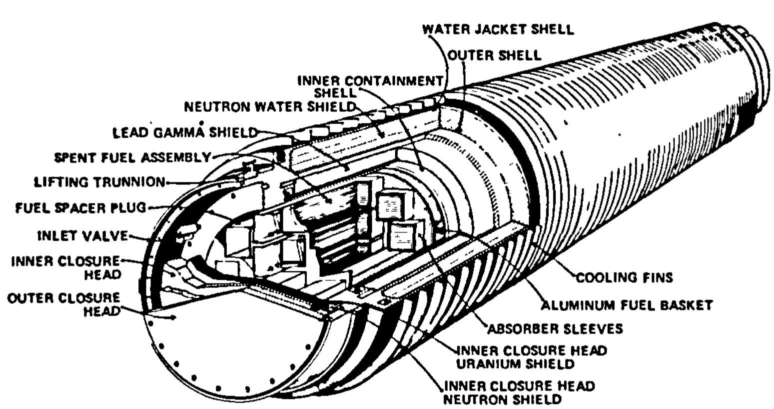
Over the past 30 years, millions of packages of radioactive material have been shipped within the United States with an exceptionally favorable safety record as compared to general experience in the transportation of hazardous materials. During this period, packages of nuclear materials were involved in some 400 reportable accidents or incidents in all modes of transportation. None resulted in serious injury or death attributable to radiation exposure.

In view of current and anticipated increases in shipments of nuclear materials, extensive studies have been undertaken in this field by NRC, other Federal agencies and the States (see Chapter 10) to assure safety and the lowest levels of exposure to the public that are reasonably achievable.

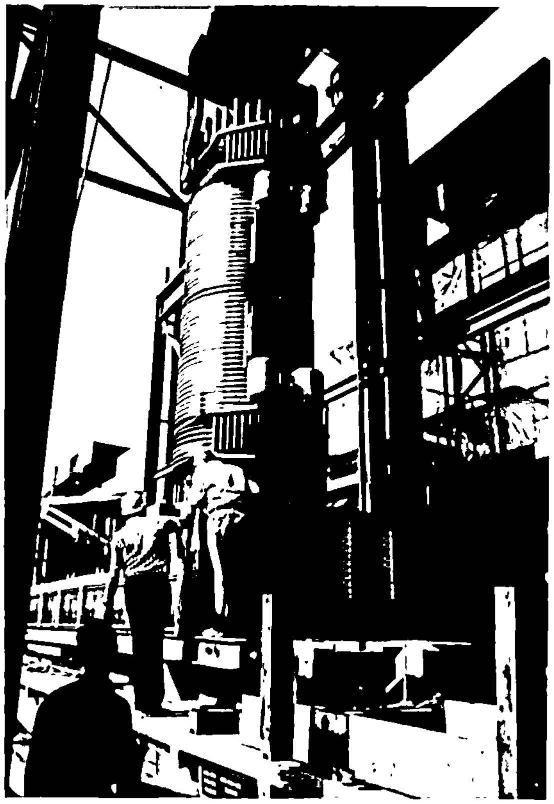
Regulation of commercial nuclear material shipments is conducted in part by NRC, the Department of Transportation, the U.S. Postal Service, and the States. NRC regulations apply to its licensees and generally specify procedures and standards for packages and shipments. (Special requirements to safeguard fissionable materials in transit are described in Chapter 5.) DOT regulates certain types of packaging, labeling and conditions of carriage. Since DOT and NRC jurisdictions overlap in providing for safety in shipment of nuclear materials in interstate and foreign commerce, by air, and by water, the agencies operate under a memorandum of understanding to avoid duplication and provide consistent, comprehensive and effective regulation. The Postal Service regulates shipments of nuclear materials by mail, and the States have regulatory authority over intrastate transport of nuclear materials not regulated by the Federal agencies.

Package Safety Reviews

The NRC-DOT agreement provides that DOT has responsibility for "Type A" packages (those for small quantities of materials which would not pose a significant hazard if released), and NRC has the review and approval authority for all "Type B" package designs (those for kinds and quantities of nuclear materials which must be safely retained in their containers under normal and accident conditions), including those for fissile and for large quantities of nuclear



The proposed NLI-10/24 irradiated fuel shipping cask, currently under review by the NRC. When fully loaded with 10 pressurized water reactor or 24 boiling water reactor fuel assemblies, it would weigh about 100 tons. It is designed with lead and water shields for gamma ray and neutron radiation protection. The fuel elements are shipped "dry" with helium used as the coolant.



This General Electric Co. IF-300 spent fuel shipping cask, being loaded onto a flatbed railroad car, weighs 70 tons when loaded with 7 fuel assemblies from a pressurized water reactor or 18 assemblies from a boiling water reactor. The cask, which uses depleted uranium and water as radiation shielding material, will be secured to a 17.5-ton holding platform which is secured to the railroad car. The fuel elements are shipped "wet" with water used as a coolant.

materials, which require prior regulatory approval.

During the period January through June 1975, the technical staff approved about 50 package designs and registered some 100 users for the use of previously approved containers for nuclear materials.

Standards and tests required of shipping containers evaluated by NRC are rigorous: proposed containers must be designed and built to withstand severe stress, shock and fire. High speed collision, impact, heat, cold, puncture/penetration, explosion, fire, and water tests are used to ensure that radiation shielding and containment remain effective even under severe accident conditions.

During the year, the NRC staff undertook safety review of a design by NL Industries, Inc., of a multi-assembly spent nuclear fuel cask. Weighing almost 100 tons, it would be the largest such cask to be used in the United States, and would necessarily be shipped by rail. The cask would be capable of transporting 10 pressurized water reactor fuel assemblies or 24 boiling water reactor fuel assemblies. Completion of the review is expected in fiscal year 1976.

AIR SHIPMENTS STUDIED

In May, as a part of a review of all licensing regulations and procedures that existed at the time the agency was established in January, the Commission announced initiation of a public rulemaking proceeding concerning the air transport of all nuclear materials, including plutonium and enriched uranium. The proceeding will deal with health and safety aspects, including packaging, and the measures taken to protect materials from theft and diversion.

While no accidents involving civilian air or other transportation of plutonium had occurred in the commercial nuclear industry, concern was expressed during the year by some public officials and others over the safety and security of such shipments over highly populated areas.

In the rulemaking notice, the Commission invited comments on whether air shipments of radioactive materials should be suspended, the relative risk of such transport compared to other transportation modes, and whether and what improvements in regulatory requirements for air shipments are needed. NRC will prepare a generic environmental impact statement on air transport of all radioactive materials which will also consider relative costs and benefits of alternative modes of transportation as required by the National Environmental Policy Act. Associated research into the accident-resistant properties of plutonium shipping packages is being carried out, and data are being collected on present shipping practices by Battelle Pacific Northwest Laboratories, through questionnaires to nuclear material shippers throughout the country.

Plutonium Air Shipment Restricted

In August, Public Law 94-79 was enacted. Among other things, it prohibits air transport of plutonium (except that contained in medical devices for individual human applications) until NRC has certified to the Joint Committee on Atomic Energy that a container has been developed and tested "which will not rupture under crash and blast-testing equivalent to the crash and explosion of a high-flying aircraft."

NRC's Office of Nuclear Material Safety and Safeguards subsequently directed all licensees authorized to possess, import, or export plutonium to suspend air shipments of plutonium except when contained in medical devices. The Agreement States took similar action with respect to their licensees.

On May 5, 1975, the State of New York filed suit against NRC and six other Federal agencies seeking a preliminary injunction against air transportation of special nuclear materials. On September 9, the Federal District Court in New York City denied the request (See "Judicial Review" in Chapter 12).

Plutonium Packages Tested

Meanwhile, tests of packages for plutonium oxide are being carried out for NRC by Sandia Laboratories in Albuquerque, N. Mex., results of which will be factored into the evaluation of risk associated with the various transportation modes.

In July, three representative packages were tested to the standards used by the Federal Aviation Administration to qualify aircraft flight recorders. FAA records of flight recorders in more than 500 aircraft incidents and accidents show that they have performed their intended function in more than 90 percent of the total cases. Two of the plutonium packages met the same survivability tests as aircraft flight recorders, and the third evidenced a small amount of leakage during the last test of the series, a 40-hour water immersion test. NRC plans to have additional tests conducted during fiscal year 1976.

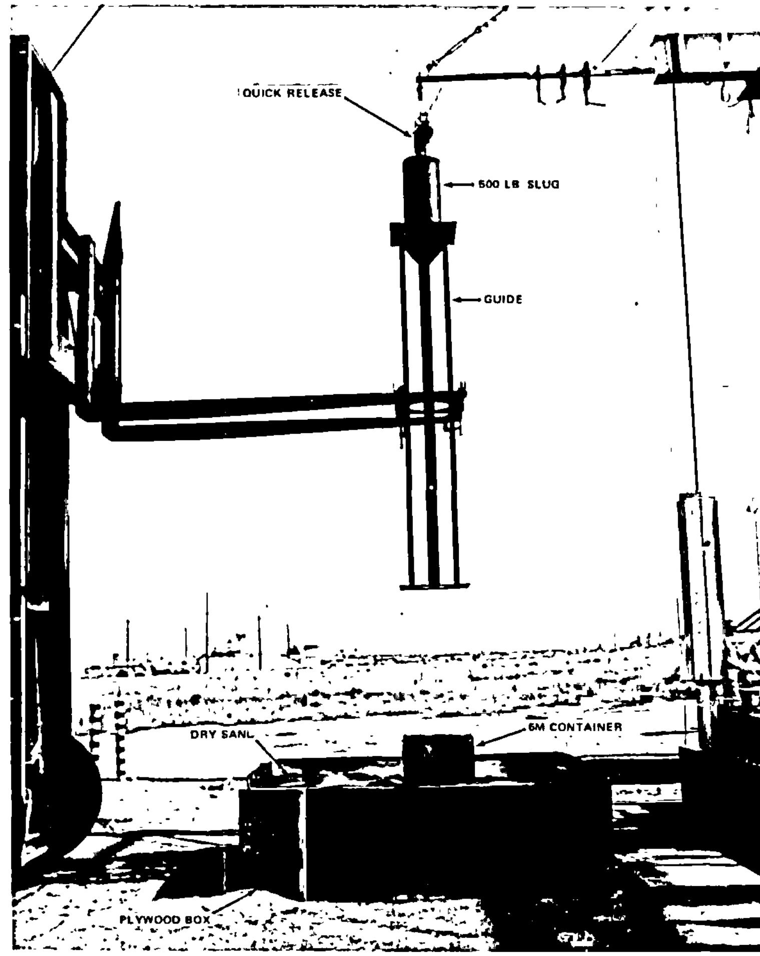
Airline Flight Attendants Study

A recent study, conducted jointly by the NRC staff and two flight attendent unions, measured actual radiation exposures received by about 100 airline flight attendants on passenger aircraft carrying packages of radioactive material. The data, which became available in March 1975, showed that radiation from the packages produced no significant addition to exposure from the combination of cosmic radiation at the altitude most commercial aircraft fly and natural background radiation at ground level. (Background radiation exposures typically range from 100 to 125 millirems a year.) As a spinoff, the data showed that cosmic ray exposures received by flight attendants at cruising altitudes range from 100 to 150 millirems in a year. The average exposure from packages was estimated to be three millirems in a year.

Airport Cargo Handlers Study

Radiation exposures received by airport cargo handlers from handling packages of radioactive material were measured at six airports —Boston, Chicago (O'Hare), Los Angeles, New York (LaGuardia), San Francisco, and St. Louis—that ship the greatest number of radioactive material packages on passenger flights. The results

Several designs of shipping containers for plutonium oxide were tested for NRC by Sandia Laboratories to determine their leak-tightness when subjected to the same test requirements used by FAA for flight recorders. The packages were subjected to severe impact, penetration, static crush, exposure to heat, and immersion in water. At right is the setup for the penetration resistance test on the side of one container; the test was also run on the top and bottom of the container. Although the outer shell of the container (shown below) was severely damaged in the series of tests, there was no damage to the inner steel vessel which would have caused release of contents.



Radiation levels at locations near a group of packages at the St. Louis Airport are measured by Washington University personnel in an NRC-supported study of radiation exposures received by airport cargo handlers. Similar studies were conducted at five other airports.

indicate that, of the most highly exposed cargo handlers, three-fourths receive minimal radiation exposures of less than 100 millirems in a year, and that no one cargo handler is likely to receive a dose of more than 500 millirems in a year.

BROAD SURVEY UNDERWAY

To calculate the total number and types of packages of radioactive material being transported annually in the U.S., NRC is conducting a survey to collect transportation information from a sample of about 2,400 NRC and Agreement State licensees and ERDA contractors. (The AEC had estimated that about one million packages of radioactive material were shipped during calendar year 1974, about three-fourths by air, consisting principally of small quantities of radiopharmaceuticals for medical diagnostic and therapeutic uses.) The results of the NRC survey will be used to assess the impact of changes in NRC transportation regulations currently under consideration and to estimate radiation exposures from normal shipments and the risk to persons and the environment from transportation accidents involving such packages. The survey will be completed in fiscal year 1976.

5

Safeguarding Materials and Plants

A Year of Planning

Throughout 1975 the Commission carried out major planning tasks, while tightening existing regulations, aimed at upgrading "safeguards"—those measures used to prevent the theft or diversion of nuclear materials or sabotage of nuclear facilities in order to protect the national security and the public health and safety.

The program of domestic safeguards for licensed nuclear materials and facilities cuts across the functions of NRC's five line organizations. It includes careful consideration of protective measures in the safety review of license applications for reactors (Chapter 2) and fuel cycle activities (Chapter 4), inspection and enforcement (Chapter 6), research (Chapter 8), and the development of appropriate standards (Chapter 9). In addition, safeguards concerns are the key element in the Commission's administration of export licensing for nuclear facilities and materials, and they play an important part in the NRC's cooperative relations with the International Atomic Energy Agency (Chapter 11).

Detailed safeguards requirements for licensed activities are set forth in the physical security regulations ("Physical Protection of Plants and Materials"), and in material control rules ("Special Nuclear Material") which pertain to the special nuclear materials (SNM) plutonium, uranium-233, and uranium enriched in the isotope uranium-235. These regulations are believed to provide an adequate basis for protecting presently licensed SNM and plants against theft, diversion or sabotage. However, the emergence of new risks, as well as technological developments, requires that the current base of regulations be continually evaluated

NRC SAFEGUARDS PROGRAM

PRINCIPAL CONCERNS
- Theft or Diversion of Nuclear Materials
- Sabotage of Nuclear Facilities

PRINCIPAL GOALS
- Prevention
- Early Detection
- Timely Response

PRINCIPAL METHODS
- Intelligence to Provide Early Warning
- Physical Protection Systems
- Materials Control and Accounting
- Coordination with Law Enforcement Authorities

and upgraded. The Energy Reorganization Act of 1974, which created the NRC and, within it, the Office of Nuclear Material Safety and Safeguards, specifically directed this Office, among other things, to carry out a review of existing safeguards of nuclear facilities and materials. This review is to comprise: (a) the monitoring, testing, and recommendation for upgrading of material accounting systems; (b) development of contingency plans for dealing with threats, thefts and sabotage of materials or facilities; and (c) a study of the need for, and feasibility of, establishing a Federal security agency to administer safeguards.

The Commission moved promptly in 1975 to tighten existing safeguards requirements, complete ongoing planning and carry out the Congressional mandates. This chapter briefly describes actions and progress in the following areas:

- Upgrading control of and accounting for special nuclear material.
- Physical protection of power reactors.
- Protection of fuel cycle facilities and materials in transit.
- Contingency planning for dealing with threats, theft and sabotage.
- Special studies on security agency and plutonium safeguards needs.

UPGRADING SNM CONTROLS

During fiscal year 1975, a special NRC task force completed an important review of industry plans for controlling and accounting for SNM. Twenty-six individual company plans were submitted by the industry to demonstrate how new and tougher regulatory requirements for material control and accounting would be met. Submittal of these plans was required by regulations imposed in October 1974.

Licenses of the 26 operators covered by the new regulations are being amended by the NRC to ensure that they follow the more stringent safeguards measures described in their approved plans. Once approved by NRC, each operator's plan becomes a binding license condition and operators will be checked regularly for compliance. The present NRC rules for SNM protection require those nuclear operators covered by the regulations to:

- Have a precise record-keeping system showing receipt, inventory, and disposition of all SNM.
- Take periodic inventories of the material on hand.
- Calculate and report to the NRC any material inventory discrepancy.
- Set up a control system to ensure that inventory discrepancy does not exceed levels allowed by the NRC.
- Maintain a measurement system to establish the material quantities entered into the record-keeping system.
- Establish material control areas.

The new regulations also specify certain other controls such as inventory methods, shipping and receiving procedures, materials storage practices, records and reports, and management controls.

Followup on Inventory Discrepancies

The principal measure of SNM control is the magnitude of inventory discrepancies. This measure is a calculated value which represents the difference between the amount of material that is supposed to be present according to the accounting records (taking into account measured receipts, transfers, and discards) and the amount of material actually found to be present during a physical inventory. The measurements required to establish the amount of material present are nearly always subject to error. A knowledge of the magnitude of these measurement errors is necessary for the proper interpretation of an inventory discrepancy.

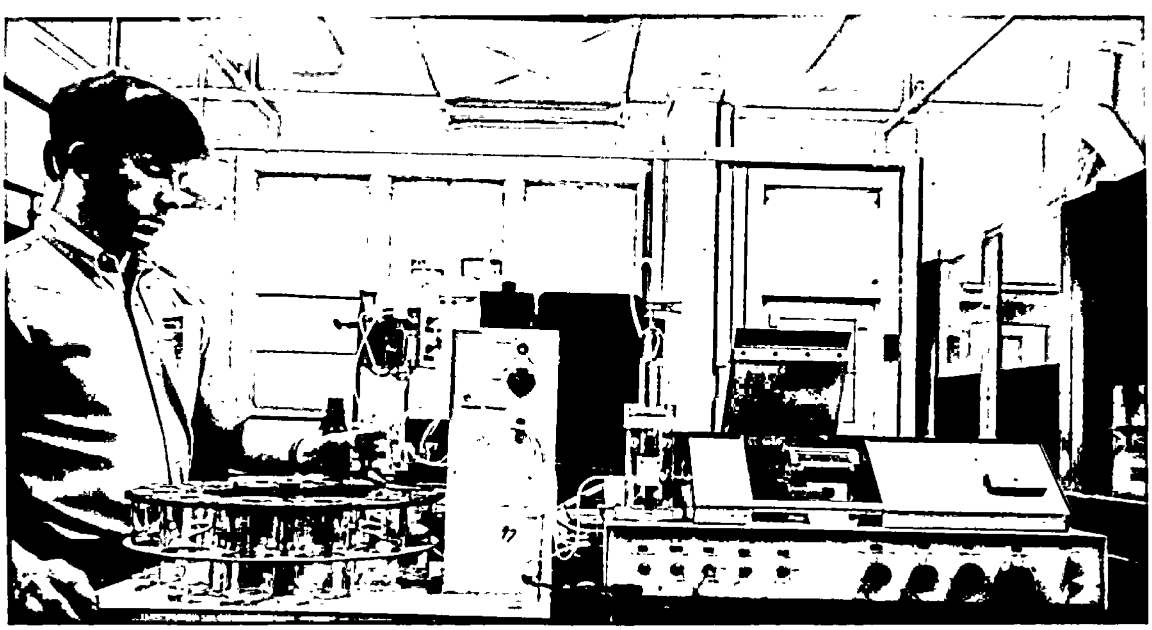
The Commission is proposing new guidelines to assure that corrective action will be taken when the amount of inventory discrepancy reaches NRC's allowable limits. Under the regulation published for public comment on July 17, 1975, explicit limits are specified for inventory discrepancies. More significantly, the new regulation would require specific actions to be taken, such as immediate reinventory, investigation of excessive inventory discrepancies or adoption of new procedures to prevent reoccurrence. In the case of a reinventory, it may be necessary in some cases to shut down the plant.

Quality Assurance for Measurements

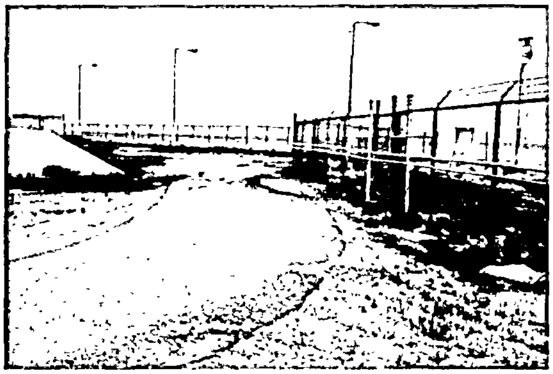
Under regulations issued in August 1975, operators must upgrade their procedures for measuring SNM to ensure that the quality of the measurement data will be acceptable for determining material on hand and detecting and analyzing inventory discrepancies.

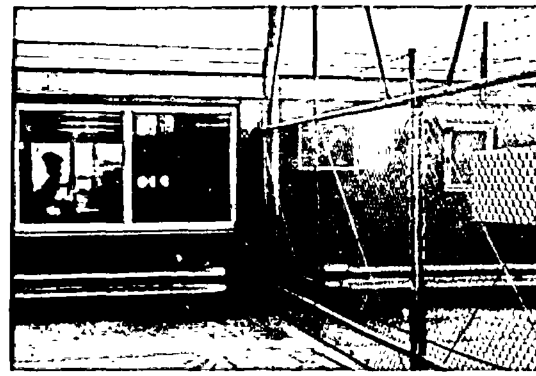
The new rules require licensees to set up management controls to assure measurement accuracy, a trained and qualified staff, a standards and calibration system, a testing system to detect and control error, a data evaluation process, program audits and periodic management reviews.

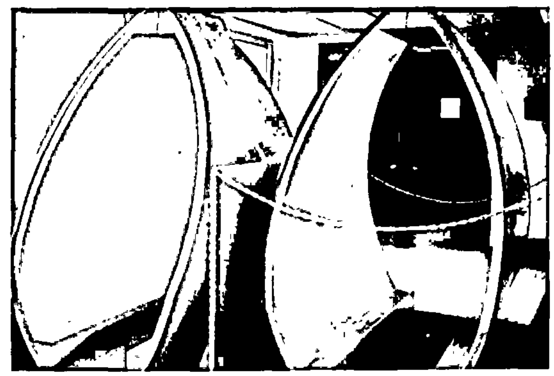
These so-called "quality assurance amendments" were adopted after a staff study showed that the need for improved measurements outweighed expected costs to the industry and the public. Licensees were given until November 11, 1975 to prepare proposed measurement control plans



The uranium content of material is determined by using an automated titrimetric procedure at the New Brunswick Laboratory. This chemical method of analysis is being used on a sample taken from a licensed facility by an NRC inspector for the purpose of verifying the special nuclear material inventory.







As a part of its program to protect a nuclear power plant against sabotage, this utility (1) has installed fencing and lighting around the plant perimeter, (2) limits vehicular traffic onto the site, and (3) monitors persons entering protected areas of the plant by use of metal detectors similar to those used at airports. Although most information relating to the construction and operation of a nuclear power plant is made available to the public by NRC, the details of a plant's physical security plan are not released for obvious reasons. These plans are carefully reviewed and must be approved by NRC, and compliance with the plans is checked during NRC inspections at the site.

to conform with the new regulations and to submit them to the NRC for approval.

PROTECTING POWER REACTORS

The subject of safeguards has become an increasingly important consideration in licensing and operating nuclear power plants. Present regulations require each applicant to include a physical protection plan when filing for a construction permit to build a power reactor. These plans are evaluated by the staff to assure that they satisfy established criteria concerning such matters as guard force organization; qualifications, training and utilization of security force personnel; designation of security areas; access controls; methods of surveillance; and communications systems. The NRC inspection and enforcement program (see Chapter 6) is designed to assure compliance with the plans through observations made during site visits.

A new reactor safeguards regulation is scheduled for issuance in fiscal year 1976 under NRC rules in 10 CFR Part 73. The proposed rule, "Requirements for the Physical Protection of Nuclear Power Reactors," was first published for public comment in November 1974. The draft rules are being revised based on comments received from the public and industry. A full staff review may result in further modifications.

Among measures being considered in the new rule are:

- More specific requirements for armed guards at plant sites (presently stipulated in Regulatory Guide 1.17).
- Search of personnel entering protected area.
- Continuous monitoring of perimeter zone.
- Guard escort for all vehicles entering perimeter.

Reactor owners would be required to submit a revised plant security plan to NRC within 90 days after the effective date of the rule.

FUEL PLANTS AND SHIPMENTS

Commercial plutonium and high-enriched uranium fuel facility operations as well as SNM in transit are subject to a comprehensive set of regulations designed to protect both the nuclear materials and the facilities from malevolent acts that could adversely affect the public welfare.

Physical protection features required for these plants include:

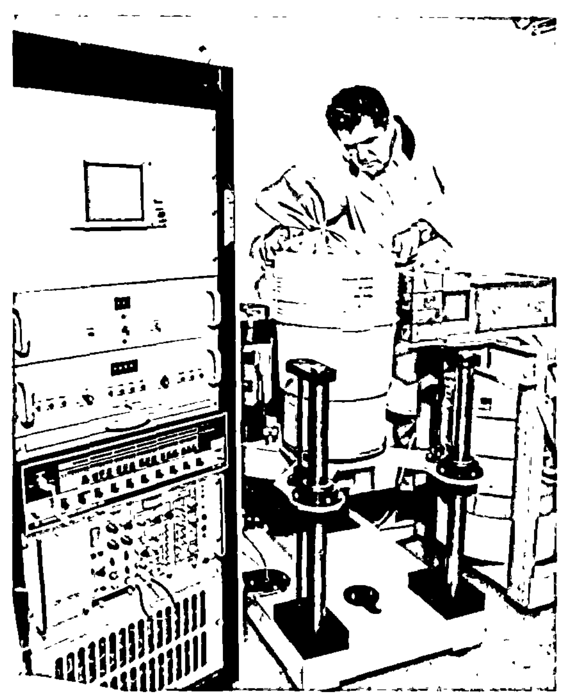
- A security force including armed guards.
- At least one guard supervisor onsite at all times.
- Written instructions to guards and watchmen.
- Erection of physical barriers.
- Intrusion alarms.
- Controlled access and search of individuals for firearms, explosives and incendiary devices.
- Exit search for concealed SNM.
- A response plan for threat assessment and action.
- Communication with response forces.

To further reinforce physical protection at nuclear facilities, additional safeguard measures—in the form of license conditions—were imposed on certain types of fuel plants during 1975. The purpose was to augment existing procedures designed to prevent diversion of high-enriched uranium and plutonium. Specifically, in addition to more stringent exit search procedures, a number of controls were applied to assure that emergency evacuations and drills could not be used as a diversion to conceal the removal of nuclear material from controlled areas. These measures, along with tightened access requirements, contribute to the effectiveness of the physical protection program.

Transportation Safeguards

Transportation of both highly enriched uranium and plutonium is regulated by NRC, since movement of the material is considered a vulnerable point in the safeguards chain. Regulatory requirements direct that all shipments be preplanned to reduce transit times to the minimum and to avoid areas of natural disaster or civil disorders. Delivery must be made to a certified receiver who accepts custody of the material. Shipments are protected by armed guards. Radio contact is provided for shipment enroute. There are special requirements for the protection of material peculiar to each mode of transportation (air, road, rail and sea).

In November 1974, the AEC published for comment proposed regulations to strengthen guard protection, require the use of seals on cargo compartments, increase the weight of containers permitted on open railroad cars, and provide advance shipping notice. The draft rules are being revised based on comments received from the



This segmented gamma scanner is one of the instruments used to measure the uranium-235 content of special nuclear material at a fuel fabrication plant by a nondestructive technique. Nuclear measurements are required by NRC regulations to assure proper accountability by the licensee of special nuclear material.

An armored vehicle and armed guards are typical of requirements for shipping significant quantities of high-enriched uranium or plutonium by road. The vehicle and guards shown here are used by one of the commercial shippers employed by nuclear firms in the United States to move radioactive materials under special protective arrangements. The armored vehicle must be equipped with radiotelephone to allow the guards to call for assistance if necessary.

public and industry and are being reviewed from a cost/benefit standpoint. The new regulation is scheduled for issuance in fiscal year 1976. (See also Chapter 9 discussion of "Safeguards Standards.")

CONTINGENCY PLANNING

The Energy Reorganization Act of 1974 charges NRC with the responsibility for developing "... contingency plans for dealing with threats, thefts, and sabotage relating to special nuclear materials, high-level radioactive waste and nuclear facilities resulting from all activities licensed under the Atomic Energy Act of 1954, as amended." In this particular area, 1975 was a year of planning and program development for the NRC. The major activities during the fiscal year consisted of organizing and staffing, clarifying major goals, and development of a program plan for implementation. In the coming year, the following results are expected:

- A better capability to assess threats against nuclear materials and facilities, based on close liaison now established with the Federal Bureau of Investigation, the Energy Research and Development Administration and other Federal agencies.
- Development of generic contingency plan models which will provide a clearly defined guide for dealing with specific types of contingencies.
- Using the generic models, specific application of contingency plans to the highest priority facilities which have material of the greatest strategic importance.

Suspected or Attempted Thefts

Section 307(c) of the Energy Reorganization Act requires the Commission to report to the Congress any "suspected, attempted, or actual thefts of nuclear materials in the licensed sector ...." None was reported during fiscal year 1975; however, three incidents of a security nature occurred.

Two of these were at the Kerr-McGee, Inc., plant in Oklahoma, one of which may have involved the surreptitious removal of trace amounts of plutonium, the other involving an individual climbing the fence and entering the protected area. The third incident involved a penetration of the protected area barrier at the Nuclear Fuel Services plant at Erwin, Tennessee, which was immediately detected. The NRC staff investigated each thoroughly and took necessary corrective actions including review of regulations to determine if present procedures are adequate.

SAFEGUARDS NEEDS EXAMINED

Security Agency Study

The Energy Reorganization Act of 1974 requires that the Director, Office of Nuclear Material Safety and Safeguards, assess the need for, and feasibility of, establishing a Federal security agency for the purpose of safeguarding civilian nuclear operations. The Act further specified that a report be submitted no later than one year following the effective date of the Energy Reorganization Act (January 19, 1975). The study was near completion at yearend.

A Federal security agency, if established, might perform a variety of activities intended to deter and prevent actual and threatened (a) theft of strategic nuclear materials, and (b) malicious release of radioactive substances into the environment.

At present, security forces include onsite personnel who are employed either by the facility licensee or its contractors, and State, local and, in some instances, Federal personnel who could be called to the site of an incident by facility operators or by persons accompanying shipments of special nuclear materials. Underlying the Congressional request for a study of the need for a security agency was concern that adequate security might be achievable only through direct Federal participation.

The primary focus of the Security Agency Study is on the question of need for a Federal security force: To what extent is there a need for direct Federal involvement in security forces? If there is a need for direct Federal involvement, what should be the role of the NRC? The Study is also addressing civil liberties implications of an NRC security agency and issues such as who would be responsible for providing necessary security forces, the use of deadly force to protect nuclear materials and facilities, security force funding, the role of NRC in recovery of stolen nuclear materials, and the need for onsite NRC inspectors for material control and accountability.

The analyses performed in the study will be based on a number of hypothetical threats involving covert and overt actions and combinations thereof. These "threats" span a range which is considered sufficient for reaching conclusions regarding the need for, and feasibility of, creating a Federal security agency.

The study distinguishes between guards and offsite reaction forces. Guards are stationed at facilities or accompany materials in transit, while reaction forces are seen as backup support in the event of a threatened or actual attack. The possible use of Federal agents in each of these three positions is being considered.

Final conclusions and recommendations will be reached after considering the inputs of contractor studies, public comments (as requested in the *Federal Register* on September 10, 1975), and inhouse safeguards expertise.

Special Safeguards Study

Safeguards issues that will be involved if the Commission permits widespread recycling of plutonium as a fuel for light-water reactors are being analyzed by the staff in a Special Safeguards Study. This study, in turn, provides important information for two other studies mandated in the Energy Reorganization Act of 1974; namely, the Nuclear Energy Center Site Survey (see Chapter 2) and the Security Agency Study.

In August 1974, the AEC had issued a "Draft Generic Environmental Statement on Use of Mixed Oxide Fuel in Light Water Reactors" (WASH-1327), a report which has become known by the acronym GESMO. (See Chapter 4.)

Shortly after publishing GESMO, AEC directed that an effort be made to spell out a safeguards program for plutonium recycle. As one of its first acts, the Nuclear Regulatory Commission reaffirmed the effort, approved the study plans, and directed expansion to include the safeguarding of all strategic special nuclear material in both plutonium and highly enriched uranium fuel cycles.

The Special Safeguards Study, expected to be completed in early 1976, uses the services of private firms under contract to consider and provide analyses of key safeguards issues. In addition, extensive use is being made of the expertise available at the Energy Research and Development Administration's national laboratories. Advice and assistance are also being obtained from within the Federal Government, especially from those agencies concerned with intelligence and security matters. The information developed from the study will be used in the safeguards supplement to the GESMO which, as discussed in Chapter 4, is being prepared on an expedited schedule.

Ensuring Compliance

Requirements that Must be Met

NRC's inspection and enforcement program is based on the precept that nuclear quality requirements are mandatory and enforceable under Federal law. NRC therefore inspects the industry quality assurance process on a continuing basis and takes enforcement action where necessary. The program is designed to assure that applicants for NRC permits and licenses, as well as existing licensees, conduct their activities in a manner that adequately protects the health, safety and security of the public and the environment in which they live. During 1975, NRC intensified its surveillance of safety-related applicant and licensee programs and of their plans and actions to safeguard nuclear materials and protect nuclear facilities.

The Office of Inspection and Enforcement, which carries out these programs on the Commission's behalf, performs three essential functions:

- Inspects materials and facility licensees to ascertain whether their quality assurance programs, plans and activities are being conducted in accordance with NRC rules and regulations and conditions of their licenses.

- Investigates incidents, accidents, thefts or diversions of special nuclear materials, and complaints or allegations from the public about NRC licensed operations or involving NRC licensed materials.
- Enforces compliance through issuance of notices of violation, imposition of civil monetary penalties, and promulgation of orders to suspend, modify or revoke licenses or to cease and desist licensed operations. Contributing field information as a basis for approving or denying permits or licenses also serves as an enforcement tool.

INSPECTIONS

Importance of Quality Assurance

A key component of excellence as it relates to safety in the construction and operation of nuclear power plants is quality assurance. This involves a planned management program of checks and controls designed to assure that plants are conceived, built, and operated to permit a high degree of confidence in their performance. Each prospective reactor licensee is responsible for developing a detailed quality assurance plan which also includes the verification of product quality from its contractors and vendors. The requirements against which licensee quality assurance plans and activities are measured are specified in NRC regulations, national codes and standards, conditions specified in permits and licenses and the applicant's or licensee's own approved operating procedures.

NRC requires that a licensee's quality assurance inspectors must be organizationally independent, professionally qualified, and present in sufficient numbers to assure thorough inspection coverage. At a typical nuclear power plant in late stages of construction, for example, out of a work force of about 2,000, roughly 100 are inspectors from the licensee and his contractors whose sole missions are to check the quality of the work underway.

NRC begins reviewing the organization and plans, six to nine months prior to the submission of an application for a construction permit, to determine that the proposed quality assurance program is fully responsive to regulations.

Scope of NRC Inspection Program

NRC inspections are of two general types: health and safety, and environmental protection inspections; and materials and facility (safeguards) protection inspections. The first covers quality assurance activities related to health, safety and environmental concerns for power and other reactors; fuel cycle facilities; architect-engineers, vendors and suppliers; and materials licensees, including universities, hospitals, research organizations, and other firms or institutions using nuclear materials. The second category deals with quality assurance in physical protection and safeguarding of special nuclear materials and facilities held or owned by licensees. Through direct observation, interviews, independent testing and review of records, NRC inspectors gather facts to ascertain compliance with approved quality assurance programs and with NRC requirements, and to establish a basis for enforcement action when such requirements are not met.

Inspection Management System

The NRC regulations, license conditions, operating procedures and codes and standards imposed on applicants and licensees as requirements also serve as the bases for NRC inspection guidance. For this purpose, the requirements are divided into discrete programs to form a computerized management information system. Inspection requirements covering light water reactor and high-temperature gas-cooled reactor inspections, for example, have been spelled out in specific procedures covering the scope and depth of inspection coverage. The same is being done for the remaining programs—materials licensee inspections, fuel cycle facility inspections, safeguards inspections, test and research reactor inspections and licensee contractor and vendor inspections. This clear definition of procedures permits the tracking of inspection activities on the management information system. NRC management, thus, will be able to quickly ascertain the status of inspection activities at a single facility as well as the overall inspection status for any program.

Structure and Staff

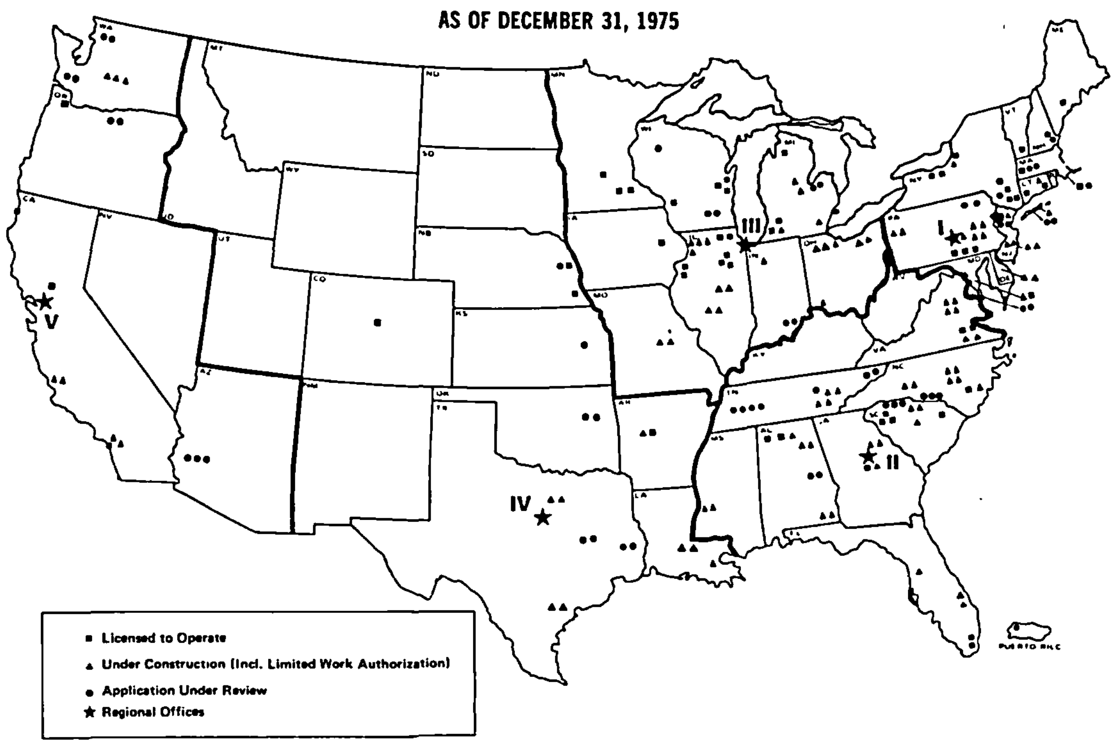
The NRC inspection staff is comprised largely of nuclear-oriented engineers and specialists, radiological and environmental safety specialists, and physical protection experts. All inspection and investigation activities, and most enforcement actions, as well, are handled by NRC regional offices, strategically located near the centers of licensee clusters. Those offices are: Region I, Philadelphia, Pa.; Region II, Atlanta, Ga.; Region III, Chicago, Ill.; Region IV, Dallas, Texas, and Region V, San Francisco, Calif. At the end of 1975, approximately 200 NRC professional personnel were assigned to inspection duties.

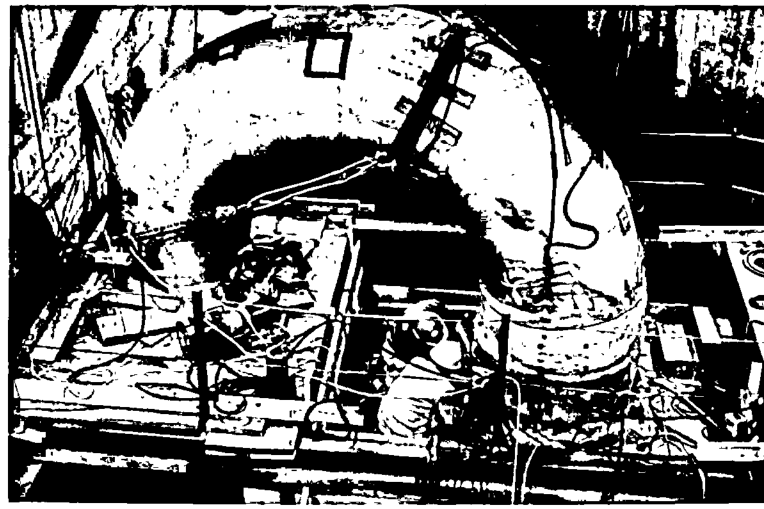
Reactor Onsite Inspections

The NRC inspection program for nuclear power plants begins with quality assurance planning and extends over the facility's entire lifetime. Based on the premise that the applicant or licensee is responsible for the design, construction and safe operation of its facility, NRC inspectors examine the licensee's efforts to obtain assurance that this responsibility is being met, and to prepare the way for corrective action if it is not.

In the period January through June, 1975, NRC's Office of Inspection and

NRC REGIONAL OFFICES
AND COMMERCIAL NUCLEAR POWER REACTORS
AS OF DECEMBER 31, 1975





NRC inspector and a licensee representative examine the stress relieving operation on a weld in the main steam line between the steam generator and turbine during construction of the Three Mile Island Unit 2 nuclear power plant. The stress relieving operation involves heating the metal on both sides of the weld to avoid brittleness.

Enforcement completed 991 health and safety inspections of power reactor facilities. Of these, 432 were pre-operating-license inspections and 559 were conducted subsequent to issuance of operating licenses. In addition, 59 inspections were conducted of research, testing and other reactor facilities. One or more items of noncompliance were revealed in 40 percent of all reactor inspections.

NRC inspections cover four phases of a nuclear power plant's life:

Preconstruction activities. Prior to docketing of an application for a construction permit, inspections focus on the prospective licensee's quality assurance program. An acceptable program must be in existence before the application will be accepted for formal NRC review. Subsequent inspections are carried out to confirm that an adequate program has, in fact, been implemented.

Construction activities. During construction, NRC inspects to verify the suitability of the materials used and the quality of fabrication. When components are received onsite, NRC inspectors spot-check to verify conformance with specifications and ensure that quality assurance procedures for handling and storage are implemented. During plant erection and the installation of components, they selectively observe activities such as welding, concrete installation, and electrical and instrument cable installation, and review the results of tests to determine whether requirements are being met.

Preoperational testing and startup. The frequency of inspections is increased significantly during preoperational testing and startup. Inspectors observe selected preoperational tests and check results to verify that components and safety systems will perform their intended functions. They also examine the operating organizational structure, training of personnel, performance of equipment and personnel, monitoring and sampling programs for radiation and effluent control, results of environmental monitoring, plans and training for emergencies, security provisions, and administrative controls for safety.

Operation activities. After routine operation begins, periodic inspections ascertain whether the licensee is operating in a safe and responsible manner in conformity with NRC requirements. Particular attention is devoted to evaluating corporate and plant management to determine whether its steps to prevent safety problems are effective, and whether it takes positive and timely corrective action in the event of abnormal occurrences.

POWER REACTOR INSPECTIONS FY 1975

Status	No. of Inspections	Average Inspections Per Reactor
Pre-Construction Phase		
Initial QA Inspection	253	2
Construction Phase		
Early Construction	357	6
Vendors of Major Components	100	–
Testing Phase		
Pre-op Testing, Final Construction, & Startup	548	18
Operation Phase		
Operating Facilities	596	16
	1854	

Offsite Vendor Inspections

In the past, utilities frequently attributed delays, malfunctions and interruptions in service to the failure of their contractors and vendors to properly control the selection, design and fabrication of hardware items incorporated into the reactor systems. NRC's response to such complaints came during 1975 in the initiation of the Licensee Contractor and Vendor Inspection Program (LCVIP) described in Appendix B to Part 50 of NRC regulations. A set of 18 relevant NRC quality assurance criteria are set forth in the regulation and these must be implemented by industry to assure that their plants will conform with requirements. Under the LCVI program, vendors are inspected directly by NRC inspectors rather than in association with licensee or utility inspectors, as previously done. This provides a more uniform application of the Commission's quality assurance requirements and at the same time reduces the need for repetitive audits and evaluations by licensees of their suppliers' generic quality assurance programs. Responsibility for product-acceptance examinations, however, continues to rest with the individual licensee or his agent.

The Licensee Contractor and Vendor Inspection program currently is carried out by a unit in the Dallas (Region IV) office. During the first six months of the LCVI program's existence, approximately 80 inspections of 69 organizations were accomplished. Based on early results, NRC hopes to extend the program to all nuclear steam system suppliers, architect-engineers and major component suppliers of safety related equipment.

Materials and Fuel Facilities

Regional inspectors conduct safety and environmental inspections of materials and fuel facility licensees (excluding reactor licensees), including spent-fuel reprocessors, fuel fabricators, waste disposal facility operators, major radiopharmaceutical firms, radiographers, and operators of medical facilities. During fiscal year 1975, 1587 such inspections were conducted, with items of noncompliance detected in 53 percent of the inspections.

Inspections are performed at frequencies determined by a classification system based on the relative weight given to safety considerations. Facility inspections focus on the evaluation of management quality assurance programs and controls over operations. A typical inspection might include a review of processing and equipment, such as filtration systems, checks for releases of radioactive effluents, and monitoring personnel to determine if regulatory requirements are being followed. The NRC effort does not duplicate or substitute for the licensee's management audit or controls. It does, however, gather—on a selective sampling basis—detailed information to ascertain whether licensees are conducting their activities with due regard to nuclear criticality control and radiological health and safety.

Safeguards Inspections

The safeguards program involves the inspection of licensees possessing strategic quantities of special nuclear material. The inspection staff conducts nuclear material control and accounting inspections of licensees which include fuel cycle facilities, reactors, research and development facilities, fuel reprocessing facilities, and universities having research reactors. Frequencies of inspections are determined by the quantity, strategic value, and accessibility of special nuclear materials which the licensees are authorized to possess. The basic elements of an inspection include a review of material controls systems and procedures, physical inventory controls, measurement controls, and records and reporting controls.

During 1975 NRC initiated a compliance inspection program to assure that approved physical security plans and strengthened regulatory requirements for nuclear power facilities had been implemented. Items of noncompliance were found in about 49 percent of the 401 safeguards inspections conducted in fiscal year 1975. All of these items have been satisfactorily resolved.

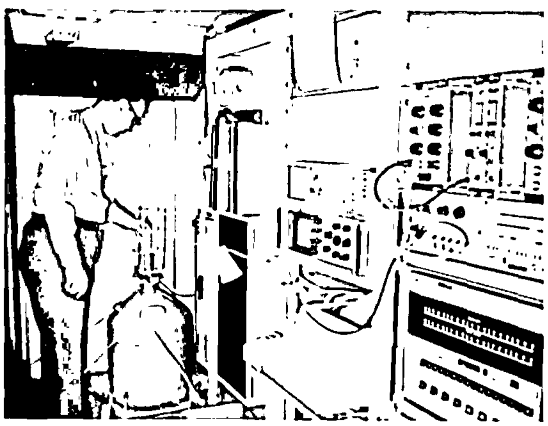
Safeguards inspection techniques. NRC increasingly relied on nondestructive measurement of special nuclear material, and on non-destructive analytical techniques for independent testing and verification of inventories. Inspectors now use equipment and computers carried in special vans to verify, through selective on-site sampling, the enriched uranium and/or plutonium content of inventory. Mobile equipment also is used to analyze low-level radioactivity in air and water effluents as part of the inspection program of confirmatory measurements, described later in this chapter. At the end of 1975, measurement vans were in use at two NRC regional offices, and an additional van was being procured for service in another region. Samples not amenable to nondestructive analysis using the mobile equipment are sent to ERDA's New Brunswick, N.J. or Los Alamos, N.M. Scientific Laboratories. In fiscal year 1975 these laboratories, under contract with NRC, analyzed 572 samples of enriched uranium and plutonium.

Materials accountability system. NRC's Nuclear Material Information System (NMIS) provides a current record of licensee nuclear material accountability data. Inspectors routinely examine material transfer documents, status reports, loss reports, and other documents and the resulting data is put into the computerized system. Corrections either to the documentation or to the NMIS information bank are made promptly.

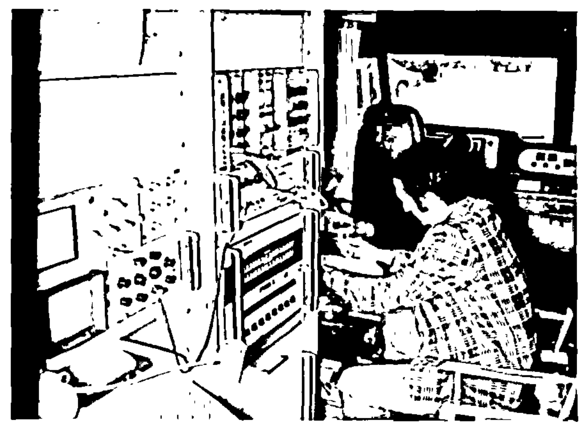
Materials in transit inspections. NRC's surveillance program for nuclear materials in transit requires that the export and import of significant quantities of special nuclear material, and not less than 5 percent of all other shipments, be monitored by NRC inspectors. The program is designed to assure that licensees have established and are maintaining required physical protection programs.

An NRC inspector (center) inventories fuel elements in a licensed fuel element fabrication facility. These fuel elements will be used in a small research reactor, such as one located at a university.

As part of the Independent Measurements Program, NRC inspectors use a Mobile Laboratory to measure samples of radioactive effluents from a nuclear power plant, checking on the accuracy of sampling routinely done by the plant's own laboratory, and providing independent data in support of an NRC inspection of the facility. The fully-equipped van is driven to the reactor site by the inspectors.



At left, an NRC inspector places a liquid radioactive waste sample in a cryogenic detector for analysis of its isotopic composition. The equipment is housed inside the Mobile Laboratory, parked at the reactor site.



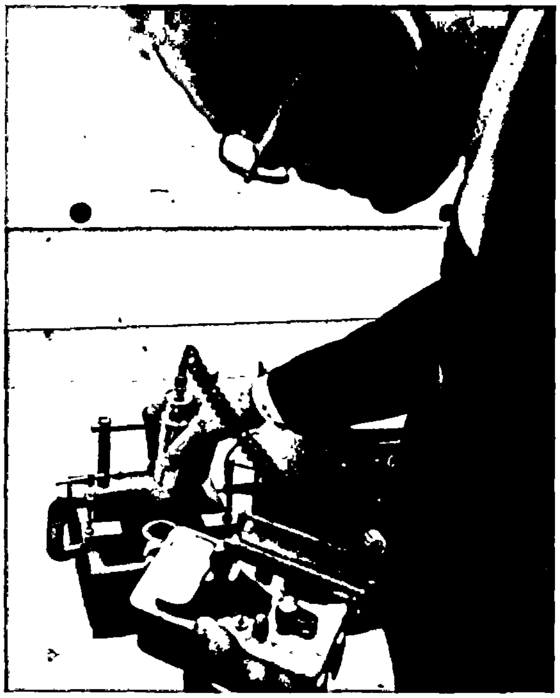
NRC inspector reviews analytical data obtained on a sample of radioactive waste from the nuclear power plant. A similar van, but with portable equipment, is used to verify measurements during safeguards inspections of materials facilities. During a safeguards inspection, the equipment is removed from the van and carried inside the facility for use.

Shipments by all modes of transportation are subjected to unannounced inspections, examination at points of origin, transfer and destination, observation and other surveillance by NRC inspectors to determine compliance and to assess the adequacy of the protection. Licensees at fixed sites subject to NRC's physical protection regulations are inspected at least annually.

Improving Inspection Effectiveness

Steps taken during 1975 to increase the effectiveness of NRC's inspection program included updating inspection requirements and inspection guidance to reflect new or revised industry codes and standards and standardized technical specifications, and the initiation of an intensive retraining program for reactor inspectors.

In addition, two "alternate" inspection programs were being studied. One involves stationing inspectors at plant locations for extended periods. Preliminary review of the results indicates that this technique may be suitable for further application. The second alternate program addresses the feasibility of using statistical sampling




During an NRC inspection of an electronic component manufacturer, the inspector in the photograph to the left uses a survey instrument to check for contamination of test equipment. At right, he surveys a radiation source, held by a licensee employee, for leakage. Licensed radiation sources are used by the manufacturer for nondestructive testing and quality control of the product.

techniques in routine inspections to assess the probability of detecting a licensee in noncompliance. Analysis of both programs will be continued into 1976.

INVESTIGATIONS

A significant part of the NRC's inspection and enforcement effort is involved in responding to reports of radiation incidents, abnormal occurrences, equipment problems, and allegations of improper or unsafe operations. Although many of these events are minor in nature and can be reviewed at a scheduled inspection, some require additional response. In these cases, a special inspection may be scheduled or a full investigation may be initiated immediately.

1975 Investigations Summary

Thirty-four investigations were conducted during the last half of fiscal year 1975. Seven involved exposures of licensee personnel as a result of radiation incidents, two involved radiation incidents without personnel exposures, seven dealt with allegations of unsafe working conditions, five with allegations of improper operations, three with alleged instances of loss or theft of material, and the remaining 10 involved other matters. In 15 of the 34 investigations, licensees were cited for failure to meet NRC requirements or license conditions.

An example of a major NRC investigation is the one conducted in connection with the March 22 electrical cable fire at TVA's Browns Ferry Nuclear Power Station near Decatur, Ala. (this event and the results of the investigation are described in detail in Chapter 7 of this report). NRC's response to the incident resulted in three separate actions: an investigation by the Office of Inspection and Enforcement, an evaluation of the safety of Browns Ferry plants by the

Office of Nuclear Reactor Regulation, and a technical and procedural review by a Special Review Group appointed for that purpose.

The investigation by the Office of Inspection and Enforcement sought to identify areas of compliance or non-compliance with license provisions and regulations relating to health and safety, and to determine the facts associated with the fire. The scope of this investigation included the events leading to the fire, subsequent fire fighting efforts, the sequence of operational events, problems experienced with nuclear steam supply systems, the interactions between operating units, and the response of the TVA groups and State and local government bodies following notification of the fire. The investigation comprised private interviews of personnel, reviews of documentation and observations, and flammability tests of the penetration sealants and cable insulation by the NRC consultants. In addition, all similarly constructed reactors were inspected to determine the adequacy of construction and operating procedures with regard to fire prevention and safety.

ENVIRONMENTAL MONITORING

The main objective of environmental monitoring is to determine if there is a buildup of radioactivity in the environment.

Each nuclear facility licensee is required to monitor major and potentially significant paths for release of gaseous and liquid radioactive effluents during normal operation. NRC inspectors check the licensee's radiological monitoring and waste systems to assure they are built as designed and operated to keep releases within regulatory limits. If a regulatory limit is exceeded, the licensee must so inform the NRC and take appropriate action.

Each power plant licensee also is required to monitor major pathways in the environment. During NRC inspections, random samples of monitoring records, procedures, and reports are examined and confirmatory measurements are made to assess the accuracy and consistency of licensee measurements of radioactivity in effluent and environmental samples. The confirmatory measurements are then further compared with NRC reference data developed in laboratory measurements.

NRC's Interagency Program

NRC enlisted the cooperation of the National Bureau of Standards, the Energy Research and Development Administration, the Environmental Protection Agency, and State health and environmental agencies in 1975 to provide corroborative evidence of the environmental and effluent radioactivity measurements submitted by licensees. An example of such cooperation is a system which permits continuous tracking of the accuracy of radioactivity measurements, established by agreement between the National Bureau of Standards and the Health Services Laboratory (HSL) of the ERDA Idaho National Engineering Laboratory. HSL functions as the NRC reference laboratory in such matters, and NRC inspectors regularly compare licensee effluent measurements with those made by HSL on identical effluent samples.

State agencies also assist in long-term, repetitive sampling to evaluate licensees' overall environmental programs. At the end of FY 1975, the 18 States participating in this program were Alabama, Arkansas, California, Colorado, Connecticut, Florida, Illinois, Maine, Maryland, Michigan, Minnesota, Nebraska, New Jersey, New York, South Carolina, Vermont, Virginia and Wisconsin. Under written contract, NRC provides these States with funds, technical support and training to assist in improving their analytical capabilities.

Another tool for the evaluation of a licensee's analytical capability is EPA cooperation with NRC under a 1975 agreement in the use of its Environmental Radioactivity Laboratory Intercomparison Studies Program at the National Environmental Research Center, Las Vegas.

ENFORCEMENT PROGRAM

All matters examined during an inspection or an investigation of licensee activities fall in one of four categories: (1) acceptable items which meet requirements, (2) items of noncompliance with regulatory requirements, (3) deviations from codes, guides, standards or commitments to the Commission, and (4) items which are unresolved pending additional information.

Enforcement action is taken to assure that persons who do not comply with regulatory requirements will act promptly to bring their programs into compliance. For recalcitrants, enforcement action must provide a proper incentive to take such corrective action and the appropriate sanction may be punitive. Notifications of deviations from approved codes, standards and guides, and from licensee commitments to the Commission, are forwarded to licensees and, if corrective measures are not properly implemented, appropriate enforcement actions are imposed.

INSPECTIONS AND INVESTIGATIONS

	FY 1974	FY 1975	FY 1976 Estimate
Reactor Inspections	1363	1960	2065
Fuel Cycle Inspections	115	148	190
Materials Inspections	1466	1439	1610
Safeguards Inspections	100	401	515
Incident Investigations	26	55	60
TOTAL	3070	4003	4440

Actions Available to the Commission

Formal enforcement actions available to the Commission include notices of violation, civil penalties, and orders.

Notices of violation are written notices to licensees, citing the apparent instances of failure to comply with regulatory requirements. (For purposes of categorization, failures to comply have been classified as violations, infractions, and deficiencies.) As an administrative action, a Notice of Violation may itself be an appropriate enforcement action in certain cases.

Civil monetary penalties may be levied by the NRC against licensees' failures to comply with requirements in licensing provisions of the Act or NRC rules, regulations, orders, or licenses. The Commission is required to issue a Notice of Violation prior to instituting proceedings to impose a civil penalty. A Notice of Proposed Imposition of Civil Penalty also must be forwarded to the licensee. The licensee is given an opportunity to file an answer. He may protest the civil penalty, deny the items of noncompliance in the Notice of Violation in whole or in part, demonstrate extenuating circumstances, show error in the Notice of Violation, or show other reasons why the penalty should not be imposed.

Issuance of orders. NRC has authority to issue orders to "cease and desist," and orders to suspend, modify, or revoke licenses. Such orders ordinarily are preceded by a written Notice of Violation to the licensee, providing him an opportunity to respond as to the corrective measures being taken. If the licensee fails to respond or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring him to show why the particular order (either of revocation, modification or suspension) should not be made effective. In some instances, where the health, safety or interest of employees or the public requires, or where deliberate noncompliance with NRC regulations is involved, the notice provision may be dispensed with and the order made effective immediately.

In addition to proceeding by way of order, the Commission also may request the Attorney General to obtain an injunction or other court order to enjoin licensees from violating the Act or any regulation or order issued thereunder.

Failure to Report Violations

NRC inspection and enforcement procedures and actions will be adjusted in 1976 to reflect a new category of violation. As discussed in Chapter 9 of this report, Section 206 of the Energy Reorganization Act of 1974 provides for civil penalties for deliberate failure by individual directors and responsible officers of licensees, facility component manufacturers and architect-engineers, and others, to report violations of NRC regulations or defects in facilities, basic components, or activities which relate or could create a substantial safety hazard. In March 1975, NRC published a proposed Part 21 to its regulations, describing proposed procedures and requirements, including inspection and enforcement, for carrying out the purpose of Section 206.

Enforcement Actions

Civil fines: During the period January 1 to June 30, 1975, NRC imposed nine civil monetary penalties as a means of enforcing compliance with NRC rules and regulations

Walworth Co., Bala-Cynwyd, Pa.; $2,500. An investigation resulting from reported overexposures revealed items of noncompliance involving personnel exposures in excess of calendar quarter limits, use of inadequately qualified radiographers, inadequate surveys of a radiographic device and failure to have an operable warning device in a radiography cell.

Virginia Electric & Power Co. Richmond, Va.; $10,000. The fine was issued as a result of noncompliance findings revealed during an inspection of the Surry I and II plant security program.

Maine Yankee Atomic Power Co., Augusta, Me.; $4,000. The fine was issued as a result of noncompliance findings revealed during an inspection of the Maine Yankee Atomic Power Plant security program.

General Electric Co., San Jose, Calif.; $6,500. Instances of repetitive or similar items of noncompliance, relating to radiation safety and personnel exposures, were revealed during an inspection of two reactors located at the Vallecitos Nuclear Center. (At year end, this fine was being contested before the Commission's Administrative Law Judge.)

X-Ray Industries, Inc., Detroit, Mich.; $1,800. An investigation conducted as a result of a personnel exposure revealed procedural and management control weaknesses which could have contributed to the exposure.

Northeast Nuclear Energy Co., Hartford, Conn. $7,500. Several instances of failure to implement or follow the site security plans were revealed during an inspection of the Millstone Point facility.

Boston Edison Company, Boston, Mass.; $12,000. An investigation at the Pilgrim plant of activities relating to the company's quality assurance program revealed three instances of noncompliance involving failure to assure that qualified personnel performed their duties properly or were present during certain nondestructive testing operations, and failure to assure that adequate audits were made of the quality assurance program and resulting records.

Gladstone Laboratories Inc., Cincinnati, Ohio; $1,150. Inspection over a period of three years indicated the need for substantial improvements in the management of the licensee's radiation safety program. The inspection leading to imposition of the fine revealed 11 items of noncompliance, most of which were similar to items uncovered in three previous inspections.

Cleveland X-Ray Inspection, Cleveland, Ohio; $1,300. Inspections uncovered 18 items of noncompliance, several of which were similar to items identified in a previous inspection. Inadequate management of the radiation safety program appeared to be the basic cause of most findings.

Orders: In addition to the civil penalty issuances discussed above, the following orders were issued during the same period.

West Virginia University, Morgantown, W. Va. On February 19, 1975, an order rescinding a previous order issued in December 1974 which required the licensee to cease use of materials held under expired licenses and to transfer such materials to persons authorized to receive it. These actions were completed in late January 1975 and verified by NRC personnel on February 10, 1975.

Green Bay X-Ray Service, Inc., Green Bay, Wisc. On February 22, 1975, an order was issued to show cause why licensed activities should not be suspended until the licensee demonstrated compliance with NRC requirements that employees be adequately trained to perform radiographic operations, and that management controls be sufficient to assure future compliance with regulatory requirements. The licensee improved the radiographic operations in question and this was confirmed by NRC inspections. On May 9, 1975, an order rescinding the February 1975 order was issued.

REPORTING

Information brought to the attention of NRC that has significant safety implications or that may indicate problems in design, materials or procedures at more than one plant, is disseminated by NRC in a Bulletin to all applicable licensees and to the public at NRC public document rooms. A Bulletin requires each licensee concerned to take actions necessary to assure a similar situation will not occur at its own facility.

7

Evaluating Operating Events

Applying Lessons of Experience

Operating experience of commercial nuclear power plants in the United States had produced, through November 1975, more than 270 reactor-years of operation without any radiation fatality or injury to plant personnel or the general public. This record is, of course, directly related to the multiple levels of protection against accidents required by the Government and provided by the regulated industry in nuclear facilities.

Complex industrial operations, nonetheless, are always subject to incidents that may involve system, component or structural failure or malfunction, personnel error, design or management deficiencies and other matters related in some way to safety. To assure that any broad trend related to safety is quickly identified and to correct specific safety-related problems, the NRC imposes extensive reporting requirements on licensees to supplement its inspection and enforcement program. When a problem arises with implications for more than one plant, NRC may issue a bulletin to licensees of all plants that may be affected, calling for such actions as immediate inspection and possible replacement of certain equipment.

Operating events that must be reported are specified by the agency. Reportable occurrences usually are not accidents in the ordinary sense of the word. Rather, they are unexpected events of actual or potential significance to the safe operation of the plant; if generic, they involve possible safety implications for other plants of similar design.

Regular plant operations are also recorded, and the record disseminated by NRC to the nuclear industry and the public. This information enables all parties concerned to avoid or correct problems and gain in expert knowledge. The data are disseminated primarily through publication of status reports, bulletins and Current Events reports. In addition, special statistical, analytical and evaluative reports are prepared and issued from time to time for NRC and industry use and for the information of the Congress and the public. All relevant data derived from nuclear plant operations are stored in a computer-based data file for expeditious retrieval and analysis in developing standards, formulating regulations, and giving general guidance to the industry.

Availability and Capacity Factors

The NRC's reporting system assembles data from nuclear power plant utilities which are also useful to the Federal Energy Administration, the Federal Power Commission, and the Energy Research and Development Administration.

As of the end of December 1975, there was a total of 56 licensed nuclear power plants in the United States, of which 51 were in commercial operation, four were in power ascension, and one was shut down. The aggregate net electrical generating capacity of these plants came to 39,000 MWe. During calendar 1975, commercial nuclear plants generated 160 million megawatt hours of electricity net, as compared with 98 million for the whole of 1974. The average availability factor (the hours a generator is in operation as a percentage of the total hours in the period) for these plants was 72.2% during 1975; this compares with 68.5% for calendar 1974. Average capacity (the net electric energy generated compared with what would have been produced at full capacity) was 58.3% for 1975, as against 57.2% for 1974.

ABNORMAL OCCURRENCES AT NUCLEAR POWER PLANTS

January – September 1975

Date	*Event Type*	*Event*	*Facility*
February 26, 1975	Single	Steam Generator Tube Failure	Point Beach 1
March 22, 1975	Single	Fire in Electrical Cable Trays	Browns Ferry 1 & 2
May 1, 1975	Single	Loss of Main Coolant Pump Seals	H. B. Robinson 2
January 25, 1975 and May 3, 1975	Recurring	Improper Control Rod Withdrawals-Maintenance	Dresden 2 Quad-Cities 1
Various: September 1974 into 1975	Generic	Cracks in Pipes at Boiling Water Reactors	Dresden 2, Quad-Cities 1 & 2, Millstone 1, Monticello, and Peach Bottom 3
April 1975	Generic	Fuel Channel Box Wear at Boiling Water Reactors	Duane Arnold, Cooper, Peach Bottom 2 & 3, Browns Ferry 1 & 2, Brunswick 2, Hatch 1, FitzPatrick, and Vermont Yankee
Various: October 1972 to May 1975	Generic	Steam Generator Feedwater Flow Instability at Pressurized Water Reactors	Surry 1, Turkey Point 3 & 4, Indian Point 2, and Calvert Cliffs 1

(Note: For the recurring and generic events, the circumstances surrounding the events varied from plant to plant.)

ABNORMAL OCCURRENCES—1975

Under Section 208 of the Energy Reorganization Act of 1974, NRC is required to " . . . submit to the Congress each quarter a report listing for that period any abnormal occurrences at or associated with any facility which is licensed or otherwise regulated pursuant to the Atomic Energy Act of 1954, as amended, or pursuant to this Act. For the purposes of this section, an abnormal occurrence is an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety . . . ."

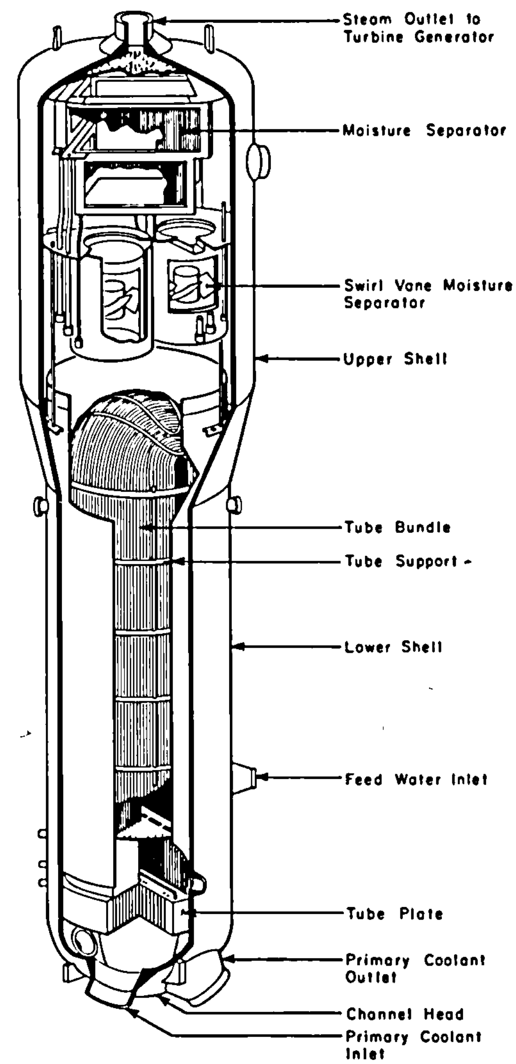
To make that determination, NRC has developed two major interim criteria, according to which abnormal occurrences are: (1) events involving an actual loss of the protection provided for the health or safety of the public; and (2) events involving major reduction in the degree of protection provided for the health or safety of the public.

None of the events occurring at operating nuclear power plants through November 1975 had any direct impact on or consequence to public health and safety, though some involved a temporary but significant reduction in the standard level of protection required. From over 1200 Licensee Event Reports received during this period, a total of seven events were considered to be abnormal occurrences under the interim criteria. Three of these were judged "single" incidents; one is of a recurrent nature; and three are generic, with implications for a number of facilities A summary of each of these follows.

Steam Generator Tube Failure

The event occurred on February 26, 1975, at the Point Beach Nuclear Plant, Unit 1, in Manitowoc County, Wis. This unit is a pressurized water reactor using primary system water to transfer heat from the reactor fuel to the secondary water system. Water in the secondary system, which is nonradioactive, is converted to steam in two steam generators by the heat from some 3,200 tubes in each generator through which primary system (radioactive) water flows. A hole developed in a 1-inch diameter tube in the "B" steam generator, resulting in a contamination of the secondary system. The leak rate from the primary to the secondary system reached an estimated maximum of 125 gallons-per-minute in about three-quarters of an hour. Licensee personnel

PWR STEAM GENERATOR



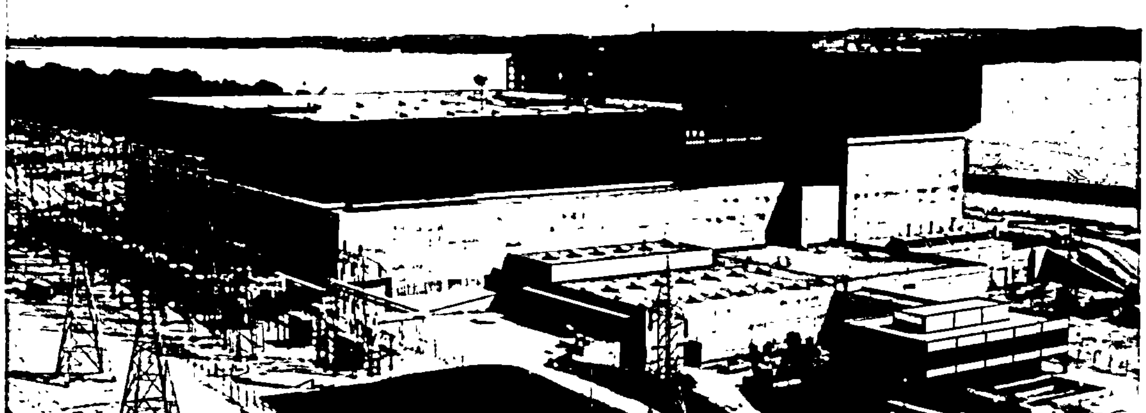
manually shut down the unit. Offsite radioactivity releases during the event were within the NRC regulatory limits and liquid releases were below maximum permissible concentration values.

The event is significant because it involved a large primary-to-secondary leak rate, the first tube failure to occur during plant operation with leakage in excess of 25 gallons-per-minute. The rate was, however, only a fifth of the rate postulated for safety design purposes and was handled within the capability of the normal primary coolant makeup system. Several radiation monitors designed to diagnose such an event did not perform as intended and caused a delay in determining that a tube failure had occurred.

The cause of the leak was identified as a buildup of sludge around the tube from a phosphate water treatment used to control tube corrosion. The build-up resulted in accelerated corrosion and subsequent tube failure. The sludge was removed, inspections performed on the tubes in both steam generators, and the defective tube was plugged, as were tubes with excessive wall thinning. For the long term, continuous removal of sludge, modification to the steam generators to minimize build-up, continuing use of a new all-volatile treatment for secondary water chemistry control, and more frequent in-service inspections have been adopted. Radiation monitoring systems and operational procedures have also been modified, and NRC has changed the specified primary-to-secondary leak rate limits to require earlier corrective action. Generic aspects of incidents of this kind were previously identified and a plant-by-plant review of the operating experience of steam generators is continuing. The Unit resumed operation on April 5, 1975.

Fire in Electrical Cable Trays

A fire occurred on March 22, 1975, at the Browns Ferry Nuclear Plant of the Tennessee Valley Authority in Limestone County, Ala. The plant contains three nuclear units, one of them under construction. The two operational units, each powered by a boiling water reactor with a



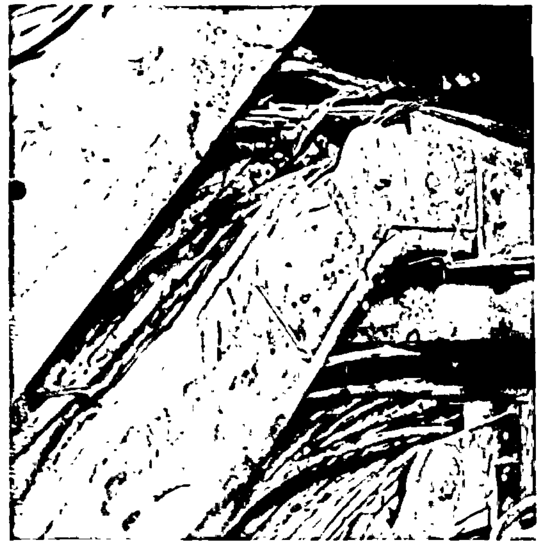
Tennessee Valley Authority's three-unit Browns Ferry Nuclear Plant located on Wheeler Lake near Decatur, Ala.

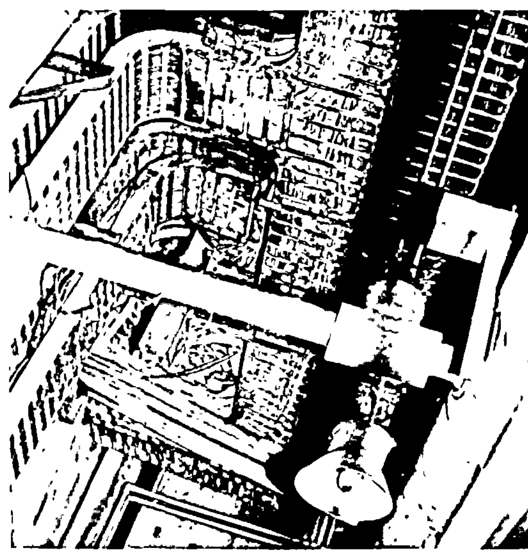
net electrical capacity of over 1,000 megawatts, and each in full power operation at the time of the fire, were shut down for an extended period as a result of this occurrence.

The fire started in an electrical cable penetration between the cable spreading room and the reactor building; the cable spreading room is located beneath the common control room for Units 1 and 2. The fire burned for about seven hours, spreading horizontally and vertically to all 10 cable trays within the penetration, into the cable spreading room for several feet, and along the cables through the penetration about 40 feet into the reactor building. The fire damage, confined to an area roughly 40 feet by 20 feet in the Unit 1 secondary containment building, affected about 1,600 electrical power and control cables.

While both units were shut down safely, normally used shutdown cooling systems and other components which comprise the emergency core cooling system (ECCS) for Unit 1 were inoperable for several hours. Other installed equipment was employed to maintain sufficient cooling capability to protect the nuclear fuel from overheating. There were no significant problems with the shutdown cooling of the Unit 2 reactor. Even though normal and emergency core cooling systems were unavailable in Unit 1 for a time, at least five alternative methods were available to provide adequate core cooling within the required time frame. There was no adverse impact on the public, plant personnel or the environment as a result of the fire; sampling indicated that airborne release rates were less than 10% of the specified limit. Ten cases of smoke inhalation and a fractured wrist were incurred by fire-fighting personnel.

The cause of the fire was the ignition of cable penetration sealing material by a candle flame, being used by a construction worker checking for air leaks. The flexible





Fire damage to electrical cables at the Browns Ferry nuclear power plant. At top, closeup view of damage in cable spreading room where the cables penetrate the wall to the reactor building. Workmen were using a candle to check for drafts, which would indicate whether the cable penetration area was adequately sealed, when the sealing material caught fire. Below, fire damage to overhead cables inside the reactor building where the fire came through the penetration.

polyurethane foam sealing material being used had not been specifically approved by the licensee's design department, nor had it been tested for this kind of application. The dangers involved in using flammable material in this manner were evidently not recognized by plant management, even though several small fires had occurred during similar testing activities at the plant. Personnel inspecting, sealing and testing the cable penetrations had not been provided with an adequate written procedural guide. Another contributing factor may have been the plant's fire-fighting techniques and equipment.

The Tennessee Valley Authority estimates that the shutdown of Unit 1 and Unit 2 will last until the spring of 1976. Detailed plans for restoration of the facilities were submitted to NRC, which reviewed the design for adequacy and continued on-site inspection activities to verify implementation of approved restoration plans. Those plans included such modifications as the addition of permanently installed water spray systems, fire detection systems and fire barriers. The new design replaces the flammable polyurethane seal with fire-resistant material and calls for added insulating material to form an effective fire stop. Administrative controls and procedures will also be revised.

NRC has thoroughly appraised conditions at the plant and effected license changes necessary to maintain the Units in a safe shutdown condition during repairs and, as these are completed, to return the Units safely to operation. A special review group within NRC is studying measures to prevent or mitigate the consequences of similar events at other plants. A Notice of Violation was sent to the licensee, listing areas of noncompliance with regulations and requiring written response detailing corrective steps to be taken and the date when full compliance will be achieved. The licensee was also asked to report to NRC on the training of personnel in fire-fighting, emergency procedures and communications, and clarification of responsibility in emergency situations. In addition, NRC instructed all licensees to review overall policies and procedures related to the possible effect of construction work on reactor operation, fire protection, and emergency shutdown, and to reevaluate electrical system design. In September 1975, NRC authorized the licensee to proceed with the proposed restoration and design modifications.

Inspection reports, technical specification changes, safety evaluations and other data developed by both TVA and NRC have been made available to the public through NRC and local Public Document Rooms. A Congressional hearing was held in September 1975 before the Joint Committee on Atomic Energy, with testimony by the NRC Chairman and other officials of the NRC and representatives of the licensee. While the Browns Ferry fire will be under scrutiny for some time to come—in all its complexity of causal factors, contributing factors, real and possible consequences and implications for all nuclear facilities—the event has already demonstrated both the importance and the effectiveness of multiple, mutually reinforcing back-up safety systems, or defense-in-depth. The potential vulnerabilities revealed by the fire will be the subject of intense analysis and will probably result in new requirements both within the industry and the agency.

Loss of Main Coolant Pump Seals

This event occurred on May 1 and 2, 1975, at the Carolina Power and Light Company's H. B. Robinson S. E. Nuclear Power Plant, Unit 2 in Darlington County, S.C. The one nuclear power unit at the site is a pressurized water reactor with a net capacity of about 700 MWe. Three primary coolant loops circulate pressurized water from the nuclear core to the steam generators. Each loop has one main coolant pump, and each pump shaft has three seals arranged in series to prevent any coolant leakage to the containment structure. The system of seals is lubricated and cooled by a water source other than the primary coolant water; normal leakage of the seal water system, called "seal water leakoff," runs through leakoff lines from the three pumps to a common line. When a first stage pump seal was found to be leaking primary coolant water, reactor power was quickly reduced to about 36% and the leaking pump was shut down. Shortly thereafter, an automatic shutdown of the plant occurred because a signal indicated that the water level in one of the steam generators was too high. Four hours later—with the other two pumps shut down and, despite attempts, not restarting—the licensee restarted the leaking pump. It was operated for about 90 minutes, during which time all three seals failed, resulting in a discharge of about 132,500 gallons of radioactive primary coolant water into the containment structure. The structure contained the leakage. However, the leak could not be isolated from the primary coolant system because the coolant loops were not designed with isolation valves. A maximum leak rate estimated at 400 gallons-per-minute occurred, greatly exceeding the postulated leakage from seals of this design. Coolant makeup was provided by a safety injection system designed for the purpose, and the fuel was adequately cooled.

Steam from the failed seals apparently affected the seals on the other pumps, which were then shut down and could not be restarted. Leakage from the failed seals was not stopped until some 16 hours after the initial shutdown of the leaking pump. During the forced plant cooldown, the normal cooldown rate prescribed in specifications of the license was exceeded. The plant was safely shut down, however, and maintained in a safe shutdown condition. Offsite release of radioactivity was within specified limits.

The event was caused by the complete failure of the seal system on a main coolant pump. That failure was compounded by failure of the pump radial bearing which damaged other seals. Procedural errors were contributing factors. The reason for the failure of the first seal, which set the train of seal failures in motion, has not been identified, although improper maintenance may explain it.

The licensee replaced all the failed parts on the affected coolant pump and all three pumps were inspected and tested before returning to service. New procedures were implemented to prevent events of this kind, with closure of the pump seal water leakoff isolation valve as the immediate action to be taken upon indicated damage to a No. 1 seal. The plant resumed operation in June 1975.

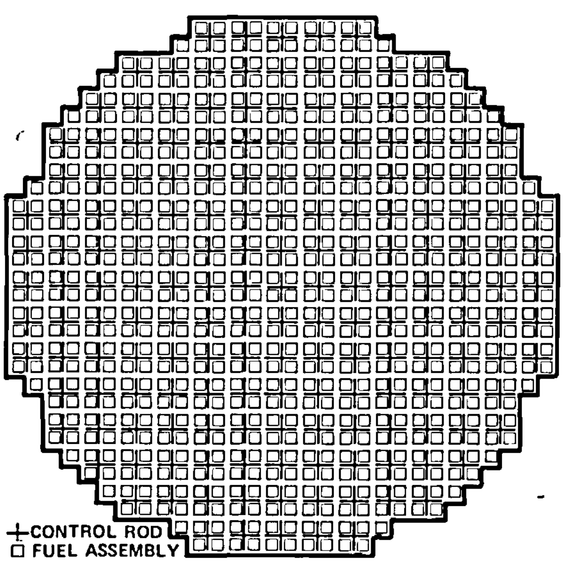
The NRC sent a Notice of Violation to the licensee, citing noncompliance with regulations, including failure to adhere to the approved procedure for plant cooldown and failure to meet commitments of the quality assurance program. A critical review of the pump seal design was performed by NRC because the reported leak rate exceeded previously stated values for that pump seal design. The review concluded that the design was adequate in that, should a No. 1 pump fail completely, the No. 2 is designed for full system pressure, thus serving as a total backup. The failure of No. 2 pump seal system in the occurrence cited was the result of continued operation under abnormal conditions, causing mechanical damage and a large leak rate. Such operation was not in accord with established procedures and more stringent controls have been applied to prevent recurrence.

Improper Control Rod Withdrawals

These two occurrences took place on January 25, 1975, at the Dresden Nuclear Power Station, Unit 2, in Grundy County, Ill., and on May 3, 1975, at the Quad-Cities Station, Unit 1, in Rock Island County, Ill. The licensee for the Dresden plant is the

Commonwealth Edison Co. and for the Quad-Cities station the licensees are Commonwealth Edison and the Iowa-Illinois Gas & Electric Co.

The Dresden 2 facility is a boiling water reactor plant with a rating of 809 MWe net. While the Unit was shut down for refueling, maintenance was being performed on the control rod drives. Through personnel errors and inadequate procedures, two adjacent control rods were withdrawn to the full-out position, in violation of the minimum separation criterion for such maintenance. The criterion is intended to prevent an unintended self-sustained nuclear chain reaction, or "inadvertent criticality." The reactor remained shut down by a safe margin; instrumentation verified that no criticality existed during the control rod withdrawal, and there was no release of radioactivity or damage to the facility. Satisfactory control was demonstrated during the performance of the shutdown margin test that is required after each refueling.



This diagram displays the top view of the matrix of control rods and fuel assemblies in a boiling water reactor core. During shutdown of a reactor for refueling, normally only one control rod may be withdrawn from the core for maintenance. Under the minimum separation criterion, a second control rod may be withdrawn if it is separated in all directions by more than two inserted control rods. Even within these limits, not more than two control rods may be withdrawn from the reactor core at one time.

Quad-Cities 1 is a boiling water reactor plant with a rating of 809 MWe net. This Unit was shut down and maintenance on the control rods was being performed much as in the Dresden incident. Two control rods were withdrawn in violation of the minimum separation criterion. In this case, the rods were separated by one inserted rod. The reactor was shut down by a safe margin; instrumentation verified that no criticality existed during the control rod withdrawals, and there was no release of radioactivity or damage to the facility.

Even if criticality had occurred in either of these two events, it is extremely unlikely that any impact on the general public would have resulted. Automatic safety features were available and functioning, such as automatic control rod insertion activated by automatic monitors. Both incidents involved personnel error and procedural inadequacy; automatic protection devices can mitigate the larger possible consequences of criticality, but, during refueling and rod maintenance plant personnel are usually on the refueling floor and could be subjected to radiation exposure in the event of inadvertent criticality. For this reason and others, the NRC places serious emphasis on maintaining strict personnel controls during refueling.

The cause of these events was, as noted, both human error and procedural deficiency which represent a potentially recurring problem. The licensees in these instances have instituted programs to improve personnel performance and management control and have modified rod maintenance procedures accordingly. The NRC investigated both events, issuing a

Notice of Violation to the licensee and imposing a civil penalty.

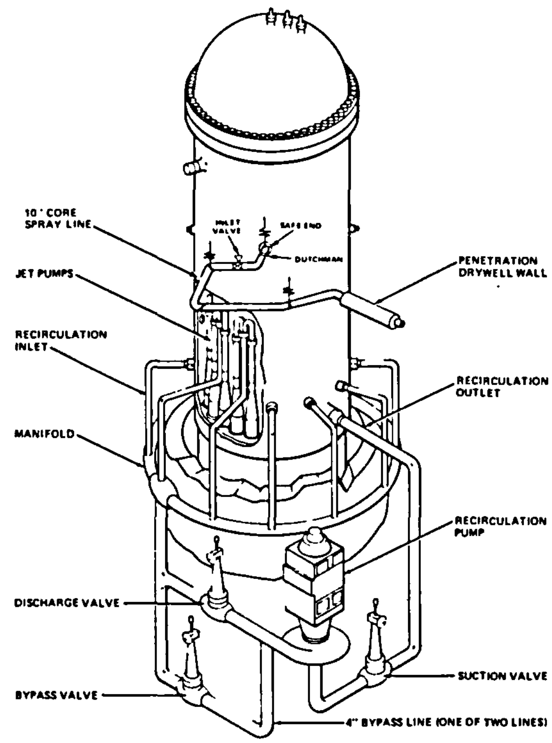
Pipe Cracks in BWRs

A series of events raising questions of a generic nature constitute this occurrence in boiling water reactors at six different facilities. Hairline pipe cracks first showed up in September of 1974 at the Dresden 2 plant, the Quad-Cities Station, Unit 2, and the Millstone Nuclear Power Station, Unit 1, in New County, Conn. Again, in December 1974, similar defects were observed at the first two facilities; and yet again, in January 1975, at those two plants and at the Peach Bottom Atomic Power Station, Unit 3, in York County, Pa.

The existence of small hairline cracks was first detected by a leakage monitoring system at Dresden 2; the cracks were discovered in 4-inch bypass pipes in the primary coolant recirculation loop. Subsequent inspection disclosed similar cracks at the other installations and in 10-inch lines in the core spray system of one BWR facility. The cracks were determined to be of the type which propagate slowly and are readily detectible before they could lead to large leaks or pipe rupture. All pipes involved were of stainless steel.

Though no immediate hazard was presented by the cracks, the fact that they affect one of the primary boundaries for the containment of radioactive material warranted prompt study and action.

After the first cracks were detected in September 1974, the Atomic Energy Commission directed that all boiling water reactor licensees with bypass systems similar in design to those found defective conduct examinations of welds in the bypass lines within 60 days. More stringent coolant leakage limits were also imposed. No new cracks were found. When new cracks that did not exist in September were found in December at the Dresden plant, the AEC directed all BWR licensees



BWR REACTOR COOLANT SYSTEM AND CORE SPRAY

to reevaluate their September findings, conduct more examinations at the next scheduled shutdown, and observe even more stringent leakage limits. Cracks were found in four reactors, the cause or causes still unknown. In early January 1975, the AEC formed a special study group to coordinate and intensify the investigation of causes.

On January 28, small cracks were found in two 10-inch core spray system pipes at Dresden 2. On January 30, 1975, the NRC directed the operators of all operating BWRs to conduct an inspection within 20 days of all circumferential welds in each core spray loop within the boundary of the reactor coolant system, plus a representative sampling of welds in other stainless steel piping. No cracks were found.

On February 5, 1975, a hearing on the NRC's discoveries and actions was held by the Joint Committee on Atomic Energy and the Senate Committee on Government

Operations; all occurrences and their possible ramifications were set out at the hearing. There had been no releases of radioactivity as a result of the cracks and, even in the remote case of pipe failure, redundant core cooling systems were available and functioning at all plants involved.

The cause of the cracking, as determined by the special NRC study group in late 1975, was corrosion resulting from a combination of stress, water chemistry and the type of material used: a type of austenitic stainless steel (type 304) which loses some of its resistance to corrosion in heat-affected zones adjacent to welds in relatively small diameter, thin-walled applications. The study group recommended a continuing program of surveillance for cracking; replacement of cracked pipes with others made of material less susceptible to the kind of corrosion that caused the initial cracking; further investigation into possible changes in operating procedures to reduce the relatively high level of oxygen contributing to corrosion in the pipes. Although additional cracks may develop in the future, the study group reported that they do not pose a threat to public health and safety because they can be detected by periodic inspection or sensitive leak detection equipment. In no instance was the structural integrity of the cracked pipes affected; all affected pipes were replaced by the licensees and, where system operation was a major factor, operational procedures were revised.

In-depth investigations of the problem initiated by the NRC are continuing, including research on corrosion susceptibility of structural materials, residual and operating stress measurements, welding and fabrication practices, and nondestructive testing. The Energy Research and Development Administration is sponsoring efforts in the same area. Foreign countries have been apprised of the action taken by NRC licensees and asked to convey results of similar examinations conducted at their BWR facilities. Cracks have been found in the core spray piping of two reactors located in Japan; NRC is in touch with other nations currently conducting tests.

Fuel Channel Box Wear

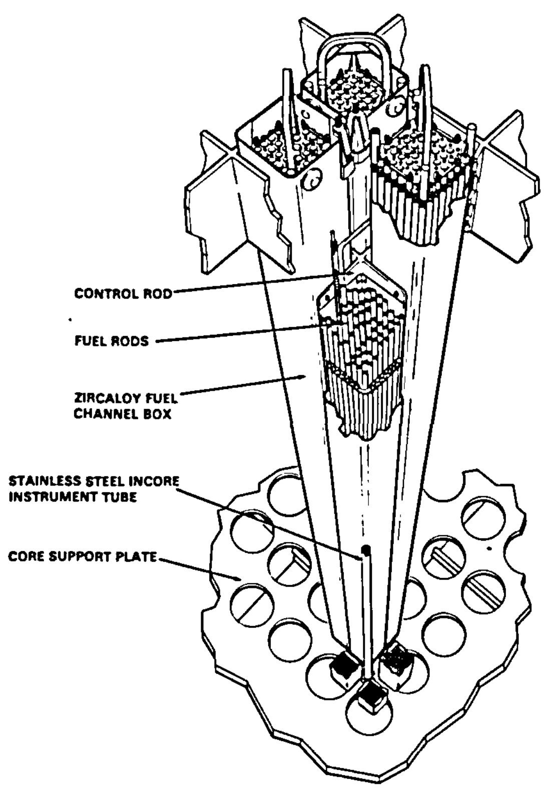
Another series of events, with generic implications, involves fuel channel box wear at boiling water reactors. First notice of a problem came on April 17, 1975, from the Duane Arnold Energy Center, Unit 1, in Linn County, Iowa. Other plants subsequently affected are: Cooper Nuclear Station in Nemaha County, Neb.; Peach Bottom Atomic Power Station, Units 2 and 3, in York County, Pa.; Browns Ferry Nuclear Power Plant, Units 1 and 2, in Limestone County, Ala.; Brunswick Steam Electric Plant, Unit 2, in Brunswick County, N.C.; Edwin I. Hatch Nuclear Plant, Unit 1, in Appling County, Ga.; James A. FitzPatrick Nuclear Power Plant, in Oswego County, N.Y.; Vermont Yankee Generating Station, in Windham County, Vt.

The General Electric Co. reported to the NRC that excessive wear and damage to some fuel channel boxes adjacent to the incore instrument tubes had been found in one class of boiling water reactor (BWR-4) by a foreign operator. The thin-walled metal fuel channel box encloses a bundle of fuel rods; one of its purposes is to guide the flow of coolant water around the fuel rods. It was determined that the wear was occurring as the result of the vibration of the in-core instrument tubes. Operation of a plant for extended periods with high wear rates could lead to penetrations of the channel wall, allowing too much of the reactor coolant to bypass certain fuel rods and thereby reducing thermal safety margins. Loose channel box fragments could also cause local coolant flow blockage and possible overheating of some of the fuel rods.

Prompt corrective action was taken in all instances cited, and there was no impact on public health and safety. The margin of safety was assured in all cases by reducing local power generation and permissible thermal-hydraulic operating limits and by reducing reactor coolant flow to decrease instrument tube vibration.

Surveillance of anomalous noise on the in-core instrument readings, indicative of vibration, was increased, and limits were placed on the permissible magnitude of such noise. In the unlikely event of an accident, existing safety systems were in readiness to assure protection of the public health and safety. There was no release of radioactivity in any of these occurrences.

The problem is significant with respect to maintaining safety margins for the reactor fuel cladding, a principal barrier for the retention of fission products



BWR FUEL ASSEMBLIES, CHANNEL BOX AND INSTRUMENT TUBE

(radioactive material formed within the fuel cladding during the fission process).

The cause of the channel box wear is, as noted, vibration over a period of time; the vibrations are set up in the incore instrument tubes by water turbulence. A high velocity flow through holes in the lower core support plate, which are intended to permit a certain amount of coolant to flow outside the channel boxes, is the basic cause.

Licensees have acted to reduce the magnitude of vibration of the tubes by reducing primary coolant flow, or they have plugged the holes in the lower core support and replaced damaged fuel boxes where excessive wear was discovered. The vendor, General Electric, is seeking a permanent correction. The NRC has conducted safety evaluations, meanwhile, at all affected plants and, in some cases, issued orders restricting operations until corrective action is completed.

Feedwater Flow Instability—Water-Hammer

Also of generic importance was a series of events involving the phenomenon called "water-hammer" in pressurized water reactors. The problem was experienced as far back as 1972 at the Surry Power Station in Surry County, Virginia; in 1973 at the Turkey Point Station, Unit 3, in Dade County, Florida; the Robert Emmett Ginna Nuclear Power Plant in Wayne County, New York; and the Indian Point Station, Unit 2, in Westchester County, New York; in 1974, again at the Turkey Point Station, in Unit 4; and in May 1975 at the Calvert Cliffs Nuclear Power Plant, Unit 1, in Calvert County, Maryland.

In pressurized water reactors, an essential part of the secondary water system (nonradioactive) is the feedwater system. This system returns water in the form of condensed steam from the main condenser to the steam generators and maintains the water inventory in the secondary system. Each PWR has at least two steam generators. Loss of the feedwater system by pipe or valve failure could affect the ability of the plant to cool down after a reactor shutdown, though auxiliary systems are provided as backup.

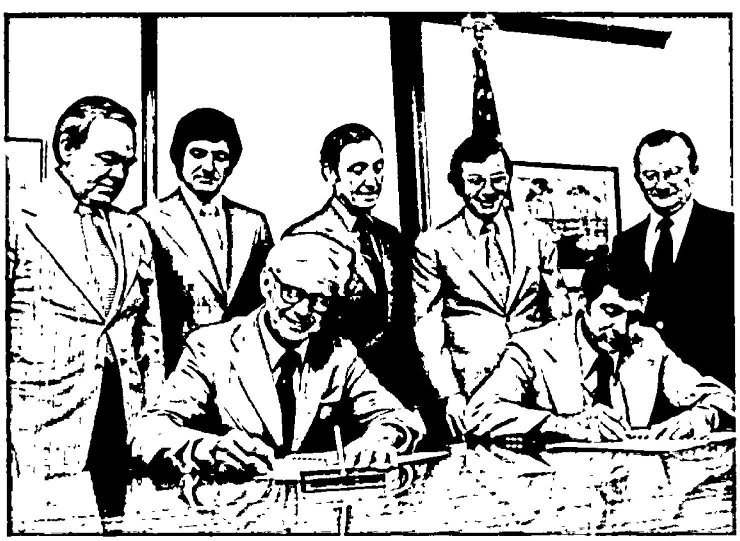
Water-hammer occurs when steam replaces water in the feedwater distribution piping (sparger) or in the feedwater inlet nozzle of the steam generator. This happens when the steam generator water level drops below the level of these components. Restarting feedwater flow causes condensation of the steam and is one of the factors inducing water-hammer. Other factors may be involved and are being sought. Feedwater flow instability, leading to water-hammer, can damage feedwater system piping and associated components; it occurred with varying severity at the above-named plants, usually after restarting feed flow following an operational adjustment required by some abnormal condition, such as a rapid change in the steam generator water level. In the remote instance that both the normal and auxiliary feedwater systems should be lost to several steam generators at once, the capability for plant cooldown could be affected. The development of design and operational modifications to reduce water-hammer to a minimum is clearly indicated. Termination of feedwater flow to several steam generators, however, has not occurred and, in none of the events cited, was radioactivity released or satisfactory safety margins compromised.

At plants where the phenonmenon has occurred, corrective actions have been taken. These include changing the feedwater piping arrangement, modifying the feedwater distribution ring or steam generator refilling procedures, or limiting refill flow rate to reduce condensation. In early 1975, the NRC contacted all PWR reactor licensees requesting a review of the potential for water-hammer in their systems and its potential consequences. Each plant's experience has been reported and a technical study made in a continuing attempt to predict the phenomenon and correct it.

Expanding Confirmatory Research

Advancing Frontiers of Knowledge

When it created a separate Nuclear Regulatory Commission, Congress recognized the need for an independent capability to develop and analyze technical information on reactor safety, safeguards and environmental protection as a basis for licensing and other decisions in the regulatory process. The Energy Reorganization Act of 1974, therefore, provided for an Office of Nuclear Regulatory Research



NRC Chairman William A. Anders (seated at right) and ERDA Administrator Dr. Robert C. Seamans met on August 8, 1975, to sign the memorandum of understanding between the two agencies for performing research at the Loss of Fluid Test facility (LOFT) in Idaho. The memorandum provides for completion of LOFT by ERDA and technical direction of research programs by NRC. The underlying provisions of the memorandum will serve as a basis for several other research agreements with ERDA. Observing the signing are, from left: Herbert J. C. Kouts, NRC; Raymond G. Romatowski, ERDA; Commissioner Marcus Rowden, NRC; Commissioner Victor Gilinsky, NRC; and Lee V. Gossick, NRC.

within NRC to perform research, characterized as "confirmatory assessment," which would relate specifically to regulatory decisions for the safe and environmentally compatible operation and protection of nuclear facilities and materials.

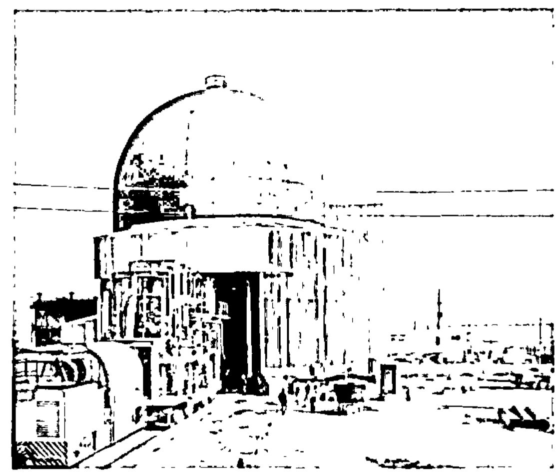
Such research, Congress directed, must be clearly distinguished from the "research and development" functions assigned by the same legislation to the Energy Research and Development Administration (ERDA). NRC would have access to all data required to assess the areas under its regulatory purview, and ERDA and other Federal agencies would cooperate with and support NRC to ensure that the research undertaken by its new regulatory research office would be fully implemented. The bulk of NRC research would be performed at ERDA facilities. In August of 1975 a Memorandum of Understanding was signed by NRC and ERDA outlining the general principles for NRC performance of research at the Loss-of-Fluid Test (LOFT) facility in ERDA's Idaho National Engineering Laboratory. The provisions of that memorandum will serve as a basis for other interagency agreements regarding the conduct of NRC research at ERDA facilities.

SCOPE OF PROGRAMS

The 1974 Act directed the transfer from AEC to NRC of resources required to carry on the Reactor Safety Research Program, and provided for other regulatory research necessary to the performance of NRC functions. In implementing these research responsibilities during 1975, NRC continued the two primary programs of Reactor Safety Research—light water reactor safety research and advanced reactor safety research—and initiated planning and coordination needed to accomplish regulatory environmental, fuel cycle and safeguards research programs.

Reactor Safety Research

The goal of the Reactor Safety Research program is to develop an independent basis and means to reliably and credibly analyze the course of events in hypothetical nuclear reactor accidents and to estimate the consequences of such accidents. Sufficient safety data exist to permit establishment of conservative requirements and safety margins for licensing nuclear power plants. NRC reactor safety research is directed to refining and reducing the allowable uncertainties in the data, in order to better define and quantify the conservative design and safety margins that must be used because of these uncertainties.



Loss of Fluid Test Facility (LOFT) at ERDA's Idaho National Engineering Laboratory, Idaho Falls, Idaho. This 55-megawatt thermal (MWt) pressurized water reactor permits NRC to study postulated nuclear reactor accidents and to derive data from them for use in analyzing hypothesized Loss of Coolant Accident conditions in full-scale nuclear power pressurized water reactors. Shown here is the LOFT reactor being shunted into its specially designed containment structure.

The Reactor Safety Research program is proceeding on two interlocking approaches: experimental programs, and analytical model development. The experimental programs generate the independent data base for developing and validating the analytical models. The models, in turn, are used to extrapolate between laboratory scales or conditions and full-scale reactors, and the validity of the extrapolation is tested through further integral experiments.

This program attempts to develop methods of analysis by which the safety of reactors can be independently assessed by NRC, and to provide information and methods needed to achieve safe operation. Both water reactor safety research and advanced reactor safety research—the two major sub-activities of the Reactor Safety Research program—are discussed in detail later in this chapter.

Environment and Fuel Cycle

The Environmental and Fuel Cycle Research program will develop an independent capability to assess the safety and environmental impact of such activities as fuel fabrication, fuel reprocessing, uranium conversion and enrichment, nuclear materials transportation, and waste management. The two focal points of the program are (1) the potential impacts on the environment of normal operation of non-reactor nuclear facilities and processes, and (2) the adequacy of safety systems to control other than normal behavior. Data and information developed will be important source material for NRC's development of regulatory guides and standards affecting the fuel cycle, and will enhance the licensing review process where there is a continuing requirement to refine and confirm estimates of the impact of plant performance on man and the environment.

Safeguards Research

The Safeguards Research program is a vital component of the overall NRC Safeguards program. Safeguards measures which presently assure adequate protection of nuclear materials and facilities must be examined and refined constantly as the nuclear industry expands and the potential for subversion and terrorist activities grows. Particular research attention is devoted to problems attendant on the projected near-term use of large quantities of plutonium and other strategic materials in private industry operations. The program, closely tied to research on the light water reactor and advanced fuel cycles, will provide important information needed for standards development, licensing activities, and subsequent energy center evaluations.

REACTOR SAFETY RESEARCH

The overall objective of the Reactor Safety Research program is to develop analytical methods that can confidently be used by NRC to assess the safety of nuclear power reactors on an independent basis. Within this objective, several specific research challenges are addressed:

- Establishing and testing, on a sound engineering base, improved analytical methods of safety analysis;
- Improving the engineering data base concerning the conditions that might trigger a reactor accident;
- Extension and improvement of independently-derived technical information against which to compare applicant or licensee safety justifications in licensing actions, and
- Reduction of present margins of uncertainty in the data and models so that the degree of conservatism applied to safety assessment may be further quantified.

Importance of Computer Codes

The methods used by NRC to assess the effectiveness of safety systems in nuclear power plants under hypothetical accident conditions are based on computer codes developed from experimental data. Computer codes are methods of solving mathematical equations—in this case, those which embody the engineering description of the ways in which the safety systems work. These codes model and analyze the important parameters of a reactor system subjected to hypothetical accidents, such as a loss-of-coolant accident (LOCA). Code development and application are given high priority in all NRC reactor safety research activities. References to the development and use of codes appear throughout this chapter, and are discussed in detail in the section titled "Analysis Development."

Water Reactor Safety Research

Water Reactor Safety Research is directed at providing a capability for independent confirmatory assessment of the safety of the current generation of nuclear plants under postulated accident conditions. The research data and analytic methods applied to the assessment of hypothetical nuclear plant accidents will result in a greater measure of confidence that the margins of safety identified in the licensing review are well defined and quantified. The program is divided into four categories: Systems Engineering, Fuel Behavior, Analysis Development, and Metallurgy and Materials.

SYSTEMS ENGINEERING

Safety research in systems engineering is addressed primarily to the study of postulated loss-of-coolant accidents in reactors and the effectiveness of emergency core cooling systems (ECCS). In general, such research is conducted through two types of tests: (1) Separate Effects tests to obtain data on those portions of a postulated accident where transient heat transfer and fluid flow phenomena are isolated, thus reducing the number of test variables and simplifying understanding of those complex phenomena, and (2) Integral Systems tests to study combined phenomena representing an entire postulated accident sequence, both to assess the significance of knowledge gained from separate effects tests and to confirm the predictive capability of computer codes.

Separate Effects Tests

Separate effects studies for pressurized water reactors (PWRs) generally conform to three sequential phases of a postulated loss-of-coolant accident: (1) the blowdown phase in which the pressurized coolant water is suddenly changed to a mixture of water and steam as the result of depressurization from a hypothetical break in a coolant pipe, (2) the steam-water mixing phase during which steam leaves the pipe-break and the emergency coolant either enters or bypasses the nuclear core, and (3) the reflooding phase in which emergency coolant enters and continues to cool the reactor core as a substitute for the lost primary coolant.

Blowdown heat transfer experiments. These experiments determine the accuracy of heat transfer correlations now used in calculating postulated accidents in PWRs. Excessively conservative values (low heat transfer) during the depressurization phase, for example, would lead to higher fuel temperature predictions during the remainder of the accident sequence than would actually occur.

Research provides data to determine coolant changes during depressurization and the time to reach critical heat flux (the overheating due to the formation of an "insulating" blanket of steam, and consequent loss of contact with coolant) as influenced by variations in power, system pressure, coolant flow and break location.

During the past year, fabrication and pre-operational testing of the Thermal Hydraulic Test Facility—a pressurized water loop at Oak Ridge, Tenn.—was completed in preparation for depressurization or blowdown tests. The loop contains an electrically heated 49-rod bundle (12-foot heated length) with flow rates up to 700 gallons per minute, and system pressure to 2,250 pounds per square inch with the capability to perform depressurization tests simulating flow from a double-ended pipe rupture. The 49-rod bundle is designed to simulate the geometry of an actual 49 nuclear fuel rod bundle in a pressurized water reactor.

Plans to begin testing with the electrically heated rod bundle were delayed by failure of a pump seal in April. The problem was solved in September and isothermal tests began. Testing of the first 49-rod bundle will continue through 1976.

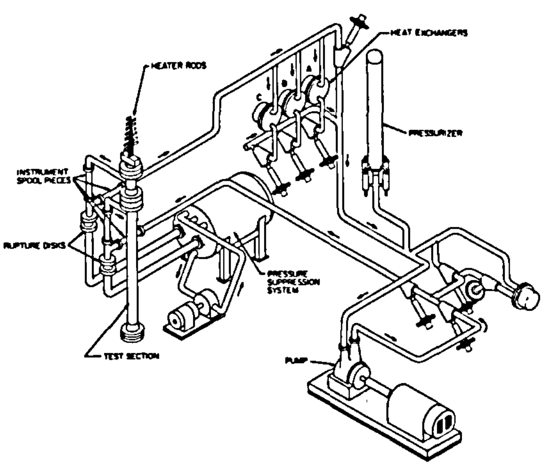


Diagram shows test section (which contains 12-foot electrically-heated 49-rod bundle), pump, pressurizer, and auxiliary equipment for conducting transient heat transfer experiments during simulated PWR depressurization from 2250 pounds per square inch pressure.

Depressurization heat transfer experiments applicable to boiling water reactors (BWRs) are conducted by the General Electric Co. (GE) in San Jose, Calif. under a program sponsored jointly by NRC, GE and the Electric Power Research Institute (EPRI). The program investigates separate effects as well as system response to a postulated loss-of-coolant accident during early depressurization. The test facility for the BWR experiments contains internal jet pumps, a steam separator and an electrically heated, 49-rod core bundle representative of a BWR fuel assembly. The test apparatus volume, flow paths, and initial operating pressures and temperatures are modeled on normal operating behavior and predicted loss-of-coolant accident behavior for boiling water reactors. Results of tests with an initial core bundle are under study. Testing with the second 49-rod bundle was completed in November 1975.

Both the PWR and BWR depressurization heat transfer programs include analytical efforts employing the latest loss-of-coolant accident computer codes. By comparing predictions with results from the tests and interpreting the reasons for differences observed, a basis will be established for further refinement of the analytical models and their further use in reactor accident analysis. Best-estimate calculations, based on the most realistic assumptions available, are compared with calculations using the conservative assumptions contained in NRC licensing criteria. A continuing comparison of results from the two types of calculations provides one measure of the degree of conservatism in the licensing models. Information obtained in these studies to date confirms that analytical methods used for licensing BWRs provide an adequate safety margin for predicting core cooling behavior during blowdown. PWR blowdown data will be obtained during the coming year, although integral system effects data obtained in the Semiscale program (see section on Semiscale) have demonstrated that analytical methods can describe potential PWR accident behavior.

Steam-water mixing tests. Separate effects tests on steam-water mixing phenomena and emergency core cooling (ECC) downcomer* penetration (into the lower region, or plenum, of the core) were conducted for NRC at the Battelle-Columbus Laboratories (BCL) and Creare, Inc. The tests were conducted with 1/15-scale models (simulating a 1,000-MWe nuclear pressure vessel)

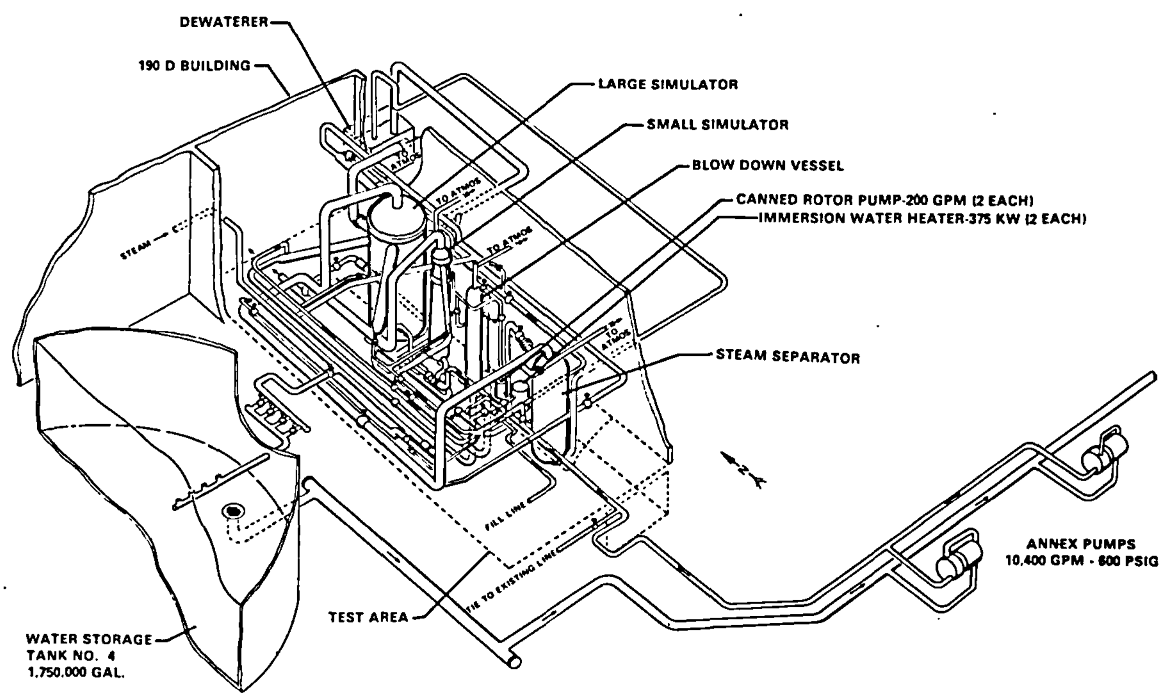
*The downcomer is the space between the nuclear core barrel and the pressure vessel.

with three intact loops and one piping circuit simulating a broken water inlet pipe ("cold leg"). Calculations show that depriving the nuclear core of the comparatively cooler inlet water leads to a higher potential temperature in the core; thus studies on the "cold leg" break are emphasized. The tests were directed at assessing the degree of ECC bypass (failure to reach the lower core region) and lower plenum filling under conditions predicated for part of a hypothetical loss-of-coolant accident.

The FY 75 results from the BCL and Creare programs have provided a data base for ECC downcomer penetration (at low pressures) which is being used to develop lower plenum filling models and bypass models. The 1/15-scale tests have corroborated limited results obtained in a 1/5-scale vessel at Combustion Engineering's Windsor, Conn., facilities while also revealing the need to simulate cold leg arrangement carefully, in subscale tests. These programs are being extended to include a cylindrical vessel and improve the transparent vessel for studies on steam flow.

The major experimental program in the area of emergency core cooling bypass and steam-water mixing phenomena, is the Plenum Filling Experiment (PFE) program at Battelle's Pacific Northwest Laboratories (PNL). This program will use both 1/5 and 4/5-scale vessels, representing the four-loop configurations of commercial pressurized water reactor systems, in simulations of steam-water flow conditions predicted for a hypothetical loss-of-coolant accident. The PFE test-bed design has been completed and some construction activities were undertaken in 1975. Funding limitations continue to pace facility fabrication activities,

PLENUM FILLING EXPERIMENT



The Plenum Filling Experiment, located at Battelle Memorial Institute's Pacific Northwest Laboratories at Richland, Washington, is NRC's major research program in the study of steam-water mixing phenomena and emergency core cooling bypass (failure of cooling fluids to reach the lower core region). The 1/5-scale and 4/5-scale vessels are used to represent the four-loop configuration of commercial pressurized water reactor systems in simulating predicted steam-water flow under hypothetical accident conditions.

however, and current schedules project the start of testing in 1978.

Basically, these tests are to provide data on the following important research topics:

Steam-water mixing—in the blowdown phase of the postulated LOCA, some of the water in the reactor vessel flashes to steam. The degree of steam and water mixing influences the motion and cooling capabilities of the water.

ECC bypass—during the blowdown phase it may be possible for steam to block the incoming emergency core cooling water and cause this water to bypass the reactor core and flow out the outlet water pipes.

Downcomer penetration—this refers to how far emergency core cooling water penetrates down into the annulus or "downcomer" between the core barrel and reactor pressure vessel.

Reflooding experiments. To produce additional data to estimate performance of nuclear reactor fuel rods during the reflood phase of a postulated loss-of-coolant accident, existing Full Length Emergency Cooling Heat Transfer (FLECHT) facilities at Westinghouse in Pittsburgh, Pa. are being utilized under joint sponsorship by NRC, EPRI and Westinghouse. These facilities are similar to those used in the initial FLECHT studies conducted several years ago by the AEC, but employ improved techniques to measure the coolant distribution below, inside, and above the 12-foot length, 100-rod bundle, which uses electric heat to represent a portion of the reactor core. Two kinds of heat distribution are used to represent different stages of reactor core lifetimes. One has the heat generation in the bundle peaked near the center. The other has the heat generation peaked near the top. Tests with the center-peaked bundles were completed in August 1975, and tests with the top-peaked bundles are scheduled for completion in May 1976.

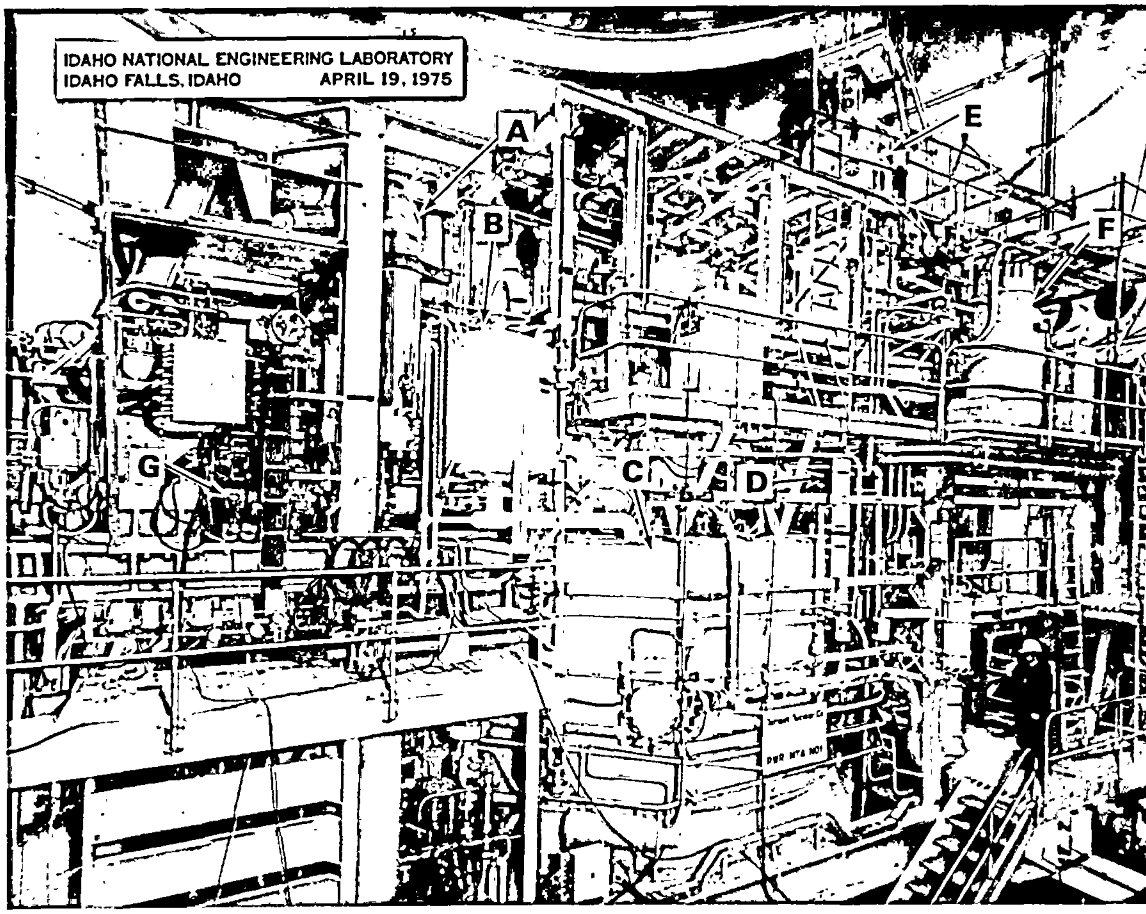
Integral Systems Tests

Integral systems studies for postulated PWR accidents now are being conducted in the nonnuclear Semiscale Facility and will be extended to nuclear core studies in the LOFT facility. These facilities and experiments are described below.

The Loss-of-Fluid-Test (LOFT) facility is a 55-megawatt thermal (MWt) pressurized water reactor at ERDA's Idaho National Engineering Laboratory. The facility is designed to accommodate study of nuclear, thermal-hydraulic, and structural processes occurring during a postulated loss-of-coolant accident in a pressurized water reactor. Through the various separate effects tests and the model development effort it will be possible to determine the effects of scaling, thereby permitting development of a relationship between the 55-MWt LOFT and a 3,500-MWt nuclear power plant. Once the basic PWR scaling is established, it will be possible to develop similar scaling for BWR's based on the various BWR Separate Effects tests.

The major objective of the LOFT integral test program is to provide data to evaluate and improve the analytical methods now used to predict the postulated LOCA response of a large pressurized water reactor. Thus, LOFT has been designed to perform a number of loss-of-coolant experiments and provide measurements of system response. These measurements will be compared with pretest predictions to check the capability of existing computer codes. The first nonnuclear test in LOFT will be run in early 1976.

The reactor coolant system has one active, heat-dissipating, operating loop which models the three unbroken loops of a four-loop plant, and a special blowdown loop which contains special quick-opening valves to simulate hypothetical pipe-break conditions postulated for a large commercial reactor accident. The blowdown loop will discharge into a suppression tank



The Loss-of-Fluid Test (LOFT) facility at the Idaho National Engineering Laboratory is now in the final stages of construction. Some of the principal features of the test assembly are: A, steam generator simulator; B, expansion tank for reactor shield cooling water; C, shield around reactor vessel; D, intact hot leg; E, steam generator; F, pressurizer; and, G, quick-opening blowdown value.

designed to provide back pressure conditions typical of current PWR containments.

The ECCS components and emergency coolant injection configurations of LOFT are similar to those used in large PWRs, and are designed to provide flow rates scaled to a large PWR. Three systems are provided for emergency coolant injection: (1) gas-driven water-filled accumulators which can inject a large volume of water into the reactor system; (2) high-pressure injection pumps which can produce a small flow of high-pressure coolant for small breaks; and (3) low-pressure injection pumps to provide large volumes of water for core cooling after a major primary system rupture experiment. The primary coolant system and ECCS are extensively instrumented, and ECCS injection points and flow rates are easily varied for experimental purposes.

The first nuclear reactor core in LOFT will be 5.5 feet long, 2 feet in diameter, and will contain 1,300 PWR-type fuel rods. Should data be desired on full-length reactor cores, the LOFT reactor is designed to accommodate cores up to 12 feet in length. Increased length and other design changes will be considered in future replacement cores. The system is instrumented to measure temperatures, flows, pressures, and coolant levels inside and outside the reactor vessel. The core fuel rods are instrumented with high temperature thermocouples* which were designed and fabricated specifically for LOFT.

The LOFT facility is now in final stages of construction. All system acceptance testing has been completed, as has steady state flow testing at reduced temperature and pressure. Steady state flow testing (referred to as hot functional tests) at 540°F and 2,250 pounds per square inch has been initiated to confirm normal operating design conditions.

A nonnuclear test series, scheduled to start early in 1976 following the hot functional tests, will provide information for evaluating the effects of core nuclear and heat transfer, steam generator heat transfer, and primary system fluid differential temperature. The nonnuclear tests also will establish system behavior with and without emergency core cooling delivery to differentiate the effect of ECC delivery on system behavior.

During the past year, planning for the nonnuclear tests was completed and detailed planning activities for the nuclear tests were initiated. Pretest predictions prepared for each nonnuclear test took into account the results obtained from the LOFT duplicative testing in the Semiscale facility.

The Semiscale facility, so named because it is a scaled model of LOFT, is a thermal-hydraulic test apparatus, basically a nonnuclear electrical simulator of a pressurized water reactor. Semiscale is designed to provide thermal-hydraulic data to aid in development of computer models to describe LOCAs and to provide similar data to be used in LOFT test planning and instrument development. Semiscale consists of a vessel (analogous to the reactor vessel), inlet and outlet water lines, pressurizer, steam generator and containment simulation system. Design, fabrication and construction activities for the 1-1/2** Loop MOD-1 system were completed during the first quarter of the fiscal year. The new MOD-1 system represents a significant improvement over previous Semiscale configurations in that it more closely models and incorporates the principal configurational features of LOFT. Major system modifications resulting from this redesign effort included:

(1) Fabrication, assembly and installation of a 1.6-MWt electrically heated 40-rod core simulator in which the 5.5-foot length of each rod is the same length as the present LOFT nuclear fuel rods;

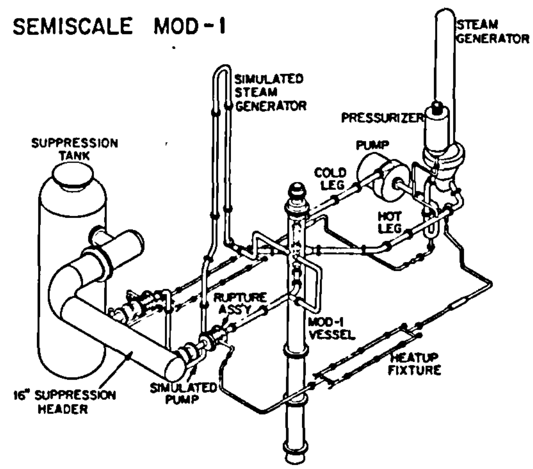
(2) Fabrication and installation of an active steam generator in the operating loop to model the heat dissipating effects taking place during a hypothetical LOCA in the three loops which are postulated to remain intact; and

(3) Fabrication and installation of pump and steam generator simulators in the loop in which the rupture is postulated to occur. The simulators were designed and scaled to those used in LOFT.

The first test series following completion of the facility in 1974 consisted of a duplication of the planned LOFT nonnuclear test series to provide planning data for the LOFT tests. Five of the seven tests in the series simulated the full rupture of the inlet cooling line of a pressurized water reactor. The tests were initiated from system coolant conditions of 540°F and 2,250 pounds per square inch. Emergency core coolant injection was a major test parameter for four of these tests. The remaining two tests, both involving emergency core coolant injection, simulated a rupture of the outlet cooling line in a pressurized water reactor. These tests were

*A thermocouple is a device for measuring temperature by electrical means.

**The term "½ loop" refers to the loop which is equipped with a simulated pump, steam generator, etc., which permits its use in simulating a pipe-break.



The Semiscale facility is a one-dimensional nonnuclear representation of a pressurized water reactor facility. The 1.6-MW core is contained in the MOD-1 vessel. The core is 5.5 feet long and 7 inches in diameter and contains 40 electrically heated fuel rods. The intact loop includes an operating pump, pressurizer, and steam generator. The loop, which is designed to be broken in simulation of a pipe break, contains a simulated steam generator and a simulated pump. The suppression tank accepts the effluent from the broken loop.

initiated from 540°F and 1,650 pounds per square inch. In all cases, test results were consistent with pre-test predictions of such parameters as flow, temperature and pressure. Data from these tests will be used in conjunction with information from the LOFT nonnuclear tests to establish the influence of physical size (i.e., Semiscale vs. LOFT vs. a commercial reactor) on postulated LOCA behavior. The tests also provided a baseline for understanding Semiscale system response without the complexity of heat addition.

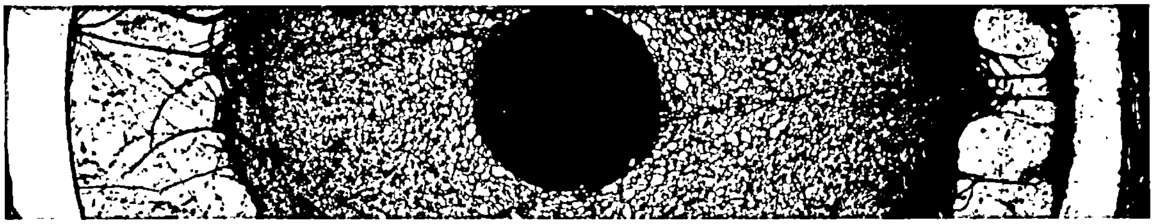
Investigations into the Semiscale system response with core heat were initiated by NRC in the spring of 1975. The first three of a six-test blowdown heat transfer series were completed before the end of June 1975. The primary purpose is to evaluate the system and core thermal response resulting from core heat addition. Results of the Semiscale blowdown heat transfer tests represent the first data from an integral system test during a loss-of-coolant experiment. Comparison of the measured results with available safety analysis codes established a capability to calculate the coupled thermal-hydraulic response of a complex system in a rapid transient.

The first three tests also demonstrated the need for greater precision in calculating the amount of fluid entering or leaving the core areas during a transient because this amount of fluid influences the amount of heat being removed from the core region. Subsequent tests in the series will examine these preliminary conclusions further.

FUEL BEHAVIOR

The escape of radioactivity from nuclear power plants is prevented, in part, by barriers designed into the structural and operational features of the plants. One such barrier is the cladding around the nuclear fuel pellets, which is affected by both the fission products themselves and the intense heat generated by the fission process. An important goal of reactor safety research is improved understanding of the response of fuel element pellets and cladding to a postulated nuclear accident. NRC examinations in this area involve a combination of laboratory studies and in-pile tests, which are experiments conducted in an operating nuclear reactor to observe actual fuel response: These activities, in turn, provide data for predicting fuel behavior in accident conditions through the development and verification of analytical codes.

Power Burst Facility. In-pile testing is conducted in the Power Burst Facility (PBF), a 40-MWt test reactor for fuel damage studies at the Idaho National Engineering Laboratory. PBF is designed to simulate such postulated accidents as flow blockage, power-cooling mismatch, and power excursions by means of a nuclear-fueled driver core which provides neutrons to heat up test fuel rods placed in a centrally located in-pile tube. There was some difficulty in the operation of this facility during initial



This photomicrograph shows a cross section of a fuel rod following a power-cooling mismatch test performed in PBF. During this successful test, the rod was subjected to power approximately 35% above peak PWR operating conditions while the coolant flow rate was reduced to only about 40% of normal.

The ziracloy cladding of the fuel rod was heavily oxidized as a result of surface temperatures exceeding 2000°F for about 10 minutes during the test. The uranium dioxide fuel pellet shows evidence of restructuring and densification caused by the high temperature (over 4300°F) reached during the test, indicated by the coarse grained structure formed from the original uniformly ultra fine grain structure. The center hole in the fuel pellet indicates the position of a thermocouple that was installed to measure fuel pellet centerline temperature.

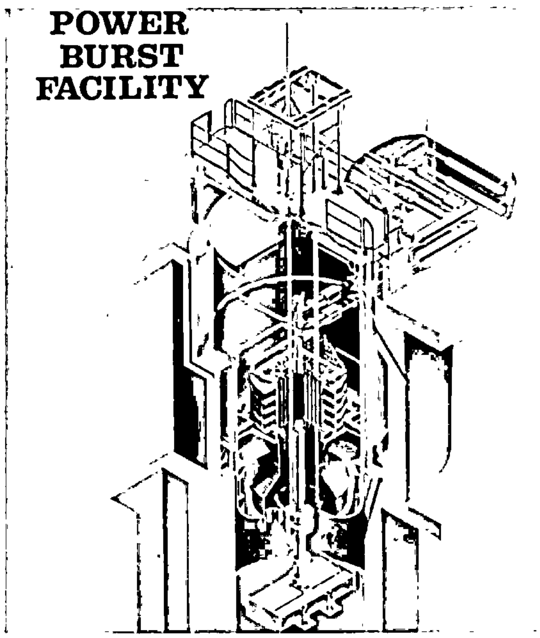
The test has allowed the selection of the best heat transfer model from among several available for use in predicting the behavior of fuel rods during postulated accidents.

operations at power last year, when it was found that the power level of PBF fluctuated above and below an established power level. A series of nuclear diagnostic tests indicated that the fluctations were probably due to the motion of the fuel pellets in the specially designed PBF fuel rods. The problem has been brought under control and stable reactor power levels up to 33 megawatts can now be attained. On more than 20 occasions, the plant and in-pile loop were operated in support of nuclear tests.

A test was performed in PBF to establish facility capabilities for measuring gap conductance by the power oscillation method. Gap conductance, which is a measure of heat transfer across the boundary between the fuel pellet and the cladding, has important implications in predicting fuel overheating during postulated loss-of-coolant accidents. The gap conductance measurement, using a power oscillation technique developed for PBF, has the virtue of not requiring thermocouples within the fuel which might influence fuel crack patterns and thus bias the changes in fuel-to-cladding conductivity related to fuel cracks.

Initial tests of a power-cooling mismatch series also were performed in PBF: a test fuel rod was tested to power levels well past the critical heat flux, where prolonged exposure can cause cladding failure for the given flow rate. Fuel rod centerline and surface temperatures and fuel rod length and internal pressure changes were used to observe the occurrence and subsequent abatement of critical heat flux. Post-irradiation examination of the fuel and cladding microstructures permitted verification of test instrumentation performance. Experimental data are being checked against pre-test prediction as an aid in fuel rod performance model development.

Halden Reactor Project. NRC participates in the Halden Reactor Project of the Organization for Economic Cooperation and Development. Halden, an international facility located in Norway, offers unique capabilities for conducting highly instrumented tests of fuel behavior and fuel changes during actual reactor operation. Ten nations participate in this program. For the U.S. portion, three existing high-burnup test assemblies were obtained for detailed study and analysis of burnup effects on mixed oxide (plutonium recycle) fuel. An experimental data report for those fuel rod tests indicated that the in-reactor thermal and mechanical response



The Power Burst Facility (PBF), located at the Idaho National Engineering Laboratory, is designed to provide experimental data on the response of fuel rods to postulated accident conditions. The PBF consists of an open-tank reactor vessel, driver core region, a central flux trap region with an in-pile tube in which the test fuel is located and a loop coolant system for providing the required system conditions in the test space. The basic objective of the PBF program is to provide information which can be used to resolve fuel behavior issues related to power reactor operation.

of mixed oxide fuel was substantially the same as that of UO_2.

During the past year, two fuel test assemblies for investigating fission gas release from fuel pellets and gap conductance were installed in the Halden reactor. These tests are designed to obtain data on fission gas release following intermediate and long term irradiation and during overpower transients.

Other highlights in the fuel behavior program. A commercial fuel assembly, supplied to the Nuclear Regulatory Commission by the Carolina Power and Light Company, will be used to study the effects of irradiation upon fuel property changes—including property measurements and fission gas release experiments. The initial version of the computer code FRAP-T (Fuel Rod Analysis Program—Transient) for prediction of fuel behavior during normal, abnormal, and accident conditions, has been completed, verified and forwarded to the Argonne Code Center for distribution. The FRAP-T documentation includes a description of the analytical models employed, material properties used in the calculations, and verification of the code through comparison of the computer code predictions with experimental data to document how well the present version of the code can predict given physical phenomena.

Experimental results obtained from laboratory programs in 1975 indicate that zircaloy cladding will undergo considerably less oxidation and deformation during a hypothetical loss-of-coolant accident than the current calculations indicate, and that about 20 percent less fission product decay heat will be produced than is presently calculated. These results begin to quantify the conservatism of the present basis of accident analysis and imply that the cladding will reach lower peak temperatures and suffer less damage than has been assumed.

ANALYSIS DEVELOPMENT

As noted early in this discussion, computer codes form the basis of nearly all research methodologies employed by NRC. Analysis done through computer codes is based on the same experiments and measurements which comprise the engineering descriptions of the methods by which safety systems work, and which, in fact, have provided tests of all parts of the codes themselves before they were constructed.

Complex digital computer codes can be used to compute the time and space dependence of important factors during total or partial accident sequences, and effects which might occur if a full-scale reactor accident ever took place. The credibility of using such codes in reactor safety assessment is based on the success achieved in using the codes to predict results of various reactor safety research separate effects and integral experiments and the validity of extrapolation to full, commercial-size systems. NRC gives code development and application high priority.

RELAP-4. Top priority has been given to improvement of the present system code, labeled RELAP-4. This code has two versions: The "evaluation model", used by the NRC staff in its licensing activities, provides conservative analysis through incorporation of NRC's acceptance criteria. The "best estimate" version, on the other hand, incorporates realistic (not necessarily conservative) mathematical descriptions of the system. This will be especially useful for code verification by comparisons between code predictions and test data.

Improvements to RELAP-4, developed over the past year, were concerned mainly with representation of two-phase phenomena, such as the relative motions of the steam-water phases which are important in loss of reactor coolant water through "small breaks" (pipe breaks which are not total).

The "evaluation model" version of the code was made available to industry and the scientific community during FY 1975. The "best estimate" version of the code is still undergoing some important revisions and improvements. It was "frozen" by the end of December 1975 and will be subjected to extensive verification that will involve comparisons of code results and test data from a variety of integral and separate effects tests. Some of these comparisons may uncover code deficiencies requiring periodic correction on some controlled schedule, approximately once every six months. This process will be repeated until all the deficiencies have been identified and corrected to the extent possible. The code also will be subjected to studies in which the uncertainties in input parameters or models are investigated to see how they influence some of the important results, such as the fuel rod's peak cladding temperature.

Advanced codes. Some known deficiencies of RELAP-4—such as inability to account for unequal velocities of steam and water, thermal non-equilibrium, and multidimensional flows—could not be removed within the present code structure. For this reason advanced loss-of-coolant accident analysis systems codes are being developed at various national laboratories. These are based on a parallel development path concept in which a variety of sophisticated modeling techniques are employed to describe the extremely complex physical phenomena occurring during a postulated accident. The most elaborate advanced LOCA code, (named TRAC) is being developed at Los Alamos Scientific Laboratory for "best estimate" analysis. Brookhaven National Laboratory is developing a somewhat simpler advanced code version (named THOR) which will subsequently be cast into a conservative form for eventual replacement of the present evaluation model version (RELAP-4 EM).

The Reactor Safety Research Advanced Code Review Group was organized in 1975 to examine the premises on which the advanced codes are based and to provide advice and guidance. It is comprised of a number of top experts in the fields of fluid dynamics, numerical analysis, two-phase flow, and heat transfer.

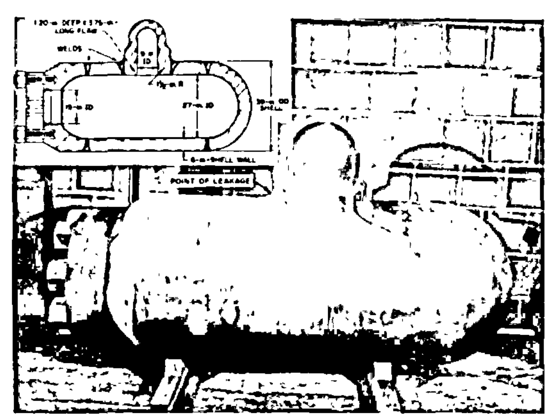
METALLURGY AND MATERIALS

Metallurgy and materials research is aimed at ensuring the continuing integrity of the primary-system pressure boundaries (vessels, components and piping) in nuclear reactors. These heavy-walled vessels, components and pipes must remain intact at all times, since failure could lead to a loss of the normal primary system water used for cooling the nuclear fuel during operation and require the injection of emergency cooling water to prevent excessive heatup of the nuclear core. The ability of the steel vessel, components and piping to retain integrity throughout operating and accident conditions is governed by (1) the material properties and the response of the steel to the reactor environment, and (2) the size and orientation of flaws in the vessel, components or pipeline. NRC research and activities in these two areas during 1975 are discussed below:

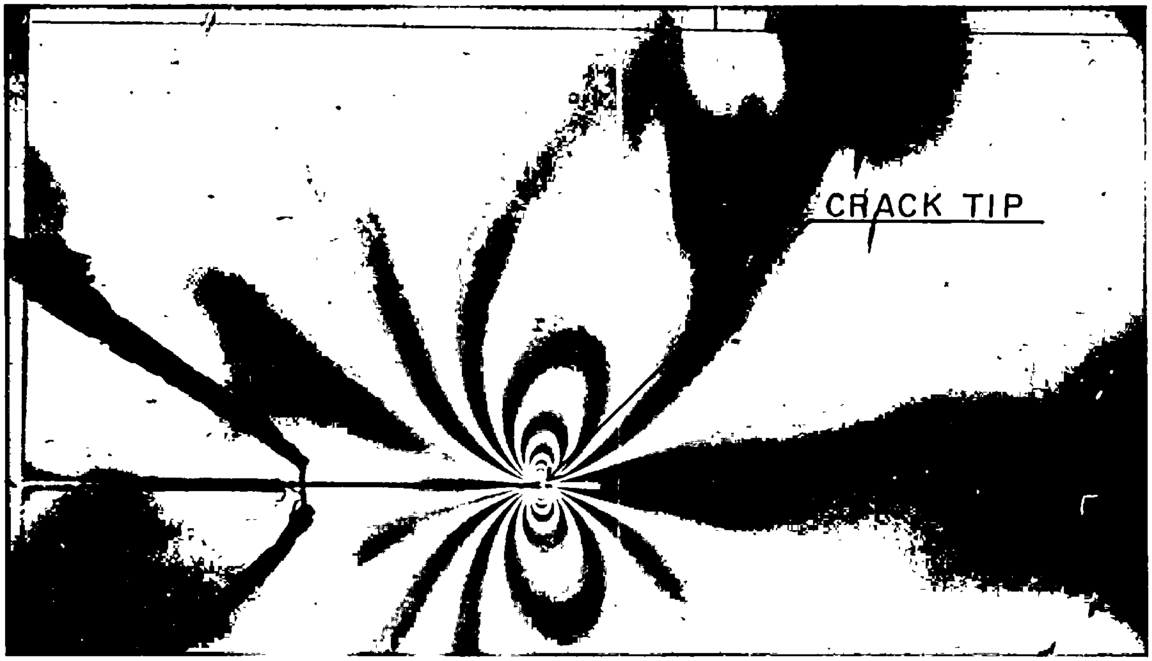
Material properties and environmental response. The research approach for this task area is to formulate analytical procedures for prediction of the behavior of reactor vessels, components or piping under operating and postulated conditions. Then, experiments are performed to test both the steel and the structures to be sure that the predictions give the correct answers. Studies are conducted to produce better understanding of the conditions under which cracks initiate and arrest, how they may grow under fatigue loading, the effects of reactor neutron radiation on properties of steel, and the consequences to vessel integrity from different stresses. The influence of flaws of different sizes is given much importance, and the experimental work is conducted at carefully regulated temperatures and stress levels to ensure that actual operational conditions are approximated as closely as possible.

Size and orientation of flaws. "Structural Integrity Under Applied Stresses" is a program dealing with the behavior of reactor components containing flaws when the components are subjected to stresses typical of operational or postulated accident conditions. Reactor performance under such conditions is predicted by a procedure called analytical fracture mechanics. Experiments are conducted in which stresses are applied to materials containing flaws of differing severity, and the results are used to validate predicted or measured fracture toughness of the material and resulting component performance.

A central activity has been the pressurization-to-failure of deliberately flawed, 6-inch thick steel pressure vessels under hydraulic loading, at the Hollifield National Laboratory. The vessels were made of A533-B and A508-C1.2 steel; some vessels also contained nozzles. Overpressures two to three times design pressure were required to cause the vessels to either leak or break, despite the presence of large flaws at both low and high test temperatures. These results have validated the failure-analysis procedures for application to reactor pressure vessels. Plans are now being formulated to extend the test program on several vessels, employing pneumatic rather than hydraulic loading—a necessary step because more energy is available from the pneumatic loading to enlarge a crack if the crack should penetrate a vessel or pipe wall. The pneumatic-loading effect will be studied in FY 1976, with a vessel that had been previously tested hydraulically, and subsequently



Thick-walled vessels of reactor pressure vessel steel have been tested hydraulically at Holifield National Laboratory to validate failure criteria for reactor safety. The integrity of this vessel was breached by leakage occurring at the base of the blanked-off nozzle—the area closest to the flaw—when the water pressure inside the vessel reached approximately 29,000 pounds per square inch. Similar vessels will be tested pneumatically during 1976.



Stress patterns are shown by refracted light of a crack in plastic propagating at 15,000 inches per second. The crack started at the slit put into the plastic at the left and continued as tension was applied to the material. This dynamic photoelasticity is used in studies of propagation and arrests of cracks at the University of Maryland.

repaired and reflawed. Two more vessels will be tested in FY 1977, including one having a nozzle.

A totally different kind of stress is applied to a reactor vessel under conditions of a postulated loss-of-coolant accident followed by emergency core cooling system operation. This is a thermal stress which would arise from the thermal shock resulting from the injection of cold ECCS water into the hot reactor pressure vessel. Such tests will be completed in fiscal year 1976 at the Holifield National Laboratory on 21-inch diameter test cylinders–intentionally flawed–in order to validate the method of analysis used to predict reactor pressure vessel behavior under thermal shock. Larger diameter steel cylinders may be tested in subsequent years. Complementary studies are being conducted at the Naval Research Laboratory to more fully characterize the specific aspects of bending and resistance to crack propagation of steel cylinders under thermal shock loading conditions.

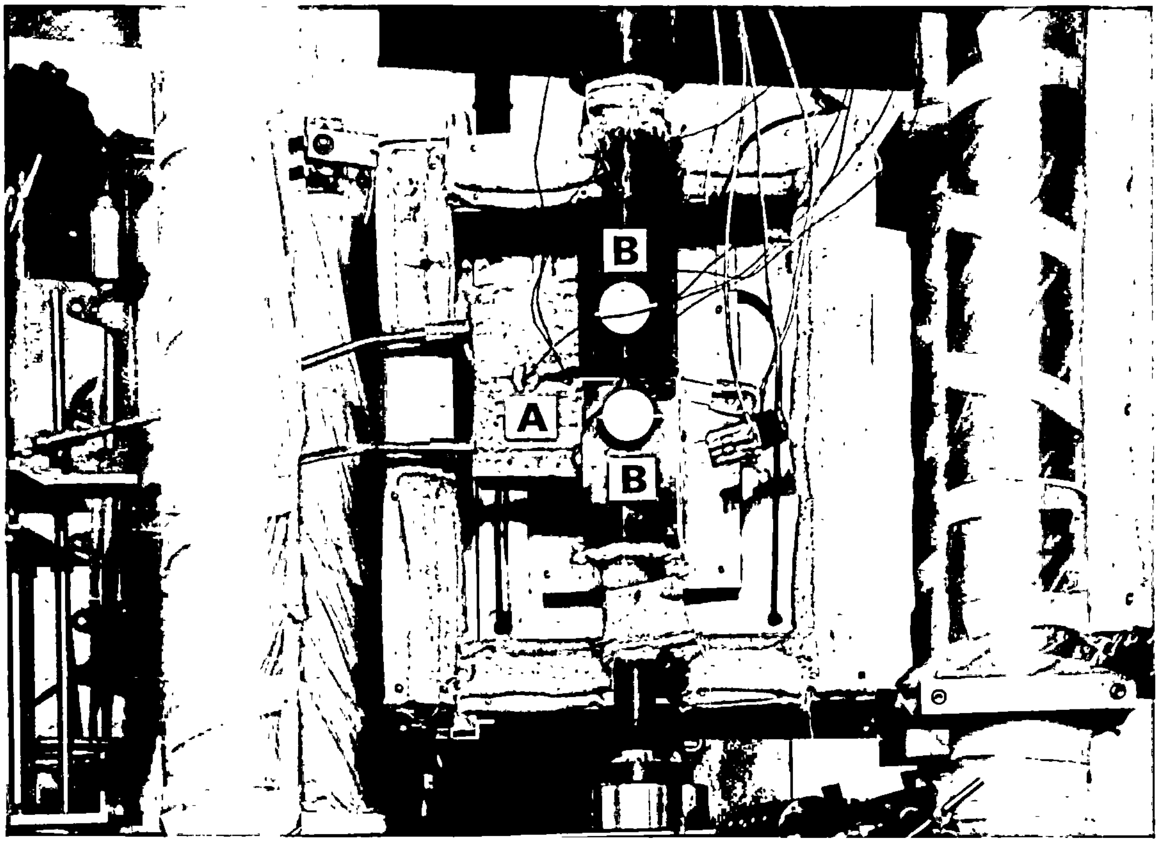
Crack arrest. If an applied stress is enough to cause a crack to grow rapidly, arrest of that crack becomes a critical safety consideration–so critical that NRC and the Electric Power Research Institute are coordinating their programs to study crack arrest. Priority has been assigned to development of a theory to characterize the dynamic propagation and arrest of a crack that will be applicable to test specimens of different geometries and to reactor pressure vessels and piping lines. Testing is being performed at Battelle Columbus Laboratories on reactor pressure vessel steels using several specimen geometries. At the University of Maryland the same specimen geometries are under test, but on birefringent plastic. The Maryland approach provides the unique advantage that both dynamic stresses and the running crack can be observed and photographed for later analysis. Excellent progress was achieved in 1975. Computer analysis was developed to describe several important specimen types, and experimental data were obtained on crack speed and energy distribution throughout specimens which should permit establishment of a standardized specimen and test procedure by the middle of 1976. Following the standardization efforts, a bank of data will be developed on the crack arrest properties of reactor pressure vessel steels in both unirradiated and irradiated conditions.

Crack propagation and arrest tests are being performed at Battelle Columbus Laboratories on reactor pressure vessel steels to develop theory applicable to reactor components. The crack is started by forcing the ram between the two pins on top of the specimen. Crack speed is measured as the crack breaks the strips across the crack path.

Radiation embrittlement. The most significant effect of irradiation on reactor pressure vessel steel is a reduction of the fracture toughness, meaning a reduced capability to remain structurally adequate under stress in the presence of a flaw. During 1975, the Naval Research Laboratory (NRL) demonstrated that residual materials remaining in the pressure vessel steel cause most of the material toughness degradation from irradiation. NRL has quantified the influence of copper on embrittlement. The NRL results were used in preparation of curves for design and regulatory use, to assure that irradiated reactor vessels are operated safely.

Fracture toughness. The ability of a material to remain structurally adequate in the presence of flaws and applied stresses is termed fracture toughness. Research is continuing to assure that the fracture toughness of reactor pressure vessel materials is fully understood and quantified for all reactor operational conditions and environments—especially the radiation environment which degrades the fracture toughness. To assure the conservatism of ASME reference fracture toughness criteria for reactor pressure vessel steels, and to verify the procedure for its use for irradiation conditions, highly irradiated specimens of reactor pressure vessel steel are being subjected to both static and dynamic toughness testing. Static testing was performed in 1975 on an irradiated, four-inch thick compact tension specimen at the Hanford Engineering Development Laboratory. Dynamic testing of additional irradiated compact tension specimens was performed at Hanford, Holifield National Laboratory, and the Westinghouse Research and Development Laboratory. A second major irradiation program was started in 1975 to complete the current NRC plan to evaluate the fracture toughness of highly irradiated reactor pressure vessel steels.

Crack growth. Existing flaws in materials can grow under repetitive or cyclic loads, from normal plant operation. Thus, an important reason for the study of flaws or crack growth is that the increased severity of the larger flaw or crack might cause the fracture toughness of the material to be exceeded, perhaps causing a component failure or leak. NRC and EPRI are conducting coordinated programs in fatigue crack growth. An important objective is to assure that the ASME code criteria related to crack growth are sufficiently conservative. Testing programs

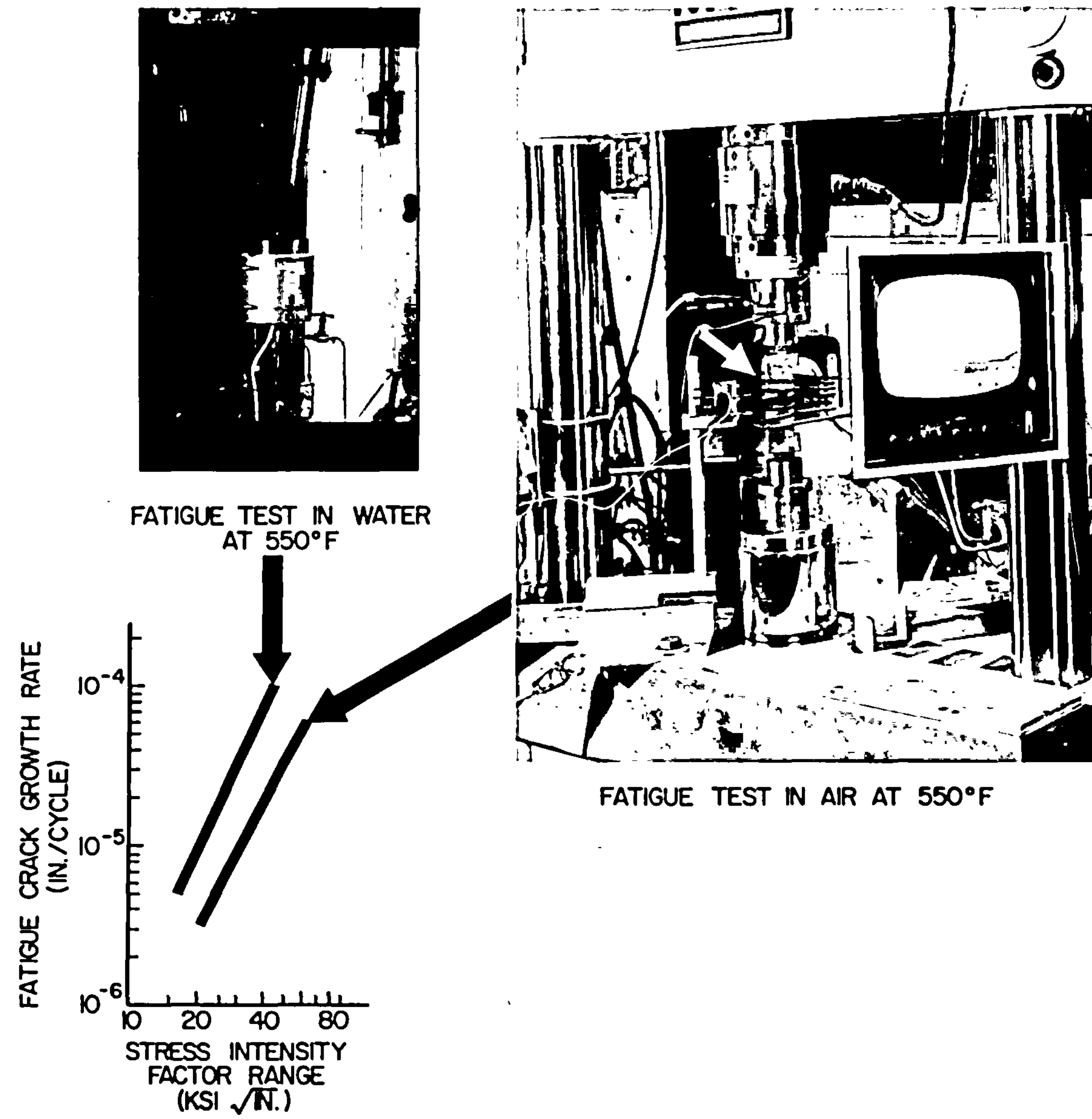


This 4-inch thick specimen of highly irradiated reactor pressure vessel steel (A) is being tested in tension and at controlled temperatures at Hanford Engineering Development Laboratory to evaluate its fracture toughness statically. Here, the pre-cracked specimen is inside an oven and is being pulled slowly from two directions (B) to measure its toughness at the point of the crack.

at the Naval Research Laboratory, General Electric Co. and Westinghouse are directed toward obtaining a better understanding of the growth of cracks under the most realistic—but still conservative—loadings and environmental conditions experienced during normal plant service. The environmental conditions addressed include neutron irradiation, high temperature, high pressure water or steam, high average stresses, and loading rates similar to those encountered in service. Experimental testing capabilities were established and neutron irradiations of fatigue crack growth specimens were completed in 1975. This will permit a full range of testing in 1976.

Non-destructive examination. This task area is aimed at improving the methods used for finding flaws in steel pressure vessels or in piping and for evaluating the severity of those flaws depending upon their size, location and orientation in the component. The most widespread technique currently used in in-service inspection is ultrasonic testing; thus, study of the capability of such inspection to detect smaller flaws with better definition is being actively pursued. Regarding fabrication inspection, the detection and location of cracks that form during the welding process are being studied with great success, using acoustic emission.

Ultrasonic testing. This procedure depends on very high frequency sound waves bounding off flaws deep within a material to indicate location and sizes of flaws much the same way that radar is used to locate

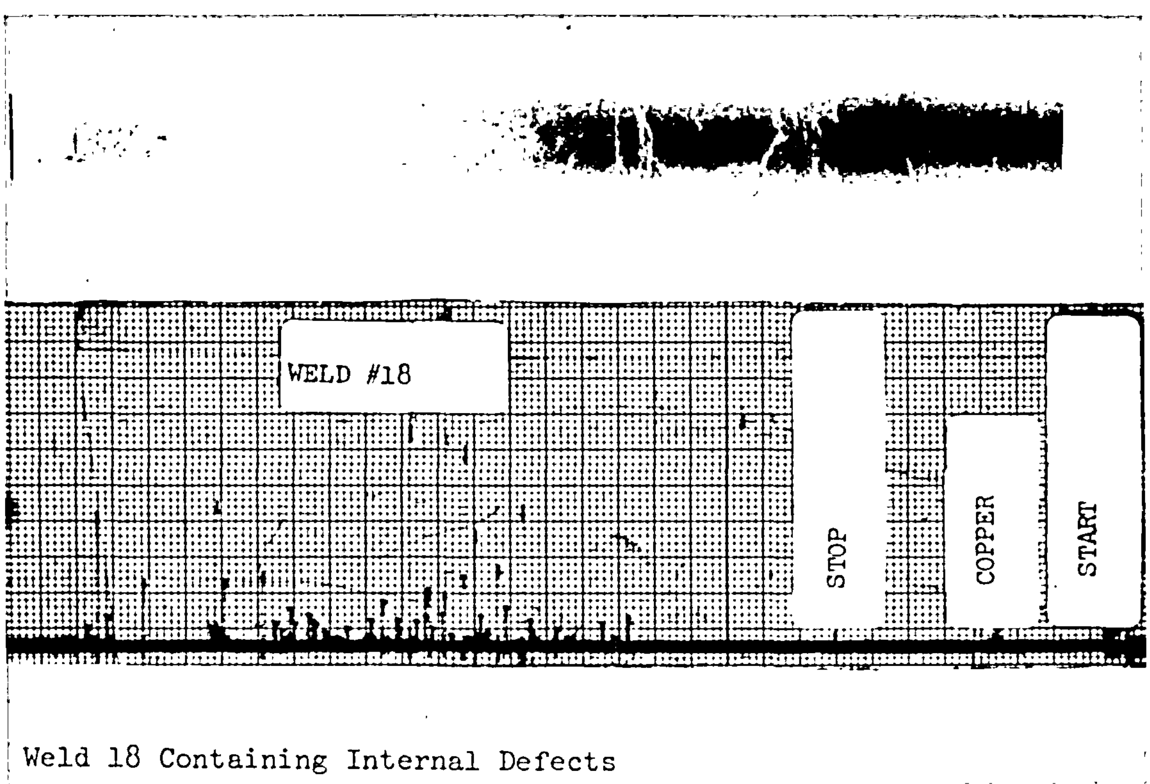


Non-destructive testing for crack-growth is accomplished using an autoclave which is a controlled temperature/pressure chamber and a load controller, which regulates the load/time factors applied to the specimen. Results are shown on a damage curve.

and track aircraft. In research conducted for NRC at the University of Michigan during 1975, a single ultrasonic transducer (which sends and receives the signals) was moved over the piece to be inspected, and the echoes stored on a computer which then correlated all the echoes from all the locations. The result was a much clearer picture of the location, depth, length and orientation of the flaw in the material, and it was possible to confirm computer-aided flaw evaluation. The existing two-dimensional system governing depth and length will be extended to three dimensions during 1976 so that data accumulation and evaluation will permit presentation of the confirmed findings of improved inspection procedures before the ASME code bodies by the end of fiscal year 1977.

Weld flaw detection. As weld metal cools and solidifies, it shrinks. If the welding conditions and machine settings are not correct, the weld puddle may crack as it shrinks. There is no guarantee that subsequent weld beads will close these cracks. As weld cracks form, they emit sound waves— acoustic emission—which can be used to warn the welder of improper welding procedures, as well as to locate the cracks precisely. Thus cracks can be found and repaired while they are accessible—not covered by subsequent weld beads. Techniques for acoustic emission monitoring of welding are being developed for NRC by the General American Transportation Corporation, which in 1975 demonstrated that acoustic emission monitoring can find flaws during welding of piping. During fiscal year 1976, the study will be extended to pressure vessel welding. The goal for the end of fiscal year 1977 is to build and prove acoustic emission monitors for use in fabrication shops, and to have documentation (for the ASME code) of the ability of acoustic emission to equal or better the performance of conventional techniques for nondestructive examination of nuclear components.

Reactor vessel surveillance. All materials and structures contain flaws. In most instances, they are so small that they have no effect on the structural integrity of components. Although reactor pressure vessels are constructed under very exacting conditions of quality control, inspection, and testing, flaws will be present, and it is important to assure that these



Weld flaw detection. Acoustic emissions—sound waves emitted when metal cracks can be monitored during welding using techniques developed and demonstrated by the General American Transportation Corp. in 1975. This read-out chart reflects an intentionally flawed weld which cracked as it cooled. Such monitoring tells the welder something is amiss in his procedures, locates the flaws and permits him to immediately correct the procedures and repair the flaw.

will not compromise the structural integrity under any postulated loading conditions. It has been noted that flaws can increase in severity under the influence of the cyclic loads from normal plant operation. As in weld flaw detection, acoustic emission occurs with flaw growth. To explore these phenomena, an operating reactor and stainless steel piping will be instrumented with acoustic emission transducers by Dunegan/Endevco in 1976. Concurrently at the Dunegan/Endevco laboratories, growing flaws will be evaluated in reactor steels under simulated operating conditions by characterizing the acoustic emission generated. Using the results of these efforts, it is possible that by 1977 a flaw in an operating reactor could be identified and located by discrimination from the background noise, and that recommendations could be made to the ASME code bodies for consideration leading to development of an automatic system.

Advanced Reactor Safety Research

Planning efforts undertaken in late 1974 led to the initiation of high priority experimental and analytical safety research program in 1975, oriented to advanced reactors—Fast Breeders (Liquid Metal Cooled-LMFBR) and Gas-Cooled (GCFBR) and High-Temperature Gas Cooled Reactors (HTGR).

The program structure derives from consideration of the methods and data needed to assess safety of reactors under a range of postulated accidents, from the anticipated loss of flow with scram to the hypothetical case of a core-disruptive accident such as might occur following a loss of flow without scram. (Scram is the action leading to rapid insertion of control rods which shut off the reactor.) In the HTGR case, the range of accidents includes the sudden depressurization (analogous to a pipe break in a light water reactor) of the primary system, steam ingress from leaks in the steam generator, and combinations of these events. Because of the unique structure of HTGRs, attention is also devoted to seismic response.

Program priorities are defined by the needs of the Offices of Nuclear Reactor Regulation and Standards Development, with recommendations of the Advisory Committee on Reactor Safeguards also used to determine priorities. Guidelines have been developed to identify interfaces of NRC and ERDA programs to minimize unnecessary duplication of effort. Extensive exchange of technical information between the two agencies ensures maximum benefit from ongoing work.

Program work is done at the national laboratories and, through contracts and subcontracts, at the Universities of Virginia, Arizona, California (Los Angeles), Illinois and Tennessee, Northwestern University and others.

FAST BREEDER REACTOR RESEARCH

The radiological source used in LMFBR site assessment is based on the potential leakage of co-agglomerated aerosols of sodium oxide and uranium-plutonium oxide from the containment.

The experimental program to verify the source and mode of transport of aerosols generated in postulated LMFBR accidents made considerable progress in 1975 at Holifield National Laboratory and at Battelle Memorial Institute (BMI).

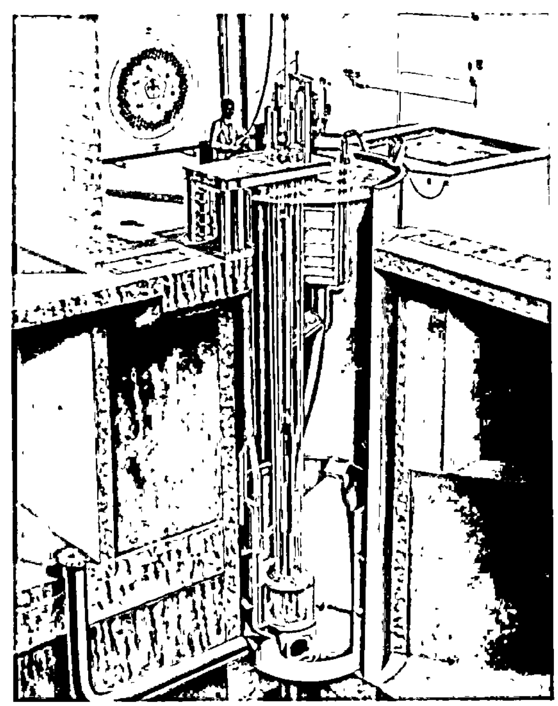
Although this program is basically long range in character, some early results have been identified which impact the review of the Clinch River Breeder Reactor (CRBR), a joint government-commercial facility. Prime among these is an independently derived, consistently conservative method for assessing the transport of aerosols in single and multiple containments. This is a key step in assessing the dose at the site boundary from a postulated accident.

This simulated fuel assembly is typical of those used in capacitor discharge vaporization tests at the Arnold Engineering Development Center for the Holifield National Laboratory's LMFBR aerosol program. During the tests, the fuel is heated drastically to see how much would vaporize. The resultant aerosol is studied to develop information on the size of radioactive particles in the aerosol, the rate of travel, and how the particles would be deposited if released in a postulated reactor accident.

At HNL, reliable methods were developed to vaporize gram quantities of uranium dioxide vapor—a very high-temperature gas—using equipment at the Arnold Engineering Development Center. The vapor source will be used at HNL to form test aerosols in vessels fabricated during 1975. These studies will also affect the CRBR review. Analytical studies at BMI complemented the experimental work by creating an improved code for predicting aerosol transport. In 1976 and 1977 the predictions of this code will be tested in test vessels and in the reactivated Nuclear Safety Pilot Plant at HNL.

Programs to verify predictions of limited fuel-coolant interactions were started at Sandia Laboratories in 1975 with the first in-pile test, a calibration run, occurring late in the year in the Annular Core Pulse Reactor. This in-pile program will extend measurements into the range characteristic of severe postulated LMFBR accidents.

Test versions of a code to study hypothetical core disruptive accident (HCDA) energetics—the SN-method, Multi-field, Multiphase, Eulerian Recriticality (SIMMER) code—were produced at Los Alamos Scientific Laboratory in 1975; these will be "frozen" in 1976 to permit extensive numerical studies for safety assessment as well as to reveal code deficiencies. An initial version of a code POOL (so named because of the support pool boiling which might occur) described the so-called "transition" or boil-up regime characteristic of fuel dispersal in a large accident, at Argonne National Laboratory (ANL). Both SIMMER and POOL are expected to impact the CRBR review. Work plans for a benchmark code to determine the effects of disturbances on coolant flow and heat transfer—the Super System Code—were finished at Brookhaven National Laboratory. Scoping studies at



Section view of the Annular Core Pulse Reactor located at Sandia Laboratories, Albuquerque, N.M. This pool type reactor is used for studies on reactor fuel behavior during simulated reactor accidents. A test fuel rod is inserted in the reactor core and exposed to multiple power pulses for periods as short as 1-2 milliseconds.

ANL showed that postulated accidents for large commercial LMFBRs are not likely to be much more severe in nature than those currently being studied for the Fast Flux Test Facility (FFTF) and CRBR. Studies at the University of California (Los Angeles) showed that discrepancies in studies of fuel-coolant interactions could be traced to inadequate measurement of the temperature distribution in the stagnant fluid.

GAS-COOLED REACTOR RESEARCH

Studies of fission product chemistry and graphite oxidation in laboratory loops at Brookhaven National Laboratory are leading to increased accuracy in the assessment of fission product transport in HTGRs and of the safety margins associated with vital structures such as the core support posts. Analytic studies at Los Alamos and Holifield National Laboratories are concentrating on models of the response of the reactor power and heat transfer to disturbances in the heat transport system. HTGRs are unique in that response of the reactor to such disturbances is heavily dependent on the response of the steam generator; the reactor nuclear core responds relatively slowly because of the high heat capacity of the graphite. The Los Alamos effort is aimed at an over-all code, Composite HTGR Analysis Program (CHAP), studying the entire system and incorporating simplified models of each component. Such a code is suitable for comprehensive reviews of power plants. This is the first code of this nature which is independent of the vendor's efforts. An early version is to be available for use in fiscal year 1976.

The effort at HNL centers on an extremely precise model of the steam generator. Its aim is to serve as a highly accurate benchmark for the simpler methods. Codes such as CHAP and the improved steam generator model are needed to assess the safety of an HTGR when there is an unexpected transient such as sudden loss of pressure in the primary system. Other analytic efforts include predictions of fission product release under accident conditions, structural response of the core and components to seismic events, and studies of the failure margin of prestressed concrete reactor vessels.

The response of HTGRs to earthquakes is important in assessing a given site; and HTGRs have structural properties well outside the normal range of reactor engineering practice. Independent evaluation of the proposed design and test data requires both analytic and experimental work. The prestressed concrete reactor vessels are new to this country, but have been used extensively for reactor containment in Europe.

Experimental work has been initiated to verify predictions of seismic response; analytical work is proceeding at the University of Illinois on improved methods of assessing the power distribution in the nuclear core under accident conditions. Special methods are being studied at Northwestern University to examine problems associated with the prediction of power distribution, taking into account the gas coolant passages.

ENVIRONMENTAL AND FUEL CYCLE RESEARCH

NRC research programs in the areas of health and environmental impact, fuel cycle site safety, and safeguards entail objectives as follows:

- Develop the technical information needed to ensure that measures taken for the protection of health, safety and the environment are adequate and to make certain that unnecessary or unwarranted requirements are not imposed;
- Develop improved methods for predicting and assessing health, safety and environmental impacts;

- Provide reliable and effective methods for evaluating sites for nuclear power and fuel cycle facilities;
- Provide reliable and efficient health and environmental protection monitoring methods;
- Identify and fill gaps in health and environmental knowledge toward improved NRC effectiveness in the performance of regulatory responsibilities;
- Assure that pertinent new research data are made available to appropriate NRC offices for application to health and environmental issues; and
- Contribute to greater efficiency in the licensing process by improving prediction, monitoring and measurement methods by validating the technical bases for regulatory actions.

Health and Environmental Research

Environmental and radiological considerations pervade the entire nuclear regulatory process. Issues concerning the possible effects of radionuclides and other polluting substances evolved by the nuclear industry are of paramount importance to national planners. Although there exists a considerable body of knowledge about human health factors, environmental systems and processes, and about radiation and other pollutants, this knowledge in some cases must be extended and refined. More years of research and study have been given to nuclear safety than to any other modern technology, yet many important issues require further examination. These include: radiation effects at low dose levels; the interactions and impacts of nuclear facility effluents with the air, land, water and biota of the environment; the interactions of nuclear plant effluents with those of other industries and their cumulative environmental impacts; the reliability of prediction and assessment methods for choosing sites, and the socioeconomic impacts of constructing and operating nuclear facilities. NRC research projects in these fields, other than reactor safety, address four primary areas:

Health. Studies that deal with the assessment of radioactivity sources, exposures and protection measures for industry workers and the public.

Environmental processes and impacts. Studies that assess, confirm or improve predictions of the fate and effects in the air, land and water environments of radionuclides, heat and other pollutants.

Socioeconomic and regional systems. Methodology for assessment of socioeconomic impacts from the nuclear industry as required by the National Environmental Policy Act. Also, methodologies for ensuring that environmental capacities and limitations can be assessed on a regional basis.

Instrumentation and Monitoring. Testing of monitoring data, equipment and techniques to ensure proper application to regulatory needs.

Fuel Cycle Safety Research

With the growth of nuclear power generation, there will be a corresponding increase in the number of supporting fuel cycle facilities. This aspect of the research program will develop information required to assess the safety of these facilities and operations, and will include both management of radioactive wastes and transportation of radioactive materials. The fuel cycle research program has been divided into three broad categories: facility safety, radioactive waste management and transportation safety.

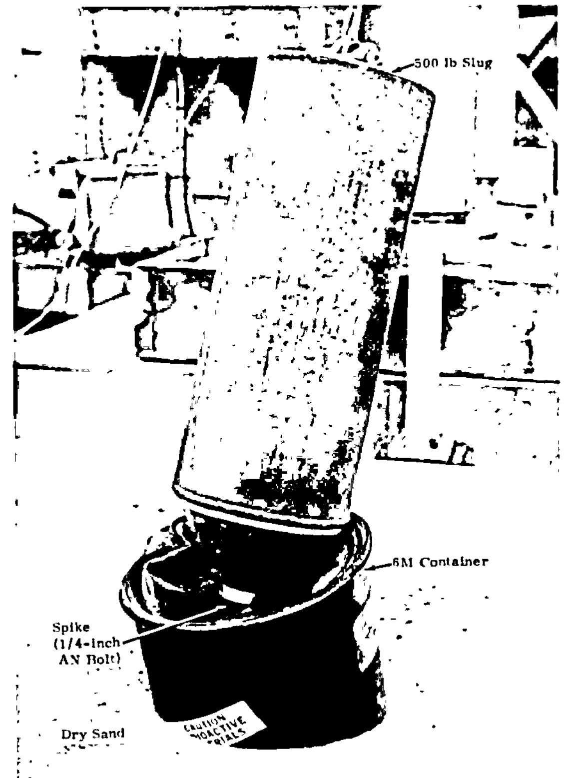
Facility safety. In support of both the licensing and standards functions of NRC, studies are under way to validate the predicted performance of gaseous and liquid effluent treatment systems and safety systems designed to control releases of radioactive materials to the environment. These efforts will consist of measurement programs conducted in operating facilities as well as selected experiments and will include testing of components important to maintaining plant safety. Efforts are also being directed toward more precise estimates of the potential for, and consequences of, fuel cycle facility accidents.

Waste management. Radioactive waste management represents an increasingly important part of the Commission's research program. Under provisions of the Energy Reorganization Act, NRC is responsible for licensing facilities for the storage of radioactive waste materials. Research in the waste management area is intended to provide the Commission with information required to assess the safety of such activities, establish appropriate design criteria and siting requirements, and develop criteria for the safe management of the various types of radioactive wastes generated in nuclear facilities.

Programs are under way to develop information on the physical and chemical characteristics of radioactive wastes that can affect the safety of waste storage systems and can influence their movement through the environment. Other work deals with identifying various waste management options—combinations of waste treatment and storage, their related costs and their potential impact on man and his environment. This information is required not only for licensing review but also by provisions of NEPA which require preparation of an environmental impact statement for each licensing action.

Transportation. NRC has recognized that as more sophisticated shipping systems are proposed for transportation of radioactive material, existing safety-verification analytical methods will require significant upgrading. A research program comprising two major elements of such upgrading is in progress.

The first major element is to provide the licensing staff with verified analytical methods to evaluate the performance of shipping packages under normal and accident transport environments. In 1975, NRC initiated a major effort to determine the adequacy of present standards when applied to packages for shipment of plutonium by aircraft. In support of this effort, two experimental projects are designed to test the capabilities of several different plutonium packages and verify the analytical models used to predict package response.



This container designed for shipping plutonium, has just been subjected to an impact-penetration test at the Sandia Laboratories, Albuquerque, N.M. The container had been resting on the surface of the sand when the 500-pound slug with a spike on its nose was dropped on it from a height of 10 feet. Although the outer package was damaged, the inner container remained secure from any leakage.

The first project subjected several packages to severe test conditions similar to those required by the Federal Aviation Administration in its acceptance of aircraft flight recorders. Other tests include impact, crush, similation fire, and water immersion. Through these tests, NRC is attempting to correlate the demonstrated high survivability of flight recorders with the performance of these shipping containers.

The second project, a test series still in progress, involves subjecting these same containers to high velocity impacts in order to establish the approximate point of gross structural failure of the package. Actual performance of these containers is being correlated with analytical models which predict container performance, in order to relate demonstrated package performance to specific accident environments by different transport modes.

The second element of transportation safety research is to provide a verified capability to predict the radiological consequences if a release of radioactive material should occur. Such a program is being formulated, using input on package failure modes.

Site Safety Research

The main purpose of site safety research is to help ensure that nuclear power plant sites have been properly evaluated with regard to earthquakes, tornadoes, floods, and other natural phenomena. Another purpose is to study engineering designs of power plant buildings, components, and systems to improve resistance to potential damage from natural phenomena. Information developed is used by NRC to evaluate sites during the licensing process, and to improve siting guides, standards, and criteria.

Earthquakes. The greatest emphasis in site safety research is placed on the study of the effects and distribution of earthquakes. In the western United States earthquakes occur more frequently than elsewhere in the country. Even more important, for purposes of siting nuclear power plants, is the fact that western earthquakes are commonly associated with faults that may be recognized through geological studies of rocks at the earth's surface. Because of this, much of the site safety research in the West is concentrated on the study, location and mapping of geological faults and determining the age of most recent movement on them.

Large earthquakes occur infrequently in the eastern United States, but within the past 200 years there have been a few large and many moderate-sized earthquakes. Although the pattern of earthquake occurrence in the East is more diffuse than in the West, near the sites of the large historic eastern earthquakes small earthquakes can be detected in somewhat higher concentrations than in adjacent areas. Most significantly, virtually no known eastern geologic faults have broken the surface in the past several thousand years, and few faults can be associated with historic earthquake activity at all.

Because of this lack of association between earthquake occurrence and known geologic faults, site safety research in the East is directed toward detailed study of the areas of past large earthquakes, such as Charleston, S.C., or the detection, location and understanding of small shocks (called microearthquakes) in regions of past moderate activity, such as New England. A jointly supported, multiple network of seismograph stations was initiated during fiscal year 1975 to monitor microearthquakes in those states north and east of Pennsylvania and Delaware. Funding support for this network is provided principally by NRC, the U.S. Geological Survey (USGS), the National Science Foundation, and the State of New York. Another small network is being established in the vicinity of Anna, Ohio, and coordination is maintained with similar activities being supported by the USGS in the area around New Madrid, Missouri.

Tornadoes. Meteorological studies that are a part of the site safety research program include those to characterize tornadoes and their effects, and to more accurately determine the past distribution of tornadoes by intensity throughout the country. Until recently, there has been no method of directly measuring actual tornado windspeeds, hence, estimates of maximum and upper limit velocities have been deduced from visual observations, damage assessments, or theoretical models. The NRC program includes efforts to develop methods of direct measurement of tornado windspeeds and to apply the measurements obtained to models based on meterological theory in order to improve maximum windspeed estimates used in design.

Other studies. Other meteorological studies include field tests to verify and compare different mathematical models used to describe the transport and diffusion of airborne radioactivity over different types of terrain and under varying meteorological conditions. The models are used to assess routine low-level release and postulated accident conditions for purposes of site evaluation.

Published NRC material. The results of NRC-funded site safety research during the past year have been reported in several publications. These include a Seismotectonic Map of the Eastern United States (J. B. Hadley and J.F. Devine, U.S. Geol. Survey Map MF-620, 1974) showing correlations of patterns of known historic earthquake occurrences with patterns of geologic faults and other structural features. Another publication is the *Preliminary Map Showing Recency of Faulting in Coastal California* (J. I. Ziony and others, U.S. Geol. Survey Map MF-585, 1974), a compilation of pre-existing and new information about onshore and offshore faults from the Mexican border to Point Arguello, northwest of Los Angeles.

Another report entitled *Procedures for Evaluation of Vibratory Ground Motions of Soil Deposits at Nuclear Power Plant Sites* (U.S. Nuc. Reg. Comm., NUREG 75/072, June 1975) is a detailed summary of procedures to develop design input criteria for foundation soils under earthquake loading conditions. Another report on foundation soils investigation is *Determination of Soil Liquefaction Characteristics by Large-Scale Laboratory Tests.* These tests examine the dynamic characteristics of liquefaction of a standard sand under carefully controlled, essentially boundary-free conditions, and compare the results with those obtained by standard engineering tests used for determining liquefaction potential of soils.

SAFEGUARDS RESEARCH

The Safeguards Research Program began to take shape shortly after the establishment of the Nuclear Regulatory Commission. Prior to enactment of the Energy Reorganization Act of 1974, safeguards research was performed by the AEC General Manager staff, with some technical support programs funded by the Regulatory staff. Administration of the ongoing technical support programs was assumed by NRC following its establishment, and became part of its new safeguards research program.

NRC safeguards research efforts during 1975 were directed to planning and coordinating a comprehensive program for fiscal year 1976 and beyond. A proposed program was developed with active participation of all major NRC offices, and involves a wide range of regulatory functions. The program is designed to achieve: (1) a more systematic basis and comprehensive technical rationale for NRC's safeguards policy and programs; (2) improved methods and procedures for licensing review, inspection and enforcement, central information processing and NRC emergency response activities; and (3) improved regulations and guides to assure effective safeguards implementation at licensee facilities.

Three broad categories of safeguards research are required to support NRC regulation of the nuclear industry.

- The first is directed toward integrating what is known about: (1) the characteristics of terrorists or other potential threats (resources, motivation, behavior, etc.), (2) the nuclear industry (plant design, criteria for operational viability), (3) national needs, and (4) the safeguards system (its objectives and constraints, available mechanisms, available options and their capabilities and limitations). The goal of this research will be to conceptualize, analyze and organize the results to provide a technical basis for NRC decisions on policy options.
- The second category recognizes that existing regulations can be improved before the results of the first category of research are completed. Using a recently developed framework for definition of safeguards subsystems and their objectives, alternative subsystem designs will be developed (e.g., combinations of mechanisms having a common regulatory objective). The alternatives will be evaluated in terms of cost-benefit, including their impact on plant safety and operational viability. Research priorities will be established on the basis of experience and staff judgement, but future work will be guided by the results of the system analysis and design studies discussed above.
- The third category supports the interdependent functions of writing and issuing rules and regulations and the work of the inspectors. The research requires development of acceptaince criteria and inspection techniques and devices for a wide variety of safeguards mechanisms and subsystems presently required by existing regulations.

The needs range from standard targets for testing the sensitivity of intrusion alarms, to analytical methods for assessing the overall effectiveness of integrated subsystems.

Other goals for safeguards research will be to support a safeguards information system to satisfy the requirements of the safeguards decision process at all levels of the structure, and to improve the technical capability for carrying out other NRC safeguards functions such as licensing review, evaluation and emergency response.

REACTOR SAFETY STUDY

On October 30, 1975, the Commission issued the final report of a study entitled "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (NUREG-75/014), known informally as the "Rasmussen Report." The objective of the three-year study, conducted by a team of 60 persons headed by Professor Norman Rasmussen of Massachusetts Institute of Technology, was to assess realistically the risks to the public that might arise from potential accidents in present-day U.S. light-water-cooled nuclear powerplants.

The overall conclusion of this most comprehensive risk assessment of nuclear power plants made to date is that the risks are very low compared to other natural and man-made risks. The advanced techniques used in producing this report have potential for application in the NRC's research and licensing activities.

Originally sponsored by the Atomic Energy Commission, the Reactor Safety Study report was issued in draft form (as WASH-1400) in August 1974 for public comment. The independent study group completed its work under NRC sponsorship, giving careful consideration to more than 1,800 pages of comments from 87 individuals and organizations which represented a broad spectrum of American society, a wide diversity of viewpoints, and all relevant fields of expertise.

Conclusion: Risks Very Low

Among specific conclusions in the study were the following:

- Nuclear power plants are about 10,000 times less likely to produce fatal accidents than many nonnuclear activities or events, such as fires, explosions, toxic chemical releases, dam failures, airplane crashes, earthquakes, tornadoes, and hurricanes.
- Nonnuclear accidents involving comparable large dollar-value damage are about 1,000 times more likely than nuclear power plant accidents.
- The chance that a person living in the general vicinity of a nuclear power plant will be fatally injured in a reactor accident is one in five billion per year, as compared with one in 4,000 for a motor vehicle accident and one in 10,000 for a fall. The chance that a person will be injured in a reactor accident is one in 75 million per year.
- In the event of an unlikely reactor accident, with a probability of one in a million per reactor per year, latent health effects, such as cancer fatalities and genetic defects, would be such a small percentage of the normal incidence rates that they would be essentially undetectable. Another latent health effect, thyroid nodules, would occur at about 15 percent of the normal incidence rate, so that the increase would be detectable. These nodules can be diagnosed readily and treated successfully.

Calculations Improved

The final report differs from the August 1974 draft report with regard to the consequences of accidents. Based on advice from eminent scientists in all relevant disciplines calculations were substantially revised. In most cases, the improved calculations resulted in increasing the consequences of accidents over the levels indicated in the draft, although risk levels remain very low compared to nonnuclear risks. With regard to accident probabilities, the final report differs only in minor respects from the draft report.

NRC Chairman Anders said of the final report:

"The Commission believes that the Reactor Safety Study report provides an objective and meaningful estimate of the public risks associated with the operation of present-day light water power reactors in the United States. The final report is a soundly based and impressive work. Its overall conclusion is that the risk attached to the operation of nuclear power plants is very low compared with other natural and man-made risks. The report reinforces the Commission's belief that a nuclear power plant designed, constructed and operated in accordance with NRC's comprehensive regulatory requirements provides adequate protection to public health and safety and the environment. Of course, such regulatory requirements must be continually reviewed in the light of new knowledge, including that derived from a vigorous regulatory research program."

Significant Step Forward

Since there have been no nuclear power accidents to date which have resulted in significant releases of radioactivity to the environment, nuclear risks could only be estimated in the study. Many of the methods used in the study, including "event trees" and "fault trees," were developed by the Department of Defense and the National Aeronautics and Space Administration in the last 10 years. The specific application of these methods in the reactor safety study represents a significant step forward in risk assessment capability.

Following publication of the final NRC report, copies were made available for public inspection in the Commission's public document room at 1717 H Street, N.W. in Washington, D.C., and the NRC's five regional offices in Philadelphia, Pa.; Atlanta, Ga.; Chicago, Ill.; Dallas, Texas, and San Francisco, Calif. (Copies may be purchased from the National Technical Information Service, Springfield, Va. 22161.)

Use of Techniques to Be Studied

The publication of this report raises the question of how the advanced techniques employed can be used in the future in connection with NRC's licensing responsibilities.

As part of NRC's ongoing effort, it is planned that the insights gained in the study will be used to identify the relative importance of various contributions to potential reactor accident risks. This knowledge can be used to achieve a better balance of effort in the safety reviews of reactor plants, and can eliminate from these reviews itmes that are not contributors to the risk. This will reduce the effort spent by the industry and the government in the licensing of reactors and should result in a decrease in licensing time and in a stabilizing of the process.

Plans over the next year call for the Offices of Nuclear Regulatory Research and Nuclear Reactor Regulation to cooperate on the:

(1) Identification of areas that are potentially fruitful in this regard,
(2) Analysis of each identified area to determine a suitable course of action to be taken, and
(3) Implementation of appropriate modifications to reactor technical review procedures.

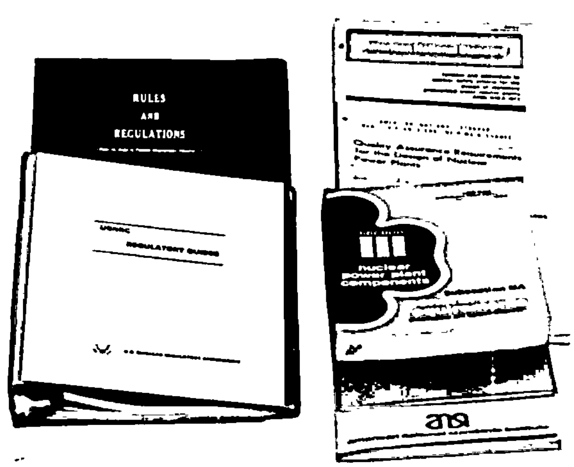
Developing Nuclear Standards

Guides for Effective Regulation

Standards are basic to the Nuclear Regulatory Commission's comprehensive program for the control and safe use of nuclear energy. Developed by the Office of Standards Development, they govern protection of the public and nuclear industry workers from radiation, safeguarding nuclear materials and plants, and protection of the quality of the environment. Thus, standards activity cuts across the range of NRC's day-to-day concerns. Many significant standards are discussed in other chapters of this annual report.

In setting forth safety requirements, including quality assurance requirements, for the design, construction, and operation of nuclear reactors, standards provide the mechanism for codifying sound engineering practice and the lessons of experience.

The standards development function also provides a mechanism for resolving frequently recurring technical issues through generic rulemaking, provides a forum for



all segments of the public to provide input to proposed standards, and clearly establishes NRC's bases for inspection and enforcement of licensed operations.

Objectives and priorities for standards development are established on the basis of (a) Commission policy direction, (b) current licensing issues, (c) inspection and enforcement need, (d) need to stabilize and accelerate the regulatory process, (e) operating experience, (f) research results, and (g) input from industry and the public.

Types of Standards

Three types of standards are used by the NRC: (a) NRC regulations, established by the Commission and published in Title 10, Chapter 1, of the Code of Federal Regulations, which set forth both general and specific requirements that must be met; (b) NRC guides, developed by the NRC staff and issued to guide license applicants and licensees; and (c) "consensus" standards, developed by recognized national professional standards organizations, often with NRC participation.

Guides describe methods found or developed by the NRC staff to be acceptable for meeting portions of the Commission's regulations. Some of these guides describe accident assumptions which the NRC staff uses in safety analyses and evaluations. Other guides set forth best engineering practice to accomplish a particular task—for example, design and construction requirements for nuclear components. Many of the guides refer to standards developed in the national standards program, which is coordinated by the American National Standards Institute(ANSI) with participation by the NRC staff. These consensus standards, if found acceptable after a thorough, independent review by the NRC staff, are endorsed in regulations or regulatory guides.

The development of NRC regulations includes a review of alternative courses of action and their costs and benefits. In addition, NRC recently has initiated work to assess the impact and value of the methods recommended in new NRC guides. All guides are issued initially for public comment, then reissued to reflect substantive comments received, particularly on how to achieve the same degree of safety in a simpler or less costly way.

Scope of NRC Standards

The major responsibilities assigned to the Office of Standards Development are:

- *Nuclear power plant standards* to define safety requirements and acceptable practices for design, procurement, construction, operation, and decommissioning of light-water-cooled reactors and advanced reactors, including high-temperature gas-cooled reactors and liquid metal fast breeder reactors.
- *Safeguards standards* for physical protection of nuclear materials and facilities and for control of nuclear materials.
- *Site and environmental standards* for site designation criteria and procedures for certifying the acceptablity of proposed nuclear facility locations; site safety requirements; environmental requirements for radiological and non-radiological effects; and radiation protection in the area of public radiological health and environmental effects.
- *Fuel cycle facility standards* to define safety requirements for light-water reactor fuel cycle facilities, advanced reactor fuel cycle facilities, and waste storage and disposal facilities.
- *Occupational health standards* in the area of radiation protection.
- *Transportation and product standards* to establish safety requirements for transportation, in conjunction with the Department of Transportation, and rules for use of radioactive materials in medical, industrial, and consumer products and applications.

This chapter summarizes NRC standards activities during 1975 in each of the above areas (except transportation, which is described in Chapter 4). Also briefly described are national nuclear standards efforts and current licensing issues being addressed by the NRC standards staff.

NRC regulations proposed and made effective during the period January through June 1975 are summarized in Appendix 5. Regulatory guides issued during the same period are listed in Appendix 6.

NUMBER OF REGULATORY GUIDES ISSUED BY CATEGORY

Subject Category	FY 1973	FY 1974	FY 1975	FY 1976 (Est.)
Nuclear Power Plant Design and Engineering	37	47	23	25
Nuclear Fuels and Materials	11	12	8	9
Materials and Plant Protection (Safeguards)	10	29	18	18
Environmental and Siting	3	5	5	10
Radiation Protection	7	12	11	12
TOTAL	68	105	65	74

POWER REACTOR STANDARDS

Nuclear power plant standards efforts during 1975 were directed principally to the following areas: (1) system, component, and structural criteria; (2) quality assurance; (3) qualification testing; (4) implementing actions required by the Energy Reorganization Act of 1974; (5) inservice inspection and surveillance; and (6) improved guidance in licensing review procedures.

In a number of instances, the guidance in these areas is based on national standards developed through the sponsorship of national technical societies under the broad direction and coordination of ANSI (see later discussion). NRC staff personnel participate as members of the committees and working groups.

System and Component Criteria

A proposed revision to the Codes and Standards Rule, which was published for comment in September 1974, relates the applicable edition and addenda of the American Society of Mechanical Engineers' (ASME) Boiler and Pressure Vessel Code to the docket date for a construction permit application rather than to the issuance date of the construction permit. This change would permit normal commercial procurement practice for nuclear components, which includes specifying applicable codes when the component is ordered.

Modifications to this code are generally first introduced by "code cases." Component manufacturers have repeatedly requested that NRC provide prompt indication of its approval of applying such code cases to component fabrication. Regulatory Guides 1.84, "Code Case Acceptability—ASME Section III Design and Fabrication," and 1.85 "Code Case Acceptability—ASME Section III Materials," have been developed and are being promptly revised to indicate NRC acceptance of code cases. Two revisions of each guide were issued during 1975. Procedures have been developed to ensure issuance of revisions six times per year, following the ASME Council meetings that approve new cases.

Regulatory Guide 1.75, "Physical Independence of Electric Systems," was revised in January 1975, so that postulated events leading to the failure of one safety-related electric system would not cause failure of any other necessary system. Also, a revised guide on "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants" (Regulatory Guide 1.81) was issued in January. Its purpose is to make certain that sharing of such systems would not cause a degradation of safety-related systems.

A guide, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants" (Regulatory Guide 1.96), was issued in May 1975, to describe an acceptable basis for implementing NRC's regulation on the design of such systems, to see to it that the total site radiological effects do not exceed the guidelines of NRC regulations on reactor site criteria in the unlikely event of a loss-of-coolant accident.

Regulatory Guide 1.95 was issued in February 1975 to describe acceptable design features for the protection of the operators of control rooms at nuclear power plants in the event of an accidental chlorine release.

Quality Assurance Standards

An amendment to the NRC's "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," was placed in effect on February 19, 1975. This changed the criterion on "Organization" to clarify the permissible organizational relationships among and within the various organizations involved in implementing a quality assurance program. It was the first substantive amendment to the quality assurance criteria since they became effective in 1970. Additional guidance was issued, including "Quality Assurance Requirements for the Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (Regulatory Guide 1.94), in April; and a revision of the guide, "Quality Assurance Requirements for the Design of Nuclear Power Plants" (Regulatory Guide 1.64) in February.

NRC standards, regulations and guides serve as bases for inspections by field inspectors throughout the lifetime of a nuclear reactor, and all safety related components must meet rigid quality assurance criteria. Here, an NRC inspector checks welding on a process water tank dome.

Qualification Testing Standards

Increasing emphasis is being placed on developing standards and guides for design verification by qualification testing and analysis. Standards for qualification testing in two crucial areas—active mechanical components and electrical components—are being emphasized. In the mechanical area, the staff is working closely with two industry 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. In the electrical area, work is continuing under contractual arrangements with Sandia Laboratories to perform tests to determine possible synergistic effects under postulated accident conditions and to perform fire tests on electric cable trays. These latter tests will provide additional information concerning physical separation criteria for the protection of redundant safety equipment against loss from a common cause (for example, fire).

Reporting Defects/Noncompliance

Section 206, "Noncompliance," of the Energy Reorganization Act requires that certain individuals who become aware of defects in basic components or of basic components that failed to comply with specified laws, rules, and regulations must report such defects or failure to comply to the NRC. A proposed rule on such reports to implement the mandate was published for comment on March 3, 1975. The extensive comments received are being reviewed.

Inservice Inspection and Surveillance

The proposed revision to the Codes and Standards Rule discussed above would require that any new inservice inspection and testing requirements (Section XI of the ASME Boiler and Pressure Vessel Code) becoming effective during the service lifetime of a facility be adopted to the degree practicable, within limitations of design and access.

Revision 1 to Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," was issued in July to reflect public comments.

Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," which was issued in July, provides data to be used for programs to monitor the effects of neutron radiation on reactor vessel steel.

Reactor Containment Standards

Prestressed concrete construction is being used increasingly for reactor containment buildings. Inspection for possible degradation of the highly loaded steel prestressing tendons is required to ensure that containment functional capability will be maintained through the operating life of the plant. One of two methods of protecting the tendons against environmental attack is usually employed – either filling the tendon duct with grease or filling the duct with cement grout.

Several guides covering prestressed concrete containments were under development in fiscal year 1975. One guide, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures" (Regulatory Guide 1.107), was issued in November. Two others were expected to be issued early in 1976. One is Revision 2 to Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures." The other guide will outline for the industry the types of commercial prestressing systems that have been accepted in the past.

In January 1975 a new Code was published – ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments – which became effective for use on July 1. It covers the quality assurance, design, construction, and testing requirements for concrete reactor vessels and concrete containments.

Safety Analysis Report Format

Revision 2 to Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," was issued in September. In order to provide interim guidance for applicants during preparation of this revision, 31 of 38 regulatory guides in the 1.70 series were issued in advance of publication of the revised format. Each guide identified the information needed to make a portion of the Standard Format consistent with the Standard Review Plans being developed by the Office of Nuclear Reactor Regulation. As such, the 1.70 guides were essentially sections of Revision 2 of the Standard Format and represented about half the updated information needed by the NRC to conduct a complete review in accordance with the Standard Review Plans.

SAFEGUARDS STANDARDS

A major activity of the Office of Standards Development has involved several

An NRC inspector (at left) makes an independent check at a licensed fuel fabrication plant to verify the licensee's inventory of fissionable material. Under new amendments to NRC regulations, published in 1975, licensee control and accountability measures for special nuclear materials were substantially strengthened.

rulemaking actions to improve the protection of significant quantities of special nuclear material (SNM) from theft and to strengthen the security of nuclear facilities against acts of sabotage that could endanger the public by exposure to radiation. (See also Chapter 5, "Safeguarding Materials and Facilities.")

SNM Accountability Standards

During 1975, two amendments to NRC regulations on special nuclear material accountability were published. For some time licensees have been required to maintain material balances in order to account for material received, shipped, discarded, and maintained on inventory. Measures had not been prescribed, however, to ensure that a licensee's material control and accounting system would provide high quality measurement data to determine accurately the quantities of SNM on inventory and to distinguish an actual loss of material from measurement uncertainty.

One of the amendments to the SNM regulations, issued as an effective rule in August, identified specific requirements for planning, establishing, and maintaining an adequate measurement control program. The program must include organizational controls for management of measurement quality, training and performance qualifications requirements, a standards and calibration system, a quality testing system for the determination and control of systematic and random errors, an evaluation system for collection and statistical analysis of the data, and program audits and management reviews. The programs must be in effect nine months after publication or 30 days after NRC approval, whichever is later.

A second amendment to the SNM regulations, published for comment in July, dealt with actions to be required of licensees in the event that a large discrepancy (sometimes referred to as material unaccounted for — "MUF") is discovered between the quantity of material assigned to a licensee according to shipping records and the results of a physical inventory of his holdings. Under the proposed amendment, if the discrepancy exceeds the limits specified in the regulations, the licensee would be required to take immediate action appropriate to the amount of the discrepancy.

Security Standards

Many public comments have been received regarding two proposed new requirements for the added protection of nuclear power plants and SNM in transit. These proposed amendments to the regulations on physical protection of plants and materials were published for comment in November 1974. If adopted, they would require reactor operators to establish a physical security organization, provide access control to and within the nuclear power plant using qualified and trained guards, and establish other physical security measures including communications, liaison with local law enforcement authority, and visitor restrictions for nonemployees. High-enriched uranium-235, uranium-233 and plutonium shipments in transit would be subject to additional safeguards measures.

To allow ample time for NRC inspection teams to verify that these or other required protection measures for SNM in transit are actually being provided, NRC issued an effective rule change in November 1975 that would require licensees to notify the appropriate NRC Regional Office at least seven days in advance of a planned SNM shipment and again upon arrival of such shipments.

The review, consideration, and resolution of the comments received on these proposals have constituted a major staff activity in 1975.

Regulatory Guides on Safeguards

Commensurate with NRC's greater interest in physical protection measures and in industrial standardization of appropriate means of implementing regulatory requirements, the NRC has initiated a number of actions in an effort to quantify and improve both current practice and the state-of-technology. NRC has stepped up its activity in industrial standards-writing efforts.

A comprehensive compilation of all the needed standards (referred to as the NRC Body of Safeguards Standards), annotated to indicate priority, status, and potential source (appropriate standards-writing organization), was prepared for, submitted to, and discussed with the American National Standards Institute (ANSI). This body of standards will be updated as necessary in guiding future efforts of both industry and NRC standards organizations.

In January through June 1975, the NRC issued 15 regulatory guides dealing with the protection of plants and materials. Of special note is Regulatory Guide 5.52, "Standard Format and Content for the Physical Protection Section of a License Application (For Facilities Other Than Nuclear Power Plants)," which is indicative of NRC's continuing program to standardize the license application review process to reduce the time required to accomplish such reviews.

SITE AND ENVIRONMENT

The site standards program is composed of three complementary efforts: site designation, site safety, and environmental standards.

The site designation standards work includes writing guides and standards to facilitate separating the certification of sites from the licensing of the nuclear power plant. The program also includes State or multi-State and NRC interaction in the resolution of technical and procedural siting issues on a regional basis. In addition, it includes development of emergency preparedness standards and responsibility for rulemaking in connection with the power reactor siting regulation of the NRC.

Public comments received on three important documents issued during 1975 — Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations"; WASH-1355, "Nuclear Power Facility Performance Characteristics For Making Environmental Impact Assessments"; and WASH-1361, "Safety-Related Site Parameters" — indicate that these documents fulfill their purposes towards implementing the site designation concept. Another regulatory guide, in preparation, on early site review will provide applicants for early site reviews with the format and description of information required by the staff.

In the area of site safety, the program includes writing standard site assessment procedures that effectively interface with power plant design standards so that a uniform and reasonable degree of conservatism is provided. The program includes standard methods for nonradiological and offsite hazard assessment, specification of combined natural events, and application of meteorological data.

The environmental standards work includes environmental technical specifications for reactors, equivalent biological risk (health effects) assessment for fuel cycle facility siting, and rulemaking to establish a monetary equivalent for population exposures to radiation. The program also includes development of guides to implement Appendix I to 10 CFR Part 50, the recently adopted effluent control regulation for nuclear power plants (see Chapter 3), and technical reports treating the state-of-the-art of effluent control for fuel cycle facilities.

Revisions to Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," provide current guidance on the interface of the Federal Water Pollution Control Act Amendments (FWPCA) with NRC licensing actions and definitive guidance on requirements for information on economic and social costs and benefits needed to fulfill NEPA requirements.

Site Designation Standards

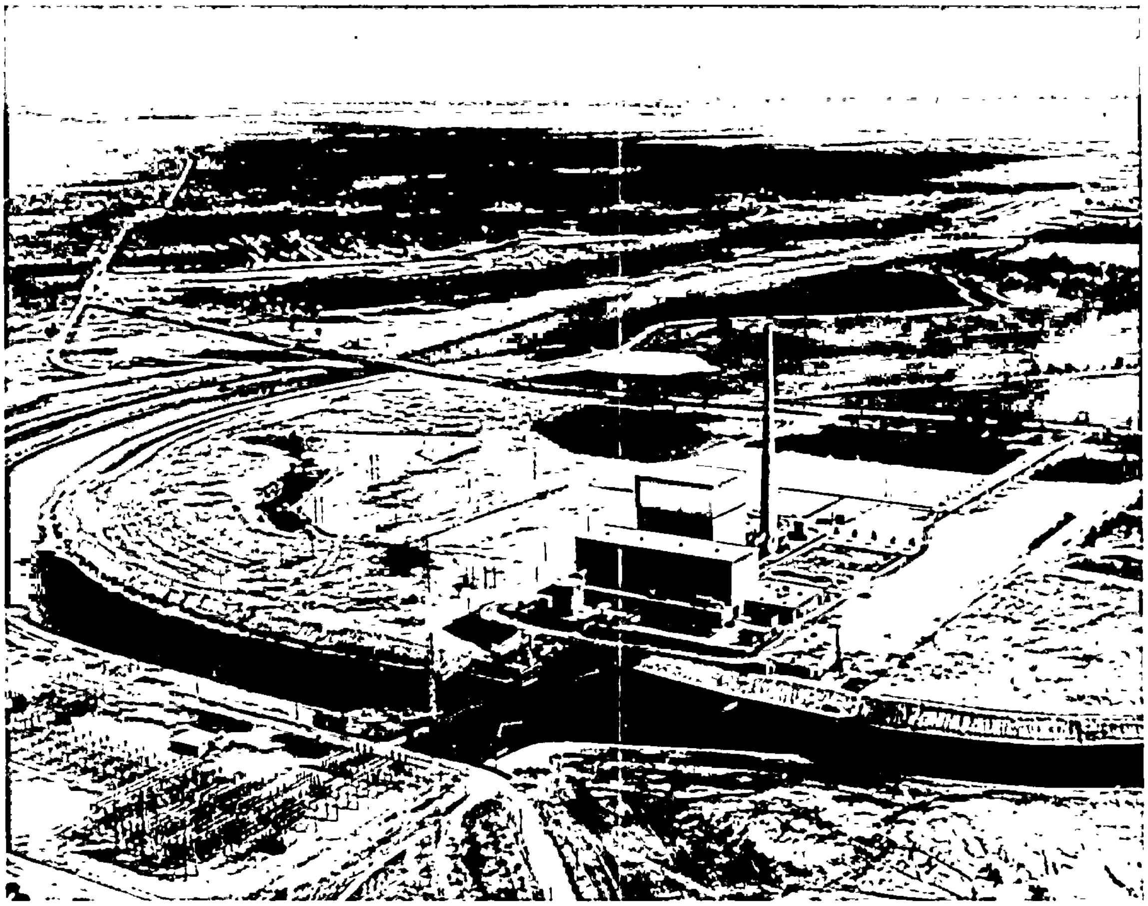
The site designation standards program is shifting emphasis from the codification of NEPA procedures to the standardization of regional siting issues. The principal mission of this program is to provide methods and procedures for effective and efficient decision-making in the selection of sites for nuclear power plants and related fuel cycle facilities. Within this program lies the responsibility for standards development to implement the "early site review" and "designated siting" concepts of NRC. These concepts are a key part of the Commission's effort to encourage and speed standardization of the nuclear power option and to increase the effectiveness and timeliness of the licensing process. Principal work over the next several years involves identification and study, on a regional scale, of selected technical and procedural siting issues of concern to local, regional, and Federal levels of government. An important part of the overall program is development of demonstration programs with States and NRC which is discussed in Chapter 10.

The considerations in developing more effective decision-making on regional siting fall into two broad categories: technical concerns and institutional mechanisms and interactions. Table 1 lists specific concerns that the NRC has identified as worthy of further study to produce practical results.

The common thread through these topics is the need to address the public concern regarding reactor siting at all levels of government for health, safety, and environment, and for an adequate supply of power. Ultimately the thread runs to resolution of siting decisions in a timely manner with effective participation by the public and all levels of government. A common goal is designation of sites in advance of an identified need for the power within a particular service area. The resulting bank of power plant sites would then logically fit into planning by utilities and governments to meet the needs of their consumers and constituents.

Site Safety Standards

The site safety standards program is principally aimed at resolving the interface between structural design standards and siting criteria related to manmade and natural events that could affect safe operation of nuclear facilities – for example, earthquakes, floods, offsite explosions, and air crashes. NRC is now initiating a systematic examination of the existing regulatory base



Development of nuclear plant/site-designation and environmental standards involves consideration of both radiological and nonradiological factors, as well as close cooperation with other government agencies at Federal, State and regional levels. Progress towards improving the mechanisms for such coordination during 1975 included the accomplishment of memoranda of understanding between NRC and the Environmental Protection Agency, Energy Research and Development Administration, the U.S. Army Corps of Engineers and others, as well as development of close working relationships with several State regulatory organizations.

to provide a uniform and balanced degree of conservatism in specifying siting criteria related to these phenomena.

Environmental Standards

The environmental standards program includes the promulgation of regulations and guides for control of reactor and fuel cycle facility effluents. Both radiological and nonradiological environmental effects are included. In the latter category are such issues as environmentally based technical specifications for the operation of nuclear facilities, form and content of environmental reports submitted to NRC in connection with licensing applications, and performance standards for protection of the aquatic and terrestrial environment from the effects of NRC-licensed activities.

This program also includes coordinating NRC interaction with the International Commission on Radiological Protection, the National Council on Radiation Protection and Measurements, the Environmental Protection Agency, and the National Academy of Sciences in matters concerning

Table 1.

REGIONAL SITING CONCERNS

Technical

- Radiological Assistance
- Emergency Preparedness
- Regional Geology
- Quantitative Assessment of Alternative Sites (Weighted Site Selection Methodology)
- Cumulative Impact
    1. Radiological
    2. Nonradiological
        a. Effluents
        b. Socioeconomic
- Land Resources Management
- Nuclear/Nonnuclear Incremental Costs and Benefits
- Technical Assistance to States
- Need for Power Certification Regional/National
- Characterization of Regions
- Regional Environmental Report

Institutional Mechanisms and Interactions

- Interstate Advance Notice of Intent to Select Site
- Taxes to Apportion Benefits/Burdens
- Rights of Interstate Intervention or Veto Authority
- Joint Hearings
- Common Legislation
- Multilateral State Agreements
- Interstate Rate Setting (Severance or Extraction Taxes)

environmental protection and radiation standards.

A new rule on effluent control is discussed in Chapter 3. The Office of Standards Development is devoting a major effort toward producing several regulatory guides concerning aspects of the new regulation.

FUEL CYCLE STANDARDS

During the period January through June 1975, six standards and guides were issued on special technical considerations involving fuels and materials facilities. These were:

- A proposed amendment, "Plans for Coping with Emergencies," the regulations on special nuclear materials, published June 27, 1975.
- Regulatory Guide 3.26, "Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants."
- Regulatory Guide 3.37, "Nondestructive Examination of Welds in the Liners of Concrete Barriers in Fuel Reprocessing Plants."
- Regulatory Guide 3.28, "Welder Qualification for Welding in Areas of Limited Accessibility in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants."
- Regulatory Guide 3.29, "Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants."
- Regulatory Guide 3.30, "Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants."

Major effort also was expended in developing other standards and guides in the fuels and materials area. Among those under development at the end of the year were proposed amendments to NRC regulations covering requirements for the storage of spent fuels from light-water-cooled reactors at independent spent fuel storage installations and guidance for the preparation of license applications related to such installations. Also being developed were regulatory requirements and guidance for the licensing of private uranium enrichment plants.

Engineering studies continued during the year to provide bases for needed standards and guides on seismic and decommissioning criteria for fuel cycle facilities.

Other activities included technical assessments related to as-low-as-reasonably-achievable effluent releases with respect to fuel reprocessing plants, uranium mills, enriched uranium fuel fabrication plants, and plutonium processing and fuel fabrication plants.

Increased emphasis was given to needed standards and guides related to the long-term storage and ultimate disposal of radioactive waste — particularly high-level radioactive waste from fuel reprocessing plants — in view of the requirement of the Energy Reorganization Act of 1974 that NRC license ERDA facilities used primarily for the receipt and storage of high-level radioactive wastes generated under licensed and ERDA activities.

OCCUPATIONAL HEALTH RULES

Prenatal Radiation Exposure

Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure," was issued in March 1975, to assist licensees in informing workers about possible health risks to the children of women who are exposed to radiation during pregnancy. An appendix to the guide gives workers who might be affected by the rule the information necessary to make informed decisions. The guide was issued in connection with a proposed rulemaking action that would specifically require such instruction.

Personnel Monitoring Reports

Certain Commission licensees (power reactor operators, radiographers, nuclear fuel processors or reprocessors, and major byproduct material processors) are required to report annually the number of workers who received radiation doses within specified ranges from licensed radioactive material under their control. On May 30, 1975, the Commission published in the *Federal Register* a proposed amendment to the regulations on standards for protection against radiation that would extend this requirement to all Commission licensees.

NATIONAL STANDARDS

The American National Standards Institute (ANSI) is a federation of leading trade, technical, and professional organizations, government agencies, and consumer groups. Its principal function is to act as a clearinghouse to coordinate the work of standards development in the private sector, which is currently carried on by nearly 400 different organizations. Its procedures make possible competent, efficient, and timely development of standards. Duplication and overlap are minimized, and a neutral forum is provided to consider and identify standards needs. American National Standards are approved on a basis that ensures a national consensus of consumers, government agencies, manufacturers and scientific, technical, and professional organizations.

The overall management of nuclear industry standards is the responsibility of ANSI's Nuclear Standards Management Board (NSMB). An NRC staff member serves as an executive director of the ANSI-NSMB executive committee and has a direct managerial role in the ANSI-NSMB nuclear program in such areas as establishing a coordinated nuclear standards program.

The NRC staff works with the national standards program to help establish priorities so that the efforts of the working groups are concentrated on developing standards that can be most useful in protecting the public health and safety. More than 190 standards are now referenced in NRC regulations and guides.

The actual writing of standards is done by experts, most of whom are members of

Table 2. SOCIETIES PARTICIPATING IN NUCLEAR STANDARDS DEVELOPMENT

American Board of Health Physics
American Concrete Institute
American Industrial Hygiene Association
American Institute of Chemical Engineers
American National Standards Institute
American Nuclear Society
American Public Health Association
American Society of Mechanical Engineers
American Society for Nondestructive Testing
American Society of Quality Control
American Society for Testing and Materials
Association of Neutron Radiographers
Conference of Radiation Control Program Directors
Health Physics Society
Institute of Electrical and Electronics Engineers
Institute of Nuclear Materials Management
Instrument Society of America
National Academy of Sciences – National Research Council
National Council on Radiation Protection and Measurement
National Fire Protection Association
National Institute of Occupational Safety and Health
Nuclear Energy Liability Insurance Association
Society for Nuclear Medicine

the pertinent technical and professional societies. Approximately 210 NRC staff members participate in the development of nuclear standards, with most of the standards working groups organized under technical and professional societies. The technical and professional societies with which the NRC Standards Development Office interacts are indicated in Table 2.

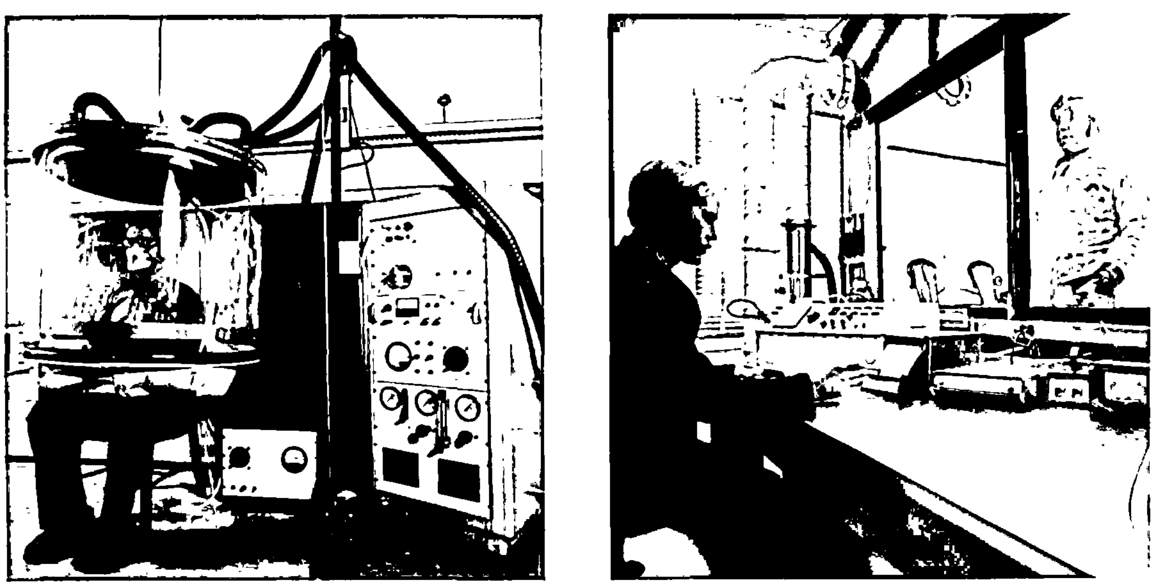
ADDRESSING CURRENT ISSUES

- *Protection of SNM in Transit*
  Regulations are under consideration to strengthen the protection of special nuclear material while it is being transported between plants.
- *Clearances for SNM Access*
  A program is under consideration to provide clearances for personnel having access to or control over special nuclear material in the commercial fuel cycle.
- *Respiratory Protection Program*
  The Commission is preparing amendments to regulations for protection of workers from airborne radioactive materials. The amendments would simplify the regulations for respiratory protection and would provide guidance on acceptable respiratory protection programs.
- *Early Site Review*
  NRC is studying establishment of a formal early site review procedure to improve the efficiency of the licensing process. The early site review procedure basically extends the time scale over which the licensing process can proceed and removes the construction permit review and hearing phase from the critical path.
- *Transportation of Plutonium by Air*
  The current public concern over plutonium being transported in and out of Kennedy Airport has raised several important questions. The staff is in the process of developing a generic environmental impact statement for air transportation of plutonium that will form the basis for rulemaking (see Chapter 4).
- *Steam Generator Tube Problems*
  The problems several pressurized water reactor plants have had with corrosion and erosion of steam generator tubes, particularly identified with the Palisades Plant (see Chapter 7), made the SD staff focus on the need for standards to deal with this problem. The staff developed a guide for surveillance of steam generator tubes and is working with the ASME Code Group, Section XI, on inservice inspection of nuclear plant components to include requirements in its future revisions.

- *BWR Pipe Cracks*
Within the last year, several boiling water reactors have experienced cracking in stainless steel piping containing reactor coolant (see Chapter 7). The need for upgrading standards for inservice inspection to ensure that these cracks can be identified in a timely manner before they become major safety problems is an area that is under evaluation by Standards Development. The NRC staff has had discussions with the ASME Code group, encouraging them — with NRC participation — to cover the inservice inspection of stainless steel piping.

- *Failure of Components to Function*
Reports from NRC inspectors in the field have indicated that many active components — particularly pumps and valves — have not functioned when called on in tests or during operation. The ASME Code principally deals with ensuring structural integrity and not the functioning of components. The staff has undertaken an industry development program, with two ANSI groups, to develop standards to provide greater assurance that pumps and valves will operate when called on. The major focus is presently on qualification testing.

- *QA Standards for Manufacturers*
No quality assurance standards have been developed specifically for reactor component manufacturers. Often this has led to uneven application of quality assurance requirements since the guidance available is not tailored to specific quality needs of the given product. The staff has undertaken, primarily through the industry standards development program, to develop standards applicable specifically to component manufacturers.



Occupational health standards for the protection of licensee personnel from atmospheres contaminated with airborne radioactive materials are set to strictly limit personnel exposures. Exposures normally are limited through engineered controls such as ventilation equipment. When such controls cannot be reasonably applied, or while they are being installed, respiratory protective equipment might have to be worn to limit personnel exposures. Quantitative information required to evaluate such equipment is obtained in tests done for the NRC such as those shown here. At left, an employee inside a portable test chamber at the Los Alamos Scientific Laboratory wears an air purifying respirator. At right, a supplied-air suit is tested. In each test, equipment efficiency is measured as subjects perform exercises simulating working conditions.

10

Cooperating with the States

Partnership in Safety

NRC continued to administer the long-standing program whereby certain nuclear regulatory authority is transferred to States ("Agreement States") having qualified radiation protection programs, and the agency expanded other wide-ranging cooperative and informational activities to keep pace with the States' growing interests in nuclear matters and to enlist their support in carrying out the Commission's Congressional mandates.

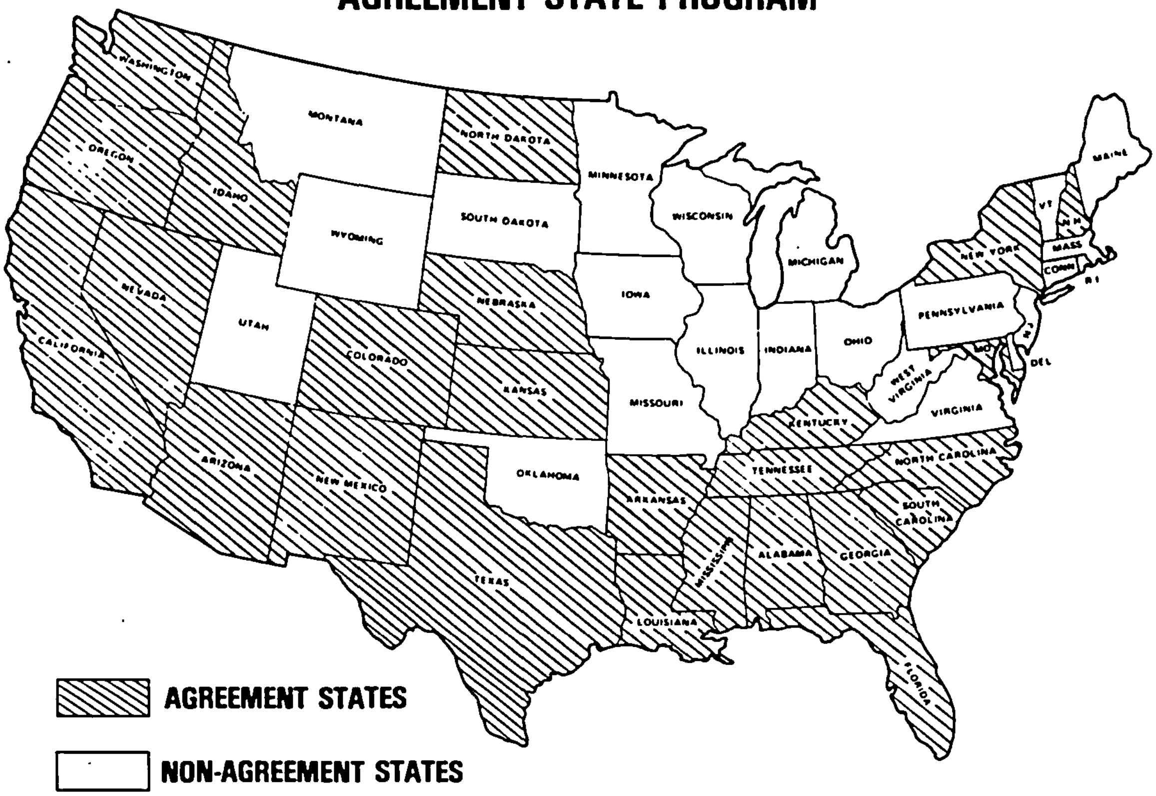
Interaction with States during 1975 was especially apparent, for example, in seeking coordination in the nuclear facility licensing process, in cooperative surveillance of nuclear material shipments, and in assisting the States to develop effective radiological emergency response plans.

The objective of NRC's cooperative activities with the States is to prepare them to take on greater responsibilities in controlling radioactive materials, establish coordination in considering nuclear power plant sites, and maintain open channels of communication to exchange advice and comment on nuclear matters of mutual interest.

Increased importance has been assigned to liaison with State executive and legislative branches, including initiation of a system to track nuclear-related legislation and activities in all States in order to keep abreast of events that could have a significant impact upon the nuclear power program.

In addition, the Commission maintains liaison with regional and national-level State organizations. Technical liaison and collaborative programs have been established with the Conference of Radiation Control Program Directors, Western and Southern Interstate Nuclear Boards, the National Governors Conference, the Conference of State Legislatures, and the National Association of Regulatory

AGREEMENT STATE PROGRAM



Utility Commissioners. During the year, NRC participated with other Federal agencies and the States, through the Conference of Radiation Control Program Directors, in improving and revising regulations for State radiation control programs. NRC also participated in Conference task forces to (a) develop bonding and perpetual care requirements for nuclear licensed facilities, (b) consider disposal of radioactive wastes, (c) improve State regulatory program effectiveness, and (d) identify training needs for State personnel.

AGREEMENT STATES PROGRAM

Section 274 of the Atomic Energy Act authorizes the Commission to enter into agreements with States whereby they assume regulatory authority over radioisotopes, source materials, and special nuclear materials in quantities not sufficient to form a critical mass (a mass capable of supporting a self-sustaining chain reaction). Before concluding an agreement, NRC must find that the State's radiation program is adequate to protect public health and safety and is compatible with that of the NRC. Redeterminations of adequacy and compatibility are made annually by special NRC reviews and exchanges of statistical and technical information with each Agreement State.

At the end of 1975, there were 25 Agreement States exercising regulatory jurisdiction over a total of approximately 10,500 nuclear material licenses, in addition to about 8,500 such licenses administered directly by NRC. The Agreement States were: 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 Washington.

During the year, active negotiations were underway with Illinois, Michigan and New Jersey preparatory to entering into regulatory agreements.

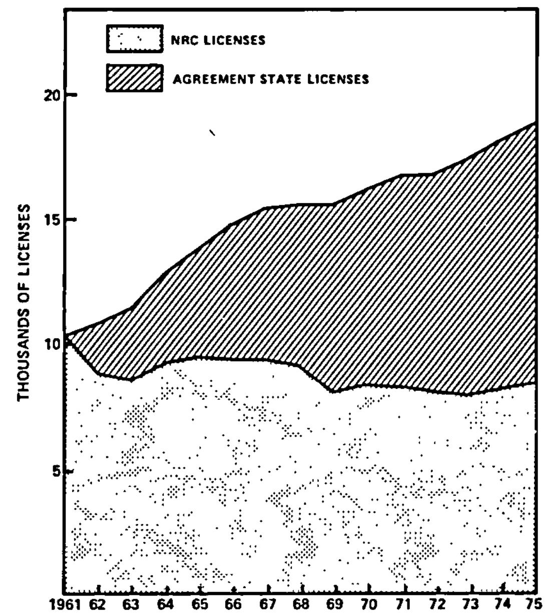
Maintaining Compatibility

To promote an orderly regulatory pattern, NRC conducts a program of cooperation with the Agreement States. This agreement program includes the provision, by NRC, of technical training courses for State personnel; exchange of statistical information on licensing and inspection activities and incidents; exchange of current information on regulations, licensing, inspection, and enforcement practices; technical information; and consultation and technical assistance on special regulatory problems.

Program Reviews. NRC's annual review of each Agreement State program covers six major program elements: organization, administration, personnel, regulations, licensing, and compliance. NRC provides comments and recommendations to assist in maintaining compatible programs and assuring protection of the public health and safety. Based on the reviews and other information, all 25 Agreement State programs were determined to be adequate for the period January-December 1974 (the formal NRC redetermination of adequacy and compatibility of all Agreement State regulatory programs is made on a calendar year basis).

To avoid duplicating regulation, the U.S. Department of Labor accepts NRC's certification that Agreement State radiation control programs are adequate to protect the health and safety of the public and radiation workers. Thus, the Department of Labor does not assert its own regulatory authority under the Occupational Safety and Health Act for Agreement State licensees.

NUCLEAR MATERIALS LICENSES IN EFFECT
1961-1975



DATA AS OF END OF CALENDAR YEARS

Maxey Flats Task Force. Special NRC assistance was provided to one Agreement State after Kentucky State inspectors discovered that the commercial radioactive waste disposal site at Maxey Flats, Ky., was contributing radioactivity to the environment. At the request of Governor Julian M. Carroll, NRC conducted a review of the State-licensed waste disposal ground. The NRC task force—including hydrology and radiological assessment experts—visited the site, studied pertinent data and met with various technical officials to determine if offsite leakage had occurred and, if so, whether the public health and safety had been compromised.

Upon consideration of all available information, the task force reported that no significant hazards were found. The report found that the State and the licensee had been taking appropriate action to improve water management at the burial site so as to minimize migration of radioactive material. The Governor, following review of the findings, indicated the NRC report was fully responsive to his request.

Lee V. Gossick, NRC's Executive Director for Operations, talks with State officials during the NRC-sponsored course on "Regulatory Practices and Procedures" held in Bethesda, Md., in September 1975.

Training State Personnel

NRC conducts training programs for State personnel to assist States in preparing for agreement status and to help existing Agreement States train new radiation control staff. Training during the year included two courses on "Medical Use of Radionuclides for State Regulatory Personnel," one given by Baylor College of Medicine, the other by Georgetown University Medical Center. A total of 37 State personnel attended these courses. A course in "Safety Aspects of Industrial Radiography for State Regulatory Personnel" was given by Louisiana State University and was attended by 15 State personnel. In addition, an orientation course in "Regulatory Practices and Procedures" was given at NRC Headquarters, attended by 20 persons and an "Inspection Procedures" course was presented at the NRC Region III office for 12 selected State personnel. A 10-week course in "Health Physics and Radiation Protection" was given by Oak Ridge Associated Universities for 16 State staffers, and a new course on the regulatory aspects of the use of radioactive material for gas and oil well logging was co-sponsored by the NRC and the State of Texas, with 31 participants.

Annual Meeting

NRC also conducts an annual meeting with all Agreement States to consider regulatory matters and policies of common interest. Following the 1974 meeting, the Agreement States made several recommendations, and NRC has taken action on all of the State recommendations with the exception of one: a recommendation that NRC initiate legislation to bring accelerator-produced and naturally occurring radioactive material under its jurisdiction. This recommendation, which would require amendment of the Atomic Energy Act, is under study by NRC staff.

OTHER NRC-STATE ACTIVITIES

Collaborative Monitoring

Nineteen States participate with NRC in monitoring low-level radioactive emissions at the point of release at nuclear power plants within their borders. The collaborative program is carried out under written contracts whereby NRC provides some funds, technical support and training to improve the State's analytical capabilities. Participating States at the end of fiscal year 1975 were Alabama, Arkansas, California, Colorado, Connecticut, Florida, Illinois, Maine, Maryland, Michigan, Minnesota, Nebraska, New Jersey, New York, Pennsylvania, South Carolina, Vermont, Virginia, and Wisconsin.

Essentially the same States assist NRC in comparing the analysis of duplicate environmental samples taken from NRC licensed facilities on a long-term repetitive basis to evaluate the overall quality of the environmental program at nuclear power plant sites. (See further discussion in Chapter 6.)

Surveillance of Nuclear Shipments. In view of overlapping responsibilities of NRC, the Department of Transportation (DOT), and the States in the matter of transport of radioactive materials, short term (3 months) surveillance contracts or agreements have been executed between NRC, DOT, and the States of New Jersey, Minnesota, South Carolina, New York, Louisiana, Illinois, Oregon, Missouri and Texas, and the City of New York.

These jurisdictions will collaborate with NRC and DOT in developing a mutually acceptable surveillance program within their jurisdictions. It was expressly agreed that the Federal agencies may participate in any activity conducted under the arrangements, and that all data developed will be shared freely and promptly.

The purpose of these arrangements is to develop and maintain information concerning transportation of radioactive materials, institute a surveillance program over the transportation of a specified sampling of radioactive materials, identify and report any situations that are not in compliance with existing rules and regulations, and work together in assessing the transport regulations.

Under an agreement with NRC and DOT, Illinois Department of Public Health personnel surveyed transport company facilities to determine whether the day-to-day handling of packages containing radioactive material represents a health hazard to those workers who come in contact with them. Measurements of radiation levels were made on the surfaces of packages at airline and trucking company freight terminals, and inside and outside of trucks used to transport radioactive materials. (Photos by Mary Hock, Illinois Dept. of Public Health.)

Emergency Response Plans

Assuring that State and local governments have effective plans for responding to any radiological emergency at nuclear facilities has been of increasing concern to NRC and other Federal agencies which have emergency preparedness responsibilities. Although the likelihood of a serious accident at a licensed nuclear facility having an impact on public health and safety is very low, NRC and other agencies consider it prudent to plan offsite emergency actions in support of fixed nuclear facilities. Typically, these plans address such matters as evacuation, medical support, emergency communications and accident assessment.

As the "lead agency" in the Federal interagency program spearheading this effort, NRC during 1975 actively assisted State and local governments in developing and improving such emergency response plans. Other Federal agencies engaged in this program are the Environmental Protection Agency, the Defense Civil Preparedness Agency, the Energy Research and Development Administration, the Federal Disaster Assistance Administration, the Department of Transportation, and the Department of Health, Education, and Welfare. Responsibilities are assigned by the Federal Preparedness Agency of the General Services Administration.

During the year, NRC formally reviewed 25 State radiological emergency response plans, following up in each case with specific guidance for improvement. A "Field Training Cadre," made up of representatives from NRC and other Federal agencies, visited 17 States to provide direct assistance. Similar "cadres" evaluated actual field tests of emergency plans in six states.

A planning guidance document, "Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans in support of Fixed Nuclear Facilities" (WASH-1293), was published in December 1974, and distributed to State and local governments in early 1975. This document is the basic standard.

Training Programs. In the area of training, the "Federal Interagency Course in Radiological Emergency Response Planning" was established by NRC and other Federal agencies at the Defense Civil Preparedness Agency Staff College in Battle Creek, Mich. Approximately 125 State and local government emergency planning personnel have attended the course. It is planned to continue the course through fiscal year 1976 and beyond.

NRC and other Federal agencies are developing a second course in "Radiological Emergency Response Operations" to be offered to State and local government personnel during 1976.

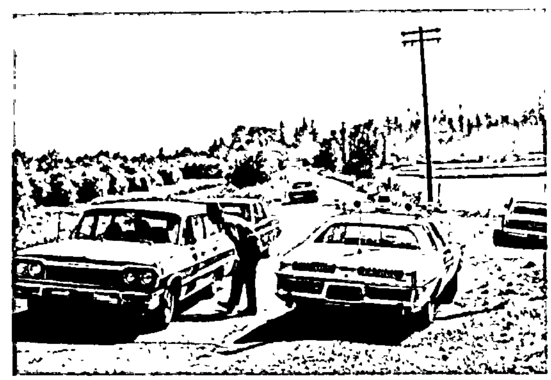
Coordination in Licensing and Siting

During the year, NRC continued efforts to reduce the licensing time for nuclear power plants by reducing the duplication of State and Federal activities through increased coordination. Twenty-three States have enacted legislation that provides for varying degrees of regulation of location and construction of both power plants and transmission lines. Other States are working on similar legislation.

In an effort to reduce overlaps between the respective NRC and State responsibilities and streamline the licensing process, the NRC was instrumental in establishing a Federal-State Siting Coordinating Steering Committee and three working groups composed of representatives of NRC and other Federal agencies with energy responsibilities, as well as States with ongoing power plant siting programs.

As a result of these efforts, a joint Federal-State power plant siting conference was held in 1975 and attended by approximately 70 persons representing 27 States and six Federal agencies. A number of

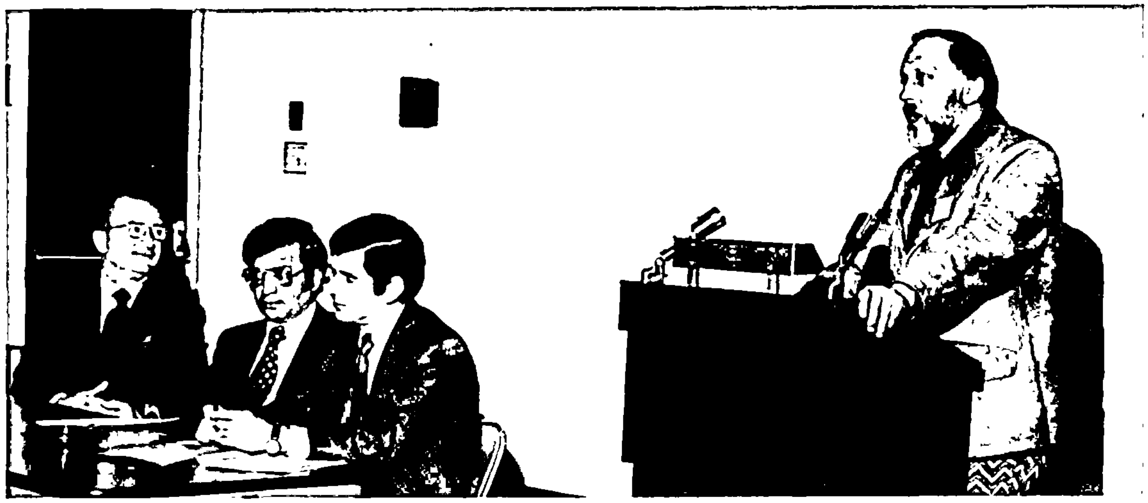
Federal, State and local personnel join Portland General Electric Co. in a test of emergency procedures for the Trojan Nuclear Plant near St. Helens, Ore. Testing the utility's radiological emergency response plan was one of the NRC's requirements before an operating license could be issued for the plant. Participants included health agencies in Oregon and Washington, State and local police, ambulance and hospital facilities, and U.S. Coast Guard and Weather Bureau personnel. Above, the Emergency Control Center staff operates from the plant during the drill. At right, a Columbia County Sheriff's deputy informs a motorist of the exercise at a temporary roadblock in the area.



recommendations developed at the conference are currently being implemented by NRC, including arrangements for NRC and States to keep each other informed of important stages in the review of an application for a new nuclear power plant, appointment of State and NRC liaison officers, and joint Federal-State efforts to coordinate hearings. A second conference is planned during 1976.

NRC also works closely with States to insure that respective rules and regulations are compatible and, to the maximum degree possible, complementary. In the spring of 1975, NRC began providing detailed briefings to the States on its own rules, regulations and policies.

The Commission has continued to work with the State of Maryland in efforts to hold joint hearings on the proposed Douglas Point Generating Station. (These hearings have been delayed by the applicant's reassessment of its need for the facility.) Discussions are continuing with Oregon and New



William Luch, Chairman of the Oregon Nuclear and Thermal Energy Council, delivers the keynote address at the Federal-State Conference on Power Plant Siting, sponsored by NRC on April 7-9, 1975. At left are Lee V. Gossick, Commissioner Victor Gilinsky, and Herbert H. Brown of NRC.

York on the possibility of holding joint hearings on applications for facilities within those States.

An important part of the overall program for the early review and designation of nuclear power plant sites is development of demonstration programs. An ongoing program with Maryland has helped to point the way. Under the Maryland Power Plant Siting Act of 1971, the State has the responsibility to acquire sites for energy-related facilities. NRC is working jointly with Maryland to test a methodology for site selection and evaluation. That program has led to expansion of the concept of resolution of issues on a regional basis by a recently initiated joint program with the Southern Interstate Nuclear Board (SINB) and the NRC. This program has progressed to the point of identifying important issues that show promise of being resolvable on a multi-State basis.

The participation of a representative of the National Governor's Conference on the SINB Steering Committee of State representatives and the participation of representatives of other interested Federal agencies in planning meetings point toward broad State-Federal cooperative efforts at this early stage in developing the program.

Cooperating with Other Nations

Toward Global Responsibility

The pace of NRC interaction with foreign governments and international organizations accelerated markedly in 1975 due to the broadened responsibilities assumed by the new agency. Focal points of activity include power reactor safety and research, advice about abnormal occurrences at operating nuclear power plants, plant physical security and safeguards for fissionable nuclear materials, and licensing and regulating actions and recommendations.

POLICY AND GOALS

The NRC goals of safe and secure operation of nuclear facilities and accountable use of nuclear fuels are shared with an increasing number of foreign governments and international organizations. As more nuclear power reactors become operational each year in the U.S. and other countries, new questions arise concerning safety related to construction and operation of nuclear facilities or the use, storage, disposal, and accountability of nuclear materials.

Consistent with the agency's mandate to protect the public, the environment, and national security, the NRC coordinates broad international cooperative activities through programs of reciprocal regulatory information exchange and cooperation in development of standards, and administers export and import licensing responsibilities. In addition, the NRC engages in technical information exchange agreements on reactor safety research.

Short range goals are to increase the number of countries participating in the information exchanges and to increase the volume and expand the scope of the information received from other countries.

Professor Ezio Clementel, President of the Comitato Nazionale per l'Energia Nucleare of Italy, and NRC Commissioner Richard T. Kennedy exchange copies of the technical exchange arrangement at the signing ceremony on May 29, 1975.

As information made available through correspondence and personal visits broadens knowledge of overseas safety problems and practices, the international programs are expected to make continuing contributions both to U.S. safety activities and to the safety of nuclear power operations throughout the world.

EXCHANGE OF INFORMATION

Two kinds of exchange agreements are negotiated and implemented by the NRC—one covering information exchange on general regulatory safety matters, and the other arranging participation in discrete cooperative programs of safety research.

Information Exchange Arrangements

Building on a regulatory safety information exchange program started in 1974, NRC, during its first year of operation, concluded bilateral arrangements with four additional foreign governments—the United Kingdom, Italy, Denmark and West Germany—bringing to nine the number of such arrangements. Previous arrangements were signed with Japan, France, Switzerland, Sweden and Spain.

The objective of these arrangements is to make available the operational and other safety and reliability-related data being accumulated in the U.S. and foreign nuclear power programs. The projected growth in worldwide power reactor operation over the next four years alone is expected to generate more operational experience than all the reactor operating experience to date. This new information will be of great value in updating reliability estimates, confirming research findings, and performing other safety and economic analyses.

Through these bilateral arrangements, it has already been possible for NRC to offer direct assistance in resolving specific technical problems. And, by receiving timely information on the operating experience of foreign reactors, NRC has gained insight into U.S. operating problems. The pipe-cracking problems observed in U.S. and Japanese boiling water reactors early in 1975 illustrate the possibility of international collaboration in assessing and correcting power reactor safety problems.

Significant NRC publications and announcements are distributed through the exchange arrangement administrator, and foreign information is received by the NRC's International Office. Exchanges related to standards development provide a common base for protecting the health and safety of all people.

Cooperation in Research

NRC has concluded technical information exchange agreements relating specifically to reactor safety research with the governments of France, West Germany, Italy, Japan, Sweden, and Denmark. Negotiations for such agreements were also started with the governments of the Netherlands and Belgium. These arrangements facilitate cooperation in the field of reactor safety research through exchange of technical reports, visits of technical experts, and organization of specialist meetings and conferences.

During 1974, the exchange of light-water reactor safety research reports was initiated with West Germany, Japan, and France. Reports from these countries are now routinely sent to the Energy Research and Development Administration's Technical Information Center at Oak Ridge, Tenn., for reproduction and distribution to scientists and engineers in industry, government laboratories and universities in the United States.

Individuals and teams of foreign technical experts from participating countries have visited NRC research experts and facilities to discuss detailed research results and to improve mutual understanding of light-water reactor safety problems. Over 40 foreign technical experts attended the second annual Light-Water Reactor Safety Research Information Meeting (sponsored by the former AEC) in September 1974 at Germantown, Md. U.S. scientists and engineers have, in turn, visited European and Japanese establishments.

In November 1974, following the initiative of Secretary of State Henry Kissinger, the International Energy Agency (IEA) was formed, representing many of the industrial nations of Europe and Japan. A major objective of the new organization is to provide a mechanism for cooperation in the field of energy research and development. Reactor safety research was designated as one of nine areas of energy research and development where multi-national cooperation was encouraged. Four NRC research programs—the Loss-of-Fluid Test (LOFT); Power Burst Facility (PBF) operated by Aeroject Nuclear Corporation in Idaho; the Plenum Fill Experiment (PFE) conducted by Battelle Pacific Northwest Laboratory; and the Heavy Section Steel Technology (HSST) conducted at Holifield National Laboratory—were suggested by the U.S. as important projects for international cooperation within the framework of the IEA. (See Chapter 8 for descriptions of research programs.)

The first of a series of agreements for cooperation on the LOFT program within the framework of the IEA was signed by NRC and the Federal Republic of Germany on June 20, 1975. The West German Government agreed to contribute approximately $4 million to the LOFT program in return for the opportunity to have its technical experts participate in the program over a four-year period. Other countries expressing an interest in participating in a cooperative LOFT program included Austria, Japan, Spain and Sweden.

International cooperation on PBF, PFE and HSST research programs will be on a reciprocal basis whereby foreign governments will contribute data and results of comparable reactor safety research programs from their countries. Countries expressing an interest in cooperation in the PBF program are Austria, Denmark, Italy, Germany, Japan, Spain and Sweden; for the PFE program, Austria, Denmark, Italy, Germany and Japan; and for the HSST program, Denmark, Germany, Italy, the United Kingdom, Sweden and Spain.

Since reactor safety is a universal concern, the past year's progress in international co-operation in such research represents an important initial step toward achieving the maximum benefits from improved under-standing of reactor safety phenomena.

Foreign Visitors

Concurrent with increases in the number of bilateral arrangements completed, the number of operating foreign reactors, and the number of countries with operating reactors, has been the largescale increase in the number of foreign visitors to NRC.

Mutual interest in technical reactor safety, structural integrity, material safety and accountability, standards development, and organizational issues attracted 392 visitors representing 28 countries and three international organi-zations in calendar year 1975. Meetings were scheduled with appropriate NRC technical and managerial staff members after identification of visitor areas of concern. Visitors thereby received informa-tion from NRC while providing the NRC staff with valuable insights into technical problems, research programs, and public opinion in their countries and organizations.

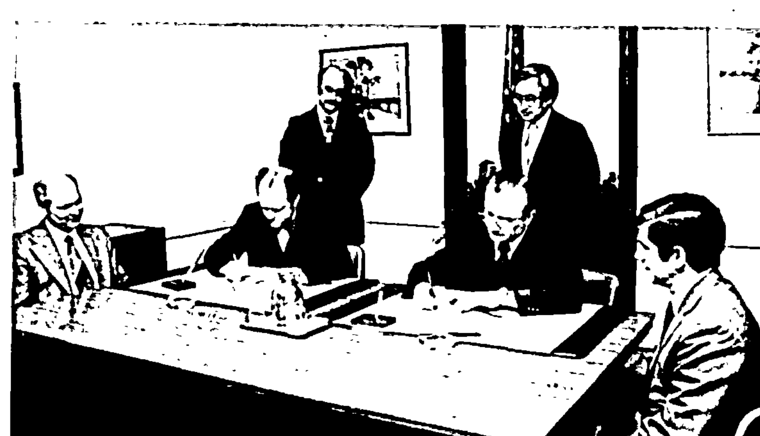
INTERNATIONAL ATOMIC ENERGY AGENCY

NRC continues to work closely with the International Atomic Energy Agency (IAEA), a self-governing agency under the aegis of the United Nations, which is respon-sible for international nuclear energy activi-ties, including nuclear safety and safeguards.

During the year, NRC contributed sub-stantial support to the IAEA's intensified program for the collation and preparation of internationally acceptable codes and guides on the siting, safety, and reliability of nuclear power plants. In addition, NRC furnished IAEA with a large number of U.S. nuclear safety documents to be used as models and examples for international safety codes and guides.

NRC experts participated in other IAEA activities, including efforts to strengthen international safeguards procedures and update recommendations to member coun-tries on the protection of nuclear facilities and materials from terrorist attacks.

In 1975, NRC also supported IAEA training activities, especially those directed towards training technical and administra-tive manpower in developing countries which are planning or constructing their first nuclear power plants. In April and



As part of an international cooperative program on energy research and development, NRC is negotiating a series of agreements with other coun-tries for cooperation in NRC-sponsored research programs. The first such agreement on the LOFT program was signed on June 20, 1975. Shown participating in the signing ceremony are: seated, Heinz Seipel, Germany; W. Schmidt-Kuester, Ger-many; Lee Gossick, USNRC; and Herbert Brown, USNRC; standing, K. Wiendieck, German Embassy; and Walter Kato, USNRC.

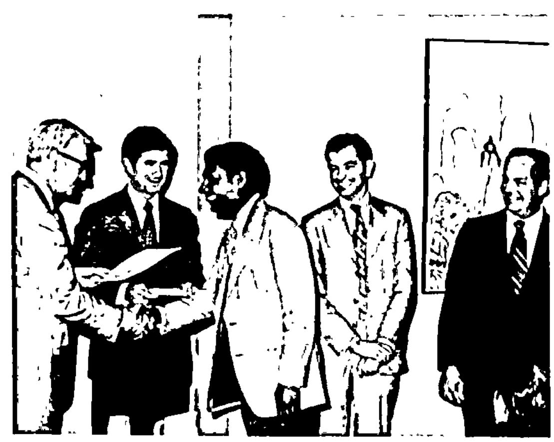
May 1975, NRC hosted and co-sponsored with IAEA a three-week course on the principles and techniques of regulating nuclear power for public health, safety, and environmental protection. Thirty-one foreign nuclear safety officials attended the course. NRC also prepared the regulatory/safety aspects of a syllabus for a 15-week course on nuclear-power-project planning and implementation to be held at ERDA's Argonne National Laboratory twice during 1976.

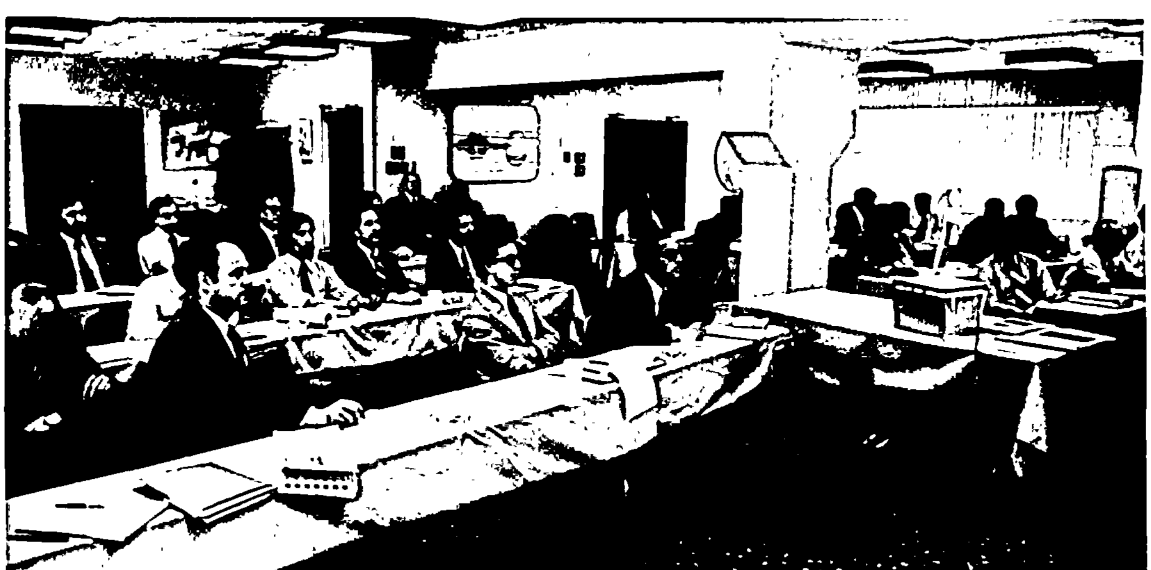
IAEA Safeguards

NRC participated with the Department of State, Arms Control and Disarmament Agency and ERDA in (1) continuing efforts to develop a U.S.-IAEA safeguards agreement for application of IAEA safeguards in the U.S., (2) preparation of a Presidential report to Congress outlining ways of strengthening IAEA safeguards through allocation of funds made available under the Foreign Assistance Act of 1974, and (3) a meeting with the Safeguards Policy Committee of the Atomic Industrial Forum to discuss the draft U.S.-IAEA Agreement and its anticipated impact on U.S. industry.

Under an anticipated U.S.-IAEA agreement, the international agency would implement IAEA safeguards controls at certain U.S. nuclear facilities. Affected plants would be mainly commercial operations—

Commissioner Richard T. Kennedy presents a certificate to K. V. Mahadeva Rao of India at the conclusion of the Training Workshop on Methods and Technical Bases of Nuclear Energy Regulation on May 16, 1975. From left, Commissioner Kennedy; Herbert H. Brown, USNRC; Mr. Rao; Morris Rosen, IAEA; and Joseph D. Lafleur, Jr., USNRC. This NRC-IAEA co-sponsored course was attended by officials from Bolivia, Brazil, Bulgaria, Chile, Czechoslovakia, Egypt, Ghana, Greece, India, Iran, Israel, Korea, Mexico, Pakistan, Philippines, Poland, Singapore, Spain, Thailand, Turkey, and Yugoslavia. In addition to NRC and IAEA lecturers, the course included guest lecturers from eight other countries.





defense-related facilities are exempt. The U.S. has volunteered to apply IAEA safeguards control to domestic operations to demonstrate willingness to accept the same controls that nonnuclear-weapons countries are obliged to accept under the Treaty on the Non-Proliferation of Nuclear Weapons. The purpose of the IAEA safeguards program is timely detection of the diversion of significant quantities of nuclear material or facilities from peaceful uses to weapons use, and deterrence of such diversion by the risk of early detection. The U.S.-IAEA Safeguards Agreement is expected to be approved in 1976. NRC will be the primary contact with IAEA for licensed activities.

NRC staff participated in the following IAEA safeguards meetings:

(1) A panel of experts working under IAEA sponsorship to revise and update the "Recommendations for the Physical Protection Of Nuclear Materials."

(2) An IAEA advisory group meeting on "Systems of Accounting for and Control of Nuclear Material."

(3) An international symposium on safeguarding of nuclear material.

EXPORT CONTROL

NRC has the responsibility under the Energy Reorganization Act and the Atomic Energy Act for issuance of export licenses for nuclear facilities and materials.

Because of the increasing dangers of international nuclear proliferation, the Commissioners have devoted their personal attention to problems surrounding the export of nuclear materials.

NRC, in carrying out its export licensing program, assures that export of source and special nuclear material and nuclear facilities will be made only to those countries which accept adequate international safeguards controls. In addition, NRC determines that those countries receiving significant quantities of plutonium or high-enriched uranium have adequate physical security programs. In view of the implications inherent in exporting and importing such materials, the Commissioners require that proposed export-import actions involving significant quantities be referred by the NRC staff to the Commission for its review prior to approval. In addition, guidelines were developed to instruct the staff in the review of such applications. Because of international safeguards implications, primary attention has been focused on exports.

In performing its export control functions, NRC has developed working relationships with the Executive Branch agencies having nuclear export responsibilities so that NRC's export licensing decisions are based on the most complete available information. Thus, under the procedures that have been worked out, NRC retains its independent role in the export process without duplicating the duties and responsibilities of the Executive Branch. Information and assessments, principally received from the State Department, the Arms Control and Disarmament Agency, and ERDA, are generally of two types: technical data relating to safeguards (both materials accounting and physical security), and information of a more general policy nature bearing on the relation of the export to United States non-proliferation objectives. These data, in addition to information and assessments developed independently by NRC, are considered by NRC in making its final determination on the issuance of the export license.

Export licensing actions totaled 277 during the entire fiscal year 1975, including 46 for byproducts, 68 for source material, 160 for special nuclear material, and three for reactors. All nuclear materials exported from the U.S. are governed by international safeguards designed to detect and deter diversion of the materials to military purposes.

12

Responding to Public Concerns

Public Participation in Regulation

Recognizing that regulation of nuclear energy involves vital interests of the whole society, the Nuclear Regulatory Commission has made openness and candor with the public a cardinal principle of operation. This principle governs virtually every activity of the agency, benefiting the public, the industrial users of nuclear energy and the NRC. The public gains awareness of the issues and the chance to play a direct role in resolving them; the industry gains knowledge of specific public concerns and the chance to accommodate them early in its planning; the NRC gains exposure to a broad range of views and the chance to earn public confidence in the independence and fairness of the regulatory process.

The specific means by which the public is kept informed of NRC's licensing activity and enabled to participate at various phases of the process are set forth in this chapter, together with a review of Commission decisions and court actions undertaken in 1975.

INFORMING THE PUBLIC

A major vehicle for keeping the public informed of NRC activities is the public announcement—of Commission actions, proposed or effected changes in regulation, availability of documents, dates and locations for licensing hearings and rulemaking proceedings, safety-related incidents at licensed nuclear facilities, and enforcement actions. *Federal Register* notices also are issued on many of these same actions.

The Office of Public Affairs handles numerous inquiries from the news media and the general public and, when appropriate, arranges for reporters to interview members of the Commission and/or senior staff members on matters of more than routine interest. Press conferences are conducted on some of these actions. In addition, members of the Commission and the senior staff frequently speak or conduct business outside of Washington and often meet with local newsmen or hold news conferences on these occasions. NRC staff members also respond in appropriate detail to written requests for information from the public, whether addressed directly to the NRC or referred from Congressional offices or the White House. Testimony before Congressional committees by members of the Commission and senior staff is another important means for making information available to the public.

Most documents originated by NRC, or submitted to it for consideration, are

The NRC Public Document Room at 1717 H Street, N. W., Washington, D. C., contains approximately 3,000,000 documents and is used by over 3500 people a year. Also available to users of the PDR are typewriters, copying services, and a microfiche reading machine.

At left, Mrs. June Allen, President of the North Anna Environmental Coalition, examines documents concerning the nuclear power facilities being constructed by the Virginia Electric & Power Co. The documents are provided by the NRC and are housed in the "Local Public Document Room" at the University of Virginia Library in Charlottesville. This is one of the more than 130 Local PDRs established by NRC to aid the public in areas where nuclear power plants are proposed for construction and operation. Gregory Johnson and Laura Brubaker (at right), of the University's library staff, organize and maintain the documents and assist users of the PDR.

placed in the Commission's Public Document Room in Washington, D.C., for public inspection; the range of documents available goes well beyond requirements of the Freedom of Information Act. In addition, documents related to a specific licensing proceeding or licensed operation are made available in the local public document room established in the vicinity of each proposed or existing nuclear facility. More than 100 such rooms are now open to the public. Further, a listing of those operational events at nuclear power plants which must be reported to NRC is placed in each public document room every two weeks.

NRC also submits a quarterly report to Congress of "abnormal occurrences" at licensed nuclear facilities, as required by law. "Abnormal occurrences" are unscheduled events which the Nuclear Regulatory Commission considers significant from the standpoint of public health and safety (see Chapter 7). In addition, interruptions in power generation, malfunctions or other incidents with noteworthy safety implications are made public by the NRC, whether or not the licensee makes public announcement of it.

A major, ongoing effort is the review of all regulations, policies and procedures in effect when the NRC was established on January 19, 1975, to assure that they properly reflect all of the Commission's regulatory responsibilities and are fully responsive to the legitimate interests of the public. This review seeks to provide all interested parties—utilities, environmental and public interest groups, manufacturers, suppliers and others—with the opportunity to make NRC aware of their views. Informal meetings are arranged between interested parties and the Commissioners or senior staff; proposed regulations and guides are circulated for comment; more official events, such as meetings of the Advisory Committee on Reactor Safeguards, are opened to the public; and public rulemaking hearings are held to deal with broad safety and environmental issues on a generic basis.

FORMAL PARTICIPATION

Informing the public is only the beginning of the NRC's commitment to openness; providing the public an opportunity to be heard and giving full consideration to every valid concern is equally essential. The mandatory public hearings of the Atomic Safety and Licensing Board (see below) on each application for a construction permit, as well as the non-mandatory hearings on an application for an operating license (held on request), afford several options for actual public participation. Notice of a hearing on a given application is published well in advance in the *Federal Register* and posted in a document room near the proposed site for a nuclear facility, together with a copy of the full application. Interested persons or groups are invited to petition the Licensing Board for the right to participate in the hearing by: (1) submitting a written statement at the hearing; (2) making an oral presentation at the hearing; or (3) becoming an "intervenor" in the proceeding with full participatory rights, including cross-examination of other participants. Should the Licensing Board disallow a petition, appeal may be made to the Atomic Safety and Licensing Appeal Board (see below) by the petitioner; in some instances, the Commission may rule on a petition. Ultimately a petitioner may seek a ruling in the appropriate Federal Court of Appeals and the Supreme Court of the United States.

To facilitate public participation, hearings of the Licensing Board are commonly held in communities near each proposed facility site. When intervenors are to be involved in the hearing, they participate fully in prehearing conferences with all other interested parties for the exchange of data and identification of issues in contention.

Like its predecessor, the Atomic Energy Commission, the Nuclear Regulatory Commission has continued to receive requests from members of the public for financial assistance to help defray the costs of participation in licensing proceedings. Competitive proposals to conduct an independent study of this question were solicited in 1974 by the AEC, from among which the Nuclear Regulatory Commission selected a Washington, D.C. law firm to develop data and background information which would serve to focus the issues to be explored in any future rulemaking proceeding. The study report, entitled "Policy Issues Raised by Intervenor Requests for Financial Assistance in NRC Proceedings" (NUREG-75/071) was completed in July 1975. Following issuance of the report and before deciding what further steps should be taken, NRC has invited public comments on the report. In particular, the Commission has asked for views and recommendations on the following issues:

(1) Whether a tentative conclusion by the AEC that statutory authority for financial assistance exists is correct;
(2) Whether provision of financial assistance in some or any of the forms discussed in the study report is desirable as a matter of policy choice;
(3) If financial assistance is deemed desirable, what priorities should be observed and what specific rules should govern grants of assistance;
(4) Whether there are preferable alternatives to financial assistance.

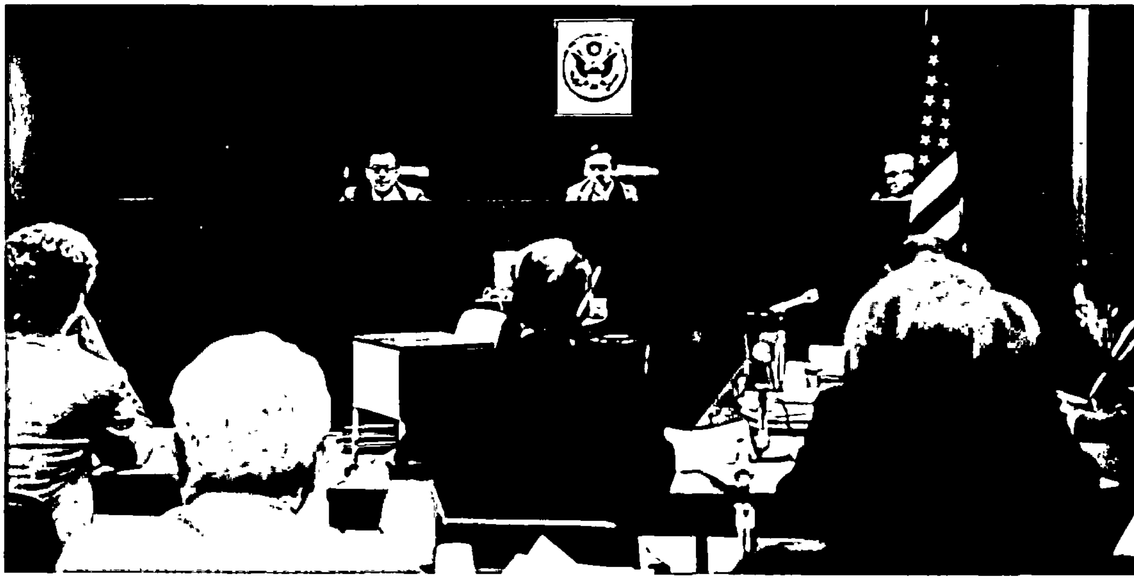
ATOMIC SAFETY AND LICENSING BOARDS

Public participation in the licensing process reaches fruition in proceedings conducted by Atomic Safety and Licensing Boards, for it is here that the public may

OPPORTUNITIES FOR FORMAL PUBLIC HEARINGS IN NRC PROCEEDINGS

Type of Proceeding	*Opportunity for Hearing*	*Purpose of Hearing*	*Criteria for Granting Hearing*	*Unit Deciding To Hold Hearing*
RULEMAKING Proceeding	Prior to issuance of final rule.	To determine whether a proposed rule should be adopted.	At the discretion of the Commission.	Commission (which may decide to hold informal or "hybrid" hearing).
MANUFACTURING LICENSE Proceeding*	Mandatory hearing prior to issuance of manufacturing license.	To determine whether a license authorizing the manufacture of a production or utilization facility of a particular design should be issued.	Mandatory hearing on safety and environmental issues.	Mandatory hearing before Licensing Board.
CONSTRUCTION PERMIT Proceeding*	Mandatory hearing prior to issuance of construction permit	To determine whether a particular production or utilization facility should be constructed at a particular site and, where indicated, to resolve adverse antitrust matters.	Mandatory hearing on safety and environmental issues; on antitrust matters, upon request by interested persons or Attorney General or at discretion of Commission.	Mandatory hearing before Licensing Board.
OPERATING LICENSE Proceeding*	Prior to issuance of operating license.	To determine whether a particular production or utilization facility should be permitted to operate; antitrust review where significant changes have occurred since previous antitrust review.	Request by any person whose interest may be affected by proceeding who raises genuine issue of material fact, and at discretion of Commission; in addition, in the case of antitrust review, there must be determination by the Commission that significant changes have occurred.	Commission, Appeal Board or Licensing Board, as appropriate.
MATERIALS LICENSE Proceeding	Either prior to or after issuance of materials license.	To determine whether a particular materials license should be issued or remain in effect.	Request by any person whose interest may be affected by proceeding and at discretion of Commission.	Commission, Appeal Board, Licensing Board or Administrative Law Judge, as appropriate.
SHOW CAUSE Proceeding (to modify, suspend or revoke a license or for other appropriate action).	Prior to issuance of final Commission Order.	To determine appropriate action to be taken.	Upon demand by person cited in Show Cause Order or by request of other persons whose interest may be affected, upon making requisite factual showing.	Commission

*An opportunity for hearing is also provided prior to issuance of amendments to manufacturing licenses, construction permits and operating licenses which involve significant hazards considerations. If there are no significant hazards considerations, opportunity for hearing may be provided after such amendments are issued.



Hearings to determine whether a construction permit should be issued for a proposed nuclear power plant are conducted by three-member atomic safety and licensing boards. The hearings are open to the public. In addition to participation by the utility involved and the NRC staff, participation in the hearing may include testimony, questions, or statements from members of the public, private organizations, or other Federal, State or local agencies.

place its concerns, information, and conclusions on the record before an independent tribunal.

It is a requirement of the Atomic Energy Act of 1954 that no construction permit for a nuclear power plant and related facilities may be issued until a public hearing has been held on the application. This hearing is conducted before a Licensing Board authorized to issue a decision on the application (known as an "Initial Decision") which, subject to the NRC's review and appellate procedures discussed below, usually becomes the final NRC decision. Although a notice of hearing inviting public participation is published shortly after receipt of a construction permit application, the hearing itself takes place after completion of the NRC staff's safety or environmental review. Ample notice of the proceeding is given to the public, State and local agencies, and other interested groups.

Additionally, the Atomic Energy Act requires that, before a nuclear power plant or related facility may be licensed to operate, or before certain license amendments may be issued, an application must be filed and an opportunity for hearing must be provided. Thus, members of the public, State and local agencies, and other interested groups can cause a hearing to be held at this stage of the licensing process, within certain legal requirements. Public participation is also invited in proceedings instituted by the NRC staff.

The Atomic Energy Act also requires that, under certain circumstances, a determination must be made by NRC as to whether any activity licensed by it may create or maintain a situation inconsistent with the antitrust laws, and that the NRC take appropriate action should this determination be affirmative. While the procedures laid down by the Act for this review are more complex than those outlined in the preceding paragraph, a similar opportunity to trigger a hearing is provided.

Each of the Boards that conduct these hearings consists of three members drawn from the Atomic Safety and Licensing Board Panel—a body of legal, technical, environmental, and other experts appointed by the Commission. As of December 31, 1975, the Panel included 17 full-time and 46 part-time members. Of these 63 members, 20 are lawyers, 16 environmentalists, 14 engineers, 9 physicists, 3 economists, and 1 chemist. (See Appendix 3 for names of members.) All members are chosen for their recognized experience, achievement, and independence. Assignments to a given Licensing Board are based on the kinds of issues involved in the application to be considered. Separate hearings may be conducted by a Board on the technical aspects of an application and environmental questions, and 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.

Public hearings before the Licensing Boards consumed approximately 375 days during calendar year 1975, as compared with 222 days in 1974. Boards issued 17 Initial Decisions authorizing licenses during 1975. Because of the increasingly complex issues raised by public participants and other parties, the Boards found it necessary during this period to rule on a number of procedural matters before hearing evidence and rendering an Initial Decision. Significant procedural determinations by the Licensing Boards are published, together with Initial Decisions, in the official Commission reporter, *Nuclear Regulatory Commission Issuances,* which is available to the public.

Antitrust hearings played an increasing role in the work of the NRC's Boards. One full-scale evidentiary hearing begun in December 1974 was still in progress at year end. Two antitrust proceedings were settled by the interested parties with ASLB approval in April, and the first Initial Decision following a full-scale antitrust hearing was issued in July. In December 1975 another full-scale evidentiary hearing began.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

The Appeal Boards were first established in 1969 under the Atomic Energy Commission to exercise the Commission's authority and perform its review functions in facility licensing proceedings. In 1972, AEC created an Appeal Panel from which three-member Appeal Boards for individual proceedings would be selected by the Chairman of the Panel. In accordance with the Energy Reorganization Act of 1974, the functions of the Appeal Boards were specifically transferred to the Nuclear Regulatory Commission. (See Appendix 3 for current membership of the Appeal Panel.)

The decision of a Licensing Board on a petition to intervene and its "Initial Decision" to approve or disallow a permit or license are subject to appeal. The Appeal Board may base its review on exceptions filed by one of the interested parties or proceed on its own initiative. The Licensing Board itself may certify a question or refer a ruling to the Appeal Board for a decision at any point in the Licensing Board hearing. Decisions of the Appeal Board are not subject to further appeal by interested parties within the administrative process, though judicial review may be sought in the Federal courts. In some cases the Commission may review an Appeal Board action, but only on its own initiative and not at the behest of any party. Ordinarily the decision of the Appeal Board represents the final order of the Nuclear Regulatory Commission in any licensing matter.

During 1975, Appeal Boards completed or undertook review of 85 matters. (Published decisions during that period are numbered ALAB-252 through ALAB-303.)

All except the first four of these decisions were published in the NRC's monthly publication entitled *Nuclear Regulatory Commission Issuances* (The first four decisions, which were rendered during January 1975, prior to the reorganization, appear in the last volume of the AEC's similar publication *Regulatory Adjudication Issuances.*) In order to assist readers in identifying and locating various issues discussed by the Appeal Boards, these published opinions included, for the first time, brief summaries of the rulings and headnotes of significant legal issues. The summaries and headnotes (which also appear with respect to published decisions of the Commission and the Licensing Boards) were prepared under the direction of the Appeal Panel staff.

The opinions rendered by Appeal Boards during 1975 covered a broad range of both substantive and procedural issues, involving the safety, environmental, and antitrust aspects of reactor construction and operation. Among the more significant highlights during the first six months of 1975 were the following:

(1) In the *Limerick* decision (facility at Pottstown, Pa.), the Appeal Board treated in considerable detail the required environmental review of alternative methods of providing water for the cooling system.

(2) Delays in the projected dates of operation for several proposed reactors formed a backdrop for several Appeal Board opinions during this period. The *Nine Mile Point* decision (facility at Scriba, N.Y.) spelled out standards for considering the need-for-power issue, including the weight which may be accorded to econometric demand analyses and the criteria by which such analyses should be evaluated. The *Douglas Point* (Md.) memorandum outlined the circumstances under which—given a delay in projected need for a facility (and hence in projected construction and operation)—early hearings on certain issues might nevertheless be held. And in the *Vogtle* (Ga.) proceeding, the Appeal Board on its own initiative dealt with the question of the continued validity of construction permits when the utility has decided to postpone construction of the facility for an extended period.

(3) In the *San Onofre* case (facility at San Clemente, Cal.), the Appeal Board elaborated on the criteria for determining whether an exclusion area complies with applicable regulations.

(4) In the *Prairie Island* proceeding (facility at Red Wing, Minn.), the Appeal Board began exploration in depth of issues related to the reliability of the reactors' steam generator tubes.

Appeal Boards have authority to conduct evidentiary hearings but had no occasion to do so in 1975. An appeal Board plans to hold such a hearing in early 1976 in connection with its examination of the steam generator tube issues in *Prairie Island.* In addition, the Commission in August 1975 designated an Appeal Board to conduct hearings on certain seismic issues which had been raised in connection with the Indian Point facility. These hearings should take place in April 1976.

COMMISSION REVIEW

The Commission has been actively engaged in its quasi-adjudicatory responsibilities since its inception. On January 27, 1975, it issued its first Memorandum and Order, affirming an Appeal Board decision in the *Prairie Island* proceeding which allowed cross-examination by an intervenor on any issue in controversy which, though not initially raised by him, is one in which he has a discernible interest. The Commission said that its opinion "underscored the fundamental importance of meaningful public participation in its adjudicatory process."

On April 30, 1975, the Commission announced its decision in the "As Low As

Reasonably Achievable" rulemaking proceeding. The proceeding, begun in 1971 by the Atomic Energy Commission, culminated in the regulation described in Chapter 3 under the heading, "New Rule on Effluent Control" (see page 43).

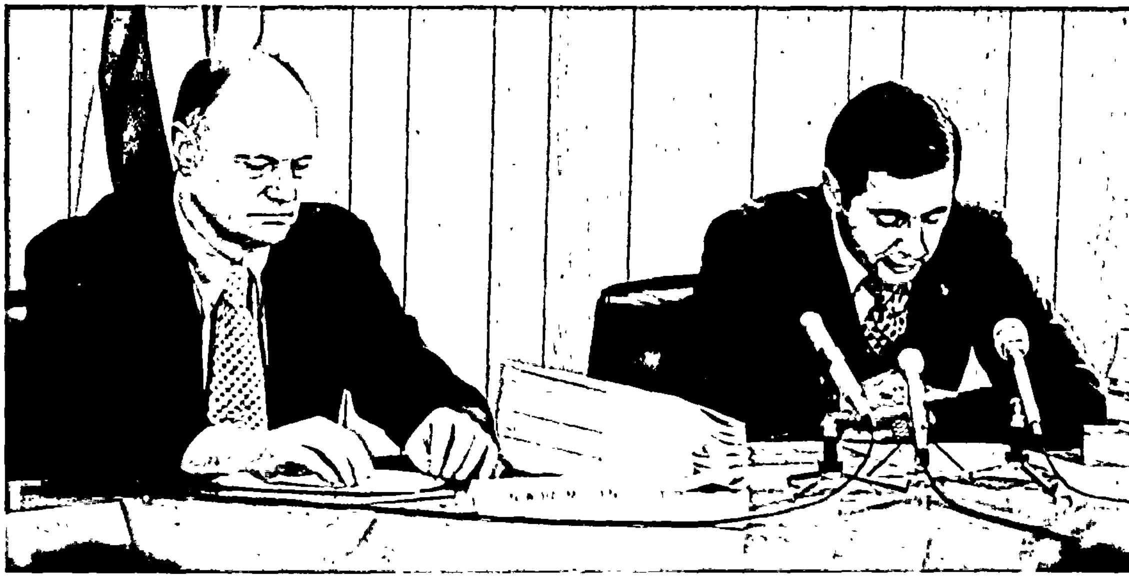
In a decision construing its rule concerning non-timely filings of petitions to intervene, the Commission held that Erie County, New York should be allowed to participate as an intervenor in the ongoing licensing proceeding on the Nuclear Fuel Services' *West Valley* Reprocessing Plant. Though it found that the County had not shown good cause for its untimely filing of a petition to intervene, other factors—particularly the possibility that the County's interests may be unique and might be unrepresented without intervention—persuaded the Commission to order that the petition be accepted.

In August the Commission issued several decisions and also announced a rulemaking. One of these decisions dealt with a unique problem arising in the *Catawba* (South Carolina) construction permit proceedings. In adopting emergency core cooling system (ECCS) regulations in 1973, the Atomic Energy Commission had provided a one-year transition period, until December 28, 1974, during which construction permits could be granted upon a showing of compliance with the previously effective Interim Acceptance Criteria. Compliance with the new ECCS regulation would, in all cases, be required at the operating license stage. The *Catawba* application, filed in June 1972, failed to result in issuance of construction permits before the cut-off date because of the reopening of the once-closed record on issues unrelated to ECCS. After hearing oral arguments on factual and procedural considerations unique to this case, the Commission determined that application of the December 28, 1974, cut-off date would not serve the purposes for which that cut-off date was originally established. Therefore, the Commission granted a waiver of application of that date, thus authorizing issuance of construction permits for the two units. (See Chapter 2.)

On August 4, 1975, the Commission granted a request of the Citizens' Committee for the Protection of the Environment for review of a decision by the Acting Director of Licensing (as delegate of the Director of Regulation) not to institute a show cause proceeding under 10 CFR 2.202 and 2.206, affecting Consolidated Edison Company of New York's *Indian Point* site. Noting that its current regulations do not specifically provide for Commission review of the Director's refusal to institute a show cause proceeding, the Commission announced interim criteria by which it may review this kind of decision, pending rulemaking to develop more precise standards for the exercise and review of prosecutorial or enforcement functions.

On August 11, 1975, the Commission granted Consumers Power Company's petition for a declaratory order regarding the current validity of a previously issued but unused license amendment authorizing up to a full-core load of mixed oxide fuel at its *Big Rock Point* (Michigan) Nuclear Plant. The Commission, noting a series of intervening developments regarding plutonium recycle, determined as a matter of discretion that a review required by the National Environmental Policy Act is a prerequisite to the use of the amendment in this case. The decision distinguished this particular matter from the broader issues involved in generic environmental inquiry into plutonium recycle and noted that its decision in this specific matter would not foreclose any of its options in the ultimate resolution of the generic inquiry.

At the end of August, the Commission initiated a rulemaking looking to resolution of a major issue inherited from the AEC: requests by intervenors in Commission



Press conferences with NRC Commissioners and concerned staff are held when there is substantial public interest in a specific NRC activity. One such occasion was the press conference held on October 30, 1975, concerning the completion of the final report on the Reactor Safety Study. Professor Norman C. Rasmussen (seated left) and Chairman William A. Anders lead the conference.

proceedings for financial assistance. This topic is discussed earlier in this chapter under "Formal Participation" (page 162).

In a Memorandum and Order issued on December 2, 1975, the Commission approved a stipulation settling differences among parties contesting the installation of a permanent cooling system for Unit 3 of the Indian Point Nuclear Generating Station. In essence, the stipulation calls for installation of closed-cycle cooling for Unit 3 after five years of operation, subject to possible reopening of the question pursuant to the stipulation and the Commission's rules of practice. The Commission noted its rule encouraging fair and reasonable settlement of contested initial licensing proceedings, pointing out, however, that NEPA requirements must be—and, in this case, had been—met.

JUDICIAL REVIEW

Federal court actions involving the Nuclear Regulatory Commission during 1975 include nine cases decided, four cases argued and awaiting decision, and two cases pending.

Cases Decided During 1975

Carolina Environmental Study Group v. AEC, et al., 510 F.2d 796 (D.C. Cir. 1975). On January 21, 1975, the Court of Appeals for the District of Columbia Circuit affirmed a final order of the AEC (now NRC) granting Duke Power Co. a construction license to build two nuclear reactors. Petitioners had contended that the Commission failed to comply with the National Environmental Policy Act because it inadequately considered Class 9 accidents and alternative sources of power in the impact statement. They had also claimed a denial of due process because of the AEC's dual role in promotion and regulation of nuclear power. The Court approved the Commission's treatment of Class 9 accidents as hypothetical occurrences whose probability was extremely remote.

The Conservation Society of Vermont, et al. v. AEC. On April 17, 1975, a three-judge district court dismissed a complaint which challenged the constitutionality of the Atomic Energy Act on the ground that it vested both developmental and regulatory functions in the AEC. The action was dismissed because the Energy Reorganization Act of 1974 had rendered the issues moot.

Ralph Nader, et al. v. NRC, et al. On May 30, 1975, the Court of Appeals for the District of Columbia Circuit affirmed the Commission's denial of a petition seeking a shutdown or derating of 20 nuclear power plants previously licensed for operation. The Court agreed with the Commission that a determination of compliance with the Acceptance Criteria is equivalent to a determination of compliance with Criterion 35. It pointed out that the former were detailed specifications implementing the more general Criterion 35.

Porter County Chapter of the Izaak Walton League of America, Inc., et al. v. AEC, et al. On April 21, 1975, the United States Court of Appeals for the Seventh Circuit set aside the NRC construction permit for the nuclear power reactor proposed for the Bailly site on Lake Michigan in Northern Indiana. The Court denied the Government's petition for rehearing or rehearing *in banc.* In so ordering, the original three-judge panel observed "We find nothing in the Atomic Energy Act of 1954 or the Energy Reorganization Act of 1974 which would prevent NIPSCO from applying for another license or would prevent NRC from considering such new application."

On June 16, 1975, NRC requested that the Solicitor General authorize the filing of a petition for a writ of certiorari in the Supreme Court. The Solicitor General filed a memorandum supporting the suggestion of Northern Indiana Public Service Co. that the case be summarily reversed. On November 11, 1975, the Supreme Court reversed the judgment of the Seventh Circuit and remanded the case for consideration of questions which that court had not decided. The Supreme Court held that the Seventh Circuit had erroneously rejected the NRC's interpretation of its own power plant siting regulations.

The State of New York v. NRC, et al. (S.D.N.Y., 75 Civ. 2121). The State of New York claimed that several Federal agencies, including NRC, had violated the National Environmental Policy Act by transporting and allowing the transport of special nuclear material, particularly plutonium, by air without having first prepared an Environmental Impact Statement. The agencies agreed to prepare an impact statement and re-examine pertinent regulations. The State of New York sought a preliminary injunction suspending air transport of the material pending completion of the impact statement and associated rulemaking procedures. The petition was

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denied, the Court noting that air shipment was inherently a better protection from theft than surface transport and that, with the safety record of such shipments in the past, the State's contention that irremediable harm could ensue without immediate suspension was not convincing.

On August 9, 1975, Congress enacted Public Law 94-79, prohibiting the licensing by NRC of any air shipments of nuclear material except that contained in certain medical devices designed for individual human use. The restriction will remain in force until NRC has certified to the Joint Committee on Atomic Energy that a safe container has been developed. (See Chapter 4 of this report for details.)

Environmental Coalition on Nuclear Power, et al. v. NRC, et al. (3rd Cir., 75-1421). Petitioners sought review of an Order issuing construction permits for Limerick Generating Station, Units 1 and 2, of the Philadelphia Electric Co. They contended that the Commission's review concerning cooling water for the facility was not in accordance with the National Environmental Policy Act. In its brief, the Commission contended that its environmental review was complete and that Limerick was economically and environmentally sound when operated on the basis of natural river flow. As the Commission further stated, alternative methods of operation which would increase the plant's economic efficiency will not be authorized until and unless they too can be shown to comply with NEPA. On November 12, 1975, two days after it heard oral argument, the court issued an order denying the petition of review.

York Committee for a Safe Environment, et al. v. NRC (D.C. Cir., 74-1923). On December 9, 1975, the United States Court of Appeals for the District of Columbia, rejecting numerous other objections raised by the petitioners to the grant of an operating license for the Peach Bottom Atomic Power Station, Unit 2, remanded the case to the Commission for an individualized analysis of the costs and benefits of reducing routine radioiodine releases in accordance with the Commission's "as low as reasonably achievable" regulations. The Court rejected the petitioners' contentions that the licensee's emergency plans did not comply with the Commission's regulations and that the Commission had violated the National Environmental Policy Act by giving inadequate consideration to the transportation of spent fuel and to the cumulative effects of the operation of existing and planned reactors in the vicinity of Peach Bottom.

Citizens for Safe Power, Inc., et al. v. NRC, et al. (D.C. Cir., 74-1186). Contending that the Atomic Energy Act requires, beyond compliance with safety regulations implementing the Act, a consideration of "residual" risks, petitioners sought review of an order granting an operating license for the Maine Yankee facility. The Commission responded that the Act does not require such additional consideration because the regulations themselves embrace a consideration of such risks. Moreover, such residual risks were expressly considered in review under the National Environmental Policy Act. On December 24, 1975, the Court issued its opinion holding that the Commission's action complied with the Atomic Energy Act and was an adequate assessment, in the context of NEPA review, of the residual risk.

Hudson River Fisherman's Assoc., Inc., et al. v. NRC (2d Cir. No. 75-4212). Petitioners challenged the Appeal Board's Order interpreting—and approving as interpreted—a stipulation which they executed jointly with the other parties to NRC's adjudicatory hearing concerning issuance of an operating license for Unit 3 of the Indian Point Nuclear Generating Station. Following the Commission's December 2, 1975 Order (see page 168) approving the stipulation and vacating portions of the Appeal Board's decision, the court dismissed the suit on December 10, 1975.

Cases Argued and Awaiting Decision

North Anna Environmental Coalition v. NRC, et al. (D.C. Cir., 75-1312). Petitioners seek review of a Commission determination allowing Virginia Electric and Power Co. to continue construction of four reactors at the North Anna site. After a hearing, the Commission determined that a geological fault on which the reactors were sited was not a "capable fault" and therefore posed no danger to the public health and safety. The court heard oral argument on November 20, 1975.

Lloyd Harbor Study Group v. NRC, et al. (D.C. Cir., 73-2266). Petitioners seek review of an order granting a construction permit for the Shoreham nuclear facility. At issue in the case is whether nuclear fuel cycle effects attributable to this particular plant—which are identical to those from all other reactors and which were the subject of an ongoing generic rulemaking proceeding—had to be considered in the environmental impact statement. Another issue, identical to the question decided in *Carolina Environmental Study Group,* is whether the National Environmental Policy Act requires a detailed analysis of the consequences of Class 9 accidents. The case has been briefed and argued and awaits action by the Court.

Natural Resources Defense Council, Inc., et al. v. NRC, et al. (D.C. Cir. Nos. 74-1385 and 74-1586). These two cases were consolidated for oral argument. In number 74-1586, the petitioners challenged the Commission's issuance of a rule prescribing the manner of accounting, in individual licensing cases, for the environmental consequences of the uranium fuel cycle activities attributable to the particular plant. In number 74-1385, petitioners challenged the Commission's issuance of the Vermont Yankee operating license. Relying on NEPA, they attack the Commission's treatment of environmental effects of fuel cycle activities attributable to Vermont Yankee operation.

Aeschliman, et al. v. NRC, et al. (D.C. Cir. 73-1776) and *Saginaw Valley Nuclear Study Group, et al. v. NRC, et al.* (D.C. Cir. 73-1867). Petitioners in these companion cases seek review of Commission orders granting construction permits for Units 1 and 2 of the Midland facility in Michigan. At issue is the Commission's decision to evaluate emergency core cooling systems and impacts of the nuclear fuel cycle by generic rulemaking rather than in separate adjudications. By order of the Court the cases were held in abeyance pending determination of the related fuel cycle issues in the above-mentioned cases brought by the Natural Resources Defense Council.

Other Pending Cases

City of Cleveland, Ohio v. NRC (D.C. Cir. No. 75-2115). The City of Cleveland filed, on November 17, 1975, a petition to review a decision of the Atomic Safety and Licensing Appeal Board in a consolidated proceeding involving the Davis-Besse Nuclear Power Station (Unit 1) and the Perry Nuclear Power Plant (Units 1 and 2). The Appeal Board held that appointment of a Special Master, with the parties' agreement, to decide certain discovery matters did not violate an NRC prohibition against redelegation of the Licensing Board's authority. The Appeal Board also pointed out that the parties had agreed to be bound by the Special Master's discovery rulings.

Carolina Environmental Study Group v. NRC, et al. (U.S.D.C.W.D.N.C., No. C-C-73-139). This suit challenges the granting of a construction permit to Duke Power Company for the McGuire facility in North Carolina. The issues it had in common with the suit of the same name in the Court of Appeals for the D.C. Circuit have been dismissed. The remaining issue is the constitutionality of the limitation of liability in the Price-Anderson Act. Memoranda on that issue have been filed; argument was heard on December 18, 1975. The Court ordered that a further hearing be held.

Though the NRC is not a party to it, special mention is merited of the Supreme Court's grant of certiorari in the case of *Train v. Colorado Public Interest Research Group*, by which the Court has accepted the Government's petition for review of an appellate court decision. The issue before the Court is whether nuclear waste material, regulated and controlled through licenses issued by the NRC, is a "pollutant" within the meaning of the Federal Water Pollution Control Act, which required the Environmental Protection Agency to establish regulations and controls over pollutants by administering a detailed permit program. The U.S. Court of Appeals for the Tenth Circuit, reversing the district court decision, answered this question in the affirmative. The EPA has appealed this decision to the Supreme Court, which heard oral argument from the parties on December 9, 1975. (See also Chapter 3.)

Members of the Commission briefed members of Congress on NRC's licensing and inspection program during visits to nuclear power plants in August. Shown above during a visit to the Rancho Seco nuclear plant are Commissioner Rowden, Chairman Anders, Congressman John McFall, Congressman John Moss, and E.K. Davis of the Sacramento Municipal Utility District. Members of the staffs of Congressmen Robert Leggett and Paul McCloskey also participated in the tour. Commissioner Rowden and Congressman Clair Bergener toured the San Onofre Plant along with members of the staffs of Congressmen Del Clawson and Jerry Patterson. Commissioner Kennedy was joined at the Fort Calhoun plant by Congressmen John McCollister, Berkley Bedell, and Thomas Harkin, and a member of Senator Richard Clark's staff.

Appendix 1

Nuclear Electric Generating Units In Operation, Under Construction, Or Planned

(As of June 30, 1975)

The following listing includes 243 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 June 1975, representing a total capacity of approximately 243,000 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.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
ALABAMA						
Clanton	Alan R. Barton Nuclear Plant: Unit 1	1,159	BWR	UR	Alabama Power Co.	1983
Clanton	Alan R. Barton Nuclear Plant: Unit 2	1,159	BWR	UR	Alabama Power Co.	1984
Clanton	Alan R. Barton Nuclear Plant: Unit 3	1,159	BWR	UR	Alabama Power Co.	1985
Clanton	Alan R. Barton Nuclear Plant: Unit 4	1,159	BWR	UR	Alabama Power Co.	1986
Decatur	Browns Ferry Nuclear Power Plant: Unit 1	1,067	BWR	OL	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant: Unit 2	1,067	BWR	OL	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant: Unit 3	1,067	BWR	CP	Tennessee Valley Authority	1976
Dothan	Joseph M. Farley Nuclear Plant: Unit 1	829	BWR	CP	Alabama Power Co.	1976
Dothan	Joseph M. Farley Nuclear Plant: Unit 2	829	PWR	CP	Alabama Power Co.	1977

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
ALABAMA (Continued)						
Scottsboro	Bellefonte Nuclear Plant: Unit 1	1,213	PWR	CP	Tennessee Valley Authority	1980
Scottsboro	Bellefonte Nuclear Plant: Unit 2	1,213	PWR	CP	Tennessee Valley Authority	1981
ARKANSAS						
Russelville	Arkansas Nuclear One: Unit 1	850	PWR	OL	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One: Unit 2	912	PWR	CP	Arkansas Power & Light Co.	1977
ARIZONA						
Winterburg	Palo Verde Nuclear Generating Station: Unit 1	1,238	PWR	UR	Arizona Public Service Co.	1982
Winterburg	Palo Verde Nuclear Generating Station: Unit 2	1,238	PWR	UR	Arizona Public Service Co.	1984
Winterburg	Palo Verde Nuclear Generating Station: Unit 3	1,238	PWR	UR	Arizona Public Service Co.	1986
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	430	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.	1981
San Clemente	San Onofre Nuclear Generating Station: Unit 3	1,140	PWR	CP	So. Calif. Ed. & San Diego Gas & Electric Co.	1982
Diablo Canyon	Diablo Canyon Nuclear Power Plant: Unit 1	1,084	PWR	CP	Pacific Gas & Elec. Co.	1976
Diablo Canyon	Diablo Canyon Nuclear Power Plant: Unit 2	1,106	PWR	CP	Pacific Gas & Elec. Co.	1977
Clay Station	Rancho Seco Nuclear Generating Station: Unit 1	913	PWR	OL	Sacramento Municipal Utility District	1975
*	Central Valley: Unit 1	1,128	BWR	A/O	Pacific Gas & Elec. Co.	1981
*	Central Valley: Unit 2	1,128	BWR	A/O	Pacific Gas & Elec. Co.	1982
*	San Joaquin Nuclear Project 1	1,300	—	A/O	LADWP, PG&E, SCE, SDE&G, CDWR	1982

* Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
CALIFORNIA (Continued)						
*	San Joaquin Nuclear Project 2	1,300	—	A/O	LADWP, PG&E, SCE, SDE&G, CDWR	1983
*	San Joaquin Nuclear Project 3	1,300	—	A/O	LADWP, PG&E, SCE, SDE&G, CDWR	1985
*	San Joaquin Nuclear Project 4	1,300	—	A/O	LADWP, PG&E, SCE, SDE&G, CDWR	1987
*	Vidal	1,500	—	A/O	Southern Calif. Ed.	Indef
Clay Station	Rancho Seco Nuclear Generating Station: Unit 2	1,100	—	A/O	Sacramento Municipal Utility District	1984
*	Sundesert 1	1,160	—	A/O	San Diego Gas & Elec. Co.	1985
*	Sundesert 2	1,160	—	A/O	San Diego Gas & Elec. Co.	1988
COLORADO						
Platteville	Fort St. Vrain Nuclea Generating Station	330	HTGR	OL	Public Service Co. of Colorado	1975
CONNECTICUT						
Haddam Neck	Haddam Neck Generating Station	575	PWR	OL	Conn. Yankee Atomic Power Co.	1968
Waterford	Millstone Nuclear Power Station: Unit 1	652	BWR	OL	Northeast Nuclear Energy Co.	1971
Waterford	Millstone Nuclear Power Station: Unit 2	828	PWR	CP	Northeast Nuclear Energy Co.	1975
Waterford	Millstone Nuclear Power Station: Unit 3	1,156	PWR	CP	Northeast Nuclear Energy Co.	1979
DELAWARE						
Summit	Summit Power Station: Unit 1	770	HTGR	UR	Delmarva Power & Light Co.	1981
Summit	Summit Power Station: Unit 2	770	HTGR	UR	Delmarva Power & Light Co.	1984
FLORIDA						
Florida City	Turkey Point Station: Unit 3	666	PWR	OL	Florida Power & Light Co.	1972
Florida City	Turkey Point Station Unit 4	666	PWR	OL	Florida Power & Light Co.	1973
Red Level	Crystal River Plant: Unit 3	825	PWR	CP	Florida Power Corp.	1976
Ft. Pierce	St. Lucie Plant: Unit 1	810	PWR	CP	Florida Power & Light Co.	1975
Ft. Pierce	St. Lucie Plant: Unit 2	810	PWR	UR**	Florida Power & Light Co.	1980

*Site not selected.

**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe	Type	Status	Utility	Scheduled Operation
FLORIDA (Continued)						
*	Unnamed: Unit 1	1,300	PWR	A/O	Florida Power Corp.	Indef
*	Unnamed: Unit 2	1,300	PWR	A/O	Florida Power Corp.	Indef
South Dade	South Dade 1	1,200	PWR	A/O	Florida Power & Light Co.	1980
South Dade	South Dade 2	1,200	PWR	A/O	Florida Power & Light Co.	1980
GEORGIA						
Baxley	Edwin I. Hatch Plant: Unit 1	786	BWR	OL	Georgia Power Co.	1975
Baxley	Edwin I. Hatch Plant: Unit 2	795	BWR	CP	Georgia Power Co.	1978
Waynesboro	Alvin W. Vogtle, Jr. Plant: Unit 1	1,113	PWR	CP	Georgia Power Co.	199X
Waynesboro	Alvin W. Vogtle, Jr. Plant: Unit 2	1,113	PWR	CP	Georgia Power Co.	199X
ILLINOIS						
Morris	Dresden Nuclear Power Station: Unit 1	200	BWR	OL	Commonwealth Edison Co.	1960
Morris	Dresden Nuclear Power Station: Unit 2	809	BWR	OL	Commonwealth Edison Co.	1970
Morris	Dresden Nuclear Power Station: Unit 3	809	BWR	OL	Commonwealth Edison Co.	1971
Zion	Zion Nuclear Plant Unit 1	1,050	PWR	OL	Commonwealth Edison Co.	1973
Zion	Zion Nuclear Plant Unit 2	1,050	PWR	OL	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Station: Unit 1	800	BWR	OL	Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co.	1972
Cordova	Quad-Cities Station: Unit 2	800	BWR	OL	Comm. Ed. Co.-Iowa-Ill. Gas & Elec. Co.	1972
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	UR**	Commonwealth Edison Co.	1980
Byron	Byron Station: Unit 2	1,120	PWR	UR**	Commonwealth Edison Co.	1981
Braidwood	Braidwood: Unit 1	1,120	PWR	UR**	Commonwealth Edison Co.	1980
Braidwood	Braidwood: Unit 2	1,120	PWR	UR**	Commonwealth Edison Co.	1981
Clinton	Clinton Nuclear Power Plant: Unit 1	933	BWR	UR	Illinois Power Co.	1981

**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
ILLINOIS (Continued)						
Clinton	Clinton Nuclear Power Plant: Unit 2	933	BWR	UR	Illinois Power Co.	1984
Savannah	Carroll County Station: Unit 1	1,120	—	A/O	Commonwealth Edison Co.	1984
Savannah	Carroll County Station: Unit 2	1,120	—	A/O	Commonwealth Edison Co.	1985
INDIANA						
Westchester Town	Bailly Generating Station	645	BWR	CP	Northern Indiana Public Service Co.	1980
Madison	Marble Hill: Unit 1	1,112	PWR	A/O	Public Service of Indiana	1982
Madison	Marble Hill: Unit 2	1,112	PWR	A/O	Public Service of Indiana	1984
IOWA						
Pala	Duane Arnold Energy Center: Unit 1	535	BWR	OL	Iowa Elec. Light & Power Co.	1974
*	Iowa Power Unit 1	1,000	BWR	A/O	Iowa Po. & Lt. Co.	1983
KANSAS						
Burlington	Wolf Creek	1,150	PWR	UR	Kansas Gas & Elec. Co.	1982
LOUISIANA						
Taft	Waterford Steam Electric Station: Unit 3	1,113	PWR	CP	Louisiana Power & Light Co.	1980
St. Francisville	River Bend Station: Unit 1	934	BWR	UR	Gulf States Utilities Co.	1981
St. Francisville	River Bend Station: Unit 2	934	BWR	UR	Gulf States Utilities Co.	1983
MAINE						
Wiscasset	Maine Yankee Atomic Power Plant	790	PWR	OL	Maine Yankee Atomic Power Co.	1972
Sears Island	Central Maine	1,200	PWR	A/O	Central Maine	1983
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	CP	Baltimore Gas & Elec. Co.	1977
Douglas Point	Douglas Point Generating Station: Unit 1	1,178	BWR	UR	Potomac Electric Power Co.	1985
Douglas Point	Douglas Point Generating Station: Unit 2	1,178	BWR	UR	Potomac Electric Power Co.	1987

*Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
MASSACHUSETTS						
Rowe	Yankee Nuclear Power Station	175	PWR	OL	Yankee Atomic Elec. Co.	1961
Plymouth	Pilgrim Station: Unit 1	670	BWR	OL	Boston Edison Co.	1972
Plymouth	Pilgrim Station: Unit 2	1,180	PWR	UR	Boston Edison Co.	1982
Turners Falls	Montague Unit 1	1,150	BWR	UR	Northeast Nuclear Energy Co.	1986
Turners Falls	Montague Unit 2	1,150	BWR	UR	Northeast Nuclear Energy Co.	1988
MICHIGAN						
Big Rock Point	Big Rock Point Nuclear Plant	75	BWR	OL	Consumers Power Co.	1965
South Haven	Palisades Nuclear Power Station	700	PWR	OL	Consumers Power Co.	1971
Lagoona Beach	Enrico Fermi Atomic Power Plant: Unit 2	1,093	BWR	CP	Detroit Edison Co.	1980
Bridgman	Donald C. Cook Plant: Unit 1	1,060	PWR	OL	Indiana & Michigan Elec. Co.	1986
Bridgman	Donald C. Cook Plant: Unit 2	1,060	PWR	CP	Indiana & Michigan Elec. Co.	—
Midland	Midland Nuclear Power Plant: Unit1	458	PWR	CP	Consumers Power Co.	1982
Midland	Midland Nuclear Power Plant: Unit 2	808	PWR	CP	Consumers Power Co.	1981
St. Clair County	Greenwood Energy Center: Unit 2	1,200	PWR	UR	Detroit Edison Co.	1984
St. Clair County	Greenwood Energy Center: Unit 3	1,200	PWR	UR	Detroit Edison Co.	1986
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
Red Wing	Prairie Island Nuclear Generating Plant: Unit 2	530	PWR	OL	Northern States Power Co.	1975
MISSOURI						
Fulton	Callaway Plant: Unit 1	1,120	PWR	UR	Union Elec. Co.	1981
Fulton	Callaway Plant: Unit 2	1,120	PWR	UR	Union Elec. Co.	1983
MISSISSIPPI						
Port Gibson	Grand Gulf Nuclear Station: Unit 1	1,250	BWR	CP	Mississippi Power & Light Co.	1979

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
MISSISSIPPI (Continued)						
Port Gibson	Grand Gulf Nuclear Station: Unit 2	1,250	BWR	CP	Mississippi Power & Light Co.	1984
NEBRASKA						
Fort Calhoun	Fort Calhoun Station: Unit 1	457	PWR	OL	Omaha Public Power District	1973
Fort Calhoun	Fort Calhoun Station: Unit 2	1,136	PWR	A/O	Omaha Public Power District	1983
Brownville	Cooper Nuclear Station	778	BWR	OL	Nebraska Public Power District	1974
*	NPPD-2	1,100	—	A/O	Nebraska Public Power District	1986
NEW HAMPSHIRE						
Seabrook	Seabrook Nuclear Station: Unit 1	1,200	PWR	UR	Public Service of N.H.	1980
Seabrook	Seabrook Nuclear Station: Unit 2	1,200	PWR	UR	Public Service of N.H.	1981
NEW JERSEY						
Toms River	Oyster Creek Nuclear Power Plant: Unit 1	640	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.	1982
Salem	Salem Nuclear Generating Station: Unit 1	1,090	PWR	CP	Public Service Elec. & Gas Co.	1976
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.	1982
Salem	Hope Creek Generating Station: Unit 2	1,067	BWR	CP	Public Service Elec. & Gas Co.	1984
Little Egg Inlet	Atlantic Generating Station: Unit 1	1,150	PWR	UR	Public Service Elec. & Gas Co.	1985
Little Egg Inlet	Atlantic Generating Station: Unit 2	1,150	PWR	UR	Public Service Elec. & Gas Co.	1987
*	Atlantic Generating Station: Unit 3	1,150	PWR	A/O	Public Service Elec. & Gas Co.	1990
*	Atlantic Generating Station: Unit 4	1,150	PWR	A/O	Public Service Elec. & Gas Co.	1992
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	CP	Consolidated Edison Co.	1975

*Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
NEW YORK (Continued)						
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.	199?
Ontario	R. E. Ginna Nuclear Power Plant: Unit 1	490	PWR	OL	Rochester Gas & Elec. Co.	1970
Brookhaven	Shoreham Nuclear Power Station	819	BWR	CP	Long Island Lighting Co.	1978
Scriba	James A. FitzPatrick Nuclear Power Plant	821	BWR	OL	Power Authority of State of N.Y.	1975
Long Island	Jamesport: Unit 1	1,150	PWR	UR	Long Island Lighting Co.	1982
Long Island	Jamesport: Unit 2	1,150	PWR	UR	Long Island Lighting Co.	1984
Somerset	Unnamed: Unit 1	1,220	BWR	A/O	N.Y. State Elec. & Gas Co.	1984
Somerset	Unnamed: Unit 2	1,220	BWR	A/O	N.Y. State Elec. & Gas Co.	1986
Sterling	Sterling Power Project: Unit 1	1,150	PWR	UR	Rochester Gas & Elec. Co.	1982
*	Greene County Nuclear Power Plant	1,191	PWR	A/O	Power Authority of State of N.Y.	1983
*	Mid-Hudson East 1	1,300	—	A/O	Empire State Power Resources	1987
*	Mid-Hudson East 2	1,300	—	A/O	Empire State Power Resources	1989
*	Mid-Hudson West 1	1,300	—	A/O	Empire State Power Resources	1990
*	Shoreham West 1	1,300	—	A/O	Empire State Power Resources	1987
*	Shoreham West 2	1,300	—	A/O	Empire State Power Resources	1989
*	St. Lawrence 1	1,300	—	A/O	Empire State Power Resources	1988
*	St. Lawrence 2	1,300	—	A/O	Empire State Power Resources	1990
NORTH CAROLINA						
Southport	Brunswick Steam Electric Plant: Unit 2	821	BWR	OL	Carolina Power & Light Co.	1975
Southport	Brunswick Steam Electric Plant: Unit 1	821	BWR	CP	Carolina Power & Light Co.	1977
Cowans Ford Dam	Wm. B. McGuire Nuclear Station: Unit 1	1,180	PWR	CP	Duke Power Co.	1978
Cowans Ford Dam	Wm. B. McGuire Nuclear Station: Unit 2	1,180	PWR	CP	Duke Power Co.	1979

*Site not selected.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
NORTH CAROLINA (Continued)						
Bonsal	Shearon Harris Plant: Unit 1	900	PWR	UR†	Carolina Power & Light Co.	1984
Bonsal	Shearon Harris Plant: Unit 2	900	PWR	UR†	Carolina Power & Light Co.	1986
Bonsal	Shearon Harris Plant: Unit 3	900	PWR	UR†	Carolina Power & Light Co.	1988
Bonsal	Shearon Harris Plant: Unit 4	900	PWR	UR†	Carolina Power & Light Co.	1990
Davie Co.	Perkins Nuclear Station: Unit 1	1,280	PWR	UR	Duke Power Co.	1983
Davie Co.	Perkins Nuclear Station: Unit 2	1,280	PWR	UR	Duke Power Co.	1985
Davie Co.	Perkins Nuclear Station: Unit 3	1,280	PWR	UR	Duke Power Co.	1987
*	Carolina P&L: Unit 8	1,150	PWR	A/O	Carolina Power & Light Co.	199?
*	Carolina P&L: Unit 9	1,150	PWR	A/O	Carolina Power & Light Co.	—
*	Carolina P&L: Unit 10	1,150	PWR	A/O	Carolina Power & Light Co.	—
OHIO						
Oak Harbor	Davis-Besse Nuclear Power Station: Unit 1	906	PWR	CP	Toledo Edison-Cleveland Elec. Illum. Co.	1976
Oak Harbor	Davis-Besse Nuclear Power Station: Unit 2	906	PWR	UR	Toledo Edison-Cleveland Elec. Illum. Co.	1983
Oak Harbor	Davis-Besse Nuclear Power Station: Unit 3	906	PWR	UR	Toledo Edison-Cleveland Elec. Illum. Co.	1985
Perry	Perry Nuclear Power Plant: Unit 1	1,205	BWR	UR**	Cleveland Elec. Illum. Co.	1986
Perry	Perry Nuclear Power Plant: Unit 2	1,205	BWR	UR**	Cleveland Elec. Illum. Co.	1982
Moscow	Wm. H. Zimmer Nuclear Power Station: Unit 1	810	BWR	CP	Cincinnati Gas & Elec. Co.	1979
Moscow	Wm. H. Zimmer Nuclear Power Station: Unit 2	1,179	BWR	A/O	Cincinnati Gas & Elec. Co.	1984
Berlin Hgts.	Erie: Unit 1	1,200	—	A/O	Ohio Edison Co.	1983
Berlin Hgts.	Erie: Unit 2	1,200	—	A/O	Ohio Edison Co.	1984
OKLAHOMA						
Inola	Black Fox: Unit 1	1,150	BWR	A/O	Public Service Co. of Oklahoma	1983
Inola	Black Fox: Unit 2	1,150	BWR	A/O	Public Service Co. of Oklahoma	1985

†Exemption granted to allow some early work at site.
*Site not selected.
**Limited work authorization issued

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
OREGON						
Prescott	Trojan Nuclear Plant: Unit 1	1,130	PWR	CP	Portland General Elec. Co.	1976
Arlington	Pebble Springs: Unit 1	1,260	PWR	UR	Portland General Elec. Co.	1983
Arlington	Pebble Springs: Unit 2	1,260	PWR	UR	Portland General Elec. Co.	1986
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.	1981
Pottstown	Limerick Generating Station: Unit 2	1,065	BWR	CP	Philadelphia Elec. Co.	1982
Shippingport	Shippingport Atomic Power Station: Unit 1	90	PWR	—[1]	Duquesne Light Co. &ERDA	NA
Shippingport	Beaver Valley Power Station: Unit 1	852	PWR	CP	Duquesne Light Co. & Ohio Edison Co.	1975
Shippingport	Beaver Valley Power Station: Unit 2	852	PWR	CP	Duquesne Light Co. & Ohio Edison Co.	1981
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	CP	Metropolitan Edison Co.	1978
Berwick	Susquehanna Steam Electric Station: Unit 1	1,050	BWR	CP	Pennsylvania Power & Light Co.	1980
Berwick	Susquehanna Steam Electric Station: Unit 2	1,050	BWR	CP	Pennsylvania Power & Light Co.	1982
Fulton	Fulton Generating Station: Unit 1	1,160	HTGR	UR	Philadelphia Elec. Co.	1984
Fulton	Fulton Generating Station: Unit 2	1,160	HTGR	UR	Philadelphia Elec. Co.	1986
RHODE ISLAND						
No. Kingston	New England: Unit 1	1,200	PWR	A/O	New England Power Co.	1982
No. Kingston	New England: Unit 2	1,200	PWR	A/O	New England Power Co.	1983
SOUTH CAROLINA						
Hartsville	H.B. Robinson S.E. Plant: Unit 2	700	PWR	OL	Carolina Power & Light Co.	1971

[1]Operable but OL not required.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
SOUTH CAROLINA (Continued)						
Seneca	Oconee Nuclear Station: Unit 1	871	PWR	OL	Duke Power Co.	1973
Seneca	Oconee Nuclear Station: Unit 2	871	PWR	OL	Duke Power Co.	1974
Seneca	Oconee Nuclear Station: Unit 3	871	PWR	OL	Duke Power Co.	1974
Broad River	Virgil C. Summer Nuclear Station: Unit 1	900	PWR	CP	So. Carolina Elec. & Gas Co.	1979
Broad River	Virgil C. Summer Nuclear Station: Unit 2	900	—	A/O	So. Carolina Elec. & Gas Co.	1990
Lake Wylie	Catawba Nuclear Station: Unit 1	1,153	PWR	UR**	Duke Power Co.	1981
Lake Wylie	Catawba Nuclear Station: Unit 2	1,153	PWR	UR**	Duke Power Co.	1982
Cherokee County	Cherokee Nuclear Station: Unit 1	1,280	PWR	UR	Duke Power Co.	1984
Cherokee County	Cherokee Nuclear Station: Unit 2	1,280	PWR	UR	Duke Power Co.	1986
Cherokee County	Cherokee Nuclear Station: Unit 3	1,280	PWR	UR	Duke Power Co.	1988
TENNESSEE						
Daisy	Sequoyah Nuclear Power Plant: Unit 1	1,148	PWR	CP	Tennessee Valley Authority	1977
Daisy	Sequoyah Nuclear Power Plant: Unit 2	1,148	PWR	CP	Tennessee Valley Authority	1977
Spring City	Watts Bar Nuclear Plant: Unit 1	1,177	PWR	CP	Tennessee Valley Authority	1978
Spring City	Watts Bar Nuclear Plant: Unit 2	1,177	PWR	CP	Tennessee Valley Authority	1979
Oak Ridge	Clinch River Breeder Reactor Plant	350	LMFBR	UR	U.S. Government	1982
Hartsville	TVA Plant 1: Unit 1	1,233	BWR	UR	Tenn. Valley Authority	1980
Hartsville	TVA Plant 1: Unit 2	1,233	BWR	UR	Tenn. Valley Authority	1981
Hartsville	TVA Plant 2: Unit 1	1,233	BWR	UR	Tenn. Valley Authority	1981
Hartsville	TVA Plant 2: Unit 2	1,233	BWR	UR	Tenn. Valley Authority	1982
*	TVA/CE No. 1	1,300	PWR	A/O	Tenn. Valley Authority	1982
*	TVA/CE No. 2	1,300	PWR	A/O	Tenn. Valley Authority	1983
*	TVA/GE No. 1	1,233	BWR	A/O	Tenn. Valley Authority	1983
*	TVA/GE No. 2	1,233	BWR	A/O	Tenn. Valley Authority	1983

* Site not selected.

**Limited work authorization issued.

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station: Unit 1	1,150	PWR	CP	Texas P&L, Dallas P&L, Texas Elec. Service	1980
Glen Rose	Comanche Peak Steam Electric Station: Unit 2	1,150	PWR	CP	Texas P&L, Dallas P&L, Texas Elec. Service	1982
Jasper	Blue Hills Station: Unit 1	918	PWR	UR	Gulf States Utilities Co.	1985
Jasper	Blue Hills Station: Unit 2	918	PWR	UR	Gulf States Utilities Co.	1987
Wallis	Allens Creek: Unit 1	1,213	BWR	UR	Houston Lighting & Power Co.	1980
Wallis	Allens Creek: Unit 2	1,213	BWR	UR	Houston Lighting & Power Co.	1982
Bay City	South Texas Nuclear Project: Unit 1	1,250	PWR	UR	Houston Lighting & Power Co.	1980
Bay City	South Texas Nuclear Project: Unit 2	1,250	PWR	UR	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	788	PWR	OL	Va. Electric & Power Co.	1972
Gravel Neck	Surry Power Station: Unit 2	788	PWR	OL	Va. Electric & Power Co.	1973
Gravel Neck	Surry Power Station: Unit 3	859	PWR	CP	Va. Electric & Power Co.	1983
Gravel Neck	Surry Power Station: Unit 4	859	PWR	CP	Va. Electric & Power Co.	1984
Mineral	North Anna Power Station: Unit 1	898	PWR	CP	Va. Electric & Power Co.	1977
Mineral	North Anna Power Station: Unit 2	898	PWR	CP	Va. Electric & Power Co.	1977
Mineral	North Anna Power Station: Unit 3	907	PWR	CP	Va. Electric & Power Co.	1980
Mineral	North Anna Power Station: Unit 4	907	PWR	CP	Va. Electric & Power Co.	1981
WASHINGTON						
Richland	N-Reactor/WPPSS Steam	850	GR	—[1]	Wash. Public Power Supply System	1966
Richland	WPPSS No. 1 (Hanford)	1,218	PWR	UR	Wash. Public Power Supply System	1980
Richland	WPPSS No. 2 (Hanford)	1,103	BWR	CP	Wash. Public Power Supply System	1978

[1]Operable but OL not required. (A gas-cooled, graphite-moderated reactor owned by the U.S. Government)

Site	Plant Name	Capacity (Net MWe)	Type	Status	Utility	Scheduled Operation
WASHINGTON (Continued)						
Satsop	WPPSS No. 3	1,242	PWR	UR	Wash Public Power Supply System	1981
Richland	WPPSS No. 4	1,218	PWR	UR	Wash. Public Power Supply System	1982
Satsop	WPPSS No. 5	1,242	PWR	UR	Wash. Public Power Supply System	1983
Sedro Wooley	Skagit Nuclear Power Project: Unit 1	1,277	BWR	UR	Puget Sound Power & Light Co.	1982
Sedro Wooley	Skagit Nuclear Power Project: Unit 2	1,277	BWR	UR	Puget Sound Power & Light Co	1985
WISCONSIN						
Genoa	Genoa Nuclear Generating Station (LaCrosse)	50	BWR	OL	Dairyland Power Coop.	1971
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	541	PWR	OL	Wisconsin Elec. Power Co.	1974
Durand	Tyrone Energy Park: Unit 1	1,150	PWR	UR	Northern States Power Co.	1985
Ft. Atkinson	Koshkonong Nuclear Plant: Unit 1	900	PWR	UR	Wisconsin Elec. Power Co.	1983
Ft Atkinson	Koshkonong Nuclear Plant: Unit 2	900	PWR	UR	Wisconsin Elec. Power Co	1984
PUERTO RICO						
Arecibo	North Coast Nuclear Plant: Unit 1	583	PWR	UR	Puerto Rico Water Resources Authority	1981

Appendix 2

NRC Organization

COMMISSIONERS

William A. Anders, Chairman
Marcus A. Rowden
Edward A. Mason
Victor Gilinsky
Richard T. Kennedy

The Commission Staff

Peter L. Strauss, General Counsel
Benjamin Huberman, Director, Office of Policy Evaluation
John A. Harris, Director, Office of Public Affairs
Carlton C. Kammerer, Director, Office of Congressional Affairs
Thomas J. McTiernan, Director, Office of Inspector and Auditor
Samuel J. Chilk, Secretary of the Commission

Other Offices

Advisory Committee on Reactor Safeguards, William Kerr, Chairman
Atomic Safety & Licensing Board Panel, James R. Yore, Acting Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

EXECUTIVE DIRECTOR FOR OPERATIONS

Lee V. Gossick, Executive Director for Operations
William J. Dircks, Assistant Executive Director for Operations
Stephen H. Hanauer, Technical Advisor

Staff Offices

Daniel J. Donoghue, Director, Office of Administration
Howard K. Shapar, Executive Legal Director
Robert J. Friedman, Controller
Edward E. Tucker, Director, Office of Equal Employment Opportunity
Barrett J. Riordan, Director, Office of Planning and Analysis
Joseph D. Lafleur, Jr., Acting Director, Office of International and State Programs
William G. McDonald, Director, Office of Management Information and Program Control
Seymour H. Smiley, Director, Office of Special Studies

PROGRAM OFFICES

Benard C. Rusche, Director, Office of Nuclear Reactor Regulation
Kenneth R. Chapman, Director, Office of Nuclear Material Safety and Safeguards
Herbert J.C. Kouts, Director, Office of Nuclear Regulatory Research
Robert B. Minogue, Director, Office of Standards Development
Donald F. Knuth, Director, Office of Inspection and Enforcement

NRC Regional Offices

Region I Philadelphia, Pa., James P. O'Reilly, Director
Region II Atlanta, Ga., Norman C. Moseley, Director
Region III Chicago, Ill., James G. Keppler, Director
Region IV Dallas, Texas, E. Morris Howard, Director
Region V San Francisco, Calif., Robert H. Engelken, Director

The Energy Reorganization Act of 1974 specified that, below the Commission level, there would be an Executive Director for Operations, and three regulatory or "line" offices: the Offices of Nuclear Reactor Regulation, Nuclear Material Safety and Safeguards, and Nuclear Regulatory Research. During the transition phase of the organization's development, NRC determined that two additional offices were needed at the same level to perform functions not specifically mandated by the legislation. (See organization chart in Chapter 1.)

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 performs licensing functions associated with nuclear power plants. NRR reviews applications for construction permits and operating licenses for power, test and research reactors, and is responsible for the issuance of licenses for operators and senior operators at licensed facilities. NRR also is responsible for the detailed technical safety and environmental evaluation of both applications and the operating facilities themselves, as well as for review of generic safety issues associated with reactor safety, containment safety, site safety, and engineering. A continuing NRR responsibility is the review and improvement of the licensing process.

The Office of Nuclear Material Safety and Safeguards is responsible for 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 was established in recognition of the importance of confirmatory assessment to the nuclear regulatory process. RES is responsible for the reactor safety research program, and, during FY 1975, plans were initiated to increase such research to the level necessary to assure an independent technical basis for NRC licensing activities. RES also is responsible for contracting research in such fields as safeguards, health effects associated with the nuclear fuel cycle, environmental impact of nuclear power, criticality control, waste treatment and disposal, and transportation of radioactive materials.

The Office of Standards Development develops regulations, criteria, guides, standards, and codes pertaining to nuclear health and safety and environmental protection in the siting, design, construction and operation of nuclear reactors and other facilities, and to the management, protection and use of nuclear materials held by NRC licensees.

The Office of Inspection and Enforcement inspects material and facility licensees to determine if operations are conducted in compliance with provisions of applicable licenses and the Commission's rules; determines that requirements for the docketing of applications and the issuance of construction permits or operating licenses have been met; investigates accidents, incidents, allegations of improper action and possible diversion of special nuclear material; and enforces the rules. IE also acts on behalf of the NRC in supervising and administering the Commission's five regional field offices, located as follows: Region I, Philadelphia, Pa.; Region II, Atlanta, Ga.; Region III, Chicago, Ill.; Region IV, Dallas, Texas, and Region V, San Francisco, Calif.

The Commission Staff

The Office of the Secretary provides administrative and logistic support services required for the discharge of the Commission's roles and missions and implementation of Commission decisions and other actions. In the discharge ot this overall secretariat function, the Office forecasts and schedules the conduct of Commission business; records Commission meetings; supervises and operates the NRC paperwork system; maintains status of NRC actions; supervises and administers the NRC Washington, D.C. Public Document Room; operates the Commission Correspondence and Records Facility; maintains the official docket of the Commission; coordinates protocol activities at Commission level; provides administrative service by furnishing the NRC Federal Advisory Committee Management Officer; operates the classified document control system for Commissioners and other offices reporting to the Commission; and performs additional functions as assigned by the Commission.

The Office of the General Counsel is the NRC's chief legal officer and the legal advisor to the Commission, providing legal opinions, advice, and consultation in connection with the quasi-judicial responsibilities of the Commission and in the development of substantive policy matters. GC represents the Commission in matters relating to litigation, and, in cooperation with the Department of Justice, in court proceedings affecting NRC programs. The office also provides legal advice on legislative matters of concern to NRC, drafts legislation, prepares and reviews testimony, and prepares and transmits statements or views requested on proposed legislation.

The Office of Policy Evaluation reports directly to the Commission and provides a point for independent review of proposed and existing policies and programs. PE has a special responsibility to assure that proposed policies reaching the Commission are complete, balanced and understandable, that they take proper and timely cognizance of NRC's technical, economic and social concerns, that they are compatible with prior decisions and assumptions or—if they are not—the nature and implications of change are clearly presented to the Commission. PE also coordinates and periodically prepares, in response to Commission request, issue papers and studies of special interest to the Commission.

Office of the Inspector and Auditor conducts investigations and inspections to verify the integrity of

NRC operations; investigates allegations of NRC employee misconduct, and complaints in the areas of equal employment opportunity and civil rights; assists the Commission in carrying out its financial management responsibilities by developing policies and standards governing financial and management audit programs; administers the Commission's day-to-day audit activities; serves as the point of contact with the General Accounting Office on matters affecting the agency's financial and management audit functions; and maintains liaison with the Department of Justice and other law enforcement agencies, including coordination and handling of criminal referrals.

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 assistance to the Commission and senior staff on congressional matters, coordinates interagency congressional relations activities, and is the principal liaison with congressional committees and members of Congress.

Support Staff

The Office of Administration provides personnel administration; security and classification; technical information; facilities and materials license fees; contracting and procurement; rules, proceedings and document services; data processing; building management; printing and reproduction; records management; housekeeping functions, and support for the local public document rooms. Directs management and administrative support programs.

The Office of the Controller develops and maintains the Commission's financial management program, including policies, procedures, and standards of accounting, budgeting, pricing, contract finance, automatic data processing equipment acquisition, and accounting for capitalized property. Prepares reports necessary to the financial integrity, efficiency, and management of NRC direct and contract operations and to safeguarding of NRC funds. Administers financial functions for the agency and maintains liaison with the General Accounting Office, Office of Management and Budget, Treasury Department, General Services Administration, and other agencies, congressional committees, and industry. The Controller provides a resource planning and evaluation function which designs systems and develops criteria for use in developing NRC program planning; evaluates the critical relationship between resource allocation and program performance; develops a coordinated and comprehensive Five-year Plan; and conducts manpower productivity analyses.

The Office of the Executive Legal Director represents the NRC staff in proceedings involving the licensing of nuclear power reactors and other major nuclear facilities; the enforcement of Commission license conditions and regulations; rulemaking, and antitrust matters in connection with licensing power reactors and other types of nuclear facilities. Provides legal advice in administrative and operational matters such as contracting, claims, financial management, and personnel administration.

The Office of Equal Employment Opportunity plans and administers the NRC Equal Employment Opportunity program, assuring, through consultation with other offices and advice to the Commission, that appropriate measures are taken to meet Federal goals for the employment and advancement of minority personnel.

The Office of International and State Programs provides direct program support to the NRC staff for international relations, state relations, and emergency preparedness programs. (At year-end, NRC was in the process of establishing separate Offices for International Programs and for State Programs.)

The Office of Management Information and Program Control provides an integrated management information and control system for program planning and the reporting and analysis of schedules and performance of programs accomplished by the other offices of NRC.

The Office of Planning and Analysis provides independent overall analyses of programs, issues, policy options and alternatives, and coordinates NRC's cost-benefit policy.

The Office of Special Studies conducts the Congressionally mandated Nuclear Energy Site Survey, due to the Congress in FY 1976.

Other Offices

The Energy Reorganization Act provided for the continued functioning of three independent bodies within the NRC structure which play key roles in the nuclear power plant licensing process:

Advisory Committee on Reactor Safeguards, an independent group of 15 eminent scientists and engineers who review and report their recommendations on all applications for construction and operation of nuclear power reactors and related matters.

Atomic Safety and Licensing Board Panel: Three-member licensing boards drawn from the Atomic Safety and Licensing Board Panel—made up of lawyers and others with expertise in various technical fields—conduct public hearings on construction permit applications and on operating license applications.

Atomic Safety and Licensing Appeal Panel: Three-member appeal boards selected from this panel review decisions of the licensing boards, as required.

Appendix 3

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 and has provided continuing input into the Commission's Reactor Safety Research Program. The members are appointed for four-year terms by the Commission. The Committee annually designates its own chairman and vice chairman. As of December 31, 1975, the members were:

Prof. WILLIAM KERR, *Chairman,* Professor of Nuclear Engineering, University of Michigan, Ann Arbor, Mich.

Dr. DADE W. MOELLER, *Vice Chairman,* Professor of Engineering in Environmental Health, Head of Environmental Health Sciences Dept. and Associate Director, The Kresge Center for Environmental Health, School of Public Health, Harvard University, Boston, Mass.

JOHN H. ARNOLD, Consultant, Air Products and Chemicals, Inc. Allentown, Pa.

MYER BENDER, Manager of Engineering, Holifield National Laboratory, Oak Ridge, Tenn.

Dr. SPENCER H. BUSH, Senior Staff Consultant, Battelle Memorial Institute, Pacific Northwest Laboratory, Richland, Wash.

Dr. MAX W. CARBON, Professor, Nuclear Engineering Department, University of Wisconsin, Madison, Wis.

LUCIAN W. FOX, Research Manager, Environmental Analysis and Planning Division, Savannah River Laboratory, Aiken, S.C.

Dr. HERBERT S. ISBIN, Professor of Chemical Engineering, University of Minnesota, Minneapolis, Minn.

Dr. STEPHEN LAWROSKI, Senior Engineer, Chemical Engineering Division, Argonne National Laboratory, Argonne, Ill.

Dr. DAVID OKRENT, Professor, School of Engineering and Applied Science, University of California, Los Angeles, Calif.

Dr. MILTON PLESSET, Professor, Department of Engineering Science, California Institute of Technology, Pasadena, Calif.

Dr. CHESTER P. SIESS, Professor and Head of Civil Engineering, University of Illinois, Urbans, 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 December 31, 1975, the Licensing Board Panel was composed of the following members and professional staff:

JAMES R. YORE, *Acting Chairman, 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. B. BRIGGS, Senior Research Engineer, Holifield National Laboratory, Oak Ridge, Tenn.

*Denotes full-time ASLBP members and staff

GLENN O. BRIGHT, ASLBP Engineer, Bethesda, Md.*

Dr. A. DIXON CALLIHAN, Retired Union Carbide Corp. Physicist, Oak Ridge, Tenn.

Dr. E. LEONARD CHEATUM, Director of Institute of Natural Resources, University of Georgia, Athens, Ga.

HUGH K. CLARK, Retired E. I. duPont de Nemours & Co. Attorney, Kennedyville, Md.

Dr. RICHARD F. COLE, ASLBP Environmentalist, Bethesda, Md.*

FREDERIC J. COUFAL, ASLBP Attorney, Bethesda, Md.*

Dr. FREDERICK P. COWAN, Retired Brookhaven National Laboratory Physicist, Stuart, Fla.

Dr. FRANKLIN C. DAIBER, Department of Biological Sciences, University of Delaware, Newark, Del.

RALPH S. DECKER, Retired Engineer, U.S. Atomic Energy Commission, Cambridge, Md.

Dr. DONALD P. de SYLVA, Assoc. Prof. of Marine Science, Rosentiel School of Marine and Atmospheric Science, University of Miami, Miami, Fla.

MICHAEL A. DUGGAN, College of Business Administration, University of Texas, Austin, Texas

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 of Center for Population Studies, University of Minnesota, Minneapolis, Minn.

JOHN H. FRYE, III, ASLBP Legal Assistant, Bethesda, Md.*

JOHN M. FRYSIAK, ASLBP Attorney, Bethesda, Md.*

MICHAEL GLASER, Partner in Law Firm of Glaser and Fletcher, Washington, D.C.

ANDREW C. GOODHOPE, Retired Federal Trade Commission Administrative Law Judge, Wheaton, Md.

Dr. DAVID B. HALL, Los Alamos Scientific Laboratory, Los Alamos, N.M.

Dr. CADET HAND, Director, Bodega Marine Laboratory, University of California, Bodega Bay, Calif.

Dr. DAVID L. HETRICK, Professor of Nuclear Engineering, 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 of 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, Holifield National Laboratory, Oak Ridge, Tenn.

Dr. WALTER H. JORDAN, Retired Senior Research Advisor & Physicist, Holifield 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 in Law Firm of Laurence, Laurence and Neilan, Arlington, Va.

ROBERT M. LAZO, ASLBP Attorney, Bethesda, Md.*

Dr. J. V. LEEDS, Jr., Professor of Environmental and Electrical Engineering, Rice University, Houston, Texas

GUSTAVE A. LINENBERGER, ASLBP Physicist, Bethesda, Md.*

Dr. LINDA W. LITTLE, Assoc. Professor, Department of Environmental Sciences & Engineering, University of North Carolina, Chapel Hill, N.C.

Dr. STANLEY LIVINGSTON, Retired Associate Director, AEC National Accelerator Laboratory, Sante Fe, N.M.

Dr. EMMETH A. LUEBKE, ASLBP Physicist, Bethesda, Md.*

EDWARD LUTON, ASLBP Attorney, Bethesda, Md.*

Dr. JOHN R. LYMAN, Retired Professor of Oceanography, University of North Carolina, Chapel Hill, N.C.

Dr. MARVIN M. MANN, ASLBP Technical Advisor, Bethesda, Md.*

Dr. WILLIAM E. MARTIN, Senior Ecologist, Battelle Memorial Institute, Columbus, Ohio

Dr. KENNETH A. McCOLLOM, Associate Dean, College of Engineering, Oklahoma State University, Stillwater, Okla.

GARY L. MILHOLLIN, Associate Professor, Catholic University of America School of Law, Washington, D.C.

MARSHALL E. MILLER, ASLBP Attorney, Bethesda, Md.*

Dr. HUGH PAXTON, Los Alamos Scientific Laboratory, Los Alamos, N.M.

Dr. THOMAS H. PIGFORD, Professor of Nuclear Engineering, University of California, Berkeley, Calif.

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*Denotes full-time ASLBP members and staff

Dr. PAUL W. PURDON, Chairman, Department of Civil Engineering, Drexel University, Philadelphia, Pa.
THOMAS W. REILLY, JR., ASLBP Attorney, Bethesda, Md.*
Dr. FORREST J. REMICK, Director of Institute of Science and Engineering, Pennsylvania State University, University Park, Pa.
DOUGLAS V. RIGLER, Partner in Law Firm of Foley, Lardner, Hollabough & Jacobs, Washington, D.C.
Dr. ERNEST O. SALO, Professor, Fisheries Research Institute-WH-10, College of Fisheries, University of Washington, Seattle, Wash.
DAVID R. SCHINK, Department of Oceanography, Texas A & M University, College Station, Texas
CARL W. SCHWARZ, Partner in 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 Assoc. Prof., Fisheries Research Institute, University of Washington, Seattle, Wash.
JOSEPH F. TUBRIDY, Attorney at Law, Washington, D.C.
Dr. LEONARD W. WEISS, Department of Economics, The University of Wisconsin, Madison, Wis.
JOHN F. WOLF, Attorney in Law Firm of Lamensdorf, Leonard & Moore, Washington, D.C.
JAMES R. YORE, ASLBP Executive Secretary, Bethesda, Md.*

Atomic Safety and Licensing Appeal Panel

An Atomic Safety and Licensing Appeal Board was established by the Atomic Energy Commission, effective September 18, 1969. The Board was delegated the authority to perform the 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. The Appeal Board was organizationally separated from the Atomic Safety and Licensing Board panel on December 16, 1971.

In view of the increase in the number of proceedings subject to administrative appellate review, the Atomic Energy Commission on October 25, 1972, established the Atomic Safety and Licensing Appeal Panel 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, exercised solely on its own motion, of decisions and actions of Appeal Boards.

The Appeal Panel on December 31, 1975, 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.
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.
PAUL GAUKLER, Legal Intern, Appeal Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.

In addition to the permanent members, also available to serve as Appeal Board members for specific proceedings are:

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 by the Atomic Energy Commission 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 Director, Division of Fuel Cycle and Material Safety serves as Committee Chairman.

*Denotes full-time ASLBP members and staff.

RICHARD E. CUNNINGHAM, *Acting Chairman, ACMI,* Acting Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Bethesda, Md.

JOHN E. CHRISTIAN, Head, Bionucleonics, Purdue University, West Lafayette, Ind.

FRANK H. DE LAND, Chief, Nuclear Medicine Department, Veterans Administration Hospital, Lexington, Ky.

Dr. DAVID E. KUHL, Professor of Radiology, University of Pennsylvania School of Medicine, Philadelphia, Pa.

Dr. JAMES L. QUINN, III, Director, Nuclear Medicine Department, The Wesley Memorial Hospitals, Chicago, Ill.

Dr. HENRY N. WAGNER, JR., Professor of Radiology and Radiological Science, Johns Hopkins Medical Institution, Baltimore, Md.

Dr. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.

Dr. CHARLES D. WEST, Professor of Medicine and Biochemistry, University of Utah School of Medicine, Salt Lake City, Utah

Dr. JOSEPH B. WORKMAN, Associate Professor of Radiology, Duke University Medical Center, Durham, N.C.

Fiscal Year 1975
NRC Financial Statements

Balance Sheet at June 30, 1975

ASSETS		
Cash:		
Funds in U.S. Treasury	$49,400,846.99	
Cash on hand	24,750.00	
		$49,425,596.99
Accounts Receivable:		
Federal Agencies	21,413.98	
Other	7,719.41	
Miscellaneous Receipts	214,325.38	
		243,458.67
Plant:		
Completed Plant and Equipment	4,499,861.16	
Less-Accumulated Depreciation	780,329.06	
		3,719,532.10
Advances and Prepayments:		
Federal Agencies	167,506.00	
Other	521,560.21	
		689,066.21
Total Assets		
		$54,077,653.97

LIABILITIES AND NRC EQUITY		
Liabilities:		
Funds held for Others		$ 943,645.99
Accts. Payable & Accrued Expenses:		
Federal Agencies	$17,397,953.94	
Other	5,674,340.48	23,072,294.42
Accrued annual lv. of employees		3,555,185.47
Total Liabilities		$27,571,125.88
NRC Equity:		
Additions:		
Funds Appropriated-net	$101,703,397.28	
Deductions:		
Net Cost of Operations	70,608,819.45	
Non-reimbursable transfers (AEC)	3,350,769.08	
Funds returned to US Treasury	1,237,280.66	
Total Deductions	75,196,869.19	
Total Equity		26,506,528.09
Total Liabilities & Equity		$54,077,653.97

Statement of Operations
Fiscal Year 1975 (January 19, 1975 through June 30, 1975)

Personal Services:		
Standards Development	$ 1,238,979.50	
Nuclear Reactor Regulation	6,264,940.09	
Inspection and Enforcement	4,904,368.95	
Program Direction and Administration	5,705,907.56	
Nuclear Material Safety & Safeguards	1,250,699.95	
Nuclear Regulatory Research	953,433.37	$20,318,329.42
Personnel Benefits:		
Standards Development	128,882.86	
Nuclear Reactor Regulation	622,241.59	
Inspection and Enforcement	612,545.74	
Program Direction and Administration	518,876.92	
Nuclear Material Safety & Safeguards	140,570.97	
Nuclear Regulatory Research	84,256.21	2,107,374.29
Benefits for former personnel - IE		13,454.79
Program Support:		
Standards Development	1,670,683.23	
Nuclear Reactor Regulation	4,721,748.42	
Inspection and Enforcement	1,204,712.69	
Program Direction and Administration	995,051.63	
Nuclear Material Safety & Safeguards	2,309,960.32	
Nuclear Regulatory Research	28,588,596.61	39,490,752.90
Administrative Support:		
Standards Development	408,312.10	
Nuclear Reactor Regulation	1,499,051.04	
Inspection and Enforcement	1,133,267.55	
Program Direction and Administration	2,024,552.99	
Nuclear Material Safety & Safeguards	418,781.26	
Nuclear Regulatory Research	137,340.15	5,621,305.09
Travel of Persons:		
Standards Development	62,016.47	
Nuclear Reactor Regulation	309,417.80	
Inspection and Enforcement	525,735.01	
Program Direction and Administration	325,582.00	
Nuclear Material Safety & Safeguards	42,416.14	
Nuclear Regulatory Research	83,273.06	1,348,440.48
Indemnities and miscellaneous:		
Nuclear Reactor Regulation	3.55	
Inspection and Enforcement	395.39	
Program Direction and Administration	18.03	
Nuclear Material Safety & Safeguards	50.00	466.97
Increase in annual leave accrual:		
Standards Development	40,351.68	
Nuclear Reactor Regulation	201,978.12	
Inspection and Enforcement	157,694.02	
Program Direction and Administration	315,144.86	
Nuclear Material Safety & Safeguards	41,945.72	
Nuclear Regulatory Research	36,959.66	794,074.06

Statement of Operations (Continued)

Depreciation expense:			
Inspection and Enforcement		15,046.24	
Program Direction and Administration		76,850.91	91,897.15
Refunds of annual license fees			2,785,144.96
Equipment write-offs			55,890.49
Total Cost of Operations			72,627,130.60
Less Revenues:			
Reimbursable work for other agencies		7,219.36	
Fees (to be deposited with US Treasury):			
Licenses		193,010.00	
License Indemnities		1,631,521.01	
Civil Penalties		31,150.00	
Facilities Review services	$153,510.00		
Miscellaneous services	1,900.78	$2,011,091.79	$2,018,311.15
Net Cost of Operations			$70,608,819.45

Appendix 5

Rules and Regulations

The regulations of the Nuclear Regulatory Commission are contained in Title 10, Chapter I, of the Code of Federal Regulations. Effective and proposed regulations concerning licensed activities, and certain policy statements relating thereto, which were published in the *Federal Register* during the second half of fiscal year 1975, are set forth below.

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Environmental Effects of Transportation of Radioactive Materials to and from Nuclear Power Plants—Part 51

On January 6, 1975, an amendment to Part 51 was published, effective February 5, 1975, incorporating the consideration of environmental effects associated with the transportation of nuclear fuel and wastes in individual cost-benefit analyses for light-water-cooled nuclear power reactors.

Discovery; Issues Not Raised by Parties—Part 2

On January 17, 1975, amendments to Part 2 were published, effective February 18, 1975, which will (1) change Commission discovery procedures as they relate to the attendance and testimony of AEC personnel at hearings and staff responses to written interrogatories and (2) provide that while Atomic Safety and Licensing Boards are neither required nor expected to look for new issues not raised by the parties, in extraordinary circumstances where the board determines that a serious safety, environmental, or common defense and security matter was not raised by the parties to a hearing, the board may exercise its discretion to examine and decide such matters; Atomic Safety and Licensing Appeal Boards may give appropriate consideration to such matters.

Licensing of Duplicate Nuclear Power Plants; Review of Standard Nuclear Power Plant Designs—Parts 2 and 50

On January 17, 1975, amendments to Parts 2 and 50 were published, effective February 18, 1975, implementing two approaches to standardization of nuclear power reactors identified by the Commission in a statement issued on March 5, 1973: (1) the "Reference System" concept under which an entire facility design or major fraction thereof can be identified as a standard design to be used in multiple applications and (2) the "Duplicate Plant" concept under which simultaneous AEC staff review of the safety-related parameters of duplicate plants to be constructed by a utility or a group of utilities may be conducted.

Group Licensing for Certain Medical Uses—Part 35

On January 20, 1975, amendments to Part 35 were published, effective immediately, enabling physicians to obtain certain radiopharmaceuticals in prepared form from radiopharmacists instead of making their own preparation from reagent kits. An editorial change was also made.

Quality Assurance Criteria—Permissible Organizational Relationships—Part 50

On January 20, 1975, an amendment to Part 50 was published, effective February 19, 1975, clarifying the intent of Criterion I, "Organization," of Appendix B, with regard to permissible organizational relationships.

Miscellaneous Amendments—Part 140

On February 19, 1975, amendments to Part 140 were published, effective March 21, 1975, which increased the amount of privately available nuclear energy liability insurance.

Implementation of the Freedom of Information Act—Parts 2 and 9

On February 24, 1975, amendments to Parts 2 and 9 were published, effective immediately, establishing procedures to be followed by persons seeking records from the NRC and the actions to be taken by the NRC with respect to such requests. These amendments reflect certain provisions contained in the 1974 amendments to the Freedom of Information Act.

Energy Reorganization Act; Revision of Chapter I to Reflect Organizational and Procedural Changes

On March 3, 1975, amendments to 10 CFR Chapter I were published, effective immediately, reflecting the division of the Atomic Energy Commission into two separate agencies, the organization of the Nuclear Regulatory Commission, the licensing and the related regulatory authority of the NRC as to certain ERDA facilities and activites, and the prohibition against sex discrimination in Title IV of the Energy Reorganization Act.

Establishment of Interim Amount of Financial Protection and Indemnity Fee—Part 140

On March 21, 1975, NRC published in the Federal Register the interim level of financial protection and the annual indemnity fee established for the Barnwell Nuclear Fuel Plant, effective April 20, 1975.

Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents—Part 50

On May 5, 1975, amendments to Part 50 were published, effective June 4, 1975, specifying design and operating requirements for nuclear power reactors to keep levels of radioactivity in effluents "as low as practicable" and providing numerical guides for design objectives and technical specification requirements for limiting conditions for operation for light-water-cooled nuclear power reactors.

Population Center Distances—Part 100

On June 24, 1975, an amendment to Part 100 was published, effective immediately, to restore and make clear the intended meaning of the current and consistent siting practice of the Commission. Public comment was invited.

Storage and Control of Licensed Materials—Part 20

On June 25, 1975, an amendment to Part 20 was published, effective immediately, to clearly convey the intention that constant control be maintained over all licensed radioactive materials in unrestricted areas.

Group Licensing for Certain Medical Uses—Part 35

On June 25, 1975, amendments to Part 35 were published, effective immediately, to include in § 35.100 the use of iodone-125 as seeds for interstitial treatment of cancer. An editorial correction was also made.

REGULATIONS AND AMENDMENTS PROPOSED

Radiation Protection; Implementation of NCRP Recommendations for Lower Radiation Exposure Levels for Fertile Women—Parts 19 and 20

On January 3, 1975, proposed amendments to Parts 19 and 20 were published for comment that would incorporate the intent of the recommendation of the National Council on Radiation Protection and Measurements (NCRP) in Report No. 39 that the radiation exposure to an embryo or fetus be minimized.

Use of Depleted Uranium in Industrial Products or Devices; Proposed General License—Part 40

On January 10, 1975, proposed amendments to Part 40 were published for comment which would authorize the manufacture, import, transfer, and use of depleted uranium contained in industrial products for mass-volume applications with minimum regulatory controls necessary to provide adequate safety in use and to exercise control over disposal or abandonment.

Reports to the Commission Concerning Defects and Noncompliance; Proposed Requirements—Parts 2, 21, 31, 35, and 40

On March 3, 1975, proposed amendments to Parts 2, 21, 31, 35, and 40 were published for comment implementing Section 206 of P.L. 93-438, the Energy Reorganization Act of 1974.

Advance Notice of Certain Shipments of Special Nuclear Material—Part 73

On April 4, 1975, proposed amendments to Part 73 were published for comment requiring that the Commission be informed seven days in advance of a shipment of quantities of special nuclear material and that it be notified upon arrival of such shipments.

Personnel Monitoring Reports—Part 20

On May 30, 1975, proposed amendments to Part 20 were published for comment which would extend to all NRC specific licensees the requirements for submission of an annual statistical summary report of estimated whole body radiation doses.

Radioactive Material; Packaging and Transportation By Air—Parts 71 and 73

On June 2, 1975, proposed amendments to Parts 71 and 73 were published for comment concerning the air transportation of radioactive materials, including packaging.

Plans for Coping With Emergencies—Part 70

On June 27, 1975, proposed amendments to Part 70 were published for comment which would codify the information required from an applicant in emergency plans submitted to the Commission in an application for a license to possess and use special nuclear material in fuel processing and fuel fabrication plants.

Appendix 6

Regulatory Guides

In November 1970, the Regulatory staff initiated a series of safety guides to assist applicants in determining the acceptability of specific safety-related features of water-cooled nuclear power plants. This series was expanded in December 1972 into a broad framework for guidance in all areas of Regulatory responsibility. In addition to providing guidance to license applicants, one purpose of the guides is to describe and make available to the public methods for implementing specific parts of the regulations that are acceptable to the NRC staff and, in some cases, to describe techniques used by the staff in evaluating specific problems or accidents. Issuance of each guide is accompanied by a public announcement and a *Federal Register* notice. Single copies of guides may be obtained by writing to the Office of Standards Development, Nuclear Regulatory Commission, Washington, D. C. 20555. The following new or revised guides were issued by NRC in fiscal year 1975 (January-June 1975).

Division 1—Power Reactor Guides

1.16 Reporting of Operating Information (Revision 3).

1.20 Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Revision 1).

1.26 Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Revision 2).

1.64 Quality Assurance Program Requirements for the Design of Nuclear Power Plants (Revision 1).

1.70.1 Additional Information—Hydrological Considerations for Nuclear Power Plants (Revision 1).

1.70.9 Additional Information—Design of Seismic Category I Structures.

1.70.10 Additional Information—Wind and Tornado Loadings.

1.70.11 Information for Safety Analysis Reports—Quality Assurance During Operations Phase.

1.70.12 Information for Safety Analysis Reports—Reactor Materials.

1.70.13 Information for Safety Analysis Reports—Code Cases Applicable to Reactor Coolant Pressure Boundary Components.

1.70.14 Information for Safety Analysis Reports—Emergency Planning.

1.70.15 Information for Safety Analysis Reports—Industrial Security for Nuclear Power Plants.

1.70.16 Information for Safety Analysis Reports—Missile Barrier Design Procedures.

1.70.17 Information for Safety Analysis Reports—Hydrologic Engineering.

1.70.18 Information for Safety Analysis Reports—Mechanical Systems and Components.

1.70.19 Information for Safety Analysis Reports—Steam Generators.

1.70.20 Information for Safety Analysis Reports—Reactor Coolant Pressure Boundary Materials and Inservice Inspection.

1.70.21 Information for Safety Analysis Reports—Reactor Vessels.

1.70.22 Information for Safety Analysis Reports—Instrumentation and Controls.

1.70.23 Information for Safety Analysis Reports—Seismic Qualification of Instrumentation and Electrical Equipment.

1.70.24 Information for Safety Analysis Reports—Environmental Design of Mechanical and Electrical Equipment Qualification Tests and Analyses.

1.70.25 Information for Safety Analysis Reports—Inservice Inspection of ASME Code Class 2 and 3 Components.

1.70.26 Information for Safety Analysis Reports—Metallic Materials for Engineered Safety Features.

1.70.27 Information for Safety Analysis Reports—Radioactive Waste Management.

1.70.28 Information for Safety Analysis Reports—Steam and Feedwater System Materials.

1.70.29 Information for Safety Analysis Reports—Meteorology.

1.70.31 Information for Safety Analysis Reports—Pump Flywheel Integrity.

1.70.32 Information for Safety Analysis Reports—Reactor Water Cleanup System.

1.70.33 Information for Safety Analysis Reports—Initial Test Programs.

1.70.34 Information for Safety Analysis Reports—Fuel System Design.

1.70.35 Information for Safety Analysis Reports—Internally Generated Missiles.

1.70.36 Information for Safety Analysis Reports—Electric Power.

1.70.37 Information for Safety Analysis Reports—Pressurizer Relief Discharge System.

1.70.38 Information for Safety Analysis Reports—Training.

1.75 Physical Independence of Electric Systems (Revision 1).

1.81 Shared Emergency and Shutdown Electric Systems for Multi-Unit Plants (Revision 1).

1.84 Code Case Acceptability—ASME Section III Design and Fabrication (Revisions 1 and 2).

1.85 Code Case Acceptability—ASME Section III Materials (Revisions 1 and 2).

1.87 Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595 and 1596((Revision 1).

1.91 Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites.

1.94 Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.

1.95 Protection of Nuclear Power Plant Control Room Operators Against An Accidental Chlorine Release.

1.96 Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants.

Division 2—Research and Test Reactor Guides

None.

Division 3—Fuels and Materials Facilities Guides

3.26 Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants.

3.27 Nondestructive Examination of Welds in the Liners of Concrete Barriers in Fuel Reprocessing Plants.

3.28 Welder Qualification for Welding in Areas of Limited Accessibility in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants.

3.29 Preheat and Interpass Temperature Control for the Welding of Low-Alloy Steel for Use in Fuel Reprocessing Plants and in Plutonium Processing and Fuel Fabrication Plants.

3.30 Selection, Application, and Inspection of Protective Coatings (Paints) for Fuel Reprocessing Plants.

Division 4—Environmental and Siting Guides

4.1 Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants (Revision 1).

4.2 Preparation of Environmental Reports for Nuclear Power Plants (Revision 1).

4.10 Irreversible and Irretrievable Commitments of Material Resources.

Division 5—Materials and Plant Protection Guides

5.16 Standard Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Nuclear-Grade Plutonium Nitrate Solutions and Plutonium Metal (Revision 1).

5.26 Selection of Material Balance Areas and Item Control Areas (Revision 1).

5.29 Nuclear Material Control Systems for Nuclear Power Plants (Revision 1).

5.31 Specially Designed Vehicle with Armed Guards for Road Shipments of Special Nuclear Material (Revision 1).

5.32 Communication with Transport Vehicles (Revision 1).

5.39 General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations.

5.40 Methods for the Accountability of Plutonium Dioxide Powder.

5.42 Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations.

5.43 Plant Security Force Duties.

5.44 Perimeter Intrusion Alarm Systems.

5.47 Control and Accountability of Plutonium in Waste Material.

5.48 Design Considerations—Systems for Measuring the Mass of Liquids.

5.49 Internal Transfers of Special Nuclear Material.

5.51 Management Review of Nuclear Material Control and Accounting Systems.

5.52 Standard Format and Content for the Physical Protection Section of a License Application (For Facilities Other Than Nuclear Power Plants).

Division 6—Product Guides

6.4 Classification of Containment Properties of Sealed Radioactive Sources Contained in Certain Devices to be Distributed for Use Under General License (Revision 1).

Division 7—Transportation Guides

7.3 Procedures for Picking Up and Receiving Packages of Radioactive Material.

7.4 Leakage Test on Packages for Shipment of Radioactive Materials.

7.5 Administrative Guide for Obtaining Exemptions from Certain NRC Requirements Over Radioactive Material Shipments.

Division 8—Occupational Health Guides

8.13 Instruction Concerning Prenatal Radiation Exposure.

Division 9—Antitrust Guides

None

Division 10—General Guides

10.1 Compilation of Reporting Requirements for Persons Subject to NRC Regulations.

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