U.S. Nuclear Regulatory Commission

1986 Annual Report

NRC

June 15, 1987

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

Dear Mr. President:

This Annual Report for 1986 of the United States Nuclear Regulatory Commission is forwarded for your transmittal to the Congress, as required by Section 307(c) of the Energy Reorganization Act of 1974.

The report is devoted mainly to coverage of events and activities occurring in fiscal year 1986, with additional treatment of events after that period where circumstances warranted.

Respectfully

Lando W. Zech, Jr.

Lando W. Zech, Jr.
Chairman

NRC

U.S. NUCLEAR
REGULATORY
COMMISSION

1986
Annual Report

PREVIOUS REPORTS IN THIS SERIES

1975 NRC Annual Report, published April 1976
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NUREG-0400, *1977 NRC Annual Report,* published April 1978
NUREG-0516, *1978 NRC Annual Report,* published February 1979
NUREG-0690, *1979 NRC Annual Report,* published March 1980
NUREG-0774, *1980 NRC Annual Report,* published March 1981
NUREG-0920, *1981 NRC Annual Report,* published June 1982
NUREG-0998, *1982 NRC Annual Report,* published June 1983
NUREG-1090, *1983 NRC Annual Report,* published June 1984
NUREG-1145, Vol. 1, *1984 NRC Annual Report,* published June 1985
NUREG-1145, Vol. 2, *1985 NRC Annual Report,* published June 1986

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FOREWORD

By Lando W. Zech, Jr.
Chairman—U.S. Nuclear Regulatory Commission

This report to the President and Congress for fiscal year 1986 is the first Nuclear Regulatory Commission annual report to be issued during my tenure as Chairman of the NRC, a post I assumed in July 1986. The report highlights our important responsibilities as well as noteworthy events, issues and activities concerning the NRC and the nuclear industry.

The Atomic Energy Act of 1954 and the Energy Reorganization Act of 1974 assign the Nuclear Regulatory Commission its primary statutory responsibilities. The Commission believes that sound and credible Federal safety regulation is an important component in assuring the continued safe use of commercial nuclear power. We at the NRC intend to continue to pursue a policy of firm and fair regulation of the nuclear industry.

Concerning nuclear power reactors, I have three main goals which I intend to pursue during my term as Chairman of the NRC. They are: (1) to do all we can, within our regulatory responsibilities, to assure that current nuclear power plants continue to operate safely; (2) to assure quality construction for plants being built; and (3), for future reactors, to place emphasis on standardization.

The NRC also has important responsibilities for the safe handling and use of nuclear material by universities, hospitals, doctors and industry, regulating fuel cycle facilities, transportation of nuclear material and the storage of spent fuel and radioactive waste. We believe that nuclear materials, properly used, can be of major importance to our national welfare. The Commission recognizes its serious responsibilities for assuring that our licensees manage radioactive materials in such a way that the health and safety of the public is adequately protected.

In order to make the NRC more efficient and effective, I have initiated, with Commission support, a major agency reorganization, a strategic planning effort and a five-year planning, programming and budget document. The purpose of the reorganization, which became effective in April 1987, is to adjust our regulatory programs to the current and future regulatory environment and to better focus on operational safety. The strategic planning effort will better enable us to establish long term goals for the NRC and monitor our progress in achieving these goals. The strategic plan will also provide the framework for the five-year plan, which is expected to be implemented by the end of this calendar year.

The next few years will be pivotal for the nuclear industry in the United States, and will set the stage for the remainder of this century and beyond. I firmly believe that by working with a genuine commitment to excellence and quality we will meet the challenges before us to provide safe and reliable nuclear energy for the citizens of our nation.

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ENERGY REORGANIZATION ACT OF 1974, AS AMENDED

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

"...the short-range and long-range goals, priorities, and plans of the Commission as they relate to the benefits, costs, and risks of nuclear power." (See Chapter 1 for overall policy and planning guidance. Specific goals concerning nuclear power reactors are also discussed in Chapters 2 and 3; operating experience and the evaluation thereof in Chapter 4; fuel cycle concerns in Chapter 5; safeguards in Chapter 6; waste management in Chapter 7; inspection, enforcement and emergency preparedness in Chapter 8; relations with the States in Chapter 9; international relations and nuclear nonproliferation in Chapter 10; and nuclear regulatory research in Chapter 11.)

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

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

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

"(3) safeguarding special nuclear materials at all stages of the nuclear fuel cycle...." (See Chapters 6, 10 and 11.)

"(4) investigating suspected, attempted, or actual thefts of special nuclear materials in the licensed sector and developing contingency plans for dealing with such incidents...." (See Chapters 6, 8 and 11.)

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

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

Section 205 requires development of "a long term plan for projects for the development of new or improved safety systems for nuclear power plants" and an annual updating of the plan. (See Chapter 11.)

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

Section 210 directs the Commission to submit "a plan providing for the specification and analysis of unresolved safety issues relating to nuclear reactors," and to include progress reports in the Annual Report thereafter concerning corrective actions. (See Chapter 2.)

NUCLEAR NONPROLIFERATION ACT OF 1978

Section 602 requires annual reports by the Commission and the Department of Energy to "include views and recommendations regarding the policies and actions of the United States to prevent proliferation which are the statutory responsibility of those agencies...." (See Chapter 10.)

ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 170(i) directs the Commission to report annually on indemnity actions implementing the Price-Anderson Act which provides a system to pay public liability claims in the event of a nuclear incident. (See Chapter 9.)

PUBLIC LAW 96-295

Section 303 directs the Commission to report annually a statement of—

"(1) the direct and indirect costs to the Commission for the issuance of any license or permit and for the inspection of any facility; and (2) the fees paid to the Commission for the issuance of any license or permit and for the inspection of any facility." (See Chapter 13.)

PUBLIC LAW 97-415

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

PUBLIC LAW 97-415

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

1986 Highlights, Policy and Planning

CHAPTER 1

This is the 12th annual report of the U.S. Nuclear Regulatory Commission (NRC). The NRC was created by enactment of the Energy Reorganization Act of 1974 as an independent agency of the Federal Government. The five Commissioners are nominated by the President and confirmed by the U.S. Senate. The Chairman of the Commission is appointed by the President from among the Commissioners confirmed.

The mission of the NRC is to assure that non-military uses of nuclear materials in the United States—as in the operation of nuclear power plants or in medical, industrial or research applications—are carried out with proper regard and provision for the protection of public health and safety, of the environment, and of the national security. The NRC accomplishes its purposes through the licensing of nuclear reactor operations and other possession and use of nuclear materials, including transport and disposal of nuclear materials and wastes; the safeguarding of nuclear materials and facilities from theft and sabotage; and inspection and enforcement actions.

This report covers the major activities, events, decisions and planning that took place during fiscal year 1986 (October 1985 through September 1986) within the NRC or involving the NRC. The report is prepared in compliance with Section 307(c) of the Energy Reorganization Act of 1974, which requires that an annual report be submitted to the President for transmittal to the Congress.

Changes Within Commission and Senior Staff

The following changes took place at the Commission and senior staff level during the report period:

Lando W. Zech, Jr., who had served as a Commissioner since July of 1984, was appointed Chairman of the NRC in July 1986, succeeding Nunzio J. Palladino.

Kenneth M. Carr was appointed to the Commission in August 1986, filling the vacancy created when former Chairman Palladino completed his five-year term.

William C. Parler was appointed General Counsel of the NRC in July 1986, succeeding Herzel H.E. Plaine.

Guy H. Cunningham, former Executive Legal Director, was appointed Deputy General Counsel in July 1986.

Victor Stello, Jr., was appointed Executive Director for Operations of the NRC in April 1986, succeeding William J. Dircks.

James H. Sniezek was appointed Deputy Executive Director for Regional Operations and Generic Requirements in May 1986, succeeding Victor Stello, Jr.

William G. McDonald was appointed Deputy Executive Director for Operations (Information Resources Management) in July 1986.

Eric S. Beckjord was appointed Director of the Office of Nuclear Regulatory Research in September 1986, succeeding Robert B. Minogue.

Noteworthy Events of 1986

Following are some of the more important decisions and actions taken by the Nuclear Regulatory Commission during the report period. Most of the matters and facts set forth below in capsule form are treated in detail in various subsequent chapters of the report, as indicated. These are items related to the most basic business of the agency, the licensing and inspection of nuclear facilities and the uses of nuclear materials, as well as broad policy decisions taken during the report period. The full range of NRC activities, encompassing such important areas as the NRC research program, nuclear waste management, operational data collection and analysis, safeguards activity, State and international programs, judicial proceeding and litigation, and others, is covered in the succeeding 12 chapters and six appendices of this report.

Power Reactor Regulation. In the course of fiscal year 1986, the NRC issued six low-power operating licenses to six separate utilities; full-power licenses were subsequently issued, also during the fiscal year, for four of these reactors. A full-power license was issued early in the report period for a unit approved for low-power operation in the preceding fiscal year; that issuance together with the four cited above produced a total of five full-power licenses for fiscal year 1986. (There were no new applications for operating licenses, construction permits or manufacturing licenses during the fiscal year; Midland Units 1 and 2 (Mich.) were officially cancelled.) These actions brought the total of power reactors licensed to operate at low or full power in the United States to 101 (see Appendix 6). Other kinds of licensing decisions and actions—license amendment requests, requests for exemptions, public hearings, new requirements (backfits), license modification requirements,

Kenneth M. Carr, who retired in May 1985 from the United States Navy with the rank of Vice Admiral, was sworn in as an NRC Commissioner on August 14, 1986. Mr. Carr, a U.S. Naval Academy graduate, had an extensive career in the Navy's nuclear program, dating back to 1950. His last tour of duty was as Deputy and Chief of Staff to the Commander in Chief, U.S. Atlantic Fleet.

and so forth—constitute an inventory of tasks which, with the large number of reactor plants now in operation, has become very extensive. The inventory has been relatively constant in the last two fiscal years, at about 4,000 licensing actions under review. During fiscal year 1986, the staff completed about 2,650 licensing actions, of which about 70 percent were plant-specific and mostly licensee-initiated. At the close of the report period, there were 55 licenses in effect for the operation of non-power reactors; two operating license renewals were issued in fiscal year 1986. (See Chapter 2.)

Nuclear materials licenses administered by the NRC increased during the report period from about 8,900 to about 9,100. Some 14,000 licenses are administered by States which, under agreements with the NRC, have authority over certain materials. (See Chapters 5 and 9.)

Removal of TMI Core Under Way. After six-and-one-half years of extremely difficult and demanding effort, the cleanup at the Three Mile Island (TMI) Unit 2 (Pa.) nuclear power plant reached a watershed during fiscal year 1986. Removal of damaged fuel and structural debris from the reactor vessel itself got under way in October 1985. Shipment of the damaged fuel from the TMI site to the Idaho National Engineering Laboratory also commenced during the report year, where it was to be analyzed to provide a basis for future cleanup procedures and also to gain a better understanding of the original accident sequence and its implications. NRC staff continued careful monitoring at the site, approving the licensee's procedures as necessary and performing inspections of the cleanup systems and equipment. Defueling is expected to be completed by the fourth quarter of calendar year 1987, and the end of the current phase of the TMI cleanup is still projected for the third quarter of calendar year 1988. (See Chapter 3.)

Inspection and Enforcement. NRC inspections during the report period covered operations of 132 power reactor licensees, 42 fuel facilities licensees and 2,048 nuclear materials licensees. A total of 96 civil penalties were imposed, and over $3 million collected, during the period; there were also 10 enforcement orders issued during the period. (See Chapter 8.) Safeguards inspections—appraising measures taken to prevent theft or sabotage at licensed nuclear operations—were carried out for 111 power reactor units, 23 non-power reactors, and 13 fuel facilities. Nine shipments of nuclear materials were inspected for safeguards purposes. (See Chapter 6.)

Nuclear Waste Policy Developments. During the report period, the NRC defined its position on the implementation of the Department of Energy's (DOE) Final Mission Plan and the proposal for the Monitored Retrievable Storage facility. Other steps taken by the NRC to assure that milestones of the Nuclear Waste Policy Act of 1982 (NWPA) are met include comment on DOE's final environmental assessments, issuance of 10 staff technical positions giving guidance to DOE on various matters, and publication of two rulemaking proceedings amending regulations to conform with NWPA requirements.

Passage of the Low-Level Radioactive Waste Policy Amendments Act of 1985, in January 1986, gave the NRC new responsibilties in the management of low-level nuclear waste (see "Exempted Nuclear Waste," below). The NRC's participation in the national program for licensing and regulating uranium recovery facilities and associated mill tailings continued. (see Chapter 7).

Final Rule on Non-power Reactor Fuel. During fiscal year 1986, the NRC issued a final rule requiring non-power reactor licensees using high-enriched uranium to convert to low-enriched uranium, contingent upon Government funding availability. Regulatory requirements for improved physical security at non-power reactors will be considered at a later date, in light of the rule and other changes.

Backfit Policy and Practice. A revised backfitting rule (10 CFR 50.109) was approved by the Commission on

Lando W. Zech, Jr., a member of the Nuclear Regulatory Commission since July 1984, was sworn in as Chairman of the Commission on July 1, 1986. Mr. Zech (left) is shown taking the oath of office, with former Chairman Nunzio J. Palladino administering. On Chairman Zech's left are his wife, Josephine, and one of their five daughters, Carol.

September 20, 1985. Staff policy for managing plant-specific backfitting was revised accordingly and issued in final form in February 1986 (Manual Chapter 0514). Four day-long workshops with industry were held in Philadelphia, Atlanta, Chicago and Dallas to acquaint nuclear power plant owners with the provisions of the new rule and the more general requirements of the revised 10 CFR 50.109 backfit regulation.

Extensive training in the implementation of the revised backfit rule was provided for the NRC staff by means of seminars in each affected Headquarters office and in each of the five Regional offices. Each of these offices has also developed procedures to implement the final Manual Chapter. The procedures are intended to provide a uniform and consistent approach to making backfitting decisions, to encourage effective interchange between the licensee and the staff on backfitting issues, and to furnish guidance on conducting analyses of the economic and potential safety consequences of the proposed requirements.

Important elements of the revised backfit rule, and the staff policy documents now revised to implement the rule, are as follows:

(1) The Executive Director for Operations is responsible for implementation of 10 CFR 50.109.

(2) A determination will be made by the staff and approved by the appropriate NRC Office Director or Regional Administrator as to whether an issue is a backfitting requirement before further action is taken.

(3) A regulatory analysis will be made and approved before a backfitting requirement is imposed on a licensee, unless prompt imposition is deemed necessary.

(4) Following the immediate imposition of a backfitting requirement, a regulatory analysis will be performed to document its assumed safety significance (cost considerations of this analysis are to be included only insofar as they facilitate a selection from among alternatives ways of meeting the requirement).

(5) Information requests other than those pertaining to compliance issues will be evaluated to ensure that the burden of response is justified in terms of the potential safety significance of the matter at issue.

From the time the revised backfit rule was issued in September 1985 through September 1986, seven potential plant-specific backfits were proposed by the staff and 23 proposed requirements were claimed by licensees to be backfits. Of the total, 19 issues had been resolved at the close of the report period, and only one involved the actual imposition of a backfit. Most were found, on further examination, to be matters of compliance with already applicable staff positions. Some issues were dropped by the staff after additional information was received from licensees.

The charter of the Committee to Review Generic Requirements (CRGR), the staff's governing policy document for the activities of the CRGR, has also been revised to reflect the revised backfitting rule. A revised charter was approved by the Commission and issued to the staff and all nuclear power plant owners in September 1986. All generic requirements, including backfit requirements, proposed by the NRC staff related to one or more classes of reactors, must be reviewed by the CRGR. The Committee seeks to eliminate unnecessary demands on licensees by ensuring that the need for a new requirement can be demonstrated by those proposing it. (See the 1983 NRC Annual Report pp. 1-3, for full description of CRGR's structure and review process.) Through its review, the CRGR seeks assurance that a proposed requirement: (1) is necessary for the public health and safety, (2) is likely to result in a net safety improvement, and (3) is likely to have an impact on the public, industry and government which is consistent with and justified by the urgency of the need for the safety improvement to be realized.

Following CRGR review, the CRGR recommends to the Executive Director for Operations (EDO) that the proposed requirement be approved, disapproved, modified or conditioned in some way. The EDO considers CRGR recommendations, as well as those of cognizant NRC offices, in deciding whether a requirement shall be imposed. From its inception in November 1981 through September 1986, the CRGR has held 95 meetings and considered a total of 146 separate issues.

Chernobyl the Worst Nuclear Accident Ever. In 1986, the worst accident by far in the history of nuclear power plant operation took place. A graphite-moderated reactor at Chernobyl in the Ukraine exploded and burned, causing massive releases of radioactivity to the atmosphere, with serious long term health implications for people in several nations of Europe, and short term radiation sickness and death to many in the local area. The reactor is located on the Pripyat river, 81 miles north of Kiev, the capital of the Ukraine and third largest city in the Soviet Union; the Chernobyl site is 425 miles southwest of Moscow.

In the core of the fourth unit of the Chernobyl nuclear power complex, at 1:24 a.m. on April 26, 1986, two explosions occurred. The first one, caused by a buildup of steam and pressure in the reactor tubes, tore the reactor vault apart; the second, a hydrogen explosion, blew the roof off the building. The core tubes in the Chernobyl-type of reactor are surrounded by graphite, a solid form of carbon (as is coal) which, when ignited and exposed to the oxygen in the atmosphere, burns intensely. Escaping radioactive gases and debris created an airborne plume which was carried many thousands of miles in the days following the explosions.

The causal chain of the catastrophe began the day before when plant personnel undertook an experiment which led to the deactivation of certain safety signals and eventually to a shutdown by plant personnel of the steam supply to the turbines of the power plant. Shortly after the latter move was taken—and the automatic reactor shutdown that should have eventuated was deliberately prevented—the reactor was beyond human control and subsequent efforts of the operators to stabilize the situation, e.g., by inserting all control rods, were in vain.

First detection of the accident outside the Soviet Union came when instruments at Sweden's Forsmark nuclear power plant showed usually high readings of radioactivity. The fallout was also detected in Denmark, Norway and elsewhere in Europe. In the United States, the President created an Interagency Task Force to monitor the course of the accident and gather and evaluate as much information about it as could be had. The Administrator of the Environmental Protection Administration headed the task force, which comprised a large number of Federal agencies, including, of course, the NRC.

The NRC staff had been following developments since the first information became available on April 28, 1986. On May 1, 1986, an Incident Tracking Team was established by the Commission to support the EPA in assessing the impact of the accident on the U.S., and to identify the implications of the Chernobyl accident for U.S. reactor operations. Arrangements were made with western and eastern European countries, Korea, Japan, Canada and Israel to

In September 1986 at Vienna, Austria, NRC Chairman Zech chaired a plenary session of the International Atomic Energy Agency's (IAEA) General Conference. At the Chairman's right is Leonid Konstantinov, IAEA Deputy Director General. The General Conference followed an IAEA extraordinary session, from August 25 to 29, on the Chernobyl accident.

receive their radiological environmental monitoring data. U.S. utilities operating or building commercial nuclear power plants were asked to voluntarily report any data they might have on anomalous radioactivity which was probably attributable to the Chernobyl accident.

After the Incident Tracking Team finished its work, on May 14, 1986, the staff continued to collect and analyze information on the Chernobyl accident. Information was developed in a joint effort involving several Federal agencies and with contributions from industry groups. By mutual agreement, the NRC took on the function of overall administrative coordinator.

A conference to receive and review the Soviet report on the matter was held in Vienna from August 25 through August 29, 1986, and was attended by representatives from a large number of countries, including all countries with nuclear energy programs. Soviet officials gave a thorough account of the accident in all its many stages. (See detailed treatment of the Chernobyl accident, and the Vienna conference and its aftermath, under "Safety Reviews," in Chapter 2.)

After the Vienna meeting, the Federal agencies and other groups participating in the drafting of a coordinated U.S. report on the Chernobyl facts resumed work on the report. Evaluation of the full implications of the Chernobyl accident—which is a near "worst-case" scenario with many lessons in the areas of safety systems, administrative controls, containment, emergency planning, source terms, and so forth—was still in progress at the end of 1986. (Two reports were to be issued early in 1987: a fact-finding report, and an assessment of the implications of the accident for regulatory programs and practices in the U.S.). Evaluation of the Chernobyl reactor design completed as of the end of fiscal year 1986 did not identify any aspect of the accident which showed a direct, clear-cut connection with U.S. commercial nuclear power operations.

Commission Policy Statements In Fiscal Year 1986

The Commission issued seven formal policy statements during the fiscal year, dealing with a wide variety of subjects and problems. A listing in chronological order, with a description of the import of each statement, is given below.

Engineering Expertise on Shift. In a policy statement dated October 22, 1985 (effective October 28), the Commission addressed the question of how to assure adequate qualifications in the personnel on shift in nuclear power plants. This concern is one of many given a sharper focus as a result of the Three Mile Island accident in 1979. A requirement introduced in 1980 provided for a specially qualified Shift Technical Advisor to be on duty to advise the shift supervisor in the event of an abnormal occurrence or situation. Later, following certain "man-machine" studies, it was decided that the ability of the entire shift crew to diagnose unusual conditions and act accordingly should be upgraded. The latest policy offers licensees two options by which to meet staffing requirements: combining one of the required Senior Reactor Operator positions with the Shift Technical Advisor position, creating a dual position in accord with NRC qualification requirements; continuing to retain the Shift Technical Advisor position, with NRC approval. The Commission expressed its preference for the dual position, in the interest of moving engineering expertise on the scene from an advisory to an active mode.

Concurrence in DOE Guidelines. In a policy statement of November 5, 1985, the Commission considered the matter of whether changes in certain Department of Energy's (DOE) guidelines which had earlier gained NRC concurrence required reconcurrence by the NRC. The Nuclear Waste Policy Act of 1982 directs the DOE to issue general guidelines for the recommendation of repository sites for high-level nuclear waste and, in doing so, to secure the concurrence of the NRC. In July of 1984, the NRC concurred in the DOE scheme as then submitted, but in mid-1985 the DOE announced plans to make a preliminary determination on suitability of potential nuclear waste repository sites before conducting detailed site characterizations. Since this represented an alteration in the sequence of actions earlier delineated, the Commission reviewed its earlier concurrence to ascertain whether reconcurrence would be necessary under the law. The Commission decided that the change in the timing of the preliminary determination did not affect the validity of the concurrence already given.

Regulation of Advanced Nuclear Plants. This policy statement was issued July 1, 1986, to clarify the licensing context, to the extent possible, for the introduction of advanced nuclear power plant designs. To that end, three distinct objectives of the statement and the policy concerning advanced plants are set forth: (1) to encourage the earliest possible interaction among prospective applicants, vendors, other agencies of Government and the NRC; (2) to provide all concerned with the Commission's views regarding the desired characteristics of advanced reactor design; and (3) to express the Commission's intention to issue timely comment on the implications of such designs for safety, and for the regulatory process as well.

By "advanced reactor designs" are meant designs significantly different from the current generation of light water reactors under construction or in operation. The category would include some current reactor designs, such as the high temperature gas-cooled reactor, liquid metal reactors and light water reactors of innovative design. The NRC (and the AEC) has some experience in reviewing these kinds of designs and, in some cases (e.g., Peach Bottom Unit 1, Fermi Unit 1, Fort St. Vrain), licensing them. The Fast Flux Test Facility and the Clinch River Breeder Reactor was given extensive study by the NRC licensing staff, and some

While in Vienna in September 1986, Chairman Zech signed an extension and renewal of a protocol for the exchange of nuclear safety information with the People's Republic of China. Signing for the Chinese National Nuclear Safety Administration is Director Liang Sheng-Jia. Witnessing the signing for the United States are U.S. Ambassador-at-Large Richard Kennedy (standing behind the Chairman) and NRC Director of International Programs James R. Shea (standing to the Chairman's left).

research on gas-cooled reactors has been undertaken every year. The policy enunciated for the present is that the Commission expects, at the minimum, the same degree of protection of the public health and safety and of the environment from advanced designs as is required of the current generation of light-water reactors, and further expects that advanced units will provide enhanced margins of safety and/or use simplified, inherent, passive or other innovative means to perform safety functions. The new designs must also be in compliance with the NRC safety goals policy (see below).

Final Safety Goals Adopted. On July 31, 1986, the Commission issued a final Policy Statement on Safety Goals for the Operation of Nuclear Power Plants. This policy statement represents the culmination of effort that began in the spring of 1981, pursuant to recommendations of the President's Commission on the Accident at Three Mile Island. The policy statement took into account the conclusions of a two-year evaluation effort, published in mid-1985, and supporting technical evaluations. The Commission's policy statement also benefited from subsequent reviews and recommendations of the Advisory Committee on Reactor Safeguards and of NRC senior management.

The safety goals have now been adopted for use in the regulatory decisionmaking process. This statement of NRC safety policy expresses the NRC's concept and broad definition of those low levels of risk to public health and safety which the industry should strive to attain in the operations of nuclear power plants. (The goals do not apply to risk from other actions involving nuclear material or nuclear waste.) The Commission's policy statement indicates that—while current regulatory practices are believed to ensure that the basic statutory requirement of adequate protection of the public is being met—the safety goals can lead to improvements in current practices and provide a better means for demonstrating the adequacy or lack of it in current requirements. The use of safety goals, however, is not meant and will not be accepted as a substitute for compliance with NRC regulations.

The following are salient portions of the Policy Statement, which became effective August 4, 1986:

The Commission has established two qualitative safety goals which are supported by two quantitative objectives. The two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks. In doing so, the Commission stresses the fact that no death attributable to nuclear power plant operations will ever be "acceptable" in the sense that the NRC would regard it as a routine or permissible event. The policy and associated regulation deals with acceptable risks, not acceptable deaths.

The qualitative safety goals are expressed as follows:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following two quantitative objectives are to be employed in making a determination as to whether the qualitative goals have been achieved:

(1) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

(2) The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

The Commission noted that, in order to avoid the adverse consequences that could be associated with severe core damage accidents, it intends to continue to pursue a regulatory program that has as its objective providing reasonable assurance, with due consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. nuclear power plant. The Commission also proposed a general performance guideline for further staff examination in crafting a strategy for the safety goal implementation. The general guideline is as follows: consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

As of October 1986, the staff has developed an implementation framework for use of the safety goals in the regulatory decisionmaking process. Additional staff work is under way to develop specific implementation guidance in matters such as plant performance guidelines, indicators for operational performance, and guidelines for the conduct of cost-benefit analysis. It is expected that, as a separate matter, Commission approval for staff use of this specific safety goal implementation guidance will be sought about mid-1987.

Plant Personnel Fitness for Duty. The pervasive social concern with drug and alcohol abuse among persons working in safety-related capacities is shared by nuclear regulators and utilities. In a policy statement issued July 30, 1986, effective August 4, the Commission took cognizance of the problem. Noting that these abuses constitute a social, medical and safety problem affecting every segment of society, the Commission cited the need for the NRC to consider appropriate action to meet its responsibilities in this area, namely, to provide reasonable assurance that no person under the influence of any substance, legal or illegal, affecting his ability to perform his duties shall have access to the vital areas of a nuclear power plant. The Commission acknowledged the very extensive efforts within the industry to address the matter—in particular the implementation of the Edison Electric Institute's "EEI Guide to Effective Drug and Alcohol/Fitness for Duty Policy Development," the work of the Institute of Nuclear Power Operations (INPO) and of the Nuclear Utility Management and Human Resources Committee (NUMARC). In recognition of those industry initiatives, and in the hope of encouraging further progress, the Commission decided to refrain for 18 months from any new rulemaking on the subject.

Announcement on July 1, 1986, of new safety goals for nuclear power plant operations was accompanied by intensified inspection efforts at plants across the country. These photos show (above) NRC Director of Nuclear Reactor Regulation Harold R. Denton during a pre-licensing visit to Duke Power Company's Catawba plant at Lake Wylie, S.C., and (below) NRC Region I (Philadephia) inspector Kamal Manoly exiting a tank during an inspection of piping at the Seabrook plant, owned by the Public Service Company of the New Hampshire.

Exempted Nuclear Waste. Portions of the Low-Level Radioactive Waste Policy Amendments Act of 1985 call for the Commission to promulgate a plan and procedure for the timely identification of nuclear wastes "below regulatory concern." These materials may, by rulemaking, be exempted from required disposal in a licensed low-level waste disposal facility. In a policy statement dated August 28, 1986, effective October 27, the Commission put forth such a plan and procedure for expeditiously granting petitions to dispose of such waste by alternative means, without posing any undue risk to the public health and safety.

Medical Services for Accident Victims. The NRC requires, as a condition of nuclear power plant operation, that an off-site emergency plan be in place and that there be reasonable assurance that it can be implemented. One element of such a plan is pre-accident arrangement for medical services for persons who might be exposed to dangerous levels of off-site radiation. Heretofore, the Commission had required only the development and maintenance of a list of treatment facilities on which post-event ad hoc medical arrangements could be based. A remand on the question from the U.S. Court of Appeals for the District of Columbia led to issuance of a policy statement on September 12, 1986, effective September 17, giving direction to staff in dealing with medical arrangements. In brief, the NRC staff is required, in cooperation with the Federal Emergency Management Agency, to issue appropriate guidance to licensees, applicants, and to State and local authorities in arranging medical services for off-site radiation victims. Leaving the exact parameters of minimally necessary arrangements to the staff, the Commission expressed its belief that such arrangements should at least include (1) a list of local or regional medical facilities and transportation providers, with descriptions of their capabilities; (2) a good faith reasonable effort by licensees or local or State governments to facilitate or obtain written agreements with such medical and transportation services; (3) training for and provision of first aid for severely exposed individuals; and (4) a good faith reasonable effort by licensees or local or State governments to carry out drills and exercises, with simulated severely exposed individuals.

Relocation And Consolidation in 1987

The fulfillment of the long-sought objective of consolidating NRC offices and operations at a single site came within realistic reach in fiscal year 1986 when, in April, the Government Services Administration (GSA) entered into an agreement to purchase an 18-storey building in Rockville, Md. The purchase of the One White Flint North Building was consummated in November, in an agreement which includes an option for GSA to lease space in an adjacent office tower to be constructed at the same site by 1990. (Construction of the latter cannot begin, however, until approved by the Montgomery County (Md.) Council.) Purchase of the second building is envisioned at a later date (in the fifth year of a 20-year lease), contingent upon Congressional approval.

As a result of these actions—which are consistent with the general policy of housing Federal agencies in Government-owned facilities, for long term economies—the NRC Headquarters staff will begin the move in 1987 from its offices, currently situated in 10 different buildings in four widely dispersed locales in and around Washington D.C., into the single Rockville venue. The event marks the culmination of effort and planning over many years, since the inception of the agency in 1975.

Offices at the new location for the Chairman and other Commissioners, the Executive Director for Operations, and associated professional and support staff are expected to be ready for occupancy by the fall of 1987. The Office of Nuclear Reactor Regulation and elements of the Office of Administration will also be among the first NRC operations to relocate. The Commission Hearing Room, a fully equipped tiered amphitheater, is scheduled for completion by November 1987. With the construction and occupancy of the second building, a full consolidation of the NRC Headquarters staff of more than 2,500 personnel could be achieved by 1990, some 15 years from the creation of the NRC.

Sweeping NRC Reorganization in 1987

In October 1986, shortly after the end of fiscal year 1986, Chairman Zech notified all NRC personnel that he and the Executive Director for Operations (EDO), Victor Stello, Jr., were about to present the Commission, for its approval, plans for a major reorganization of the NRC. The new deployment of the agency's human and material resources will be responsive to the reality of NRC's ever-increasing engagement with operating facilities and the steadily decreasing volume of nuclear plant construction activity. In October 1974, when the NRC was established by Congressional Act, there were fewer than 50 plants in actual operation and over 70 plants under construction, with 75 new applications for construction permits pending. At the close of the current report period, there were but 19 plants still undergoing active review for operating licenses, and 101 plants licensed for operation.

The proposal was subsequently approved by the Commission and, in the following several months, the reorganization, with some revisions, was clarified for all concerned and set into motion, aiming for implementation by April 1987. The full implications of the restructuring of the NRC are properly covered in next year's 1987 annual report, but the far-reaching changes are of sufficient magnitude to warrant mention in this year's report.

Among the changes is the creation of five new offices, with a transfer of numerous functions from former offices and other organizational elements. Specifically, the Office of Inspection and Enforcement has been abolished and its

The U.S. Government's purchase, in November 1986, of a new building in Rockville, Md., marked the first step toward culmination of a 10-year effort to consolidate NRC offices at a single location. The photo shows the building during a visit by members of the Office of the Executive Director for Operations: (left-to-right) Catherine Turner, Elaine Mariggio, Donna Smith and Barbara Gabriel.

functions reassigned. The inspection activities of the agency will be divided between the Office of Nuclear Reactor Regulation and the Office of Nuclear Material Safety and Safeguards (NMSS), as appropriate. A newly created Office of Enforcement, reporting to the Deputy EDO for Regional Operations, will handle the NRC enforcement program. A new Office of Governmental and Public Affairs has been created at the Commission level, with activities of the former Offices of Congressional Affairs, of Public Affairs, of International Programs (IP) and of State Programs (SP) subsumed under it. (Some SP and IP functions have been assigned to NMSS.) The Office of Administration and Resources Management (ARM) combines and coordinates the tasks of the former Office of Administration, the Office of Resource Management and the Office of Information Resources. A new Office of Personnel has been put in place reporting directly to the EDO. And an Office of Special Projects has been set up, reporting to the EDO, with licensing and inspection authority pertaining specifically to nuclear facilities of the Tennessee Valley Authority and to the Comanche Peak (Tex.) nuclear power plants.

Besides these new structures, the scope of operations in the Office of Research and the Office for Analysis and Evaluation of Operating Data (AEOD) has been greatly expanded. The former will have new responsibilities with respect to rulemaking procedures, generic issue resolution (including Unresolved Safety Issues), and the review of full-scope probabilistic risk assessments. The latter will manage the review, analysis and evaluation of reactor plant performance, will continue to manage the NRC incident investigation program and be responsible for the incident response program and for management of the CRGR requirements (see "Backfit Policy and Practice," above); AEOD will also manage the NRC's Technical Training Center.

The rationale underlying these major moves was expressed by Chairman Zech in a message to all NRC personnel:

"Our mission is far different today than it was in 1975 when the NRC was created. We have shifted from evaluation of construction permit and operating license applications to the regulation of a maturing operational industry. As the plants presently in the final stages of construction are completed, we will have progressively less regulatory actions with large complex construction facilities and much more involvement with plant operations, maintenance, life extension and other operational issues. The new organization will focus NRC's major program offices on the day-to-day safety of operational facilities and make them more accountable for our safety programs."

Reactor Regulation

CHAPTER 2

The Office of Nuclear Reactor Regulation (NRR) is responsible for regulating operating nuclear reactors, for reviewing applications for construction permits and operating licenses for new reactors, and for issuing such permits and licenses after due consideration by the Advisory Committee on Reactor Safeguards, Atomic Safety and Licensing Boards and Appeal Boards, and the Commission. (See "The Licensing Process," on the page following.) These functions entail the resolution of both generic and specific issues with regard to public health and safety, the protection of the environment, and antitrust matters.

This chapter summarizes NRR activities during fiscal year 1986, under the following headings: Status of Licensing, Improving the Licensing Process, Human Factors, Unresolved Safety Issues, Safety Reviews, Protecting the Environment, and Antitrust Activities.

Status of Licensing

Applications for Operating Licenses, Construction Permits or Manufacturing Licenses

The NRC received no new applications for operating licenses, construction permits or manufacturing licenses during fiscal year 1986. Six utilities were issued six Low-Power Licenses (permitting fuel load at 0 percent power or low-power operation at 5 percent power) during fiscal year 1986. In addition, full-Power Operating Licenses were issued to five utilities.

Table 1 is a numerical summary of NRR activity in power reactor licensing during fiscal year 1986. Table 2 identifies the licensees and facilities licensed, with additional information.

At the close of the report period, the staff was reviewing applications for operating licenses for the 24 nuclear units still under construction; the schedules for these reviews are consistent with the projected plant completion dates. Some of these units have been delayed indefinitely. Two units in Michigan owned by Consumer Power Company—Midland Units 1 and 2—were cancelled during fiscal year 1986.

Licensing Actions for Operating Power Reactors

At the end of fiscal year 1986, 101 power reactors were licensed to operate in the United States. After operations begin, both routine activities and unexpected events at these facilities can result in a need for "licensing actions" on the part of the NRC. Routine post-licensing activities affecting the reactor operations include license amendment requests, public hearings, requests for exemption from regulations, new regulations requiring backfit modifications to operating reactors, orders for modification of a license, new generic activities, or review of information supplied by a licensee for the resolution of technical issues. These activities, and the growth in the number of operating reactors, result in a relatively large number of new actions and pending actions in the inventory. During fiscal year 1986, NRR completed about 2,650 licensing actions. About 70 percent of these actions were plant-specific and predominantly licensee-initiated. The balance were multi-plant actions which result from NRC-imposed requirements. The licensing action inventory has remained relatively constant in the last two fiscal years at a level of about 4,000 licensing actions under review.

Licensing Actions for Non-power Reactors

As of September 30, 1986, 55 non-power reactors licensed for operation by the NRC were in use for research, training and testing. Three applications for operating license renewals were being reviewed or were otherwise pending. During fiscal year 1986, the staff issued two license renewals for operation, two for fuel possession only, and approved the transfer of ownership of two additional licenses. The reviews of the three remaining renewal applications and the one conversion to possession only are scheduled to be completed during fiscal year 1987.

Table 3 summarizes fiscal year 1986 licensing actions for non-power reactors.

HEU/LEU Conversion. In February 1986, the NRC published a new regulation in the *Federal Register* that required the conversion of NRC-licensed non-power reactors that operate with highly enriched uranium (HEU) fuel to the use of low enriched uranium (LEU) fuel, excepting only those reactors serving a unique purpose. The Department of Energy will provide the fuel and other conversion-related support.

THE LICENSING PROCESS

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

As soon as an initial application is accepted, or "docketed," by the NRC, a notice of that fact is published in the *Federal Register*, and copies of the application are furnished to appropriate State and local authorities and to a local public document room (LPDR) established in the vicinity of the proposed site, as well as to the NRC public document room in Washington, D.C. At the same time, a notice of a public hearing is published in the *Federal Register* and local newspapers which provides 30 days for members of the public to petition to intervene in the proceeding. Such petitions are entertained and adjudicated by the ASLB appointed to the case, with rights of appeal by the petitioner to the ASLAB.

The NRC staff's safety, safeguards, environmental and antitrust reviews proceed in parallel. With the guidance of the Standard Format (Regulatory Guide 1.70), the applicant for a construction permit lays out the proposed nuclear plant design in a Preliminary Safety Analysis Report (PSAR). If and when this report has been made sufficiently complete to warrant review, the application is docketed and NRC staff evaluations begin. Even prior to submission of the report, NRC staff conducts a substantive review and inspection of the applicant's quality assurance program covering design and procurement. The safety review is performed by NRC staff in accordance with the Standard Review Plan for Light-Water-Cooled Reactors, initially published in 1975 and updated periodically. This plan sets forth the acceptance criteria used in evaluating the various systems, components and structures important to safety and in assessing the proposed site; it also describes the procedures to be used in performing the safety review.

The NRC staff examines the applicant's PSAR to determine whether the plant design is safe and consistent with NRC rules and regulations; whether valid methods of calculation were employed and accurately carried out; whether the applicant has conducted his analysis and evaluation in sufficient depth and breadth to support staff approval with respect to safety. When the staff is satisfied that the acceptance criteria of the Standard Review Plan have been met by the applicant's preliminary report, a Safety Evaluation Report is prepared by the staff which summarizes the results of its review regarding the anticipated effects of the proposed facility on public health and safety.

Following publication of the staff Safety Evaluation Report, the ACRS completes its review and meets with staff and applicant. The ACRS then prepares a letter report to the Chairman of the NRC presenting the results of its independent evaluation and recommending whether or not a construction permit should be issued. The staff issues a supplement to the Safety Evaluation Report incorporating any changes or actions adopted as a result of ACRS recommendations. A public hearing can then be held, generally in a community near the proposed facility site, on safety aspects of the licensing decision.

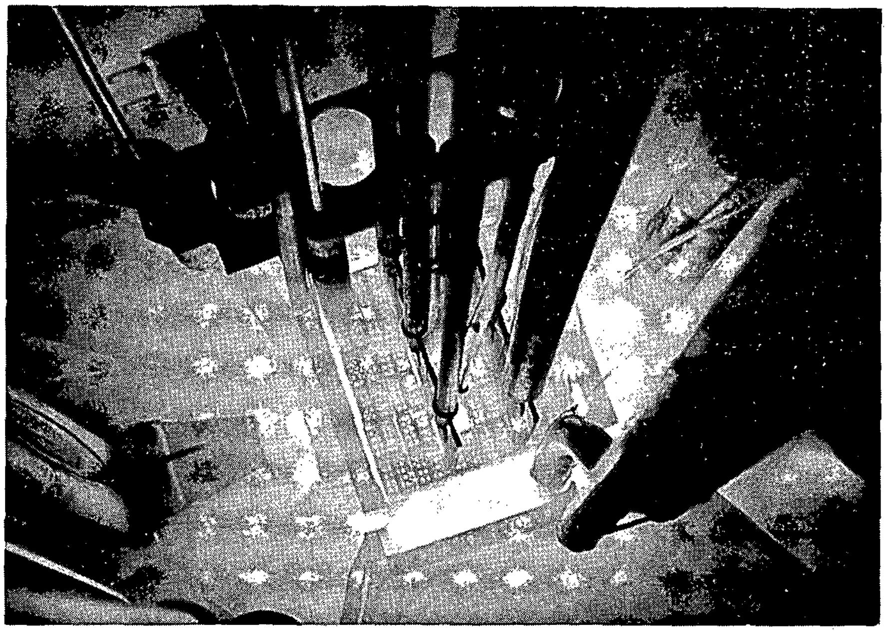
In appropriate cases, the NRC may grant a Limited Work Authorization to an applicant in advance of the final decision on the construction permit in order to allow certain work to begin at the site, saving as much as seven months time. The authorization will not be given, however, until NRC staff has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

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

The antitrust reviews of license applications are carried out by the NRC and the Attorney General in advance of, or concurrent with, other licensing reviews. If an antitrust hearing is required, it is held separately from those on safety and environmental aspects.

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

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



Shown here is the core of the Pulstar Research Reactor at the Buffalo Materials Research Center, State University of New York (SUNY) in Buffalo, N.Y. The reactor was licensed by the Atomic Energy Commission in the early 1960s and has operated at levels up to two megawatts (thermal) since that time. Some 55 non-power reactors were in operation at the end of fiscal year 1986, under license by the NRC.

At the end of fiscal year 1986, two licensees had initiated safety analyses in connection with conversion, and plans were being made to provide support for five others, commencing in fiscal year 1987. In addition, technical reviews have been initiated to evaluate and qualify the newly developed LEU fuels for use in licensed non-power reactors. It is expected that fuel qualification will be completed during fiscal year 1987.

Petition for Rulemaking Concerning Use of Graphite. The NRC received a petition during the report period to promulgate a regulation that would affect almost all of the licensed reactors that have graphite in and around their cores. The petition hypothesizes that the graphite could store sufficient energy from neutron irradiation to constitute a fire hazard.

The NRC published a notice in the *Federal Register* on September 3, 1986, requesting comments, and intends to analyze and evaluate the potential risks to the public resulting from the use of such graphite. While only one licensed power reactor contains graphite, most non-power reactors use graphite as a neutron moderator or reflector.

Special Cases

Comanche Peak. As construction of the Comanche Peak Unit 1 (Tex.) nuclear power plant neared completion, a number of complex and interrelated issues remained to be resolved before the staff could make a licensing decision. The source and nature of these issues—the most serious being allegations regarding construction adequacy and quality assurance—are described on page 10 of the *1985 NRC Annual Report.*

In response to these concerns, the applicant submitted a program plan for their resolution. The NRC staff has completed its review and issued Supplemental Safety Evaluation Report No. 13 approving the applicants' plan. The plan commits the licensee, Texas Utilities, to programs intended to demonstrate the adequacy of plant design and construction. The detailed re-verification of design and reinspection of construction effort was under way at the close of the report period. The NRC staff is currently monitoring all phases of implementation. Texas Utilities has indicated that the last review effort would be complete by August 1987. Corrective action may extend beyond that date.

Diablo Canyon. The full-power license for the Diablo Canyon Unit 1 (Cal.) nuclear power plant contained a condition requiring the licensee to re-evaluate the seismic design basis for the plant by considering information and methodologies that have become available since 1979. The licensee proposed a long-term seismic program comprising three phases: Phase I consisting of identifying the issues which needed to be addressed, Phase II of laying out the types of investigations and analyses required to address the issues identified in Phase I, and Phase III of carrying out the investigations and analyses. The licensee's program includes the generation of new data from field studies, seismic seminars at and near the plant site, offshore seismic reflection and refraction measurements, and other sources as they become available.

Major efforts since the approval of the scoping study (Phase II) consisted of:

(1) An inspection of the Diablo Canyon Nuclear Plant site by NRC staff and consultants to become acquainted with the general seismic design of the plant and the locations of existing and proposed additions in seismic instrumentation. The staff was also briefed

Table 1. Power Reactor Licensing—FY 1986

Low-Power Operating Licenses issued	6
Full-Power Operating Licenses issued	5
Safety Evaluation Reports issued	1
Draft Environmental Impact Statements issued	1
Final Environmental Impact Statements issued	1
Operating Licenses under Review	24
Applications Cancelled	Midland 1 and 2
Construction Permits issued	0
Construction Permits under Review	0
Manufacturing Licenses issued	0
Manufacturing Licenses under Review	0

on the progress made in the theoretical modeling of the plant site and the plant structures. The theoretical modeling studies are an integral part of an overall deterministic and probabilistic analysis of the site and site structures designed to establish their behavior during earthquake excitation.

(2) A field trip jointly organized by the Diablo Canyon licensee and NRC consultants from the University of Nevada/Reno to review the findings of an on-going field study. Investigations of the geologic character of the Diablo Canyon site region and its fault pattern were conducted, with special attention to the Hosgri fault.

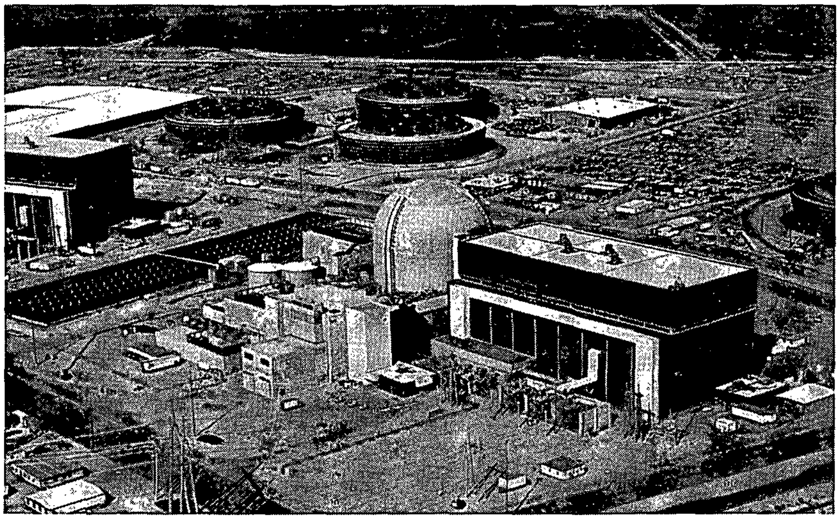
The staff and its consultants—including the U.S. Geological Survey—are continuing their independent investigations and are closely monitoring the licensee's Phase III program, which is expected to be completed by mid-1988.

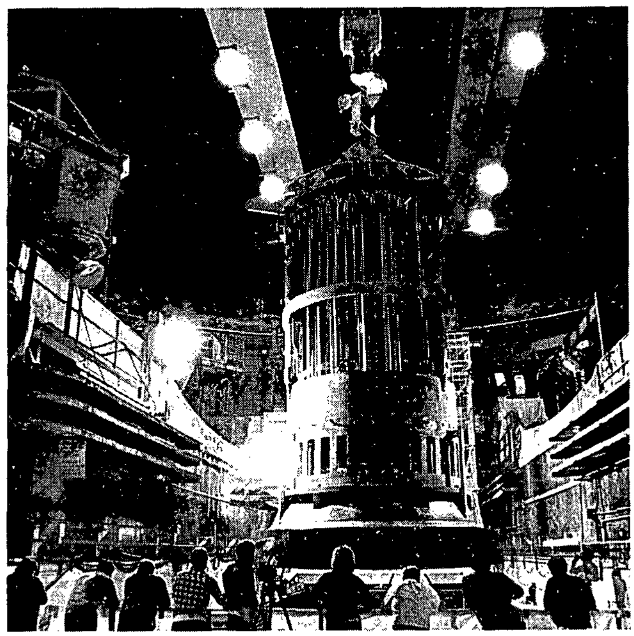
Davis-Besse. The Davis-Besse plant near Toledo, Ohio, has remained in cold shutdown status since June 9, 1985, when a loss-of-feedwater event, followed by a number of equipment failures, took place. Fact-finding by the staff was completed early in 1986, and extensive corrective measures and plant modifications by the licensee continued throughout 1986. In June, the staff completed its safety evaluation and issued its safety evaluation report, NUREG-1177. The staff and the licensee met with the Advisory Committee on Reactor Safeguards (ACRS) and the Commission several times in 1985 and 1986, pursuant to the projected restart of the Davis-Besse plant in December 1986.

Palo Verde. The Commission granted a low-power license for Palo Verde Unit 1 (Ariz.) in December 1984, a full-power license for the same unit in June 1985, a low-power license for Unit 2 in December 1985, and a full-power license for Unit 2 in April 1986. Palo Verde Unit 3 is tentatively scheduled for low-power licensing in early 1987.

Arizona Public Service Company (APS), the lead licensee for the Arizona Nuclear Power Project (ANPP) which includes these three units, faces an unusually challenging task in constructing and operating three first-of-a-kind (CESSAR System 80) nuclear units. The Palo Verde project is the licensee's first nuclear venture. In spite of some initial design problems APS was able to complete construction and obtain a full-power operating license for Unit 1 in June 1985, only nine years after receipt of its construction permit. Unit 2 was approved for a full-power license in April 1986, and the current schedule calls for fuel loading at Unit 3 early in 1987. The site, after licensing of Unit 3, will have the largest nuclear generating capacity in the United States.

Perry Unit 1. On January 31, 1986, at 11:46 a.m. EST, an earthquake of magnitude 5.0 (Richter) occurred about 10 miles south of the plant site in northeastern Ohio. The U.S. Geological Survey reported that the earthquake epicenter was at 41.65 °N and 81.16 °W. There were reports of damage near the epicenter including cracked walls, falling roof tiles and shattered windows. The earthquake, which





The Arizona Public Service Company's Palo Verde nuclear power project received a full-power NRC license for operation of Unit 2 of the three-unit station at Wintersburg, Ariz., in April 1986. Unit 1 of the complex received a full-power license in June 1985, and the third unit was scheduled for low-power licensing early in 1987. These photos show (above) Unit 1, (at right) the lowering of that unit's upper guide structure and control element drive assembly into position atop the reactor head, and (below) the station's Water Reclamation Facility, using treated waste water from seven Phoenix-area communities for reactor cooling.

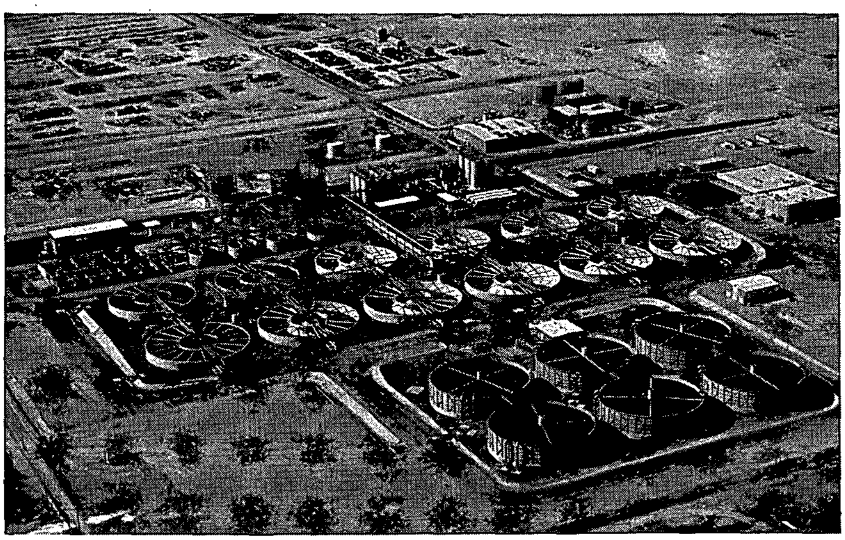


Table 2. Licenses Issued in FY 1986 for Operation of Nuclear Power Plants

Applicant	*Facility*	*Low-Power License*	*Full-Power License*	*Location*
Gulf States Utilities	River Bend	11/20/85	24 Miles NNW of	Baton Rouge, LA
Northeast Nuclear Energy Co.	Millstone 3	11/25/85	01/31/86	3 Miles WSW of New London, CT
Arizona Public Service	Palo Verde 2	12/09/85	04/24/86	36 Miles West of Phoenix, AZ
Duke Power	Catawba 2	02/24/86	05/15/86	6 Miles NNW of Rock Hills, SC
Cleveland Electric Illuminating Co.	Perry 1	03/18/86		35 Miles NE of Cleveland, OH
Public Service Electric & Gas	Hope Creek	04/11/86	07/25/86	18 Miles South of Wilmington, DE
Illinois Power Company	Clinton	09/29/86		22 Miles NE of Decatur, IL

occurred within one week of the date the Cleveland Electric Illuminating Company was scheduled to receive a low-power operating license for Perry Unit 1, was felt as far away as Washington, D.C., about 300 miles from the epicenter.

Demands on staff resources were extensive in evaluating the event and responding to concerns from certain members of Congress, local and State officials, the ACRS, and intervenor groups, and also for supporting judicial proceedings. In part, these demands resulted from the unique character of the earthquake, i.e., its high frequencies, short time duration, and low energy content. Analyses by the staff and its consultant, as well as by CEI, demonstrated that the Perry plant's seismic design has adequate safety margins to accommodate the recorded 1986 Ohio earthquake, even though the design spectra were exceeded at around 20 Hz.

Several generic questions have arisen not only from this particular earthquake but also from other small earthquakes near other nuclear power plants. The questions related to the nature and effects of high frequency ground motion on structures and components were under consideration by the NRR staff at the end of the report period, with a view to defining the extent to which currently existing programs address the adequacy of plant design criteria for accommodating earthquakes and determining whether new programs are needed to resolve this issue.

In view of these findings, it was concluded that Perry Unit 1 could be licensed to operate at levels up to 5 percent of rated thermal power, and the licensee was, therefore, issued an operating license (NPF-45) on March 18, 1986, six weeks later than initially scheduled.

There were, however, several matters, identified during the NRC staff's preliminary analyses, which needed to be completed before Perry Unit 1 could be authorized to operate at levels above 5 percent of rated thermal power. This confirmatory work was completed by the licensee in August 1986; the NRC staff's safety evaluation report on that confirmatory work is documented in NUREG-0887, Supplement No. 10, dated September 1986. (The facility was approved for full power operation on November 13, 1986, after the close of the report period.)

Rancho Seco. On December 26, 1985, the Rancho Seco (Cal.) nuclear power plant underwent a failure of the integrated control system, leading to a rapid overcooling transient. An NRC fact-finding team immediately went to the site, investigated the event in depth, and documented its conclusions in NUREG-1195. Numerous meetings were held between the licensee for the plant, Sacramento Municipal Utility District (SMUD), and NRC staff to discuss issues of concern related not only to the event itself, but also to Rancho Seco's operating problems over the past several years. Subsequently SMUD developed a comprehensive Plant Performance and Management Program to address the range of issues identified and explored with the NRC staff. The latter initiated technical reviews in a number of areas

Table 3. Licensing Actions for Non-power Reactors—FY 1986

(OL = operating license; CP = construction permit)

Non-power reactor operating licenses	55
OL renewals issued for operation	2
OL renewals issued for possession only	2
Licenses amended to possession only	1
Licenses terminated	2
CPs terminated	1
Orders issued to decommission/dismantle	2
Facilities planning decommissioning/dismantle	4
OL renewals under review	3

(e.g., system review and test program and detailed control room design review) and conducted numerous inspections of on-site activities.

At the close of the report period, the Rancho Seco facility remained in cold shutdown with maintenance, modifications, reviews, evaluations and other corrective actions under way by SMUD. The NRC staff is continuing its review of the system review and test program and other selected elements of the SMUD corrective action plan, as they are submitted by the utility. SMUD's estimate for facility restart was May of 1987.

San Onofre Unit 1. On November 21, 1985, the San Onofre Unit 1 nuclear power plant near San Clemente, Cal., underwent a partial loss of in-plant electrical power which resulted in a severe water hammer in the feedwater system. The water hammer damaged plant equipment and caused a steam and feedwater leak. Subsequently, the plant operators were able to bring the plant to a stable, cold shutdown condition without an abnormal release of radioactivity. An NRC fact-finding team was immediately sent to the site. The team's conclusion, published in NUREG-1190, was that the most significant aspect of the event was the fact that five safety-related check valves in the feedwater system had degraded to the point of inoperability in less than a year, and the condition went undetected throughout that period. The inoperable check valves permitted the severe water hammer which challenged the integrity of the plant's feedwater system. The NRC has instituted a program to address both the plant-specific and generic aspects of the event. Numerous inspections and technical reviews were carried out. After the licensee, Southern California Edison Company, corrected the check-valve problems and made other significant improvements to plant equipment, the NRC allowed the plant to resume operation in July 1986. (See Chapter 4.)

Shoreham. A low-power license was issued to Long Island Lighting Company (LILCO) on July 3, 1985, authorizing the operation of the Shoreham nuclear power plant up to 5 percent of rated power. Since that issuance, LILCO completed the low-power testing program and, on August 27, 1986, the Shoreham plant turbine generator was synchronized with the grid and produced commercial electricity for the first time. While the adequacy of the off-site Emergency Plan remained in question, the Shoreham plant was in all physical respects ready for an authorization to operate at full power.

The licensee appealed the Licensing Board's finding that full-power operation of the plant was not permissible without State and county participation in the execution of the emergency plan, a commitment which both of those governmental entities had refused to make. The Appeal Board agreed with the Licensing Board that, contrary to the licensee's claim, Federal law did not pre-empt State law in these circumstances. On the licensee's further argument that, in any event, the State and county would respond in good faith to an actual emergency, and that it was thus not necessary to rely on non-governmental personnel to execute the provisions of the plan, the Appeal Board upheld the Licensing Board's determination that the prospect of such an uncoordinated and unrehearsed response was insufficient to provide the requisite assurance of public protection. Subsequently, the Commission itself determined that the State and local governments would exert their best effort in an emergency, and sent the matter back for further evidentiary hearing and evaluation on that basis. Hearings are expected to continue through fiscal year 1987. (See Chapter 12.)

Improving The Licensing Process

Standardization

The Commission strongly endorses regulatory policies which will encourage the industry to pursue standardization of power reactor designs. It is the expectation that standardized designs will benefit public health and safety in a number of ways—concentrating industry resources on common approaches to design problems that will have wide application, stimulating adoption of sound construction practices and quality assurance, fostering constantly improving maintenance and operation procedures, and permitting more efficient and effective licensing and inspection procsses.

In this regard, the staff has proposed certain revisions to the Commission's 1978 Standardization Policy Statement, reflecting the experience the agency has acquired in implementing the statement, the applicable provisions of the Commission's Severe Accident Policy Statement and of the proposed standardization legislation, and the current views of the Commission and industry on standardization. The focus of the proposed revised policy is the Reference System Design Certification, a regulatory instrument which would fulfill the ultimate goal of licensing the construction of plants of certified designs on pre-approved sites. At the close of the report period, the staff was developing implementation guidance for the proposed revised Standardization Policy Statement. A document on the "Implementation of NRC Policy on Nuclear Power Plant Standardization" (NUREG-1225), and the Standardization Policy Statement were expected to be issued for comment in the first quarter of fiscal year 1987.

EPRI Advanced Light Water Reactor Program. The NRC is continuing to work with the Electric Power Research Institute (EPRI) on an advanced LWR standard plant program. In July 1986, EPRI submitted for NRC review the first of a 13 chapter "requirements document," treating performance specifications for a total plant in the range of 500 MWe-to-1350 MWe power output. The NRC anticipates completion of its review of all 13 chapters in 1989.

GESSAR II. On September 22, 1986, the NRC issued Amendment No. 2 to the Final Design Approval (FDA) for General Electric Company's GESSAR II Nuclear Island design. Issuance of Amendment No. 2 is a major milestone in the NRC's standardization and licensing process in that GESSAR II is the first design that successfully complies with the four licensing requirements for future plant designs identified in the Commission's Severe Accident Policy Statement. Amendment No. 2 removes the constraints on issuing construction permits and operating licenses to applicants who reference the GESSAR II design.

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

The most significant technical differences between the ABWR and the updated GESSAR II design include:

(1) A containment building similar to the Mark III design, but with a cover over the suppression pool, so that the amount of equipment exposed to the primary containment environment is minimized. The containment is also inerted.

(2) All external reactor coolant recirculation piping has been eliminated by incorporating the recirculation pumps into the reactor vessel.

(3) Fine motion control rod drives.

(4) Digital/Solid state controls.

(5) Multiplexing of control signals and the use of fiber optics.

(6) An improved reactor building design.

In June and August of 1986, GE representatives met with members of the NRC staff to discuss the staff review process for the ABWR. These discussions, continuing into fiscal year 1987, are aimed at developing an understanding of the review process, procedures and criteria for the ABWR Final Design Approval and Certification. The ABWR technical review is scheduled to begin in fiscal year 1988.

CESSAR-F SYSTEM 80. During fiscal year 1986, the staff continued its review of Combustion Engineering's application to amend the Final Design Approval (FDA) for their CESSAR-F System 80 Nuclear Steam Supply System design so as to permit referencing in new construction permits and operating license applications. The CESSAR-F FDA, issued on December 21, 1983, applied only to those plants whose construction permit applications referenced the CESSAR Preliminary Design Approval (PDA) at the construction permit stage of the licensing process. The staff's review of Combustion Engineering's amendment request was continuing at the close of the report period. A decision is expected in the second quarter of fiscal year 1987.

Westinghouse RESAR-SP 90. The staff continued its review of the Westinghouse Electric Corporation application for the PDA for its RESAR-SP/90 Nuclear Power Block design, docketed on May 19, 1984. A decision on the issuance of the PDA is scheduled for fiscal year 1987.

Westinghouse intends to pursue an FDA and a Design Certification for its RESAR-SP/90 design.

Integrated Implementation Schedules

The licenses of three operating power plants incorporate formal scheduling processes for the implementation of new and existing requirements according to their relative priorities. The staff is considering similar provisions for a number of operating facilities.

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

The staff has also solicited feedback and comment from industry groups and is presently working with an Atomic Industrial Forum (AIF) Subcommittee on Integrated Implementation Schedules. The staff expects to transmit to the Commission a proposed Policy Statement on this subject in early fiscal year 1987.

Backfitting

On September 20, 1985, the Commission published the final backfitting rule in the *Federal Register.* The staff modified NRC Manual Chapter 0514 to incorporate the provisions of the new rule. The updated Manual Chapter was issued in February, 1986, containing the following modifications: (1) information requests other than those pertaining to compliance issues will be evaluated to ensure that the burden to be imposed on respondents is justified with respect to the potential safety significance, and (2) a regulatory analysis will be performed after an immediate imposition of a backfitting requirement to document its safety significance (cost considerations of this analysis are to be included only insofar as they contribute to selecting among alternatives). Each Headquarters and Regional Office has developed implementing procedures for the final Manual Chapter.

After issuance of the final Manual Chapter, the Office of the Executive Director for Operations conducted briefings for industry representatives and training seminars for the NRC staff at Headquarters and in the Regions on the changes in the Backfit Rule. The training for the NRC staff covered the considerations necessary in identification of backfits and the process for evaluating backfits consistent with Manual Chapter 0514. The briefings included presentations by industry representatives giving the industry's view of the new Backfit Rule.

During fiscal year 1986, the staff considered 23 licensee- and seven staff-identified backfits. Of these issues, 19 have been resolved and 11 are under review. Of the licensee-identified backfits, 17 were found not to be backfits within the meaning of the Backfit Rule. The remaining six are under NRC staff review.

Priorities of Generic Safety Issues

The NRC continued to use the methodology set out in the *1982 NRC Annual Report* for determining the priority of generic safety issues. In December 1983, a comprehensive list of the issues subjected to this method was published in "A Prioritization of Generic Safety Issues" (NUREG-0933) and is updated semi-annually, with supplements in June and December. This list includes items from the TMI Action Plan (NUREG-0660) and unresolved Safety Issues (USIs), which are discussed in detail later in this chapter. The results of the NRC's continuing effort to identify significant unresolved safety issues will be included in future Supplements to NUREG-0933.

The staff identified 49 new generic issues in fiscal year 1986, 34 of which came out of the event at Davis-Besse (Ohio) in June 1985. Priorities for 28 issues listed in Table 4 were established in fiscal year 1986. NRC resolved eight safety issues other than USIs; these are listed in Table 5. A revision to the Human Factor Program Plan reduced the total number of human factors issues scheduled for resolution. Table 6 contains the schedules for resolution of all issues, excluding USIs.

Technical Specification Improvement

The Office of Nuclear Reactor Regulation initiated a Technical Specification Improvement Project (TSIP), in January 1985, to study the Technical Specifications issued as part of every power reactor operating license and to make recommendations to ensure that they are focused on important operational safety matters. Two major recommendations have come out of both the NRC and industry studies:

(1) The NRC should adopt the criteria for defining the scope of Technical Specifications proposed in the industry and TSIP reports. Those criteria should then be used by the NRC and each of the Industry Owners Groups to completely rewrite and streamline the existing Standard Technical Specifications (STS). This process would result in the transfer of many requirements from control by Technical Specifications to control by other mechanisms (e.g., the FSAR and 10 CFR 50.59, Operating Procedures, the QA Plan, or Fire Protection Plan) which do not require a license amendment or prior NRC approval when changes are made. The new STS would give greater emphasis to

Table 4. Issues Prioritized in FY 1986

Number	*Title*	*Priority*
21	Vibration Qualification of Equipment	DROP
30	Potential Generator Missiles—Generator Rotor Retaining Rings	DROP
74	Reactor Coolant Activity Limits for Operating Reactors	DROP
97	PWR Reactor Cavity Uncontrolled Exposure	Covered in III.D.3.1
111	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	Licensing Issue
112	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	Regulatory Impact
114	Seismic-Induced Relay Chatter	Covered in USI A-46
115	Reliability of Westinghouse Solid State Protection System	HIGH
122.1.a	Common Mode Failure of Isolation Valves in Closed Position	HIGH
122.1.b	Recovery of Auxiliary Feedwater	MEDIUM
122.1.c	Interruption of Auxiliary Feedwater Flow	HIGH
122.2	Initiating Feed-and-Bleed	HIGH
122.3	Physical Security System Constraints	LOW
124	Auxiliary Feedwater System Reliability	Nearly Resolved
125.I.2.a	PORV Reliability—Test Program	Covered in Issue 70
125.I.2.b	PORV Reliability—Surveillance	Covered in Issue 70
125.I.2.c	Auto Block Valve Closure	DROP
125.I.2.d	Equipment Qualification for Feed-and-Bleed Environment	Covered in USI A-45
125.II.3	Review Steam/Feed Line Break Mitigation Systems for Single Failure	DROP
125.II.4	OTSG Dryout and Reflood Effects	DROP
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During Line Break	HIGH

Number	*Title*	*Priority*
125.II.9	Enhance Feed and Bleed Capability	Covered in USI A-45
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	LOW
133	Update Policy Statement on Nuclear Plant Staff Working Hours	Licensing Issue
134	Rule on Degree and Experience Requirements	HIGH
C-4	Statistical Methods for ECCS Analysis	Regulatory Impact
C-5	Decay Heat Update	Regulatory Impact
C-6	LOCA Heat Sources	Regulatory Impact

human factors and greater clarity to the overall text. The STS would also incorporate improvements to the Bases Section for Technical Specifications.

(2) A parallel program of short-term improvements in both the scope and substance of the existing TS should be initiated, in addition to the new set of STS identified above.

An Executive Summary of the NRC's Technical Specification Improvement Program Plan was issued on April 1, 1986. The NRC is preparing a proposed Policy Statement to implement the first of the major TSIP/industry recommendations. It will serve as a basis for the NRC and industry to implement voluntary Technical Specifications improvements consistent with the recommendations of both the TSIP and the industry.

The NRC and industry have under way a parallel program of short term improvements to both the scope and substance of the existing Technical Specifications. (The need for many of these types of improvements was identified in NUREG-1024.) Much of this effort should be completed within the next year. The improvements will carry over to the STS.

Advanced Reactors

NRC interaction with the Department of Energy and its contractors continued on the review of conceptual designs for two advanced Liquid Metal Reactors (LMRs) and one advanced Modular High Temperature Gas-Cooled Reactor (MHTGR). The exchanges took place in a series of meetings to familiarize NRC staff with these concepts and to discuss key safety and licensing issues associated with the designs. In September 1986, a Preliminary Safety Information Document (PSID) on the MHTGR concept was submitted for NRC review. PSIDs for the two LMRs were submitted in November of 1986. Completion of the review of the PSIDs is scheduled for January 1988 for the MHTGR and April 1988 for the LMRs.

In July 1986, the Commission issued a final Policy Statement on Advanced Reactors. It encourages early communication between the NRC staff and advanced reactor designers and lists those safety characteristics which the Commission believes should be considered in advanced designs.

Human Factors

Staffing and Qualifications

The staff conducted a survey (SECY-86-231) of engineering expertise on shift at operating reactor plants and plants nearing operational status to assess industry response to the Policy Statement on "Engineering Expertise on Shift." The Policy Statement, issued by the Commission on September 25, 1985, identified two options acceptable to the Commission. They are: Option 1, combined Senior Reactor Operator (SRO)/Shift Technical Advisor (STA)—an individual assigned to each operating shift crew as one of the SROs who holds a baccalaureate degree in engineering, engineering technology, physical science, or a Professional Engineer's license; and Option 2, Separate STA—an individual assigned to each shift who holds a baccalaureate degree or equivalent in a scientific or engineering discipline. The staff concluded that the policy statement has effectively articulated acceptable methods for providing engineering expertise on shift and that voluntary conformance to this policy has been achieved in all but a few cases. The staff also concluded that the industry is moving toward providing degreed SRO/STAs on shift.

The Commission issued an Advance Notice of Proposed Rulemaking (ANPRM) on degree requirements for senior

Table 5. Generic Safety Issues Resolved in FY 1986

Number	*Title*
3	Setpoint Drift in Instrumentation
14	PWR Pipe Cracks
36	Loss of Service Water
61	SRV Discharge Line Break Inside the Wetwell Airspace of BWR Mark I and Mark II Containments
I.C.9	Long-Term Plan for Upgrading of Procedures
III.D.3.1	Radiation Protection Plans
HF1.2	Engineering Expertise on Shift
HF1.3	Guidance on Limits and Conditions of Shift Work

operators on May 31, 1986. The proposed rule would require, after January 1, 1991, that applicants for licenses as Senior Operators of nuclear power plants hold a baccalaureate degree in engineering or a related science from an accredited institution. The public comment period on the ANPRM expired September 29, 1986.

The staff also completed Revision 2 to Regulatory Guide 1.114, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Plant," and a conforming change to Standard Review Plan Section 13.1.2, "Operating Organization." The staff submitted the revised Regulatory Guide and Standard Review Plan for publication in the *Federal Register* to obtain public comment.

Management and Organization

The NRC has concluded that, since the management and organization of utilities licensed to operate nuclear power plants must be a responsibility of the licensee, it is not an appropriate subject for further regulatory requirements. But the NRC is developing improved methods of monitoring licensee management performance, in order to give early warning of management problems.

In keeping with this policy, the NRC has terminated work intended to provide the technical basis for formulating requirements in the field of licensee management and organization, and has instead undertaken (1) the development of licensee performance indicators, (2) improvements in the NRC's Systematic Appraisal of Licensee Performance (SALP) program (see Chapter 8), and (3) more focussed attention to particular licensees where management performance has been found to be weak.

Human Performance

A new element was added to the NRC Human Factors Program in March 1986, called "Human Performance." The point of introducing this element is to develop and implement better methods for gaining experience feedback on reactor operational events involving human error.

The first accomplishment under the new program has been the development of a procedure which the Incident Investigation Teams can use in the course of investigating and documenting the causative factors of human error by reactor operations and maintenance personnel. Training in the new investigative procedures was made a part of the first training course, given in July 1986, for members of Incident Investigation Teams.

Procedures

During the report period, the Division of Human Factors Technology of NRR (the Division of Human Factors Safety prior to the reorganization of NRR in November 1985) has been conducting oversight reviews of NRC licensing activities to ensure that current NRC policy related to procedures is applied consistently to all plants. Reports on the staff oversight reviews for licensing activities are issued to the Division Director on a quarterly basis.

The NRC is continuing to implement its long-term program for upgrading emergency operating procedures (EOPs). The program was initiated shortly after the TMI accident. The objectives of the program are to improve the technical content of EOPs and to enhance their usefulness by the application of human factors principles in making the improvements. Owners Groups, representing the four nuclear power plant vendors, have satisfactorily re-analyzed

Table 6. Generic Safety Issues Scheduled for Resolution

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
23	Reactor Coolant Pump Seal Failures	HIGH	02/88
29	Bolting Degradation or Failures in Nuclear Power Plants	HIGH	01/90
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	NEARLY RESOLVED	09/87
49	Interlocks and LCOs for Class 1E Tie Breakers	MEDIUM	09/87
51	Proposed Requirements for Improving Reliability of Open Cycle Service Water Systems	MEDIUM	08/88
65	Component Cooling Water System Failures	HIGH	02/88
66	Steam Generator Requirements	NEARLY RESOLVED	12/86
67.7	Steam Generator Staff Actions-Eddy Current Tests	MEDIUM	TBD
68	Postulated Loss of AFWS Resulting from Turbine-Driven AFW Pump Steam Supply Line Rupture	HIGH	TBD
70	PORV and Block Valve Reliability	MEDIUM	12/87
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	NEARLY RESOLVED	07/87
77	Flooding of Safety Equipment Compartments by Back-Flow Through Floor Drains	HIGH	10/87
79	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	06/87
82	Beyond Design Bases Accidents in Spent Fuel Pools	MEDIUM	08/87
83	Control Room Habitability	NEARLY RESOLVED	05/87
84	CE PORVs	NEARLY RESOLVED	TBD
86	Long Range Plan for Dealing With Stress Corrosion Cracking in BWR Piping	NEARLY RESOLVED	01/87
87	Failure of HPCI Steam Line Without Isolation	HIGH	06/88

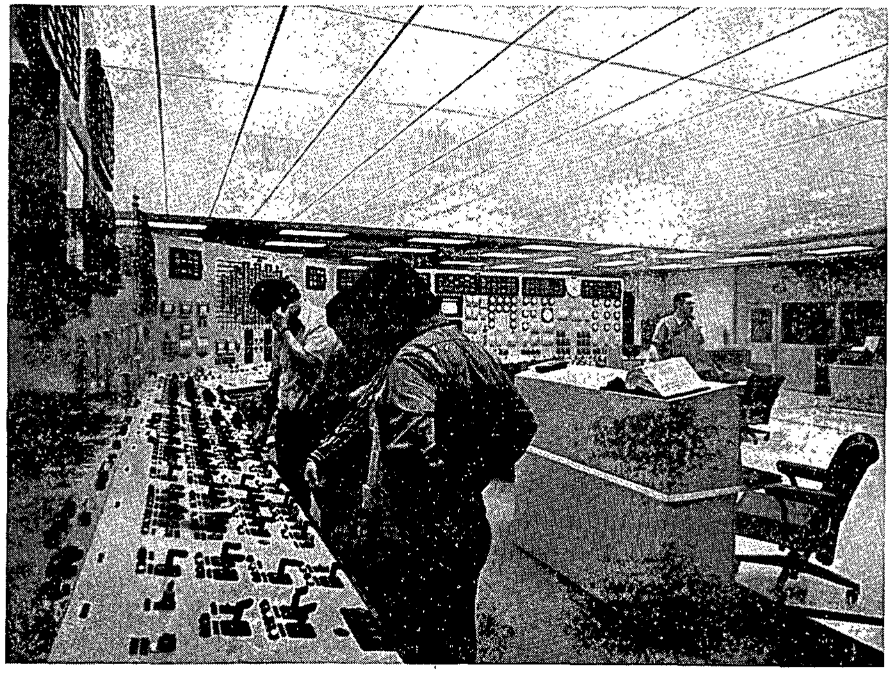
Table 6. Generic Safety Issues Scheduled for Resolution
(continued)

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
91	Main Crankshaft Failure In Transamerica DeLaval	NEARLY RESOLVED	12/86
93	Steam Binding of Auxiliary Feedwater Pumps	HIGH	09/87
94	Additional Low-Temperature Overpressure Protection For Light Water Reactors	HIGH	07/88
99	RCS/RHR Suction Line Interlocks on PWRs	HIGH	10/87
101	BWR Water Level Redundancy	HIGH	01/88
102	Human Error in Events Involving Wrong Unit or Wrong Train	NEARLY RESOLVED	10/87
103	Design For Probable Maximum Precipitation	NEARLY RESOLVED	09/87
105	Interfacing Systems LOCA at BWRs	HIGH	04/88
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	HIGH	03/88
121	Hydrogen Control for Large, Dry PWR Containments	HIGH	01/87
122.1a	Common Mode Failure of Isolation Valves in Closed Position	HIGH	TBD
122.1b	Recovery of Auxiliary Feedwater	MEDIUM	TBD
122.1c	Interruption of Auxiliary Feedwater Flow	HIGH	TBD
122.2	Initiating Feed-and-Bleed	HIGH	06/87
124	Auxiliary Feedwater System Reliability	NEARLY RESOLVED	TBD
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	HIGH	TBD
134	Rule on Degree and Experience Requirments for Senior Operators	HIGH	02/87
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	MEDIUM	09/88

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
A-30	Adequacy of Safety Related DC Power Supplies	HIGH	09/87
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	MEDIUM	08/88
B-6	Loads, Load Combinations, Stress Limits	HIGH	TBD
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	TBD
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	03/87
B-56	Diesel Reliability	HIGH	11/87
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	10/88
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	10/87
C-8	Main Steam Line Isolation Valve Leakage Control Systems	HIGH	06/88
I.A.2.6 (1)	Long-Term Upgrading of Training and Qualifications—Revise Regulatory Guide 1.8	HIGH	12/86
I.A.3.3	Requirement for Operator Fitness	HIGH	TBD
I.A.4.2 (1)	Research on Training Simulators	HIGH	12/86
I.A.4.2 (4)	Review Simulators for Conformance	HIGH	06/87
I.B.1.1	Organization and Management Long-Term Improvements		
I.B.1.1 (1)	Prepare Draft Criteria	MEDIUM	12/86
I.B.1.1 (2)	Prepare Commission Paper	MEDIUM	12/86
I.B.1.1 (3)	Issue Requirements for Upgrading Technical Resources	MEDIUM	12/86
I.B.1.1 (4)	Review Responses to Determine Acceptability	MEDIUM	12/86
I.B.1.1 (6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	MEDIUM	07/87
I.B.1.1 (7)	Issue Regulatory Guides 1.33 and 1.8	MEDIUM	07/87

Issue Number	*Title*	*Priority*	*Scheduled Resolution Date*
I.D.3	Safety System Status Monitoring	MEDIUM	02/89
I.D.4	Control Room Design Standard	HIGH	TBD
I.D.5 (3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	09/88
I.D.5(5)	Disturbance Analysis Systems	HIGH	TBD
I.F.1	Expand QA List	HIGH	12/86
II.B.5 (1)	Behavior of Severely Damaged Fuel	HIGH	06/88
II.B.5 (2)	Behavior of Core-Melt	HIGH	06/88
II.B.5 (3)	Effect of Hydrogen Burning and Explosions on Containment Structure	MEDIUM	09/87
II.C.4	Reliability Engineering	HIGH	12/87
II.E.4.3	(Containment) Integrity Check	HIGH	01/88
II.E.6.1	Test Adequacy Study	MEDIUM	05/88
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	MEDIUM	TBD
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	12/88
II.J.4.1	Revise Deficiency Report Requirements	NEARLY RESOLVED	02/87
HF 1.1	Shift Staffing	HIGH	TBD
HF 4.1	Inspection Procedures for Upgraded Emergency Operating Procedures	HIGH	09/87
HF 4.4	Guidelines for Upgrading Other Procedures	HIGH	02/87
HF 5.1	Local Control Stations	HIGH	08/87
HF 5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	09/87
HF 8 Program	Maintenance and Surveillance	HIGH	10/88

In May 1986, the NRC issued an advance Notice of Proposed Rulemaking that would require, after January 1, 1991, that Senior Operators of nuclear power plants hold baccalaureate degrees in engineering or science. Public comment on the notice was received up to nearly the end of the fiscal year.



transients and accidents and have developed generic technical guidelines for upgrading their EOPs. In accordance with NRC directives, industry has been revising the EOPs to reflect both the engineering guidance contained in the generic technical guidelines and the human factors principles.

The staff continues to evaluate industry's efforts to upgrade EOPs by reviewing Procedures Generation Packages (PGPs) from operating reactors and license applicants. The PGP describes a plant's program for adapting the generic technical guidelines to develop the technical content of plant-specific EOPs and applying human factors principles to produce EOPs which are usable by operators. By the close of the report period, nearly all plants had submitted their PGPs for review by the NRC.

To evaluate the effectiveness of the NRC's long-term program for upgrading EOPs, the staff has been auditing the implementation of PGPs at selected plants over this and the last report periods. Five audits have been performed and additional audits are planned before an assessment of the program is completed. Based on input from sources including PGP implementation audits, staff PGP reviews, and license examiners, the staff has identified certain problems that licensees are experiencing in implementing their PGPs. To alert all operators to these problems, the staff issued Information Notice 86-64. Progress by the industry in addressing the problems will be monitored by inspections, additional PGP implementation audits, and continued dialogue with industry.

During this report period, the staff issued a Temporary Instruction (TI) for use by the Office of Inspection and Enforcement in inspecting each plant's implementation of upgraded EOPs. Eventually, all plants will be inspected using the TI.

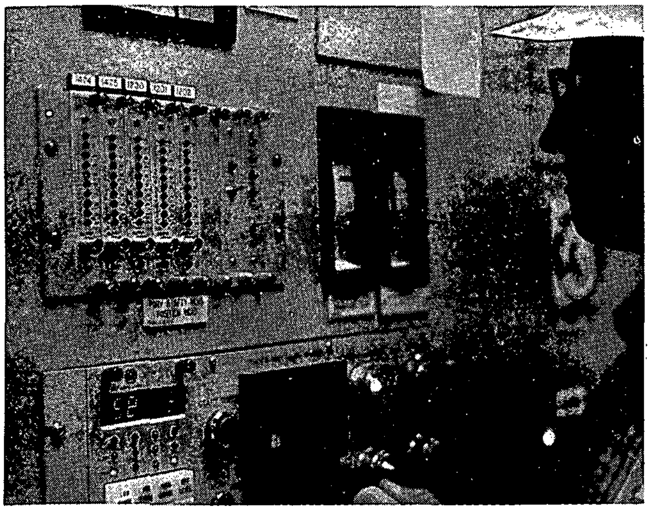
The NRC's original review of the Owners' Groups generic technical guidelines turned up certain unresolved technical issues. The staff continues working with each Owners' Group to resolve them. To date, each Owners' Group has submitted revisions to their technical guidelines, and these are in various stages of review by the staff.

Man-Machine Interface

With the reorganization of NRR early in the fiscal year, licensing reviews for "Man-Machine Interface" (MMI) activities were transferred to the new vendor-oriented licensing divisions, and the MMI activities applicable to all licensees remained in the new Division of Human Factors Technology branch for Human Factors Issues. Besides the generic MMI activities, the branch has responsibility for oversight of the vendor-specific work within the licensing divisions.

Within the three licensing divisions, reviews of detailed control room design reviews (DCRDR), Safety Parameter Display Systems (SPDS), and the data and information handling capabilities continue. During fiscal year 1986, a large number of reviews were initiated but not completed; it is expected that most DCRDR and SPDS reviews will be completed in fiscal year 1987.

To appraise the success of the SPDS programs evaluated in the licensing reviews, audits were performed on six SPDS implementations. Based on these audits, an Information Notice (86-10) was issued to the industry describing some serious weaknesses found in the systems. These were explored further with the industry in May, at an Electric Power Research Institute (EPRI) workshop on the subject. The staff



During 1986 the NRC reviewed a large number of control room designs, safety parameter display systems, and data/information handling capabilities in its continuing efforts to improve the "man-machine interface" at nuclear plants. In this photo, an NRC inspector checks safety equipment—an acoustic monitor for relief valves—at the St. Lucie nuclear power plant on Hutchinson Island, near Fort Pierce, Fla.

will follow-up industry improvements in the implementation of SPDS through audits over the coming fiscal year.

A potential issue related to weaknesses in the human factors design of control stations outside the main control room was investigated during the report period; results of the investigation and assessment of the safety significance of this concern will be made in the next fiscal year.

Under the Human Factors Program Plan, a project was instituted during the year on the subject of advanced control room designs, involving the automation of operator actions and the potential application of Artificial Intelligence. Steps have been taken to monitor industry application of these technologies and to prepare the NRC staff to conduct safety reviews of such applications. A presentation to the NRC by Kennedy Space Center personnel on their experience with the evolution of the control room for the ground launch control system gave added impetus to this effort. The space center experience has demonstrated both safety and cost benefits in increased automation. The NRC is exploring ways to introduce the technology to the nuclear industry.

In support of the International Organization for Economic Cooperation and Development, the staff developed a questionnnaire for collection and exchange of information among member countries on the use of computers in control rooms. The questionnaire will be distributed and responses analyzed in the next fiscal year.

Training

The staff continued to evaluate the results of the Institute of Nuclear Power Operations (INPO) accreditation program, in order to determine whether the industry's voluntary efforts will ensure that training is sufficiently performance-based. A two-year evaluation period ending in April 1987 is called for in the Commission's Policy Statement on Training and Qualification of Nuclear Power Plant Personnel (50 *FR* 11148), which endorsed the Training Accreditation Program managed by INPO.

As part of this evaluation effort, the staff continued to participate as observers when utilities' training programs were under examination by an INPO accreditation team. The staff has also developed criteria and procedures that will enable it, in assessing utility training programs, to ensure that those programs include the five critical elements called for by the Policy Statement. These review criteria have been published as "Training Review Criteria and Procedures" (NUREG-1220). The criteria have been applied by the staff in reviewing training programs at five utilities with accredited training programs. The results of these post-accreditation reviews were included in the "Annual Status Report on Implementation of the Commission Policy Statement on Training and Qualification"(SECY-86-119), which gives the first-year evaluation of the INPO-managed Training Accrdditation Program.

A study was also conducted during the report period of the content in the curricula of specialized educational programs for licensed nuclear reactor operators, to ascertain the extent of job-relevant knowledge taught in these programs. The results of the study were published in an "Assessment of Specialized Educational Programs for Licensed Nuclear Reactor Operators" (NUREG/CR-4411). Members of the staff have also prepared papers on training and qualification issues for a number of conferences and symposia.

Maintenance and Surveillance

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

Phase I of the MSPP, which was approved for implementation by the NRC Executive Director for Operations (EDO) in January 1985, was completed in June 1986.

Major activities completed during Phase I included:

- Development of a maintenance survey protocol to be used in collecting detailed information on maintenance programs and practices at nuclear facilities. The protocol has been employed in surveys of maintenance practices at eight facilities.

- Development, administration, and analysis of a maintenance questionnaire which was completed by nearly all Senior Resident Inspectors.

- Development of a computerized maintenance data base (MDB) containing approximately one quarter million data elements on nuclear power plant operations, maintenance and inspection, for the period 1980 through 1985.

- Development of 31 measures of plant maintenance effectiveness (analyzed data from the MDB). The measures were used to identify trends and patterns of nuclear power plant operations and maintenance performance (published in NUREG/CR-4611).

- Completion of a study of the contributions to "wrong unit/wrong train" events that involved in-depth site evaluation and root cause analysis for specific events (NUREG-1192).

The staff was also involved in supporting maintenance-related activities arising from current events and licensing actions, such as the event at Davis-Besse. The loss of feedwater event that occurred at Davis-Besse (Ohio) in June 1985 was traced, in part, to inadequate maintenance practices. As part of a broader NRC study, the staff performed two detailed surveys of the Davis-Besse maintenance program and practices, and verified many of the weaknesses identified by the licensee. Upon completion of the second survey in March 1986, the staff concluded that major improvements had been made in the maintenance program by the licensee and that the new maintenance organization was functioning with no significant identifiable weaknesses.

Major findings from the MSPP Phase I efforts are:

- While trends in plant operations showed improvement since 1980, a high percentage of failure results from improper performance of maintenance.

- Many major new maintenance programs were initiated by the industry; nevertheless, there is a wide variance in industry maintenance programs and practices, some of which appear markedly different from general trends within the industry.

- Regulatory requirements for maintenance do not adequately cover all aspects of maintenance.

- The ratio of maintenance-related Licensee Event Reports (LERs) to all LERs increased substantially in 1984.

- Inadequacies in labeling, training and procedures were the primary contributors to human error in 60 percent of the "wrong unit/wrong train" events investigated.

- A complete report of the activities, findings and conclusions of Phase I were published in the "Status of Maintenance in the U.S. Nuclear Power Industry: 1985" (NUREG-1212, Volumes 1 and 2).

Based on the results and recommendations of Phase I, the EDO approved implementation of Phase II in July 1986. The objectives of Phase II are:

- Defining the role of maintenance in safety.
- Development of recommendations for good maintenance practices.
- Evaluation of the effectiveness of industry initiatives in improving nuclear power plant maintenance.

Operator Licensing

Reactor operator licensing examinations are scheduled and administered through the NRC Regional Offices. During fiscal year 1986, NRC issued 483 new licenses and 606 license renewals for reactor operators, and 615 new licenses and 1180 renewals for senior reactor operators. Regional Office personnel also conducted requalification examinations at 19 facilities and granted 71 instructor certifications.

The staff adopted new procedures in the conduct of Regional Office audits and program reviews, in order to provide more continuity and a less disjunctive review process, while assuring consistency in all regional examinations. Revision 2 to NUREG-1021, "Operator Licensing Examiner Standards," provided explicit guidance to examiners for preparing and documenting simulator examination scenarios. The "Knowledges and Abilities Catalog for Nuclear power Plant Opeators: Boiling Water Reactors" (NUREG-1123) was issued to aid examiners in drawing up job-related examinations. A supplement for the last year's PWR catalog will be completed in fiscal year 1987 and will make the PWR catalog compatible with the BWR catalog. Uniformity in the catalogs will permit the adoption of a final revision of the "Examiner's Handbook for Developing Operator Licensing Examinations" (NUREG-1121) so that all operator licensing examinations can be prepared from the uniform set of procedures. The computerized Examination Question Bank is fully compatible with IBM-PC microcomputers and has been improved in software and other aspects. All NRC examinations are being prepared and stored on the computerized Examination Question Bank.

After analyzing over 1,700 public comments, the staff submitted for Commission action a final rulemaking package revising 10 CFR 55 and three regulatory guides (Regulatory Guide 1.8—Qualification and Training of Personnel for Nuclear Power Plants, Regulatory Guide 1.149—Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations, and Regulatory Guide 1.134—Medical Evaluation of Licensed Personnel for Nuclear Power Plants). The Commission approved the rulemaking package. The revised 10 CFR 55 requires that utilities certify to the NRC their use of a plant-reference simulator meeting the requirements of ANSI/ANS 3.5 (1985), or else propose a suitable simulation facility for NRC approval. A methodology for the evaluation of certified simulation facilities for the examination of operator and senior operators was developed and tested during the report period.

Unresolved Safety Issues

The Energy Reorganization Act of 1974, as amended, requires that the annual report of the Commission to the President and the Congress include progress reports on those items previously identified as "Unresolved Safety Issues" (USIs). Table 7 is a listing of former USIs for which a technical resolution has been achieved, and Table 8 sets forth the schedule for the resolution of USIs currently under review. These current issues are discussed in the summary that follows, with the exception of "PWR Steam Generator Tube Integrity" (Numbers A-3,4,5), whose resolution is virtually complete and has been treated at length in previous NRC annual reports.

SUMMARY OF STATUS

Systems Interactions

Adverse Systems Interactions are events that may jeopardize the independent functioning of nuclear plant systems. The staff directed considerable effort during fiscal year 1986 toward the resolution of the issue, and the initial staff evaluation of this issue is documented in an "Assessment of System Interaction Experience in Nuclear Power Plants" (NUREG/CR-4261, dated June 1986). The staff expects to issue a proposed resolution for this matter, USI A-17, early in fiscal year 1987, with final resolution near the end of fiscal year 1987.

Seismic Design Criteria

Rapid advancements in state-of-the-art technology in seismic design over the past decade have made it possible and necessary to update the NRC acceptance criteria for seismic design of structures, systems and components of nuclear plants. The Lawrence Livermore Laboratory compared NRC Seismic Design Criteria with the current state-of-the-art knowledge and published the results in its "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria" (NUREG/CR-1171, dated May 1980). Based on these recommendations and results of a staff-sponsored workshop for soil-structure interaction held in June 1986 the staff will propose modifications to related review criteria. Incorporation of the proposed changes is expected to eliminate some potential sources of unwarranted conservatism, and result in seismic design criteria that reflect an up-to-date understanding of this technology.

Station Blackout

The loss of all alternating current (a.c.) electric power (from both off-site and on-site sources) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems that do not require a.c. power supplies and on the ability to restore a.c. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event that could result in unacceptable consequences, such as severe core damage.

The staff's proposed resolution of this issue, USI A-44, which includes a proposed rulemaking and a new regulatory guide, was reviewed by the Commission and issued for public comment on March 17, 1986. The public comment period ended on June 16, 1986. The rule is scheduled to be resubmitted for review by the Committee to Review Generic Requirements in March 1987. The final rule is scheduled to be approved by the Commission and issued in September 1987.

The staff is continuing to work with the Nuclear Utility Group on Station Blackout, who are now developing detailed guidance for the use by utilities in making an assessment of their plants' capabilities.

Shutdown Decay Heat Removal Requirements

The staff is continuing to study the adequacy of systems for safely removing decay heat from a reactor core during shutdown and to assess the value and the impact of alternative measures for improving the reliability of the decay heat removal function. These studies include system reliability assessments, system engineering feasibility studies, thermal-hydraulic analyses, power plant characterizations, reviews of emergency operating procedures, and evaluation of the vulnerability of the systems to special emergencies such as fire, flood, earthquake and sabotage.

A contractor to the NRC has completed four of the seven plant studies. Two of the remaining studies are scheduled to be completed during December 1986. The seventh study has been deferred for a lack of available funding. These six studies will form the basis for the staff's assessment of current decay heat removal capability and the potential risk reduction and the cost of possible changes to plant design or operation. A technical summary of these studies and a value-impact analysis of alternatives are scheduled to be submitted to the Committee to Review Generic Requirements in March 1987 and for public comment in August 1987.

Seismic Qualification of Equipment in Operating Plants

The design criteria and methods employed for the seismic qualification of mechanical and electrical equipment in

Table 7. Formerly Unresolved Safety Issues for Which A Final Technical Resolution Has Been Achieved

Title		*Report Number*	*Date*
A-1	Water Hammer	NUREG-0927 Rev. 1 NUREG-0933 Rev. 1	March 1984
A-2	Asymmetric Blowdown Loads	NUREG-0609	November 1980
A-6	Mark I Short Term Program	NUREG-0408	December 1977
A-7	Mark I Long Term Program	NUREG-0661 NUREG-0661 Suppl.	July 1980
A-8	Mark II Containment Pool Dynamic Loads	NUREG-0808	August 1981
A-9	Anticipated Transients Without Scram	NUREG-0460 Volume 4	September 1980
A-10	BWR Feedwater Nozzle	NUREG-0619	November 1980
A-11	Reactor Vessel Material	NUREG-0744, Rev. 1	October 1982
A-12	Steam Generator and Reactor	NUREG-0577, Rev. 1	September 1982
		Coolant Pump Supports	
A-24	Qualification of Class 1E Safety Related Equipment	NUREG-0588 Rev. 1	July 1981
A-26	Reactor Vessel Pressure Transient Protection	NUREG-0224	September 1978
A-31	Residual Heat Removal	SRP 5.4.7	1978
A-36	Control of Heavy Loads Near Spent Fuel	NUREG-0612	July 1980
A-39	SRV Dynamic Loads	NUREG-0802	September 1982
A-42	Pipe Cracks in Boiling Water Reactors	NUREG-0313 Rev. 1	July 1980
A-43	Containment Emergency Sump Performance	NUREG-0897 Rev. 1	October 1985

nuclear power plants have changed significantly during the history of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and to perform intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to assure the ability to bring the plant to a safe shutdown condition following a seismic event. This issue, USI A-46, entails investigation of alternative procedures for assuring seismic adequacy of equipment in lieu of requiring qualification to current licensing requirements.

The staff evaluated the various methods available for verifying seismic adequacy of equipment in operating

Table 8. Schedule for Resolution of Current Unresolved Safety Issues

Number	*Title*	*Schedule for Issuing Staff Report "For Comment" (as of Sept. 30, 1986)*	*Schedule for Issuing Final Staff Report (as of Sept. 30, 1986)*
A-3, 4, 5	PWR Steam Generator Tube Integrity	Completed April 1985	December 1986
A-17	Systems Interactions	February 1987	September 1987
A-40	Seismic Design Criteria	March 1987	October 1987
A-44	Station Blackout	Completed March 1986	September 1987
A-45	Shutdown Decay Heat Removal Requirements	March 1987	February 1989
A-46	Seismic Qualification of Equipment in Operating Plants	Completed Sept. 1985	November 1986
A-47	Safety Implications of Control Systems	February 1987	December 1987
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns	—	September 1987
A-49	Pressurized Thermal Shock	Completed January 1986	December 1986

nuclear power plants and decided that the use of seismic experience data and of test experience data would prove the most viable and cost-effective way of doing so.

The staff concluded from its investigation of the issue, USI A-46, that there are three principal areas of concern: the adequacy of equipment anchorages and supports, the functional capability of electrical relays, and equipment unique to nuclear plants and outside the limits of the experience data base. The staff developed a proposed resolution and implementation procedure for USI A-46 which provides general guidelines for walk-through inspection of equipment anchorages and review of functional capability of electrical relays. Technical findings as documented in the draft of "Seismic Qualification of Equipment in Operating Nuclear Power Plants" (NUREG-1030) and the Proposed Resolution of USI A-46 were issued for public comment in September 1985. Final resolution for USI A-46 was expected by early 1987.

Safety Implications of Control Systems

The staff has completed systematic evaluations of the control systems typically used during normal startup, shutdown and on-line power operations of nuclear power plants, for each of the four nuclear steam supply system vendors (i.e., Babcock and Wilcox, Westinghouse Corp., Combustion Engineering, and General Electric Co.) The purpose of the studies was to identify control systems whose failure could either cause transients or accidents to become more severe than those assumed in each plant's licensing basis, adversely affect any assumed or anticipated operator action during the course of an event, cause technical specification limits to be exceeded, or cause transients or accidents to occur at a frequency in excess of those established for abnormal operational transients and design-basis accidents. Final reports detailing the staff's review of each of the designs were issued in July 1986.

These studies have identified several control system failures that could cause transients leading to steam generator or reactor vessel overfill, overcooling, overpressure or overheating events. The final reports evaluating the potential risk of these failures have been issued. In addition, various alternatives for reducing the initiating failure frequency or eliminating the failure mechanism of control systems found to be major contributors to events of concern have been analyzed.

A proposed staff resolution which includes recommendations for operating plants and for future plants is currently under staff review. The staff plans to publish for public comment a draft of the technical findings report and the proposed resolution of the issue, USI A-47, in 1987.

Hydrogen Control Measures and Effects Of Hydrogen Burns on Safety Equipment

A large quantity of hydrogen may be produced during degraded core conditions in light water reactors. The hydrogen could ignite or detonate, depending on the concentration present in the containment. Both of these occurrences can affect the design leak rate of the containment and the operation of equipment within the containment structure. An extensive research program was initiated in 1980, following the Three Mile Island accident, to control hydrogen produced by metal-water interaction in various types of containment structures and to study the effects of hydrogen combustion on safety related equipment.

Based on the research, the Nuclear Regulatory Commission published rules in 10 CFR Part 50 addressing four of the five containment types in use. These rules called for the nitrogen inerting of BWR Mark I and Mark II containments and the use of hydrogen igniters in BWR Mark III and PWR ice-condenser containments. Because of the large volume of large dry PWR containments and the dilution of hydrogen released from degraded core accidents, the staff proposed that rulemaking on these larger containment structures be deferred pending completion of both industry and NRC hydrogen research programs.

Considering projected completion of the hydrogen research programs, the promulgation of the hydrogen control regulations, and the planned lead plant implementation of hydrogen ignition systems required by the regulations on ice condenser and BWR Mark III containments, the completion date for this issue, USI A-48, is estimated to be mid-1987. The staff generic report was being prepared at the close of the report period.

Pressurized Thermal Shock

Pressurized Thermal Shock (PTS) events involve unintended rapid cooling of the steel reactor pressure vessel to a low temperature simultaneous with or followed by repressurization of the water inside the vessel. If a flaw or crack exists at a location where the vessel's inner surface has been embrittled by the neutron irradiation that occurs during normal power generation, severe PTS events could cause rupture of the vessel with potential melting of the enclosed nuclear core.

To ensure that nuclear plants do not operate with unacceptable PTS risk, the NRC promulgated a final rule on July 23, 1985, described in the *1985 NRC Annual Report*, p. 35. As specified in the rule, licensees were required to analyze their plants against the screening criterion by January 23, 1986. Judging from a partial review of these submittals and applying the most conservative method of predicting embrittlement, only one plant is predicted to exceed the screening criterion before 1993. This licensee is taking action to reduce the rate of neutron irradiation and thus postpone the point when the screening criterion is calculated to be exceeded.

Safety Reviews

TMI Action Plan

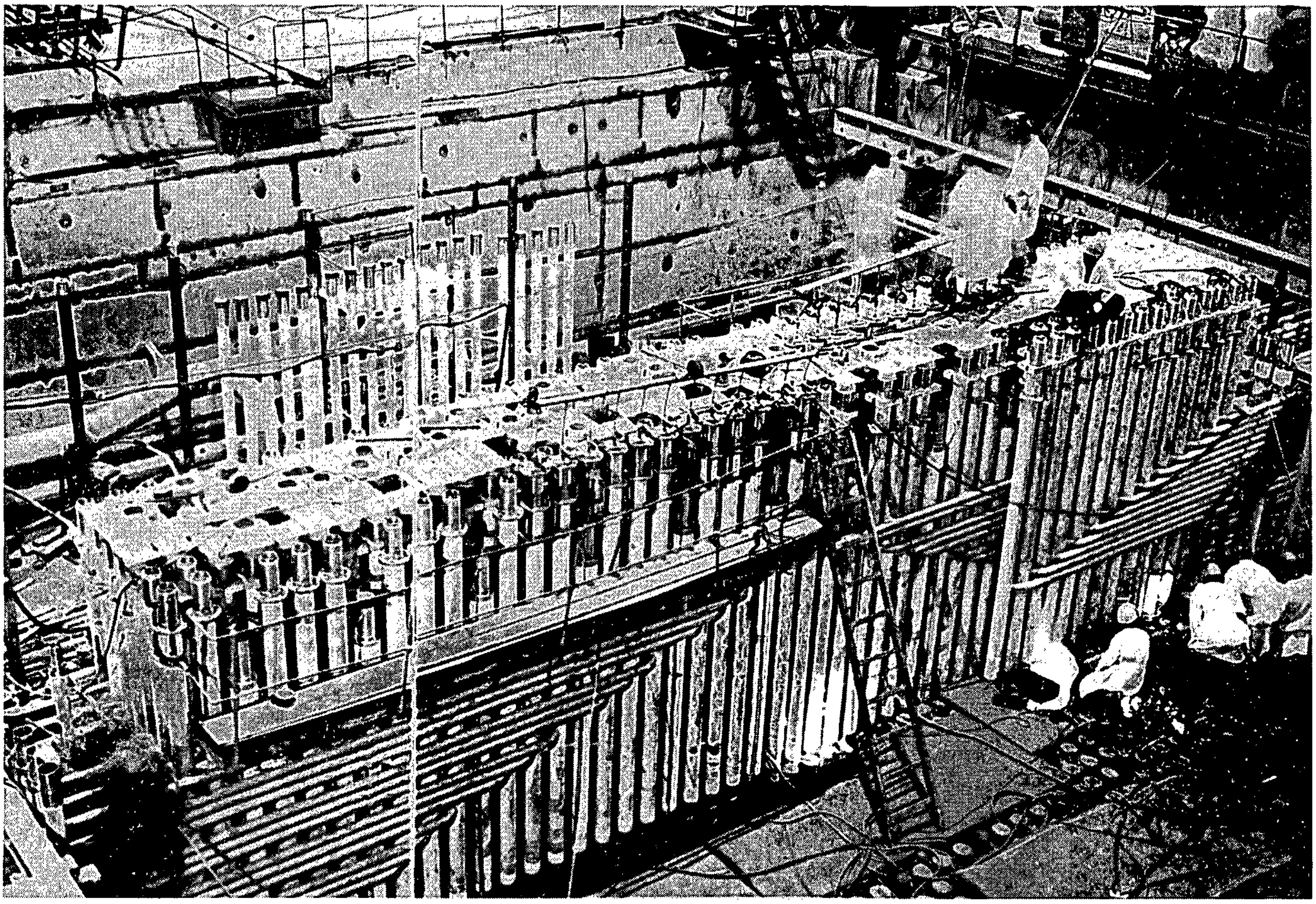
The accident at Three Mile Island Unit 2 (Pa.) in 1979 led to a thorough review of NRC regulatory and licensing requirements for nuclear power plants. The TMI Action Plan (NUREG-0660) was issued, and requirements were approved for implementation at plants in operation or under construction; these were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

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

At the end of fiscal year 1986, licensees had completed compliance with about 7,300 of the items in NUREG-0737, which represents about 85 percent of all items therein. The great majority of these actions have been verified by NRC. Of the 1,300 items still open, about 1,000 are under active review by the NRC. For the remaining 300, the NRC review has been completed but full implementation by the licensees is yet to be verified. A concentrated effort is being made to complete the review of the remaining 1,000 items as soon as possible, with a significant portion to be completed by the end of fiscal year 1987.

Follow-Up of the Accident at Chernobyl

Activities During the Accident Alert. On April 26, 1986, at about 1:24 a.m., two explosions occurred in the core of the fourth unit of the Chernobyl nuclear power plant, which is located on the Pripyat River in the Ukraine and 130 kilometers north of Kiev, the capital of the Ukrainian S.S.R.



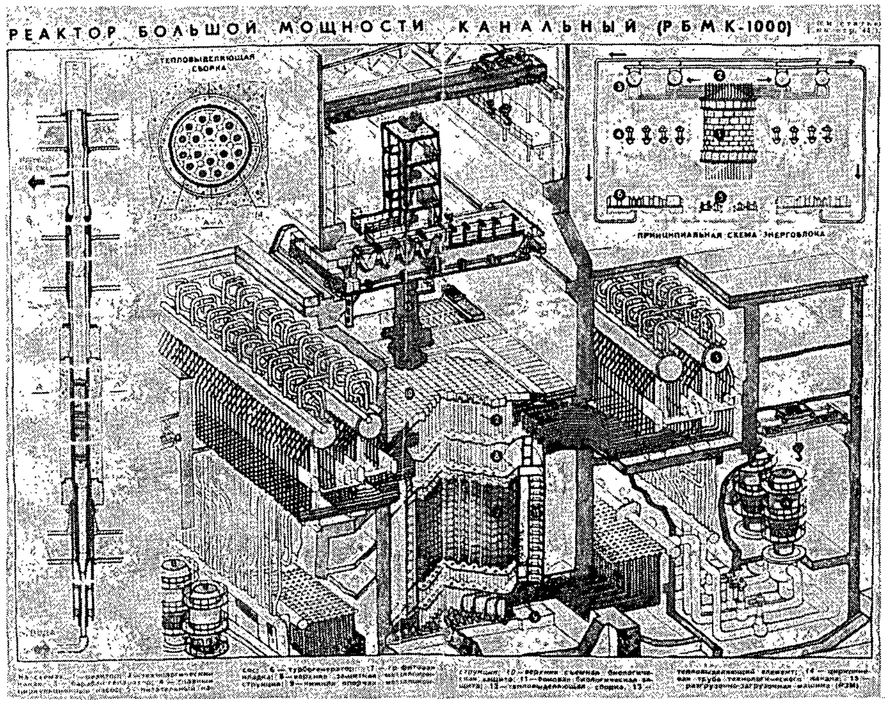
Shown is an RMBK-1000 reactor of the type involved in the April 26, 1986 accident at the Chernobyl nuclear power plant near Kiev, U.S.S.R. Information about the accident became available to the NRC on April 28 and, in addition to staff monitoring of the event, an Incident Tracking Team was established on May 1 to help assess the impact and the implications of the accident for the nation and for nuclear power generation. Collection and analysis of information on the event was continuous within the NRC, and senior staff, including Chairman Lando W. Zech, Jr., later participated in the International Nuclear Conference in Vienna to hear from the Soviet authorities and explore the matter with experts from many countries.

and third largest city in the Soviet Union. A massive fire followed the explosions, and radioactive debris was disgorged into the atmosphere. Prevailing winds carried the material into Finland and Sweden. On April 28, 1986, a high radioactivity reading was taken on a worker and also on the grounds of Sweden's Forsmark Nuclear Power plant. Within hours monitores in Denmark, Norway and elsewhere in Sweden registered increased radioactivity. These readings and wind patterns led to the conclusion that there had been a nuclear accident in the Soviet Ukraine.

Having been informed of the accident, the President created an Interagency Task Force, headed by Environmental Protection Agency (EPA) Administrator, to monitor the aftermath of the accident. Agencies represented on the task force, besides the EPA, were the NRC, the Department of Energy, the Department of State, the Central Intelligence Agency, the Food and Drug Administration, the National Oceanic and Atmospheric Administration, the Department of Agriculture, and the Federal Emergency Management Agency. The NRC staff had been following developments since the first information became available on April 28, 1986. On May 1, 1986, an Incident Tracking Team was established at the direction of the Commission, to support the EPA in assessing the impact of the accident on the U.S., and to identify the implications of the Chernobyl accident for U.S. reactor operations. The team completed its work on May 14, 1986.

A prime objective of the Incident Tracking Team was to attain a sufficient understanding of the accident and the radiological source term involved to be able to assist EPA in assessing radiological consequences for the United States. Arrangements were made with western and eastern European countries, Korea, Japan, Canada and Israel to receive their radiological environmental monitoring data.

Communication throughout with the U.S. nuclear power industry was good. As soon as the Incident Tracking Team received the daily status report on the Chernobyl incident (published by EPA), it was transmitted to the Institute of



This is a diagram of the layout of the Graphite Moderated Pressure Tube Reactor (RMBK-1000), of the kind erected at the Chernobyl site in the Ukraine. Insets show (left) water circulation on fuel cell, with cold water entering at bottom, steam-water mixture expelled at top; cross-section of a fuel assembly (top left); and a schematic of the reactor (top right).

Nuclear Power Operations for communication to U.S. utilities and vendors. Also, the Office of Inspection and Enforcement (IE) sent Notice No. 86-32 to licensees requesting voluntary monitoring for the detection of radiation from Chernobyl. In addition to supporting the Administrator of EPA with data and analyses, the team put considerable effort into preparing briefings for the NRC Chairman, the Commission, and the Advisory Committee on Reactor Safeguards, and also into answering Congressional questions and preparing Congressional testimony.

The Vienna Conference. After the Incident Tracking Team disbanded on May 14, 1986, the staff continued to collect and analyze information on the Chernobyl accident, seeking a better understanding of the causes, evolution and consequences of the entire event. Background information was developed in a joint effort involving several Federal agencies and with contributions from industry groups. Activities were coordinated through an informal *ad hoc* steering committee with representation from the agencies and several industry groups. By mutual agreement, the NRC took on the function of overall administrative coodinator.

A conference to receive and review the Soviet report on the matter was held in Vienna from August 25 through August 29, 1986, and was attended by representatives from a large number of countries, including all countries with nuclear energy programs. The large majority of these delegates were technical people. The U.S. had a strong delegation was led by Ambassador Richard Kennedy and included representatives from the U.S. agencies involved with nuclear and environmental programs. The U.S. delegation was accompanied by a number of technical consultants from universities and the National Laboratories, as well as representatives from the U.S. nuclear industry.

The Soviet Delegation was made up of some of their top scientists and engineers in technical disciplines related to the various aspects of the Chernobyl accident. Most of the Soviet delegates, including the leader, had no direct relationship with the design or operation of the Chernobyl reactor.

The central focus of the conference was on the presentation by the Soviets of information on any and all aspects related to the Chernobyl accident. The Soviet experts were impressively open and candid about the event—why it happened, how it happened, what were its consequences and general impact within the Soviet Union. The Soviet Delegation provided considerable detail about the accident scenario and its consequences, which clarified and augmented the information provided in their official report to the International Atomic Energy Agency (IAEA).

The International Nuclear Safety Advisory Group (INSAG), an advisory group to the IAEA Director General, has provided a report on the results of the Vienna Conference and plans to provide recommendations for future IAEA actions that might help enhance the reliability and safety of nuclear power plants.

U.S. Factual Report. After the Vienna meeting, the Federal agencies and other groups participating in the drafting of a coordinated U.S. report on the Chernobyl facts resumed work on the report. This work was aimed at production of a final U.S. factual report, drawing on the substantial Soviet information that had become available, to serve as a common factual reference for NRC's assessment of the implications the accident may have had for the safety regulation of U.S. nuclear power plants, and for other relevant Federal agencies' use in assessing implications with respect to their areas of responsibility.

Implications for Regulatory Activities. Evaluation of the Chernobyl accident's implications for our domestic regulatory programs and practices was in progress at the close of the report period. Reviews of the accident and the reactor design completed to date by the staff have not identified any aspects of the accident which show a direct, clear-cut connection with U.S. commercial nuclear power operations. Chernobyl is, however, a near "worst-case" warning which merits and is receiving priority attention, either to confirm that current regulatory practices and policies are sound, or to identify potentially useful improvements. NRC studies on the subjects of bypassed safety systems, certain administrative control issues, reactivity accidents, containment, emergency planning, radionuclide release, and steam explosions will be coordinated with the many on-going national and international activities. Any proposed new requirements arising from these investigations will be evaluated in accordance with the Commission's rules and practices.

Adequacy of Zone Distances. Following the Chernobyl accident, the population from a 30-km zone near the affected plant was evacuated. A great deal of attention has been focused upon this 30-km zone. Some have contrasted it with U.S. emergency planning, which entails preparations for a plume exposure pathway of about 10 miles and ingestion exposure pathway of about 50 miles. NRC plans have always recognized that public protective actions might be indicated in areas beyond 10 miles, in unusual circumstances, but that such actions need not be planned in advance, since the activity within the 10-mile zone provided for gives a base for expanding the range of protective measures, if and as needed. By this reasoning, the 30-km relocation by the Soviets would have no clear implications regarding the adequacy of the size of the U.S. planning zone. But it remains important to study and understand all the factors relevant to this issue and draw the the correct implications for regulations and practices in this area.

Source Terms. The magnitude of the source term (i.e., the various amounts, types and timings of radioactive releases) involved in the Chernobyl accident was very large, comparable in many respects to the most severe accident source terms postulated in the NRC-sponsored Reactor Safety Study, published in 1975 (WASH-1400). Since the NRC had recently completed a reassessment of the technical bases for estimating source terms, with the intent of modifying source term-based regulations, it is worthwhile to review the processes that occurred in the Chernobyl accident for these purposes:

(1) To determine whether the releases that occurred in the accident are confirmatory or contradictory to what would be predicted by the current methods.

(2) To identify any processes that may not have been previously considered.

Because of significant differences in plant design between the Chernobyl reactor and U.S. light water reactors—and therefore in the accident sequence and the chemical conditions obtaining during the accident, in comparison with U.S. reactor accidents, actual or postulated—the magnitude of the source term at Chernobyl cannot be used for validating methods of analysis applicable to light water reactors.

Nevertheless, it is possible from an examination of the various stages of radionuclide release in the Chernobyl accident to conclude that these releases can be explained on the basis of existing knowledge of severe accident phenomena and that there is no reason, based upon these results, to change the prevailing perspective on LWR source terms significantly. That does not mean that there are not some data that were not well understood or that further examination is unwarranted.

Some specific aspects of the Chernobyl accident that warrant and are receiving further investigation are:

(1) Mechanical disruption mechanisms (as distinct from actual core-melting) as potential modes of core damage.

(2) Potential dispersion of radioactive material from oxidation of uranium dioxide to higher and less dense oxide.

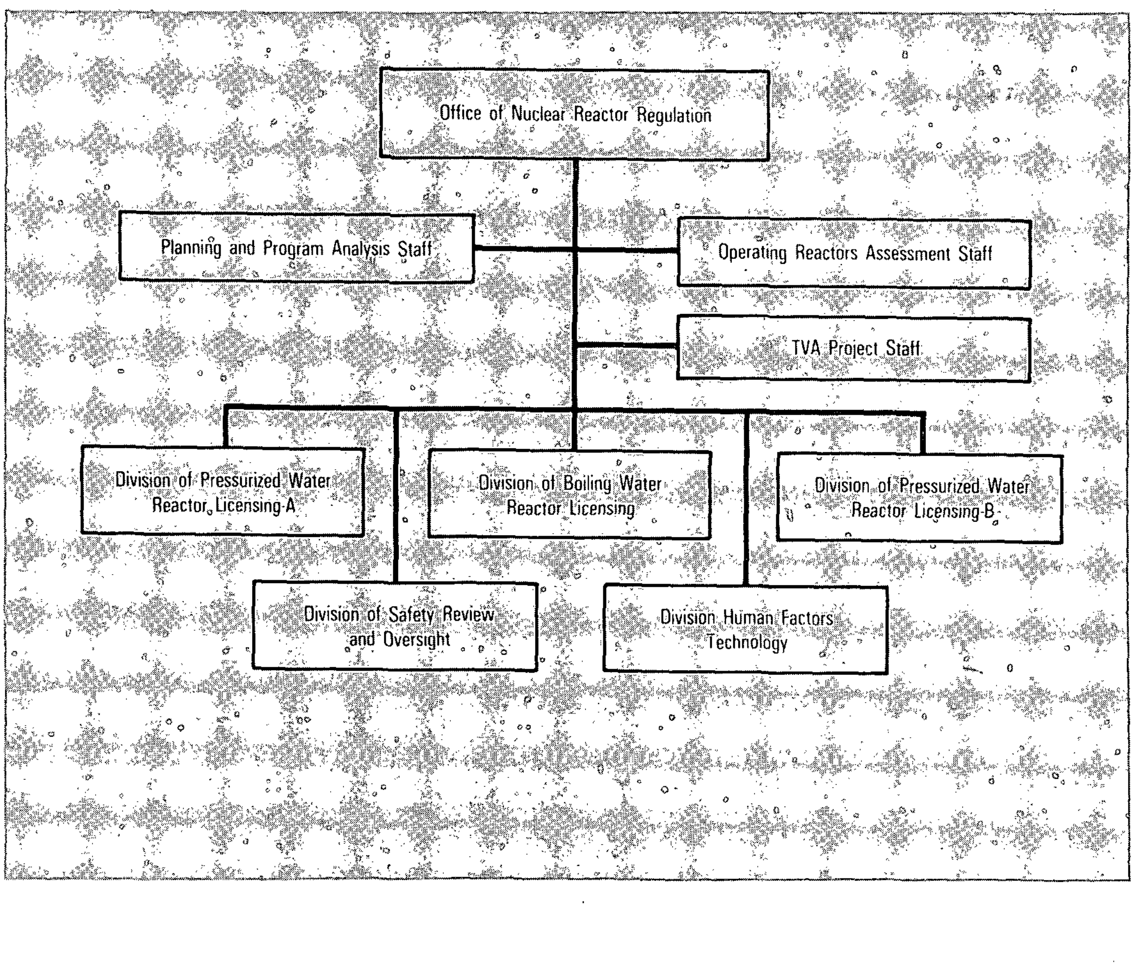
Reevaluation of TVA Operations and Management

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

Extensive technical problems at TVA nuclear plants, both operating and under construction, as well as problems with TVA management, culminated in the shutdown of all TVA operating units and lengthy delays in licensing Watts Bar Unit 1. The TVA plants were shut down in 1985. All three Browns Ferry units were shut down by March 1985 because of poor operational performance, coupled with management and equipment problems. The two Sequoyah units were shut down in August 1985 because available documentation could not substantiate environmental qualification of electrical equipment. The license of Watts Bar was delayed to allow completion of fire protection modifications, as well as to resolve certain allegations and employee concerns. The operating plant shutdowns continued throughout fiscal year 1986. The situation at TVA has been and remains one of NRC's top priority issues.

TVA recovery efforts include strengthening corporate as well as plant management by implementing major site and corporate reorganizations, by hiring experienced personnel, and by the use of experienced consultant personnel who are in turn training qualified TVA managers as their replacements. These efforts include a number of site-specific and corporate programs to resolve technical issues in such areas as design control, welding, operational readiness, and equipment qualification. The TVA created an independent Office of Inspector General, reporting to the TVA Board of Directors, to manage a confidential program for handling alleged intimidation, harassment, and wrongdoing issues. Efforts also include improvements to the quality assurance program in order to provide timely resolution of deficiencies and timely implementation of corrective action. The overall endeavor by TVA is described in their Nuclear Performance Plan.

The combined NRC Offices of NRR, IE, Region II, the newly created TVA Project Staff, as well as a Senior Management Team (SMT) of Office level Directors, are providing concentrated oversight and review of TVA efforts to recover their nuclear generation capability. The creation of the SMT represents the first time such a management body has been formed to coordinate multiple offices in the resolution of a major problem. These offices provide the equivalent of 60 to 70 people on a full time basis to give full support to the TVA effort. NRC and TVA investigations turned up a number of problems at the various sites. At Browns Ferry, the problems involved design, configuration control, weld inspections, fire protection, environmental qualification,



Reorganization of the NRC's Office of Nuclear Reactor Regulation in 1986 included the addition of a separate TVA Project Staff.

probabilistic risk assessment, licensed operator requalification, and operational and management readiness. At Sequoyah, the problems were related to design control, welding, cable installation, Technical Specification revisions and surveillance testing, alleged wrongdoing, and the effectiveness of the quality assurance program. At Watts Bar, the problems arose in verification of licensing commitments, design bases, design documents, equipment qualification, cable installation, operational readiness, essential raw cooling water system piping, alleged wrongdoing welding, and the quality assurance program. Of particular note were the number of employee concerns raised at Watts Bar, where 2,000 of the 5,000 employee concerns involved significant safety issues.

Schedules for the restart of TVA plants are expected to be keyed to TVA's progress with corrective measures at these units. Prior to any unit's restart or licensing, the NRC staff and TVA must brief the NRC Commissioners on the state of readiness for such actions as evidenced by the resolution of all safety and personnel issues.

In January 1986, TVA announced a major reorganization and restructuring which was intended to address deficiencies in management and in management-employee relations. A major part of the reorganization was the creation of the Office of Nuclear Power to establish a central management authority in TVA to be responsible for all nuclear activities. A potential conflict of interest issue arose regarding the Manager of Nuclear Power brought in under contract to direct the new office. The NRC considers resolution of this issue vital to the future of the TVA nuclear program. The purposes of the office include implementation of an effective quality assurance program; resolution of employee concerns; elimination of instances of harassment, intimidation and other wrongdoing; and strengthening of plant management at the various TVA nuclear installations.

Specific problems at operational or nearly operational TVA reactor plants were identified as follows:

Sequoyah. The TVA voluntarily shut down the Sequoyah (Tenn.) units in August 1985 because available documentation was insufficient to substantiate the environmental qualification of electrical equipment. Other problems were subsequently uncovered as a result of employee concerns and allegations as, well as difficulties at other TVA nuclear plants. These problems were identified in the areas of design control, welding, cable installation, Technical Specification revisions and surveillance testing, and quality assurance; there were also issues of intimidation, harassment and wrongdoing involved at the Sequoyah operation.

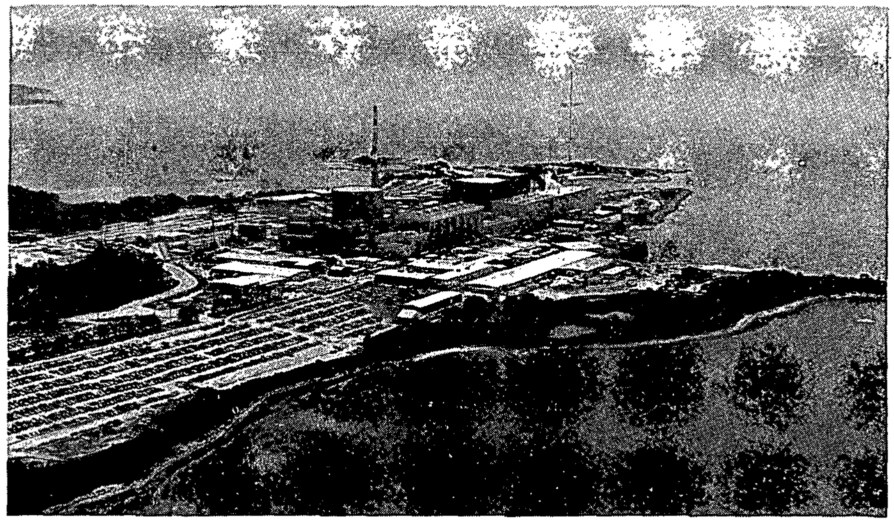
Browns Ferry. Following a prolonged period of poor operational performance coupled with management and equipment problems, tracked and monitored by the NRC Region II (Atlanta) office, all three Browns Ferry (Ala.) nuclear power plants were shut down by March 1985. NRC inspections and investigations of employee allegations, together with RVA evaluations, disclosed substantive problems in such areas as design, configuration control, weld inspections, fire protection, environmental qualification, probabilistic risk assessment, and licensed operator requalification. Rectification of these concerns and broad improvements in overall operational and managerial control were recognized as prerequisite to a resumption of operation at this site.

Watts Bar. Difficulties also arose at the third venue of TVA nuclear power plant operation, Watts Bar in Tennessee. (The Bellefonte Units 1 and 2, in Alabama, are also TVA reactor plants; they are still under construction.) Following licensing delays to carry out fire protection modifications, approximately 5,000 employee concerns were registered regarding this and other TVA facilities, of which some 2,000 had potential safety significance. Among issues raised pertaining to Watts Bar were the verification of licensing commitments, design bases, design documents, construction, and configuration control, welding, concrete, equipment qualification, cable installation, essential raw cooling water system piping, and the quality assurance program, as well as allegations of harassment, intimidation and wrongdoing.

Integrated Safety Assessment Program

In a policy statement published in the *Federal Register* on November 15, 1984, the Commission put forward a trial program to evaluate all pending licensing issues on a given operating reactor, in conjunction with operating experience, probabilistic analyses, and licensee plant improvements, so as to establish effective and efficient implementation schedules for any necessary plant modifications. This program, called the Integrated Safety Assessment Program (ISAP) was implemented in early 1985 for two volunteered plants in Connecticut: Millstone Unit 1 and Haddam Neck. (See *1985 NRC Annual Report,* pp. 36, 37).

In 1985, Northeast Utilities (agent and service organization for both licensees) completed a plant-specific probabilistic safety study (PSS) for Millstone Unit 1, together with safety assessments for the 80 licensing issues pertaining to that facility. The NRC staff completed detailed reviews of the plant's operating experience and of the PSS during 1986. The licensee has implemented immediate corrective action to minimize dominant contributors to plant risk, as identified in the PSS, and continued to implement certain plant modifications outside of the ISAP scope.



Unit 1 of the Millstone nuclear power plant in Connecticut was one of two Northeast Nuclear Energy Company plants volunteered by the utility for participation in the NRC's Integrated Safety Assessment Program (ISAP). NRC staff has recommended that ISAP be extended to other operating reactor plants.

In early 1986, a series of meetings on the Millstone Unit 1 integrated assessment program were conducted with licensee and NRC staff representatives to explore alternative means to resolve the safety issues and effect other plant improvements. Subsequently, in July 1986, the licensee submitted an integrated assessment report for Millstone Unit 1 which proposed specified actions to resolve the issues raised by the NRC and rated their relative safety importance. The NRC staff's integrated safety assessment report was to be published for peer review and comment by the end of 1986.

During 1986, the licensee also completed a plant-specific PSS and safety assessments for the 70 licensing issues for the Haddam Neck plant. The NRC staff completed detailed reviews of the plant's operating experience and the PSS. The integrated assessment meetings for Haddam Neck were scheduled for late in 1986.

The ISAP pilot efforts for both plants have been delayed from the original schedule primarily because of significant difficulties encountered during the Haddam Neck 1986 refueling outage, causing a diversion of NRC staff and licensee resources. Following issuance of the Millstone Unit 1 draft integrated assessment report, the NRC staff will forward a recommendation to the Commission for extending the ISAP experience to other operating reactors.

Probabilistic Risk Assessment

Review and evaluation of plant-specific probabilistic risk assessments (PRAs) continues to be an important activity of the NRC. Probabilistic methods of assessing levels of safety and risk have found application in an increasing variety of regulatory activities.

During fiscal year 1986, the staff completed its technical review of the PRA for the Oconee Unit 3 plant in South Carolina. The review resulted in design modifications to eliminate a vulnerability of the emergency feedwater systems to interruption of the air supply. Design changes to prevent flooding of the high-pressure injection pumps as a result of possible seismic events were also realized. The combined net effect of these changes was a substantial reduction in the risk of core melt.

Utilities are making increasing use of probabilistic analyses to justify modifications to plant design and operation. Requests for plant-specific technical specification changes, or one-time exemptions from such requirements, are routinely evaluated on the basis of risk significance. The licensee for the Seabrook plant (N.H.) has submitted a risk-based analysis to support its contention that the public health can be adequately protected with an emergency planning zone (EPZ) significant smaller than that currently required by NRC regulations. NRC review of this submittal was under way at the close of the report period.

In a continuing effort to provide PRA insights and lessons learned to the nuclear power community, NRC reports have been issued to the staff and the industry representing a synthesis of insights into plant design and operational strengths and vulnerabilities gained from PRAs. The effort is beneficial for a wide spectrum of activities—such as the performance of PRAs, inspection programs or specific safety reviews—and serves to increase awareness of identified vulnerabilities and safety issues among the staff and the industry. In addition, the insights reported help bring to light any new generic issues meriting regulatory consideration.

During the report period, the staff published an assessment (NUREG/CR-4048) of the technical feasibility of allocating reliability and risk to structures, systems, components, and operations. The study concluded that the allocation of reliability and risk is technically feasible and can provide insights into plant risk that cannot be obtained in quantitative PRA models.

The staff has reviewed the study conducted by Toledo Edison Company, licensee for the Davis-Besse plant in Ohio, on the reliability of that unit's Auxiliary Feedwater System, following the June 9, 1985 incident. The staff review identified the significant role played by operator errors, failure dependencies, support systems, and plant-specific data. The conclusions supported the staff recommendation for the integrated assessment of reliability of systems, one that considers positive and potentially negative effects of design modifications, and also considers the impact of the balance-of-plant systems and other non-safety related systems.

Probabilistic risk methods continue to prove useful in setting priorities for the resolution of generic and unresolved safety issues.

Severe Accident Policy Implementation

Severe nuclear accidents are those in which substantial damage is done to the reactor core, whether or not there are serious off-site consequences. In 1984, the staff forwarded to the Commission for review and approval a recommended "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants." The Commission approved the Policy Statement in 1985, and published it in the *Federal Register* on August 8, 1985 (50 FR 32138). A follow-up "Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information," SECY-86-76, was forwarded to the Commission on February 28, 1986.

The implementation plan comprises three major elements.

First, an integrated, systematic approach is being developed, in cooperation with industry, to perform an examination of each nuclear power plant now operating and under construction for possibly significant risk contributors (outliers) that might be missed without a systematic search. This effort is referred to as the Individual Plant Examination (IPE) program. As is the case with studies under way internationally, the emphasis is placed on containment performance in seeking a balance of accident prevention and consequence mitigation. Specific methods for conducting individual plant examinations are being developed by the IDCOR (Industry Degraded Core Rulemaking) program. Resolution of issues and approval of the IDCOR methods were to be completed in 1986. The guidelines and criteria for plant evaluations will be issued in a generic letter to licensees.

As the second element of the program, guidance will be issued on the form, purpose and role of PRAs in the licensing process for new plant applications. The guidance will address the appropriate combinations of deterministic requirements and probabilistic considerations, along with the minimum content of the PRAs. A NUREG report on the role of PRAs in new plant applications is to be issued for public comment in early 1987.

Third, those rules and regulatory practices affected by accident releases (source terms) are to be modified to reflect a sound scientifically based state of knowledge, grounded in the research into both source terms and severe accident phenomena. The implementation plan has been developed to take into consideration uncertainties in on-going severe accident research.

Proposals for rule changes and for changes in regulatory practice to incorporate the insights gained through the evaluation of severe accidents and the results of on-going research in source terms are expected over the next two years.

Fire Protection

The NRC fire protection rule (Section 50.48 and Appendix R to 10 CFR 50) for nuclear power plants became effective on February 17, 1981. It required licensees of all plants holding operating licenses issued prior to January 1, 1979, to submit plans and schedules for meeting the applicable requirements, to describe proposed modifications needed to provide alternative safe-shutdown capability, or to submit requests for exemptions from specific technical requirements of the rule, if appropriate. For plants licensed after January 1, 1979, the NRC staff uses the criteria of the Standard Review Plan (NUREG-0800), which includes the technical requirements of the fire protection rule, to review and evaluate the fire protection program. Requests for exemptions, and proposed modifications to provide alternative safe shutdown capability, continue to be received and reviewed by the staff.

The staff issued Generic Letter 86-10 to provide to licensees and applicants additional clarification of NRC fire protection guidelines and requirements in several areas of concern—such as fire area boundaries, fire barriers qualifications, automatic suppression systems, intervening combustibles, etc. The Generic Letter also provided for removing the fire protection program from individual plant technical specifications and citing it by reference in the Final Safety Analysis Report.

A number of minor nuclear plant fires did occur in 1986, with no adverse impact on the plants or the public. Three fairly extensive fires took place in 1986. Each was fully evaluated and found to have had no actual or potential adverse impact on the affected plant or the public, and no generic implication for other nuclear power plants. They caused no releases of radioactive material and no radiological exposures to the public or to plant personnel. The three fires involved (1) ignition of plastic foam material in the thermal expansion gap area between the steel containment pressure vessel and the heavy reinforced concrete biological shield wall at Dresden Unit 2 (Ill.); (2) ignition of activated charcoal adsorber material in the main condenser off-gas filter system (a non-safety system) at the Perry plant (Ohio); and (3) a fire in a large wood fill cooling tower at Browns Ferry (Ala.). In the case of the Browns Ferry cooling tower, even though it resulted in a multi-million dollar loss, the fire was of no safety concern because the tower was required only to satisfy limitations on the amount of waste heat conveyed to the river water during periods of high temperature and/or low river flow. The Browns Ferry plants were not operating at the time.

Operational Safety Assessment

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

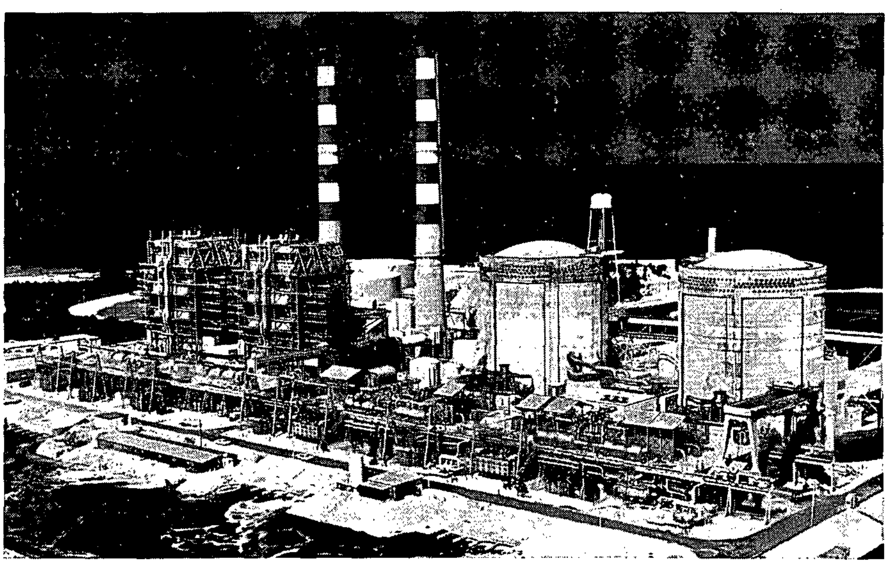
In fiscal year 1986, the staff instituted a formalized program for the assessment of major reactor incidents. Examples of investigations of operating reactor events occurring in fiscal year 1986 include:

- Water hammer event at San Onofre Unit 1 in November 1985
- Feedwater Transient Without Reactor Scram at LaSalle in June 1986
- Fire in Charcoal Offgas Beds at Perry in June 1986
- Degradation of Auxiliary Feedwater System Due to Fuse Failure at Catawba in June 1986
- Potential Failure of All Emergency Diesel Generators at Turkey Point in June 1986
- Ten-Inch Line Break and Scram with Complications at Oconee in September 1986.

Radioactive Effluents Summary and Analysis

The program for implementing Radiological Effluent Technical Specifications (RETS) in operating reactors—a program designed to keep releases of radioactive materials to unrestricted areas during normal operations as low as reasonably achievable—neared completion during 1986. Many of the licensees had implemented their RETS during fiscal years 1984 and 1985. Of the remainder, all but four implemented RETS during fiscal year 1986. It is anticipated that all operating reactors will be operating under RETS by early 1987.

Licensees submit periodic reports on radioactive effluents and radiological environmental monitoring, as part of the RETS license requirements. Semiannual reports contain detailed summaries characterizing the radioactive gaseous and liquid effluents released from the plant to the environment, and also quantify solid radioactive wastes shipped off



Under a formalized incident assessment program initiated in 1986, the NRC investigated, among other matters, a potential diesel emergency generator failure at the Turkey Point nuclear power plant near Florida City, Fla., about 25 miles south of Miami on Biscayne Bay. Turkey Point features two nuclear units, each rated at 666,000 kilowatts, and two fossil-fuel units, each rated at 381,000 kilowatts.

the site. These reports include calculations of the radiation doses from these effluent releases to members of the public off-site. The NRC annually collates these individual plant summaries in two publications: "Radioactive Materials Released from Nuclear Power Plants" and "Population Dose Commitments Due to Radioactive Releases from Nuclear Power Plant Sites."

In addition to the semiannual effluent reports, licensees submit an annual radiological environmental operating report. This report contains the results of the extensive weekly and monthly monitoring programs required by the plant's RETS and records when, if ever, radioactive contamination above natural background is detected outside the plant boundaries. The semiannual effluent reports and the annual radiological environmental operating reports for all operating plants are available for public inspection in local Public Document Rooms. (Data on radiation exposures for personnel within nuclear power plants is reported on in Chapter 11.)

Reassessment of B&W Reactors

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

The NRC's Executive Director for Operations informed the Chairman of the Babcock & Wilcox Owners Group (BWOG), by letter dated January 24, 1986, that events at B&W-designed reactors had lead the NRC staff to conclude that there is a need to re-examine the basic design requirements for B&W reactors. By letter dated February 13, 1986, the BWOG agreed to take the lead in a concerted effort to define the problems assoicated with reducing the frequency of reactor trips, and complexity of post-trip response, in B&W plants.

The NRC reassessment of B&W plants involves a review of BWOG efforts and includes an assessment of the thermal-hydraulic design, instrumentation, controls and power supplies, along with a review of operating experience and operator training and responses. The staff is also performing limited independent evaluations of the B&W plant design, including review of operating experience, evaluation of inspection results, and thermal-hydraulic analysis. The initial staff Safety Evaluation Report (SER) is expected to be issued in early 1987, appraising most of the Owners' Group activities. The remaining BWOG efforts are scheduled to be completed by April of 1987. The final staff SER supplement dealing with the entire program is expected to be issued by June of 1987.

Application of Leak-Before-Break Technology

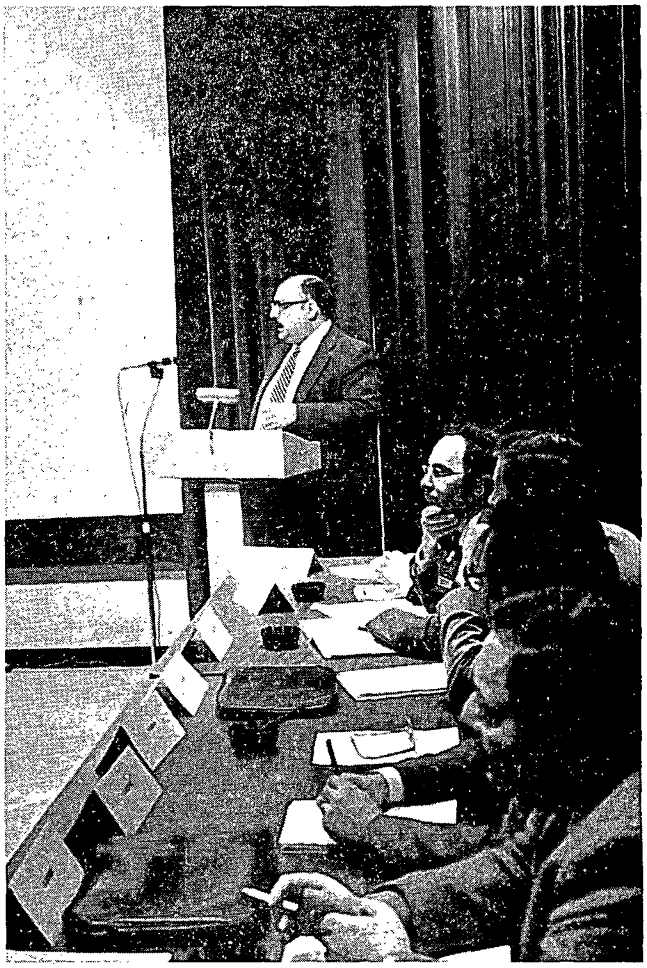
In May of 1986, a limited-scope revision of General Design Criterion 4, Environmental and Missile Design Bases, became effective. This revision was made to permit the use of advanced fracture mechanics, as an alternative method, in determining the locations of postulated pipe break in primary coolant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

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

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

Source Term Releases of Radionuclides in Severe Accidents

Radioactive releases from accidents, or "source terms," play a major role in a number of regulatory applications—such matters as plant siting evaluations, emergency planning, qualification of safety-related electrical equipment for



On January 31, 1986, a moderate earthquake, with the epicenter about 18 miles south of the Perry nuclear power plant in Ohio, set off seismic monitors in the plant but did no apparent damage to safety equipment. Both the licensee, the Cleveland Electric Illuminating Company, and the NRC conducted comprehensive inspections after the quake and confirmed that safety systems were intact. Shown here is Dr. Robert Bernero (standing), NRC's Director of BWR licensing, during a review of the utility's investigative results.

performance under accident conditions, environmental impact statements, and plant engineered safety features (ESF) to mitigate fission product releases.

Severe accident source terms were first systematically assessed as part of the Reactor Safety Study (WASH-1400) in 1975. A significant research effort has been under way to re-examine such potential releases, employing the most advanced scientific understanding. In July 1986, the NRC staff issued its "Reassessment of the Technical Bases for Estimating Source Terms" (NUREG-0956). The staff intends to use the methodology in its regulatory consideration of such releases at light-water power reactors in the U.S. The report discusses and presents the major advances in the technology for calculating accident source terms and represents years of research and analysis effort by the NRC staff and its contractors, as well as extensive peer review and public comment.

In February 1986, the NRC staff identified 10 regulatory areas for possible reassessment and revision in light of the latest research and grouped them into short-term, intermediate-term and long-term items.

The short-term items include revised discussion of accident risks in Environmental Impact Statements (EIS), reassessment of spray systems additives for PWRs, and fission product removal credit for BWR suppression pools. The revised discussion of accident risks in EIS is considered complete with the publication of the Environmental Statement for the South Texas Project, Units 1 and 2 (NUREG-1171), which contained risk assessments using the latest source terms available, as well as those adopted from the rebaselined Reactor Safety Study (RSS). Work on the remaining short-term items is in progress.

The remaining intermediate and long-term items include emergency planning, containment leak rates and integrity, control room habitability and air filtration, equipment qualification, safety issue prioritization, siting and accident monitoring. Reassessments in these areas will be addressed within the next several years as research follow-on studies to NUREG-0956 become available.

Ohio Earthquake Near Perry

On January 31, 1986, at 11:46 a.m. EST, an earthquake of magnitude 5.0 occurred about 10 miles south of the Perry plant in northeastern Ohio. The January 31 earthquake triggered in-plant seismic monitoring instruments. These instruments were fully operational at the time, although the facility itself was not licensed or operational. No damage was sustained by the plant structures, systems, components or equipment which could be attributed to the earthquake. The plant equipment and structural designs had substantial margins of safety relative to the loads and stresses induced by the earthquake affirming the seismic design basis for the plant. The NRC staff's safety evaluation detailing these findings is set forth in NUREG-0887, Supplement No. 9, dated March 1986. (See discussion under "Special Cases," earlier in this chapter.)

Transamerica Delaval Diesel Generators

During a load test on August 12, 1983, the main crankshaft failed on one of the three emergency diesel generators (EDGs) at the Shoreham nuclear power plant

(N.Y.). The EDGs at Shoreham were manufactured by Transamerica Delaval, Inc. (TDI), which has supplied 54 EDGs to 14 other nuclear power plant sites in the United States. While evaluating the Shoreham failure, the NRC staff and the industry became aware of a broad pattern of deficiencies involving critical engine components in TDI EDGs. These deficiencies appeared to stem from inadequacies in design, manufacture, and quality assurance/quality control on the part of TDI. In response to these problems, 13 nuclear utilities formed an Owners Group to establish a program for upgrading and confirming the adequacy of the TDI diesels for nuclear service. (See the *1985 NRC Annual Report,* pp. 48 and 49, for further background.)

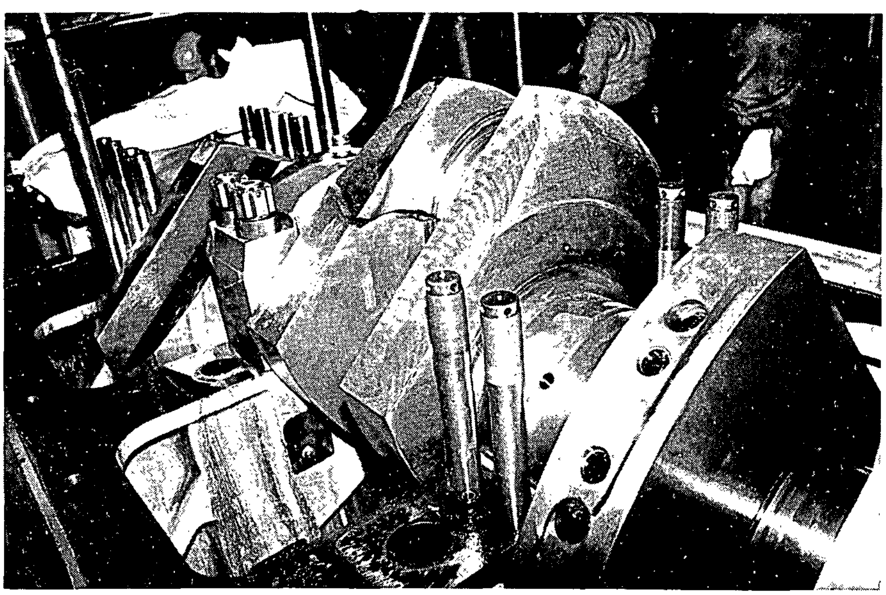
During fiscal year 1986, the staff completed its final evaluation of the Owners Group findings and recommendations and issued its final report as NUREG-1216 (August 1986). The staff concluded that implementation of the Owners Group and staff recommendations concerning quality revalidation inspections, component modifications and replacement, load and operating restrictions and precautions will establish the adequacy of the TDI diesel generators for nuclear standby service, as required by General Design Criteria 17 of Appendix A to 10 CFR 50. The staff further concluded that these actions would ensure that the design and manufacturing quality of the TDI engines are within the range normally assumed for diesel engines designed and manufactured in accordance with 10 CFR 50, Appendix B. Continued reliability and operability of the TDI engines for the life of the facilities will be ensured by implementation of an enhanced maintenance and surveillance program. The staff will evaluate each facility using TDI EDGs against the Owners Group and staff recommendations.

Pipe Cracks at Boiling Water Reactors

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

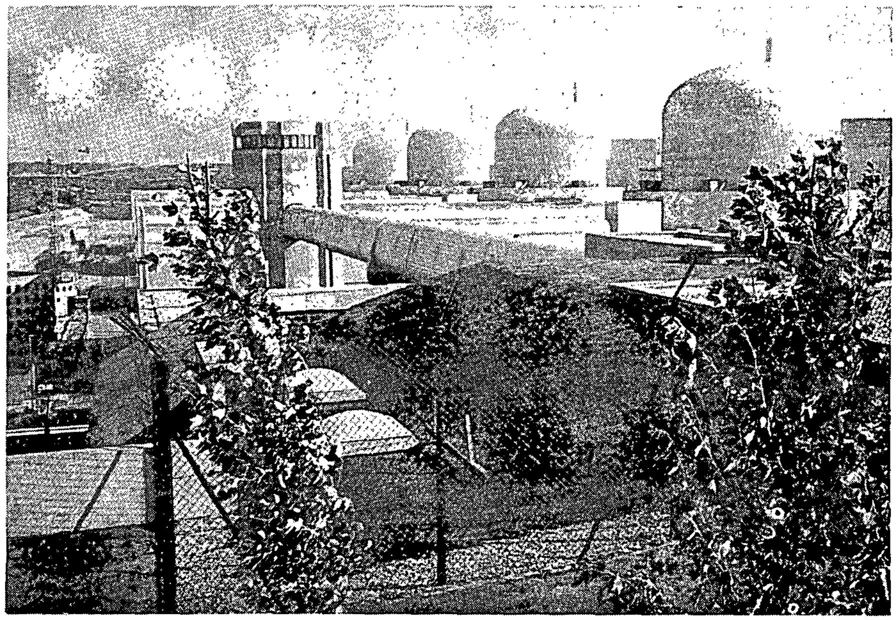
Some utilities have chosen to replace their degraded piping with more resistant material. This replacement has been completed at the following eight BWRs: Nine Mile Point Unit 1 (N.Y.), Monticello (Minn.), Pilgrim Unit 1 (Mass.), Hatch Unit 2 (Ga.), Cooper (Neb.), Peach Bottom Unit 2 (Pa.), Vermont Yankee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

The staff continued to pursue its long range plan for dealing with BWR pipe cracking during 1986. A draft Generic letter outlining the staff position regarding acceptable mitigative actions and a draft revision to NUREG 0313, the technical basis document, were issued for public comment on July 21, 1986. After the 60-day comment period, the comments were reviewed and evaluated. It was expected that all comments would be satisfactorily resolved and appropriate revisions made to these two documents by mid-November of 1986.



An NRC staff evaluation in 1986 of the findings of an industry group concluded that improved quality inspections, component modifications, and various restrictions and precautions applied to the design, manufacture and use of the Transamerica Delaval Inc. (TDI) diesel emergency generator will ensure more reliable operation thereof in the future. The main crankshaft of a TDI generator at the Shoreham nuclear power plant on Long Island, N.Y. (shown above) was cracked during a 1983 load test.

The NRC's continuing evaluation and comparison of U.S. PWR reactors with those of other countries included, in 1986, an exhaustive study of the French "standard design" P4 unit. Four such units are now operating at the Paluel plant on the channel coast, north of Le Havre.



After internal NRC review, the final resolution will be presented to the Commission for approval. It is expected that the resolution will be issued in final form early in 1987.

Generic Improvements Program For BWR Containments

Severe accident risk dominates regulatory concerns with public protection associated with transients at nuclear power plant. The fundamental objective of the Commission's Severe Accident Policy is to take all reasonable steps to reduce the chances of a severe accident and to assure the capability to mitigate the consequences of such an accident should one occur. Consistent with and pursuant to the Commission's policy to further reduce the chances of severe accidents and to mitigate their consequences, an industry initiative is under way to develop a methodology for Individual Plant Evaluation (IPE), designed to seek out risk "outliers" associated with a particular site.

While the IPE approach will satisfactorily address system reliability and containment performance for each plant specifically, several generic BWR containment enhancements have the potential for significantly mitigating the consequences of certain severe accident sequences. The generic approach has the advantage of expeditious implementation on all plants and will be responsive to the Commission's policy regarding mitigation of the consequences of severe accidents. At the close of the report period, the staff was developing a proposed Generic Letter requiring potential containment improvements. The proposed Generic Letter is scheduled to be issued in fiscal year 1987.

Comparison Between French And U.S. PWR Designs

In late 1982 and early 1983, the NRC performed a comparative analysis of the differences between the British Sizewell B design and current U.S. pressurized water reactor (PWR) technology, as typified by the SNUPPS design, with major emphasis on potentially significant safety improvements. The results of that evaluation were published in May 1983 as NUREG-0999. Because of the value of that activity and the continued interest of the NRC staff and the Advisory Committee on Reactor Safeguards in regulatory approaches to reactor safety in foreign countries, the NRC staff initiated a project in May 1985 to compare a typical French PWR design with a typical U.S. design.

Electricite de France (EDF), the French electric utility, operates about 30 "standard design" PWR units of the three-loop 900-MWe variety. In recent years, EDF developed a new four-loop 1300-MWe PWR design (called P4). Six of the P4 reactors are in operation in France, and about 12 more are under construction. The French also have developed an advanced design 1400-MWe plant (called N4), but a review of the N4 design was beyond the scope of this project.

Four of the six operating P4 plants are at the Paluel site in northern France. The Paluel plant includes a number of features that differ significantly from current U.S. PWR technology. Accordingly, the NRC staff reviewed the P4 design to identify and assess the safety significance of these design differences. This review entailed a visit to the Paluel plant and discussion with French safety authorities and EDF officials. The focus of the NRC review was on significant design differences, and associated emergency procedures.

In carrying out this project, the NRC staff reviewed relevant French reports to examine and assess differences between current French and U.S. PWR designs, the latter as typified by the Westinghouse standard design. Approximately 25 design differences were identified. Based on deterministic assessments, engineering judgment, and insights derived from generic probabilistic assessments, the staff evaluated the relative safety significance of these design differences. Two design features were judged to be of high relative safety significant compared to the other features: (1) an interconnection between the low pressure injection system and the containment spray system and associated mobile equipment, for long-term cooling following a loss-of-coolant accident; and (2) a small steam-driven electrical generator that provides power for reactor coolant pump seal cooling and selected instrumentation and controls in the event of a station blackout. These and other design differences are discussed in the final report of this project entitled "Analysis of French (Paluel) Pressurized Water Reactor Design Differences Compared to Current U.S. PWR Designs" (NUREG-1206, June 1986).

Protecting the Environment

Environmental Impact Assessment

The NRC staff prepared a draft and environmental impact assessment for South Texas Units 1 and 2 during fiscal year 1986. This action completes the environmental impact assessments for all reactors currently under review for operating licenses.

Antitrust Activities

As required by law since December of 1970, the staff has conducted pre-licensing antitrust reviews of all construction permit applications for nuclear power plants and certain other commercial nuclear facilities. In addition, applications for amendments to construction permits or operating licenses that transfer ownership interest or operating responsibility in a nuclear facility are subject to antitrust review. In fiscal year 1986, staff received two requests for amendments which required antitrust review. The review of one of the requests was completed, and it was concluded that no antitrust problems were associated with the amendment. The other request was still under review at the close of the fiscal year.

An application for an operating license is not subject to formal antitrust review unless the staff first determines that "significant changes" in the licensee's activities have occurred since the review of the application for a construction permit (see NUREG-0970, "Procedures For Meeting NRC Antitrust Responsibilities"). During fiscal year 1986, seven significant change analyses were completed. In each instance, the finding was that the changes that had occurred were not significant in an antitrust context. The staff received requests in two cases to re-evaluate the "no significant change" findings. After re-evaluation by the staff, one finding was affirmed; the other request was still under review as of September 30, 1986.

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

Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS), established by statute in 1957, provides advice to the Commission on potential hazards of proposed or existing eactor facilities and on the adequacy of proposed safety standards. The Atomic Energy Act of 1954 also requires that the ACRS advise the Commission with respect to the safety of operating reactors and perform such other duties as the Commission may request. Consistent with the Energy Reorganization Act of 1974, the Committee will also review any matter related to the safety of nuclear facilities specifially requested by the Department of Energy (DOE). Also, in accordance with Public Law 95-209, the ACRS is required to prepare an annual report to the U.S. Congress on the NRC Safety Research Program.

The ACRS reviews requests for pre-application site and standard plant approvals, each application for a construction permit or an operating license for power reactors, applications for licenses to construct or operate test reactors, spent fuel reprocessing plants, and waste disposal facilities.

Consistent with the statutory charter of the Committee, all ACRS reports, except for classified reports, are made part of the public record. Activities of the Committee are conducted in accordance with the Federal Advisory Committee Act, which provides for public attendance at and participation in Committee meetings. The ACRS membership, which is drawn from scientific and engineering disciplines, includes individuals experienced in metallurgical engineering, electrical engineering, mechanical engineering, structural engineering, reactor operations, reactor physics, and environmental health.

During fiscal year 1986, the Committee completed its annual report to Congress on the NRC Safety Research Program for fiscal year 1987, and its annual report to the Commission on the Safety Research Program and Budget for fiscal year 1988.

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

- Impacts of Natural Phenomena on Off-Site Emergency Response.
- Environmental Protection Agency Standards for a High-Level Radioactive Waste Repository.
- Consideration of Earthquakes in Off-Site Emergency Planning.
- NRC Incident Investigation Program.
- Potential Upgrade of Auxiliary Feedwater Systems in Certain Operating Plants.
- Definition of High-Level Waste.
- Reassessment of the Technical Bases for Estimating Source Terms.
- Requalification Programs for Licensed Power Reactor Operators.
- Protection Against Criticality During TMI-2 Defueling.
- Quantification of Public Health Risks.
- NRC Review of Advanced Reactor Designs.
- Definition of Low-Level Radioactive Waste.
- Support of Radiation Protection Organizations.
- Salvaging of Contaminated Smelted Alloys.
- Fire Protection Research and Fire-Related Systems Interactions.
- Support of Radiation Protection Organizations Such as the National Council on Radiation Protection and Measurements, The International Commission on Radiological Protection, and the National Academy of Sciences.
- Development of De Minimis Levels for Radiobiological Doses.
- Babcock & Wilcox Owners Group Safety and Performance Improvement Program.
- Tennessee Valley Authority's Management Reorganization and Shutdown of TVA's Nuclear Power Plants.
- Various Office of Nuclear Material Safety and Safeguards and Office of Nuclear Regulatory Research Waste Management Topics.
- Degraded Piping Research.

The Committee's activities during the report period reflected the continuing license activity within the Commission and included two reports on requests for nuclear power plant operating licenses, one report each on a request for a final design approval and a full-term license, a special report on the effect of a nearby earthquake on a licensed nuclear plant, and a report on a proposal to modify an operating license.

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

- Proposed NRC Advanced Reactor Policy Statement.
- Rulemaking for Revisions to Operator Licensing Requirements.
- Revision 3 to Regulatory Guide 1.63.
- Revision of 10 CFR Part 20, "Standards for Protection Against Radiation."
- Proposed Insider Safeguards Provisions.
- Implementation Plan for the Severe Accident Policy Statement and Regulatory Use of New Source Term Information.
- Guidelines for Boiling Water Reactor Coolant Pressure Boundary Piping.
- Broad Scope Rule Revision to General Design Criterion 4.
- Proposed NRC Safety Goal Policy.
- Resolution of USI A-17, "Systems Interactions in Nuclear Power Plants."

- Draft Commission Policy Statement on Technical Specifications.
- Proposed Revisions to Sections 9.2.1 and 9.2.2 of the Standard Review Plan.
- Emergency Planning Requirements for NRC Licensees.
- Proposed NRC Standardization Policy Statement.
- Regulatory Guide 1.114, Revision 2, "Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit."
- NRC Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel.
- Proposed Revision to the ECCS Rule in 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," and Appendix K, "ECCS Evaluation Models."
- Proposed Resolution of Generic Issue 124, "Auxiliary Feedwater System Reliability."
- Resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants."

The Committee commented in one report on the NRC Staff's proposed priority rankings for newly identified generic issues.

In order to improve its own functioning, the Committee evaluated the report of an outside panel of experts which appraised and made recommendations on the effectiveness of the ACRS. The Committee responded to these recommendations in a report to the NRC, and took a number of recommended actions, including the establishment of a subcommittee to set priorities and provide guidance for the activities of the Committee and its staff. This Planning Subcommittee has met seven times during this reporting period.

In performing the reviews and preparing the reports cited above, the ACRS held 12 full Committee meetings and 94 subcommittee meetings. Members of the Committee also participated in several conferences and visits to exchange safety-related information with foreign groups engaged in nuclear regulatory and development activities. A member of the Committee participated in the recent International Atomic Energy Agency briefing by representatives of the U.S.S.R. regarding the sequences and consequences of the accident at the Chernobyl Nuclear Station. (See discussion under "Safety Reviews," earlier in this chapter.)

In addition, the full Committee met with representatives of the Federal Republic of Germany, including the Reaktor-Sicherheitskommission, on May 9, 1986, in Washington, D.C., to discuss high- and low-level radwaste standards for radionuclide release limits (dose limits), plans for environmental monitoring of radwaste repositories, and modeling in connection with performance assessment of radwaste handling and disposal facilities to assure compliance with related standards and protection of the public health and safety.

During this year, the Committee became more actively involved in regulatory matters related to the processing and disposal of radioactive waste, at the request of the NRC. The Committee provided seven reports to the NRC during this reporting period on matters related to high- and low-level radwaste handling and disposal.

Cleanup at Three Mile Island

CHAPTER 3

Fiscal year 1986 was marked by the most significant progress yet in the cleanup of the damaged Unit 2 reactor at the Three Mile Island nuclear power plant (TMI-2) near Harrisburg, Pa., since the accident in late March 1979. Removal of damaged fuel and structural debris from the reactor vessel finally got under way in late October 1985, six and one-half years after the event.

Special defueling equipment was used to transfer core debris from the reactor vessel to safe temporary storage locations in the Unit 2 spent fuel pool. Shipment of the damaged fuel from the TMI site to the Idaho National Engineering Laboratory (INEL) also began during the fiscal year. A special drilling rig was used to take full-length core samples, which will be analyzed at INEL to provide data for future defueling planning and to develop a better understanding of the TMI-2 accident sequence and its applications.

NRC on-site staff continued to monitor the day-to-day cleanup operations conducted by the licensee, General Public Utilities Nuclear Corporation (GPUNC). The on-site staff performed numerous reviews and issued necessary approvals of the licensee's detailed defueling procedures, and conducted periodic inspections of plant cleanup systems and equipment. In conjunction with headquarters staff, the NRC TMI site staff performed safety and technical reviews of licensee proposals for major cleanup activities to continue to assure the safe, expeditious cleanup of Unit 2.

During fiscal year 1986, GPUNC performed additional video inspections in previously inaccessible regions of the reactor vessel. Decontamination and dose reduction activities were performed in parallel with defueling operations. Surveys and sampling activities were conducted in the reactor building basement, the pressurizer, and the steam generators to measure radiation fields and quantify the radioactive debris in those areas. Seven of eight reactor vessel internal vent valves were removed to provide improved access to the lower vessel head. The processing and shipment of radioactive wastes also continued, primarily in support of defueling operations.

Although considerable progress was made during the fiscal year in defueling the TMI-2 reactor vessel, some operational difficulties were encountered. GPUNC employed numerous techniques, with varying degrees of success, to combat the growth of microorganisms in the reactor coolant, which at times seriously restricted visibility in the vessel. Also, certain defueling tools could not be used to perform the intended functions, particularly those designed to break up the hard mass of fused core debris. The licensee has been able to develop new tools and techniques to resolve the difficulties encountered to date and to allow defueling to continue. Because of the delays incurred as a result of these problems, and in light of a more accurate assessment of the nature and extent of the remaining defueling tasks, the licensee has slightly revised the schedule with regard to attainment of certain cleanup milestones. Defueling activities are projected for completion in the fourth quarter of calendar year 1987, representing a three-month adjustment to the schedule projected one year ago. The completion of the current phase of the cleanup is still estimated to occur by the third quarter of calendar year 1988.

The cleanup funding situation remained stable during fiscal year 1986, with committed sources in place to fund the estimated total cost of $965 million. Through the end of the fiscal year, approximately 700 million dollars had been spent on the cleanup, leaving a total of 265 million dollars for remaining cleanup expenses. (See discussion of the financial aspects of the cleanup at the end of Chapter 9).

TMI-2 Defueling Scheme

The licensee has designed and installed unique equipment and systems to accomplish the primary goal of the TMI-2 cleanup: the removal of the damaged fuel and structural debris from the reactor vessel. During defueling activities, the reactor coolant system (RCS) is vented to the reactor building atmosphere, with RCS cooling by natural heat loss. The internals indexing fixture (IIF), installed over the open reactor vessel and filled with water to an elevation of five feet over the vessel flange, provides additional radiation shielding for defueling workers. The RCS is borated to a concentration of approximately 5000-parts-per-million to prevent recriticality of the damaged fuel in any configuration. The Defueling Water Cleanup System (DWCS) is used to process reactor vessel water to reduce activity levels and to provide the visibility necessary to conduct the remote defueling operations.

Workers perform defueling operations from a shielded defueling work platform (DWP), which is located at a height of nine feet above the reactor vessel flange, over the IIF. The platform has a rotatable 17-foot diameter surface with six-inch steel shield plates and is designed to provide access for defueling tools and equipment into the reactor vessel. The DWP supports defueling operators, especially designed long-handled tools, remote viewing equipment, and two jib cranes used to manipulate the tools. Numerous manual and hydraulically powered long-handled tools are used to perform a variety of functions, such as pulling, grappling, cutting, scooping and breaking up the core debris. These tools are used to load debris into defueling canisters positioned under water in the reactor vessel. The canisters are then sealed and transported using shielded canister transfer equipment to submerged storage racks in spent fuel pool "A" of the Auxiliary and Fuel Handling Building (AFHB). The canisters are designed and stored to prevent an inadvertent criticality event. Following dewatering to control the buildup of combustible gases, the canisters are loaded into a specially designed shipping cask and transported to a Department of Energy facility in Idaho for interim storage.

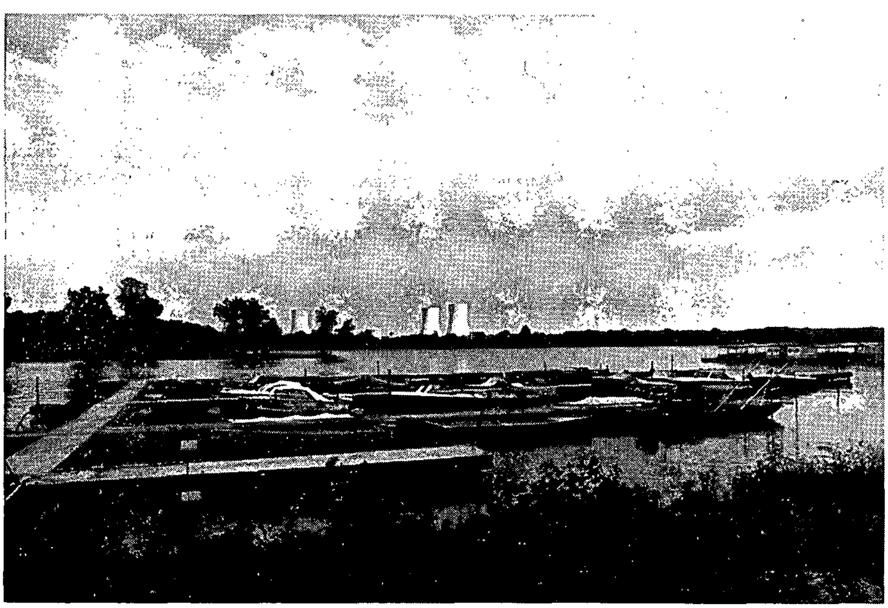
Reactor Vessel Defueling Activities

A total of 345 entries were made into the TMI-2 reactor building during fiscal year 1986, bringing the total number of entries made since the March 1979 accident to 1,047. Entries made during the fiscal year were primarily for installation and operation of defueling tools and equipment and defueling support activities.

During the first quarter of fiscal year 1986, GPUNC completed preparations for defueling and commenced preliminary defueling operations. Initial in-vessel activities involved the relocation of structural debris to allow the installation of the canister positioning system—a submerged, rotating carousel device capable of holding five defueling canisters. In December 1985, several defueling canisters were filled with debris consisting of fuel assembly end fittings, control rod spiders, and small pieces of fuel assemblies. In early January 1986, the first group of defueling canisters was sealed, dewatered, and transferred to storage racks in spent fuel pool "A" in the AFHB.

Dose rates to personnel during the initial phase of defueling were low and remained low throughout the year, averaging less than 10 mrem/hr on the DWP and less than 40 mrem/hr near the shielded canisters during transfer. The licensee discontinued the use of respirators during defueling activities, based on air sample data collected during the first month.

"Pick and place" defueling of the loose TMI-2 core debris continued through April 1986. Nearly 16 percent of the estimated total of 308,000 pounds of debris was removed from the reactor vessel before poor visibility temporarily halted defueling operations. A large population of microorganisms had rapidly developed in the RCS, clogging the DWCS filters and hindering the operators' ability to view remotely the defueling activities in the vessel. These growths, consisting of algae, fungi, bacteria, and aerobic and anaerobic organisms, proved difficult to kill in several tests. In April and May, GPUNC conducted a multi-phase program to restore reactor vessel water clarity. The program consisted of high pressure hydro-lancing to remove growths adhering to reactor vessel surfaces, the addition of hydrogen peroxide as a biocide, and the use of a high pressure positive displacement pump to kill the microorganisms. A diatomaceous earth (swimming pool-type) filter was then operated in conjunction with the letdown and makeup of batches of reactor coolant, to remove the organic material and improve the clarity of the RCS water. These techniques proved successful in restoring visibility in the vessel and were repeated as necessary to maintain water clarity throughout defueling activities for fiscal year 1986. Pick and place defueling was resumed in May, following the completion of the water treatment program.



This scene of the Goldsboro Marina on the bank of York Haven Pond shows one of the primary access points for fishermen working the Susquehanna River. The marina is just west of the Three Mile Island nuclear power plant, shown in the background.

In July 1986, the licensee conducted a core stratification sample acquisition program. Most of the loose core debris had been removed from the reactor vessel, and more data were needed to plan the defueling of the material under the hard crust layer of the damaged core. A special drilling rig was assembled on top of the DWP, and 10 full-length sampling penetrations were made from the surface of the debris bed to inches above the lower head of the reactor vessel. These samples of the reactor core (approximately 2.5 inches in diameter and eight feet long) will be analyzed at INEL, along with earlier samples of debris collected from the lower vessel head, in order to provide data on the material properties of the core debris. Video inspections of the core below the debris bed were performed through several of the bore holes created by the drilling operations. Initial inspections indicated that peripheral fuel assemblies are essentially intact below the rubble bed (or "hard crust" layer), but that the central core region consists largely of a fused mass of material.

The core drilling apparatus was modified in late July when solid face drill bits were used to perforate the hard crust layer of the core in 48 locations. These perforations, ranging in depth from a few inches to 48 inches, were made to improve the effectiveness of heavy duty defueling tools in breaking up the solidified core debris. The heavy duty tools were only marginally successful, and so the drilling rig was reinstalled at the end of fiscal year 1986, to be used as the primary tool for breaking up the hard mass of core debris. Remaining fuel assembly end fittings were removed from the top of the debris bed to clear the area for further drilling operations.

By the end of the fiscal year, approximately 57,000 pounds of core debris had been removed from the TMI-2 reactor vessel, representing nearly 19 per cent of the estimated total of 308,000 pounds in the vessel.

Waste Management

During fiscal year 1986, the Submerged Demineralizer System (SDS) and the EPICOR-II system continued to be used to process radioactive water in support of cleanup activities. The SDS was primarily used to process reactor coolant and water from the deep end of the fuel transfer canal. The EPICOR-II system was used mainly to polish effluent from the SDS and to process water from the chemical cleaning building sump. The SDS and EPICOR-II systems processed approximately 1,252,000 and 490,000 gallons of water, respectively, during the fiscal year. Twenty EPICOR-II dewatered liners were shipped to Richland, Wash., for burial during the same period.

Late in 1985, water in the fuel transfer canal, spent fuel pool "A", and miscellaneous processing tanks was treated with hydrogen peroxide to kill algae growths in those volumes. These growths were unrelated to the microorganisms later found in the RCS.

In July 1986, the licensee began operation of the newly constructed Waste Handling and Packaging Facility. The facility is used to process the increased volumes of low-level solid waste generated as a result of defueling operations. Activities conducted in the facility include sectioning, disassembly, and other size reduction operations; mechanical decontamination of equipment and tools; and packaging of solid wastes in 55 gallon drums and low specific activity boxes.

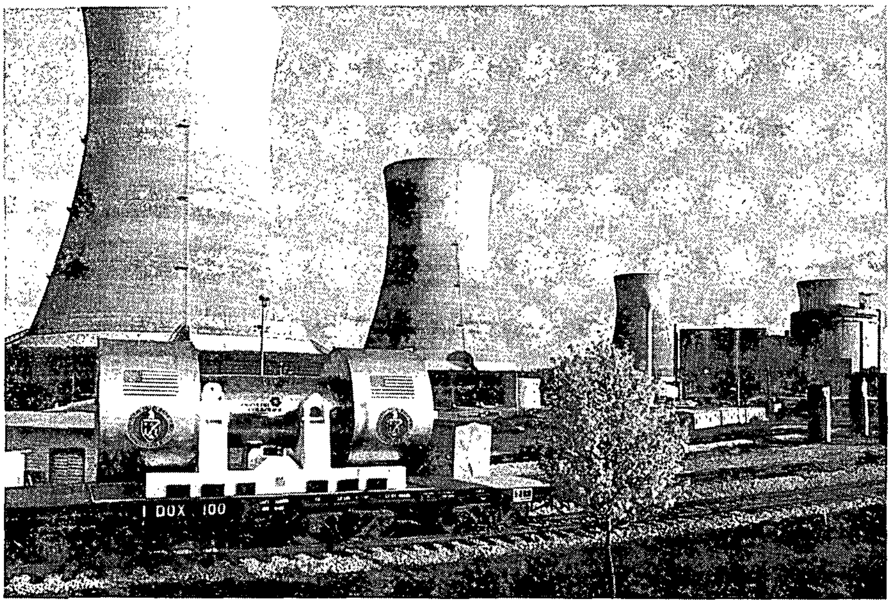
Also in July, GPUNC submitted a proposal for disposing of approximately 2.1 million gallons of slightly radioactive water, contaminated during the accident and used in subsequent cleanup operations. Of the proposed alternatives, the licensee requested approval of a method involving the forced evaporation of the water at the TMI site over a two and one-half year period. The residue from this operation, containing small amounts of the radioactive isotopes cesium-137 and strontium-90, and large volumes of boric acid and sodium hydroxide, would require solidification and disposal as low-level waste.

The licensee has petitioned the Secretary of Energy for the additional burial ground waste volume allocation necessary to implement this plan. The NRC staff was reviewing the licensee's proposal at the close of the report period and will make a recommendation to the Commission, whose approval is required prior to the initiation of any disposal option.

The first off-site shipment of fuel and debris removed from the damaged TMI-2 core took place in July of 1986. Under a previous agreement with the NRC, the Department of Energy (DOE) will take possession of the high-level waste at the TMI site boundary and is responsible for transport of the material and interim storage at the Idaho National Engineering Laboratory. In April 1986, the NRC issued certificates of compliance for the two Nuclear Packaging shipping casks to be used for shipment of the fuel debris by rail. Each cask is designed to hold seven defueling canisters; therefore, an estimated 35 to 40 trips will be necessary to ship all the TMI-2 core debris to INEL. Two additional casks were later shipped from the TMI site, so that by the end of the fiscal year, approximately 4 percent (12,000 lbs.) of the total estimated core debris had been transferred to INEL.

Decontamination and Dose Reduction

Throughout the fiscal year, GPUNC continued to perform decontamination and dose reduction activities aimed at maintaining worker radiation exposures as low as reasonably achievable (ALARA). Scabbling (the mechanical removal of a layer of concrete), water flushing, vacuuming, painting, and hands-on techniques such as wiping and scrubbing were the primary methods used to decontaminate areas in the reactor building and the AFHB. Decontamination efforts during the year helped to maintain low average



A U.S. Department of Energy (DOE) shipping cask awaits loading at the Three Mile Island nuclear power plant. The cask, licensed by the NRC, is used to transport debris from the damaged TMI reactor core for interim storage at DOE's Idaho National Engineering Laboratory, near Pocatello, Idaho.

dose rates at the 305-foot and 347-foot elevations in the reactor building (67 mrem/hr and 40 mrem/hr, respectively), and resulted in freeing over 90 percent of the area at the 281-foot elevation in the AFHB from contamination controls.

In conjunction with ongoing decontamination activities, sampling and surveys were performed in areas of the reactor building and the AFHB. Video inspections and thermoluminescent dosimeter surveys conducted in the pressurizer indicated that little fuel was deposited there as a result of the accident. Small quantities of particulate material were discovered in the steam generator upper head spaces. A robot vehicle was used to collect concrete samples of the highly radioactive reactor building basement, where dose rates typically remain in excess of 100 rem/hr, for the purpose of planning decontamination of basement surfaces. A robot device was also used to measure the high dose rates in the AFHB Seal Injection Valve Room. The measured dose rates due to gamma radiation in the room ranged from 30-75 R/hr in general areas to 300 R/hr in hot spots. The defueling work platform continued to be the lowest dose rate area in the reactor building because of special dose reduction efforts and shielding. Dose rates on the DWP averaged 8 mrem/hr for most of the fiscal year.

Advisory Panel on TMI Cleanup

The Advisory Panel for the Decontamination of Three Mile Island Unit 2—composed of citizens, scientists, and state and local officials—was formed by the NRC in 1980 to provide input to the Commission on major cleanup issues (see Appendix 2 for a list of current members). During fiscal year 1986, the panel held five public meetings in Harrisburg and Lancaster, Pa., and met twice with the NRC Commissioners in Washington, D.C. Topics addressed by the panel during the year included: TMI-2 health effects studies presented by the Pennsylvania Department of Health and local citizens, status of the ongoing defueling operations, Department of Energy plans for off-site shipment and storage of fuel, the licensee's proposal for disposal of the accident-generated water, and ongoing NRC enforcement actions.

Operational Experience

CHAPTER 4

ANALYSIS AND EVALUATION OF OPERATIONAL DATA

The NRC Office for Analysis and Evaluation of Operation Data (AEOD) was established in 1979, in one of the Commission's earliest major steps toward improving the ways in which licensee operating experience can be used to identify and resolve problems with potential safety-related implications. The Office, which reports directly to the Executive Director for Operations, carries out the collection, assessment and feedback of operational data to the NRC, the nuclear industry and the public.

AEOD's focus and role in the program is to provide a strong capability for the analysis of operating experience, independent of routine regulatory activities associated with licensing, inspection or enforcement, and to disseminate the lessons learned through effective channels.

The Office accomplishes its mission through analysis and evaluation of operational safety data associated with all NRC-licensed activities. These include the operations of commercial power reactor licensees and radioactive material and fuel cycle licensees. The Office also coordinates the overall NRC operational data program and serves as the focal point for interaction with outside and foreign organizations performing similar work.

Among AEOD's specific activities are the following:

- Screen and analyze U.S. and foreign operational events, seeking indications of potential safety problems, issuing and tracking recommendations for action by other NRC offices.
- Conduct the NRC trends and patterns program and communicate broadly the facts and significance of operating experience.
- Develop and coordinate agency guidance on Licensee Event Report requirements and monitor the effectiveness of the Nuclear Plant Reliability Data System.
- Develop and maintain computerized storage and retrieval systems for reactor and non-reactor operational data, including foreign data.
- Develop and support the procedures for and establishment of Incident Investigation Teams.
- Prepare and coordinate the quarterly Report to Congress on Abnormal Occurrences (NUREG-0090 series), as required by Section 208 of the Energy Reorganization Act of 1974; the bi-monthly Power Reactor Event reports (NUREG/BR-0051 series); the monthly Licensee Event Report Compilation (NUREG/CR-2000 series); and other feedback documents.
- Prepare reports of U.S. events for transmittal to the Nuclear Energy Agency's Incident Reporting System.
- Serve as principal point of contact with the Advisory Committee on Reactor Safeguards (ACRS), the Institute of Nuclear Power Operations (INPO), and the Nuclear Safety Analysis Center (NSAC) on matters involving the collection and evaluation of operational data.

NRC Handling of Operational Data

Domestic. On January 1, 1984, the Licensee Event Report (LER) rule (10 CFR 50. 73) became effective (See *1985 NRC Annual Report,* p. 61). The rule requires that, in its event report, the licensee describe in a reasonably complete and detailed manner all actuations of engineered safety features (ESF), including scrams (reactor shutdown); all losses of safety function at a system level; all significant systems interactions; all technical specification violations; and all significant internal and external threats to plant safety. AEOD estimated in 1983 that the number of LERs would probably decrease by 50 percent following adoption of the new rule, but that twice as much information would be provided in each LER. Thus, no change was anticipated in the licensee resources required to implement the LER rule. This estimate of the number of LERs continued to be substantially correct during fiscal year 1986.

Under contract with the Nuclear Operations Analysis Center (NOAC) at Oak Ridge, Tenn., AEOD operates and maintains the Sequence Coding and Search System (SCSS), a computerized storage and retrieval system for LER data. Its function is to encode all the relevant technical information provided by the licensee in the LER with enough "tags" to assure ready retrieval of individual items. During fiscal year 1986, about 6,100 LERs were added to the system. This figure includes those LERs provided by the licensees during the fiscal year, as well as LER data added from 1980, which was completed by the end of the fiscal year. This latest increase brought the number of LERs added to the data base (since 1980) to more than 22,500. During the report period, SCSS was made directly accessible to over 15 additional users, making a total of over 60 authorized users of the data base.

Under a "trends-and-patterns program," the NRC periodically conducts an in-depth analysis of sets of operational event data reported by commercial power reactor licensees in the LERs and through INPO's Nuclear Plant Reliability Data System (NPRDS). The program is designed to detect, through statistical and engineering analysis, those trends or patterns in incidents of low individual significance that may indicate an unrecognized safety concern. Based on the more detailed data provided under the LER rule, the 1986 program included four studies focused on events covered by the requirements for reporting reactor trips, ESF actuations, systems unavailability, and technical specifications, respectively. Two of these reports, as well as a trends-and-pattern analysis of operatoinal experience at new plants, were issued in August 1986. (These three reports are summarized later in this chapter.) During fiscal year 1986, the NRC also implemented a trends-and-patterns analysis of component level data using NPRDS data.

INPO maintains the NPRDS, industry's component failure data base, which is a voluntary initiative approved in lieu of a regulatory program. For that reason especially, the Commission has requested that a continuing NPRDS evaluation program be carried out by the NRC staff. Two semiannual evaluation reports on NPRDS progress were forwarded in January 1986 (SECY-86-35) and July 1986 (SECY-86-216). In these reports it was noted that there has been a substantial improvement in the percentage of NPRDS-reportable failures that have been submitted to the data base, and that INPO's estimate of the expected number of failure reports, when all plants are fully participating, has essentially been achieved. However, the NRC staff is concerned about the possible preferential reporting of failures documented in LERs, which could compromise the validity of the planned end-uses of the data base. The staff also feels that there still is the need for improvements in the timeliness and quality of the data. Still, the NRC staff believes that the current levels of participation in the NPRDS warrant increased use of the system as a source of reliability data.

Foreign. In fiscal year 1986, the NRC continued efforts to increase the number and usefulness of foreign experience reports that are received. By means of its program at NOAC, the NRC continues to systematically screen and assess selected foreign information for its applicability to the U.S. program and to abstract it for computerized data filing. This file now contains information on more than 6,600 foreign events.

The agency also continued it participation in the exchange of operational event information with other countries through activities involving the Nuclear Energy Agency (NEA), the International Atomic Energy Agency (IAEA), and various bilateral agreements. In September 1986, the NRC participated in the annual IAEA/NEA meetings. A number of significant technical papers were presented, and a number of events were identified with sufficient relevance to U.S. reactor operations to warrant further study. The NRC will continue to take an active part in efforts to improve the International Reporting System, in effect since the late 1970s, giving emphasis to the actual significance of reported events and to the overall quality of the reports exchanged.

In April 1986, the NRC also participated in organizing the Symposium on Reducing Reactor Scram Frequency, which was sponsored by the NEA and held in Tokyo, Japan.

Incident Investigation Program

The Incident Investigation Program (IIP) was established by the NRC's Executive Director for Operations (EDO) and approved by the Commission to assure that the NRC's investigation of significant events would be timely, thorough, well coordinated and formally administered. The scope of the IIP includes the investigation of significant operational events involving reactors and non-reactor activities licensed by the NRC. The primary objective of the IIP is, in general, to ensure that operational events are investigated in a systematic and technically sound manner, and, specifically, to gather all available information pertaining to the causes of the events—including those involving the NRC's activities—and to provide appropriate feedback regarding the lessons of the events to the NRC, the industry and the public.

With its focus on the causes of operating events and the identification of associated corrective actions, the IIP process contributes to nuclear safety by providing for a complete technical and regulatory understanding of significant events. The IIP generates two investigatory responses based on the safety significance of the operational events. Both are provided by an NRC team put together to determine the circumstances and causes of an operational event. For an event of potentially major significance, an Incident Investigation Team (IIT) is established by the EDO to investigate it. The investigation of less significant operational events is conducted by what are designated Augmented Inspection Teams (AITs), which consist of a Regional-directed team complemented by Headquarters personnel and, in some cases, by personnel from other Regions. (See Chapter 8 for further discussion of AITs.)

IIT at San Onofre Unit 1. On November 21, 1985, the San Onofre Unit 1 (Cal.) nuclear power plant underwent a partial loss of in-plant a.c. electricity while the reactor was operating at 60 percent power. Following a manual reactor-trip, the plant lost all a.c. power for four minutes and experienced severe water hammer in the feedwater system. This incident caused a leak, damaged plant equipment, and tested the integrity of the plant's heat sink. Because of the potentially important safety implications of this event, the NRC's Executive Director set up an IIT to look into it.

Results of the San Onofre IIT investigation were issued in January 1986 (NUREG-1190). The most significant aspect of the investigation was the discovery that five safety-related check valves in the feedwater system had failed. These failures had occurred in less than a year, without detection, and had jeopardized the integrity of plant safety systems. The underlying causes of the event included issues that were both plant-specific and generic.

IIT at Rancho Seco. On December 26, 1985, the Rancho Seco (Cal.) nuclear power plant underwent a loss of d.c. power within the integrated control system (ICS) while the plant was operating at 76 percent power. Following the loss of ICS d.c. power, the reactor tripped automatically on high reactor coolant system (RCS) pressure followed by a rapid overcooling transient and automatic initiation of the safety features actuation system on low RCS pressure. The overcooling transient continued until ICS d.c. power was restored 26 minutes after its loss. The event involved a number of equipment failures and personnel errors.

An AIT was sent to the site on December 27 and, based on initial observations, decided that the event was more complicated than it had at first seemed, and that it bore potentially significant generic implications. Consequently, on December 30, the EDO directed that the investigation be upgraded to an IIT dispatch. The results of the IIT activity were issued as NUREG-1195.

Some general conclusions deriving from these two IIT investigations were: (1) in each of the events, the lessons of experience had not been and were not being aggressively sought out, thoroughly assessed, and acted upon by the licensees involved; (2) operating events can impose a significant burden on the operating personnel on duty; (3) maintenance and test practices may not provide sufficiently high confidence that all plant equipment will perform reliably and predictably in off-normal situations; and (4) training, plant procedures, and instrumentation may not be providing all of the knowledge and guidance needed by the operators.

These IIT events were also classified as abnormal occurrences. Detailed descriptions of the events, their consequences, and subsequent corrective actions are provided in the "Abnormal Occurrences" section later in this chapter.

Analysis of Non-Reactor Operational Experience

In addition to the screening and analysis of reactor operating experience, the AEOD reviews the non-reactor operational experience associated with the activities and facilities licensed by the Office of Nuclear Material Safety and Safeguards (see Chapter 5) and by the Agreement States (see Chapter 9.) AEOD also conducts studies from a human-factors perspective on both reactor and non-reactor operational events, and maintains and updates the non-reactor and medical misadministration data files. From the events screened during fiscal year 1986, about 150 non-reactor events and 400 medical misadministrations were entered into the files. (See further discussion below of non-reactor data.)

Annual Report to the Commission

In April 1986, the Office submitted an annual report to the Commission (AEOD/S601) for calendar year 1985. (The two previous reports to the Commission had been submitted semiannually.) Based on its extensive screening, analysis and dissemination of operational experience, AEOD offered a number of salient observations in the report, among them the following:

- Based on an analysis of operational experience feedback programs, it can be concluded that most plant managers are making moderate, not extensive, use of their own in-house operating experience and, in general, are making less use of the large body of knowledge deriving from events and concerns that originate elsewhere in the industry. In addition, the large volume and diversity of operational experience feedback may be detracting from the effectiveness of the feedback programs.

- Despite substantial programs to learn from operating experience, and major efforts to improve plant operations and personnel proficiency, a meaningful decrease in the number and rate of abnormal occurrences has not been realized; the number of significant occurrences remains relatively constant.

- There is still a rather wide disparity in the quality of the LERs submitted by various licensees; feedback through the Systematic Assessment of Licensee Performance (SALP) process should improve the quality of LERs submitted by all licensees. (See Chapter 8 for a description of SALP.)

- In the last decade, at least eight operational events at U.S. BWRs involved actual or potential overpressurization of an emergency core cooling system. Each such event can be considered a precursor to a loss-of-coolant accident outside containment; collectively, these events indicate that the likelihood of an "interfacing loss-of-coolant accident" is higher, by at least two orders of magnitude, than had previously been assumed.

- Reactor scram frequency was about the same in 1985 as in the preceding year. However, plants showing a decreased scram rate outnumbered those showing an increase by approximately two to one. In addition, the maximum scram rate about 15 percent power decreased, and new plant operations, in the early months following initial criticality, typically underwent a higher scram rate than the older plants.

- A review of non-reactor events reported in 1985 showed that the number and type of reported events did not differ substantially from those received in other years. Most of the non-reactor abnormal occurrences reported in 1985 that reflected actual or potential health effects resulted from radiography operations.

Table 1. AEOD Reports Issued During FY 1986

Case and Special Studies Designation	*Subject*	*Issued*
C503	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	12/85
C504	Loss of Safety System Function Events	12/85
C505	Therapy Misadministrations Reported to NRC Pursuant to 10 CFR 35.42	12/85
C601	Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center	8/86
C602	Operational Experience Involving Turbine Overspeed Trips	8/86
P601	Trends and Patterns Program Plan—FY 1986-FY 1988	1/86
P602	Trends and Patterns Report of Unplanned Reactor Trips at Light Water Reactors in 1985	8/86
P603	Trends and Patterns Report of Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Plants	8/86
P604	Trends and Patterns Analysis of the Operational Experience of Newly Licensed U.S. Nuclear Power Plants	8/86
S601	AEOD Annual Report for 1985	4/86
S602	An Overview of Nuclear Power Plant Operating Experience Feedback Programs	5/86
S603	Adequacy of the Scope of IE Bulletin 86-01	6/86

Non-Reactor Case Studies

During fiscal year 1986, the AEOD issued two non-reactor case studies and two survey reports. The staff also served on an inter-office Lessons-Learned Group to report on safety implications of an event involving the rupture of an overfilled cylinder containing uranium hexafluoride. That event occurred in January 1986 at the Sequoyah Fuels Corporation facility in Gore, Okla., and resulted in the death of one plant worker and injuries to several others. The Group issued its final report (NUREG-1198) in June 1986. (This event is described further under "Abnormal Occurrences," later in this chapter; see also Chapter 2.)

The two non-reactor case studies issued during the report period are discussed below.

Therapy Misadministrations Reported to the NRC. This case study (C505), issued in December 1985, is a detailed review of 16 teletherapy misadministrations and two brachytherapy misadministrations reported between November 1980 and July 1984. Some of the findings of the study are as follows:

- Of the 16 teletherapy misadministrations reviewed, 12 could have been prevented by improved patient chart reviews or, in most cases, by independent verification of patient dose calculations.
- Adverse patient reactions were factors in prompting licensee personnel to review treatment plans in only three of the misadministrations cases, despite the fact that in at least six cases the administered dose exceeded the prescribed dose by over 50 percent.
- Many facilities may not have the quality assurance programs that are consistent with recommendations of those professional medical groups involved with radiation therapy.

Table 2. Reactor Engineering Evaluations and Technical Reviews

Engineering Evaluation	*Subject*	*Issued*
E514	Core Damage Precursor Events at Trojan	10/85
E515	Inadvertent Actuation of Safety System Due to Cross-Talk	12/85
E601	Deficient Operator Actions Following Dual Function Valve Failures	1/86
E602	Unexpected Criticality Due to Incorrect Calculation and Failure to Follow Procedures	1/86
E603	Delayed Access to Safety-related Areas During Plant Operation	2/86
E604	Spurious System Isolations Caused by Panalarm Model 86 Thermocouple Monitor	3/86
E605	Lightning Events at Nuclear Power Plants	4/86
E606	Loss of Safety Injection Capability at Indian Point Unit 2	5/86
E514	Core Damage Precursor Event at Trojan-Revision 1	5/86
E607	Degradation or Loss of Charging Systems with Swing Pump Designs	7/86
E608	Re-examination of Water Hammer Occurrences	7/86
E609	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	8/86
E610	Loss of LPCI Look Selection Logic at Millstone Unit 1	8/86

Technical Review	*Subject*	*Issued*
T512	Incorrect Plugging of Steam Generator Tubes	10/85
T513	Flooding of Safety-Related Valves in Pits	11/85
T514	Potential Loss of Component Cooling Water Due to Maladjustment of Relief Valves	11/85
T515	Residual Heat Removal Service Water Booster Pump Air Binding at Brunswick Unit 1	12/85

Table 2. Reactor Engineering Evaluations and Technical Reviews (continued)

Technical Review	*Subject*	*Issued*
T516	HPCI Overspeed Trip Loss Events and Subsequent Damage Due to Water Hammer	12/85
T601	Pressure Sensitive Temperature Switch Results in Spurious Actuation of Fire Suppression System	1/86
T602	Emergency Diesel Generator Cooling Water System Design Deficiencies at Maine Yankee and Haddam Neck	4/86
T603	Inadvertent Pump Suction Transfer and Potential Auxiliary Feedwater Pump Cavitation at Davis-Besse	4/86
T604	Events Resulting from Deficiencies in Labeling and Identification Systems	5/86
T605	Failure of Main Steam Safety Valves to Properly Reseat	6/86
T606	Inadvertent Recirculation Actuation Signals at Combustion Engineering Plants	8/86
T607	Occurrence of Events Involving Wrong Unit/Wrong Train/Wrong Component—Update through June 1986	9/86

Several recommendations are proposed, including communicating the contents of the report to the affected licensees, and encouraging and supporting the initiation of a voluntary industry-directed physical quality assurance program for radiotherapy facilities.

Rupture of an Iodine-125 Brachytherapy Source at the University of Cincinnati Medical Center. This case study (C601), issued in August 1986, is a detailed review of a unique incident that involved the reuse of high activity (3040 millicurie) iodine-125 seeds for brachytherapy treatment, i.e., treating several patients with the same set of seeds. The staff's review and analysis of the incident was undertaken to determine whether there is a generic problem with the reuse of high activity iodine-125 seeds, and to assess any associated health and safety problems. The staff concluded that (1) the risk of iodine-125 seed rupture is relatively high when the seeds are reused for several patients; (2) the consequences of this incident could have been mitigated by adequate radiation surveys of the work area, or by performance of a leak test of the seeds; and (3) a similar event could also be mitigated by radiation safety procedures designed to promptly detect a seed rupture and prevent excessive personnel exposure and facility contamination.

Each of the two survey reports described below was produced during fiscal year 1986 and contains a review of all 1985 events, as well as a five-year assessment of the 1981-1985 events.

Report on 1985 Non-Reactor Events. This report (N601), issued in June 1986, supports the conclusion that, when the events of 1981-1985 are reviewed, the number and categories of events reported (i.e., exposures, lost or stolen materials, etc.) do not vary appreciably from year to year. No marked trend in event occurrence was apparent from a statistical standpoint. A review of the 97 abnormal occurrence reports from 1981 through the third quarter of 1985 indicated that 37 of them were non-reactor events. Of these, 19 were overexposures, of which 12 were received in conjunction with industrial radiography operations.

There have been several events over the five-year period that resulted from the accidental contamination of steel, most notably the 1984 case involving radioactive steel from Mexico. These events sensitized the staff to the need to collect information on events in which radioactive material was found in—or might foreseeably be introduced into—consumer products. Five reports in this category occurred in 1985. Continuing collection of such reports provide the basis for an analysis and evaluation of their significance.

Medical Misadminstrations Reported for 1985. This report (N602), issued in June 1986, discloses that the number of therapy misadministrations reported annually from 1981 through 1985 varied from four to 12. The most frequent type of therapy misadministration involved teletherapy, ranging from nine in 1984 to one in 1985. The highest

Table 3. Non-Reactor Evaluations

Designation	*Subject*	*Issued*
N601	Report on 1985 Non-reactor Events and Five-Year Assessment for 1981-1985	6/86
N602	Medical Misadministrations Reported for 1985 and Five-Year Assessment of 1981-1985 Reports	6/86

number of events occurred in 1981 (10) and 1984 (12); there is no discernible reason for the unusually high rate of incidence in these years. For diagnostic misadministrations, the annual number reported from 1981-1985 varied from 430 in 1981 to 334 in 1983.

The total number of diagnostic misadministrations reported for each of the five years shows a downward trend. When the data are broken down by type, it is revealed that two types of misadministrations—"wrong radiopharmaceutical" and "wrong patient"—comprise over 90 percent of the total reported in any given year. In most years, these two types represent 95 percent to 98 percent of reported events. The causes reported by licensees are generally simple errors associated with (1) preparation of radiopharmaceuticals, (2) processing of nuclear medicine requisitions, and (3) patient identification.

One significant category of diagnostic misadministration is that involving millicurie doses of iodine-131. In these events, the administered dose is in the dose range that would be normal for certain kinds of therapy. Hence, the unintentional risk to the patient is comparable to the risk associated with intended iodine-131 therapy. Events in this category of misadministration occur at the rate of about three-per-year.

The annual number of diagnostic misadministrations in which the wrong pharmaceutical was used did not vary much over the five-year period, ranging from 242 in 1982 to 294 in 1985. There does not appear to be any marked trend in the number of these events reported annually. In contrast to this type of diagnostic misadministration, the number of "wrong patient" events did vary widely, ranging from 121 in 1981 to 67 in 1985. The annual number of these events showed a decrease over time, and that is evidently the reason that the total number of diagnostic misadministration reports shows a downward trend over the period.

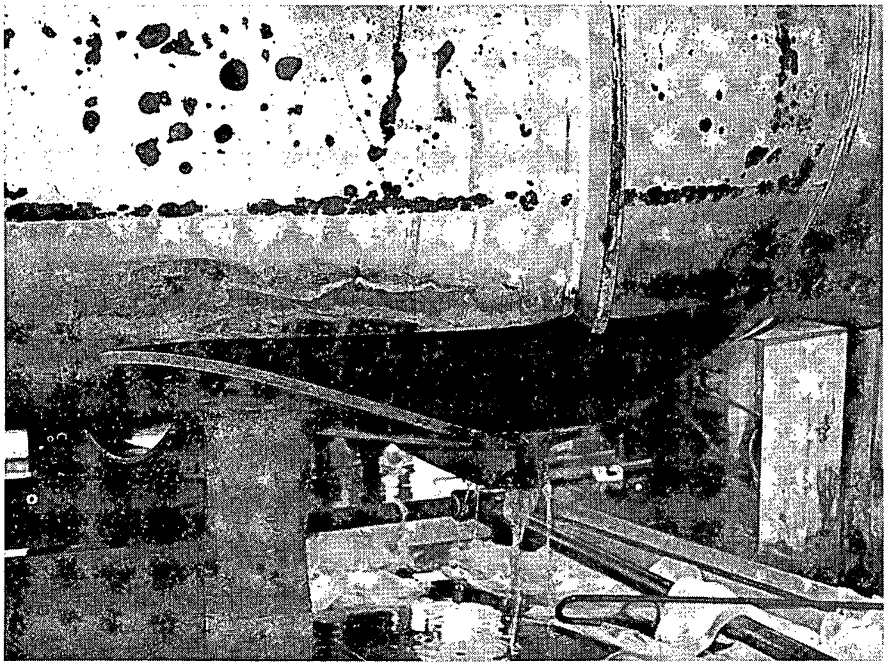
During fiscal year 1986, the AEOD staff also began a study of "source disconnect" or "failure-to-connect" events, to supplement the work already accomplished by the Radiation Steering Committee, chaired by the NRC's Office of Nuclear Material Safety and Safeguards. This committee has been examining ways to reduce radiographer overexposure. Their recommendations include developing proposed equipment standards for radiography devices and evaluating the use of different kinds of dosimetry. The AEOD effort will be to attempt to determine if there are additional lessons to be learned from the non-reactor events discussed in the survey summarized above.

ANALYSIS OF REACTOR OPERATIONAL EXPERIENCE

AEOD is responsible for screening LERs and other event-related documentation; identifying events of particular significance; conducting appropriate engineering evaluations and case studies for significant events; and formulating appropriate recommendations for action by other NRC offices. Its reviews are normally independent of and occur later than the prompt action that may be initiated by the Region or Program Offices to investigate an operating event and determine the need for immediate licensee response or generic action.

Continuing efforts are made to expand AEOD studies beyond the review of specific events and potential generic concerns to encompass broad trend-and-pattern analyses of operational data. Further, the scope of AEOD effort has been widened to take in not only the technical aspects of operational data, such as hardware and human performance, but also such important considerations as how best to use the lessons of industry experience. Addressing this latter emphasis, AEOD prepared *An Overview of Nuclear Power Plant Operating Experience Feedback Programs.* The purposes for this study were (1) to identify the characteristics of representative licensee operating experience (OE) feedback programs, (2) to ascertain how feedback documents from various NRC offices and from industry are used in such programs, and (3) to determine if there is a need for a change in the NRC's feedback program or in NRC requirements governing licensee activities related to OE feedback. The principal findings and conclusions from the study are:

(1) Although, since TMI-2, many significant and worthwhile initiatives have been taken to understand and benefit from the lessons of experience, most nuclear



This 12-foot long, 14-ton Model 48Y cylinder contained an overload of uranium hexafluoride (UF6) when it burst while being heated. The accident took place at the Sequoyah Fuels Corporation at Gore, Okla., on January 14, 1986. It led to the release, for about 40 minutes, of uranyl fluoride and hydrofluoric acid, which formed a vapor cloud that killed a plant worker and injured several others. The rupture, shown here, was 52 inches long and eight inches wide at the widest point.

plant licensees are making moderate, not extensive, use of their own in-house operating experience and, in general, are making less use of the large body of knowledge available regarding events and concerns that originate elsewhere in the industry. Licensees and the NRC should take measures to improve the effectiveness of OE reviews in correcting past and potential operational problems.

(2) The resources and attention being devoted by licensees to understanding and implementing the lessons of experience at other plants are usually considerably less than those given to in-house feedback. As a result, the benefit to be realized from the former in terms of corrective action and improved operator knowledge and capability, and the associated reduction in component failure and unplanned events, is correspondingly smaller than expected.

(3) The NRC lacks information on the range of effectiveness of industry-wide OE review activities and the degree to which plants are meeting the established NRC requirements. At this time, however, the general NRC requirements do not provide a sufficiently definitive basis to permit a meaningful evaluation of the total process by the NRC. Current industry efforts also do not provide a suitable basis for judging the effectiveness of licensee activities in this area.

(4) The large volume of OE feedback may be adversely affecting the effectiveness of the feedback programs. Duplication and overlap are dissipating resources which might otherwise be available for the effective use of OE feedback. In addition, conflicting information provided by two or more sources creates difficulties for licensees until the conflicts are resolved.

Based on this study, AEOD recommended that the effectiveness of OE activities continued to be closely monitored.

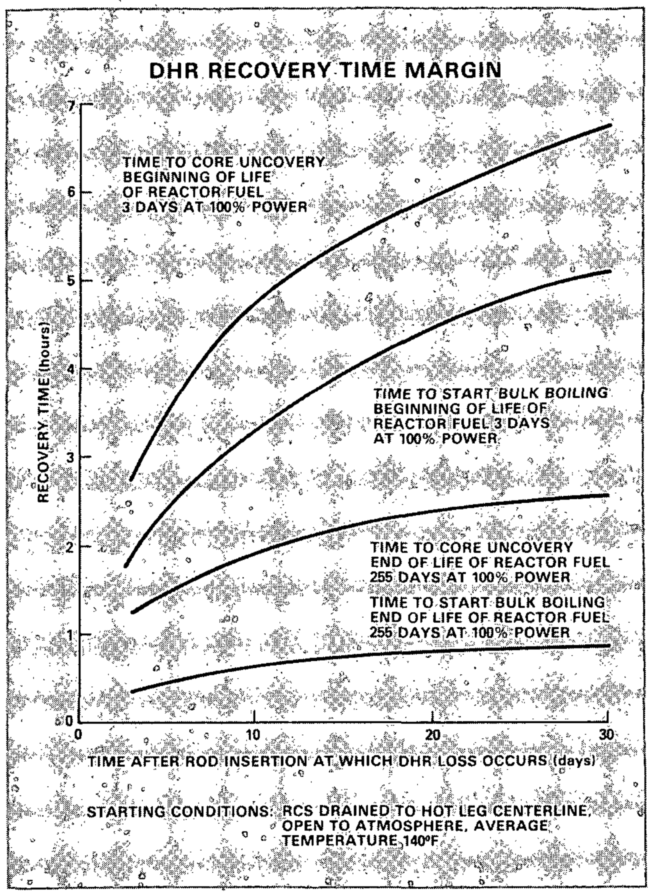
Besides the two non-reactor case studies and two survey reports already discussed, 10 reactor-related special and case studies were completed by AEOD in fiscal year 1986, as well as 25 reactor-related engineering evaluations and technical reviews. Subjects examined in the evaluations and reviews included delayed access to safety-related areas during plant operation, degradation or loss of charging systems with swing pump designs, inadvertent draining of the reactor vessel during unit shutdown, events initiated by lightning strikes at or near plants, and re-examination of events involving water hammer. Selected special studies and case studies on reactor operational experience are summarized below.

Engineering Analyses of Operational Experience

Decay Heat Removal Problems at U.S. Pressurized Water Reactors. This analysis focused on U.S. pressurized water reactor (PWR) experience involving loss of operating decay heat removal (DHR) systems. Between 1976 and 1983, there were 130 loss-of-DHR events reported, over a timeframe representing about 500 reactor-years of operation. Total loss of DHR systems under certain conditions could lead to core uncovery, and eventual fuel damage. The results of scoping analyses of total loss-of-DHR scenarios presented in the study indicate that, for certain postulated events, without timely corrective action, core uncovery could result in from one to three hours. To date, no serious damage has resulted from the loss-of-DHR events that have occurred at U.S. PWRs. However, many of the events which have occurred may be important precursors to more serious events.

AEOD's analysis of operating data indicates that the underlying or root causes of most of the loss-of-DHR system events are "human factors" deficiencies involving procedural inadequacies and personnel error. Most of the errors were committed during maintenance, testing and repair operations. The leading category of loss-of-DHR events (37 out of 130) was the automatic closure of the suction/isolation valves, and most of those resulted from human errors.

This study contains several recommendations based upon the potential safety significance of loss-of-DHR events, and administrative control of surveillance, maintenance and testing operations performed during shutdown; providing operator aids to assist in determining time available for DHR recovery and to assist operators in tracking parameters during loss-of-DHR events; upgrading the training and qualification requirements for operation and maintenance staff; requiring the use of reliable, well analyzed methods for measuring reactor vessel level during shutdown modes; modifying plant design to remove auto-closure interlocks and/or power to the DHR suction/isolation valves during period which do not require valve motion; and clarifying plant technical specifications to eliminate ambiguities associated with operating mode definitions.



Following the loss of a plant's decay heat removal system, the time available for restoration prior to core uncovery can be as short as about an hour or can extend up to many hours or days. The time available depends on the plant's operating history and status at the time of system loss. A typical time margin plot (time available for recovery of the decay heat removal function vs. time after rod insertion that the function was lost) is shown above.

Operational Experience Involving Turbine Overspeed Trips. This study was performed primarily to review and evaluate past operating experience involving overspeed trips of PWR auxiliary feedwater (AFW) turbine-driven pumps. The study also included a review of turbine overspeed trip operating experience on BWR high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) turbine-driven pumps. The analysis identified the major causes for these shutdowns and made recommendations for preventing or reducing the frequency of overspeed trips. The study was performed in response to the June 9, 1985 event at the Davis-Besse facility in Ohio in which both the main feedwater and AFW were unavailable for a 12-minute period.

Almost all of the turbines in use in AFW, RCIC and HPIC systems at operating plants are manufactured by the Terry Corporation, Steam Turbine Division, and are equipped with governors manufactured by the Woodward Governor Company. The operational conditions for these three systems are also quite similar. Accordingly, although the study was prompted by concerns about the effect of turbine overspeed trips on the reliability of PWR AFW systems, the review also included RCIC and HPCI events, so as to aid in identifying causes for AFW turbine overspeed trips. A total of 128 events involving overspeed trips of steam turbines associated with AFW, HPCI and RCIC systems were reviewed. These events occurred from January 1972 to September 1985 and were identified primarily from LERs.

The analysis of these events indicates that the predominant causes of AFW turbine overspeed trips are speed control problems associated with governors, and trip and reset problems associated with trip valves and overspeed trip mechanisms. These problems are primarily the result of inadequate performance by plant personnel, inadequate procedures, and insufficient system design considerations. The governor speed control problems are (1) slow response of the Woodward Model EG governor during quick startup, (2) entrapped oil in the speed setting cylinder of the Woodward Model PG-PL governor, (3) incorrect governor setting, and (4) water induction into the turbine. The trip and reset problems stem from the complexity of reset operations and a lack of trip position indication. To prevent or reduce the frequency of these turbine overspeed trip problems, several recommendations were developed, including:

- Licensees of PWR plants utilizing a Woodward Model EG governor for the AFW turbine should be requested to consider implementing steam bypass modifications to the AFW system to improve the turbine reliability during startup.
- In view of the number of turbine overspeed trips resulting from incorrect governor speed settings, licensees should be asked to review the adequacy of existing vendor-supplied calibration procedures used for the control system of AFW turbines.

- To assure that condensate in the steam supply line of the AFW turbine is removed before reaching the turbine, all licensees should be requested to review and verify that: (1) the steam supply line steam trap operability administrative controls are adequate; (2) the capacity of the steam traps is sufficient to remove instantaneous, rapid condensation resulting from heating the cold steam line during turbine startup; and (3) the steam supply line piping is in a configuration that will minimize the formation of condensate during a turbine cold start.

Emergency Core Cooling Minimum Flow Protection. An AEOD engineering evaluation (E511) issued in August 1985 dealt with the inadvertent closure of emergency core cooling minimum flow valves at both the Brunswick (N.C.) and Peach Bottom (Pa.) facilities, both BWR units. The valves provide an essential pump protective feature, and pump operability is generally dependent on their operability. The study found that the affected safety system trains at both the Brunswick and Peach Bottom units should have been declared inoperable when the minimum flow valves for emergency core cooling systems (ECCSs) were closed and deactivated. An evaluation of a data search for similar events clearly indicated that not all licensees may recognize the importance of minimum flow valves for ECCS pump operability. Additionally, a report from the Point Beach plant (a PWR unit is Wisconsin) stated that a design deficiency had been discovered during a review involving the minimum flow recirculation path for the safety injection (SI) pumps.

In view of these conclusions, and the design deficiency reported at Point Beach, AEOD suggested that the Office of Inspection and Enforcement (IE) issue an Information Notice to remind licensees of the importance of the minimum flow bypass line pump protection feature and the dependence of pump operability on minimum flow valve operability. On December 13, 1985, IE issued Information Notice 85-94, "Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage during a LOCA," conveying the recommended admonition.

During a review of this Information Notice, the Pilgrim (Mass.) nuclear power plant operators discovered that, under certain accident sequences, a single failure could result in all of the residual heat removal (RHR) minimum flow bypass valves being signaled to close, while all other pump discharge valves are also closed. This condition could result in no flow through the RHR pumps and could lead to the pumps running "deadheaded," with potential for pump damage in a few minutes. If this single failure occurred in conjunction with an automatic star of the RHR system, and the situation went unrecognized, RHR pump damage could occur. Such an event could disable drywell spray, shutdown cooling, torus spray, and suppression pool cooling. As a result of a loss of suppression pool cooling over a long period of time, core spray pumps could ultimately lose net positive suction head and also be unavailable. The NRC staff judged that, although the overall probability of a serious core damage accident because of this problem is low, such a single-failure vulnerability is not in compliance with the regulatory requirement for independence in ECCSs, and could compromise several important systems. It was found that nine other BWR plants also carried the potential for this single-failure problem. Therefor, on May 23, 1986, IE issued Compliance Bulletin 86-01, "Minimum Flow Logic Problems that Could Disable RHR Pumps," to all BWR licensees.

Review of the initial IE Information Notice 85-94 by licensees at several PWRs also resulted in the discovery of a design deficiency involving the minimum flow recirculation paths for SI pumps at three plants with Westinghouse-designed reactors. The design deficiency identified at these plants was essentially identical to that uncovered at Point Beach.

The concern in all four cases cited above involves a postulated small break loss-of-coolant accident (LOCA) which initiates an SI signal and starts the SI pumps. During a small break LOCA, reactor coolant system (RCS) pressure may not readily decrease below the SI pump shutoff head. A single failure resulting in the loss of the minimum flow path concurrent with SI pump actuation would cause the pumps to operate deadheaded until RCS pressure decayed below the SI pump shutoff head. The design deficiency is such that the single failure that would cause the loss of the minimum flow path will also cause the simultaneous loss of minimum flow valve position indication in the control room, exacerbating the loss of the minimum flow path. Operating the SI pumps deadheaded could result in pump damage and failure within a few minutes.

On June 6, 1986, AEOD brought these concerns involving the design deficiencies at the Westinghouse-designed PWRs to IE's attention in a memorandum action letter, "Adequacy of the Scope of IE Bulletin 86-01." AEOD suggested that IE issue either a supplemental or a separate IE bulletin to all facilities concerning the SI recirculation path deficiencies identified at the Westinghouse PWRs. Shortly after the close of fiscal year 1986, IE issued Compliance Bulletin 86-03, "Potential Failure of Multiple ECCS Pumps due to a Single Failure of Air-operated Valve in Minimum Flow Recirculation Line."

Trends-and-Patterns Analyses Of Operational Experience

Unplanned Reactor Trips at U.S. Light Water Reactors in 1985. This report analyzes 552 unplanned reactor trips that occurred in 1985 at U.S. nuclear power plants. The report defines a reactor trip as any actuation of the reactor protection system (RPS), whether automatic or manual, which results in control rod motion. Plants were included in the trip statistics contained in this analysis if they held a full power operating license, and accumulated critical hours for some portion of the calendar year 1985.

The report provides the following general observations with regard to reactor trips: (1) in terms of the overall performance of the industry, only a slight change occurred in the total trip rate from 1984 to 1985 (i.e., 5.9 and 6.0 trips-per-reactor-year, respectively); (2) hardware failures in power conversion systems—feedwater, condensate, main steam generator—dominated in 1985, and a reduction in such hardware failures would significantly reduce the number of reactor trips; (3) at power levels about 15 percent, approximately 10 percent of all reactor trips were caused by unlicensed personnel; and (4) there are a number of post-trip recovery complications attributable to equipment failures and personnel errors unrelated to the original trip cause, and these lapses may have significant safety implications.

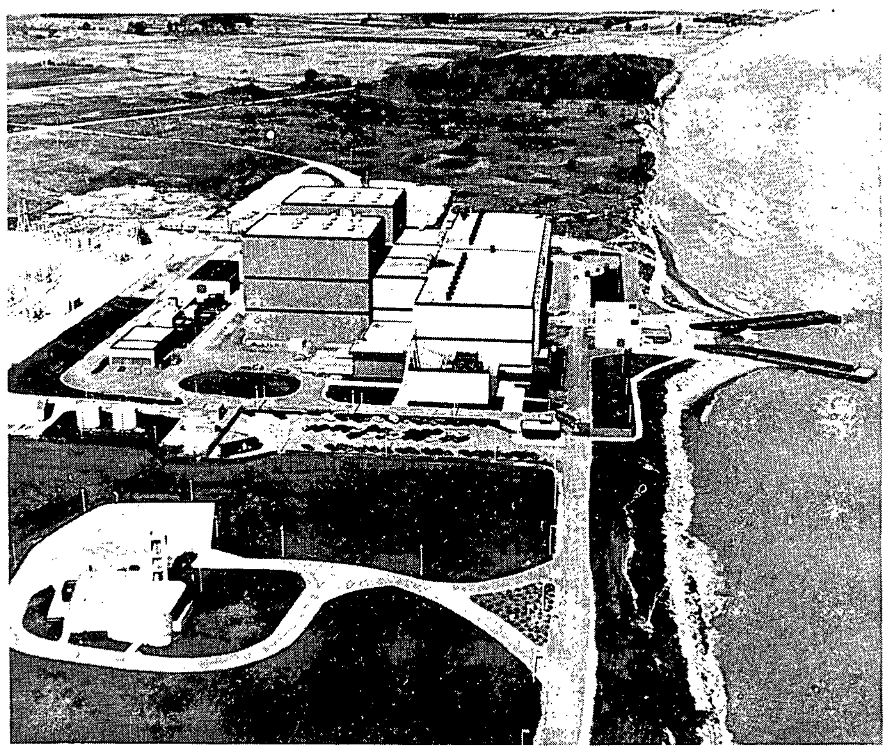
The progression of events leading to reactor trips and the post-trip response of the plant and personnel have obvious safety significance. The Commission has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee.

Engineered Safety Feature Actuations at Commercial U.S. Nuclear Power Reactors, July 1 through December 31, 1984. This analysis was limited to engineered safety feature (ESF) actuations involving systems other than the RPS. Of the 601 such actuations that occurred during the period, only 22 were deemed necessary to control a significant safety event. The rate of unnecessary actuations could be decreased by: (1) reducing the number of equipment failures during normal operation, (2) reducing the number of personnel errors during maintenance and testing, and (3) reducing spurious actuations brought about by electrical "noise spikes."

Nine units were targeted for particular scrutiny because they appear to be experiencing repeated unresolved actuations which could ultimately compromise continued equipment operability and the expectation of proper personnel response. Sustained operation (e.g., two consecutive six-month periods) with a high rate of ESF actuations may indicate a willingness to accept them in a context for which they are not appropriate. The report suggests continued attention to the specific units found to have high actuation rates and continuing problems, in order to verify that effective corrective actions are being taken.

Operational Experience of Newly Licensed U.S. Nuclear Power Reactors. During fiscal year 1986, AEOD performed an evaluation of "new plant" experience—a systematic review of operating experience data bases, with a focus on those plants that received operating licenses between 1983 and 1985. For the 19 newer units in this category, comparisons were prepared of the actuations of the reactor protection and engineered safety feature systems, security events, and miscellaneous events. The study proceeded on a plant-by-plant basis, but also with a trends-and-patterns aspect.

The goal of this new plant study was to review the performance and potential trouble areas for the new plants. This study identified certain plants and plant characteristics that appeared outside the norm during the period studied. The study also indicates that there is a wide range in the occurrence rate of the events considered, and that the maturing periods or learning curves vary substantially among these



Review of an NRC engineering evaluation issued in 1985 resulted in the discovery by operators of the Point Beach nuclear power plant at Two Creeks, Wis., shown here, of a design deficiency related to operation of the the minimum flow recirculation valves and potential effects on emergency core cooling systems. The discovery led, in part, to an issuance of reminders to all operators and, in turn, to corrective actions at a number of other plants.

plants. Further analyses to identify root causes for these events at new plants were in progress at the end of fiscal year 1986.

Loss of Safety System Function Events. This case study (C504), issued in December 1985, deals with events resulting in a total loss of safety system function (LSSF) at licensed nuclear power plants. Although the study identifies 133 instances of lost safety system function between 1981 and mid-1984, the major focus of the analysis is on 87 events (65 percent of the total) attributable to human factors, i.e., personnel errors. Over the timeframe of the study, about 0.5 LSSF events-per-year-of-reactor-operation were reported, with no significant trends observed in the rate of occurrence. The data indicate that improvement is not being made, on an industry-wide basis, in preventing LSSF events. The study also found that licensed operators, non-licensed operators and other personnel were about equally responsible for errors leading to LSSF events. A number of recommendations are put forward in the study, including a review of training and qualification programs; the monitoring of human factors data in LERs, confirming that licensees are meeting the intent of 10 CFR 50.73 requirements regarding completeness; and the collecting of LSSF event data on a continuing basis to determine to what extent LSSF events are a meaningful indicator of licensee performance.

ABNORMAL OCCURRENCES

The NRC prepares a quarterly Report to Congress on Abnormal Occurrences (NUREG-0090 series) which also serves to communicate significant event information to licensees, other Government agencies, and the public. (These reports are available from the GPO Sales Program, Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, D.C. 20013-7982. For a description of NRC's requirements under law to report abnormal occurrences, see the *1980 NRC Annual Report,* p. 82.) The reports issued during fiscal year 1986 (NUREG-0090, Vol. 8, No. 2 (April-June, 1985); Vol. 8, No. 3 (July-September, 1985); Vol. 8, No. 4 (October-December, 1985); and Vol. 9, No. 1 (January-March, 1986)) covered 10 occurrences at nuclear power plants, 16 occurrences among fuel cycle facilities and other NRC licensees (industrial radiographers, medical institutions, industrial users, etc.), and 5 occurrences at Agreement State licensees. The reports also contained updated information for certain abnormal occurrences which had been reported in previous fiscal years.

The abnormal occurrences reported during fiscal year 1986 are briefly described below. Additional details, as well as the updated information for some previously reported abnormal occurrences, are contained in the quarterly abnormal occurrence reports referenced above. Table 4 lists these reports and the occurrences covered in them.

Some of the events described below resulted in escalated enforcement actions, including civil penalties, by the NRC. (See Table 2 in Chapter 8 for listing of all civil penalties imposed during the report period, with capsule descriptions of the reasons therefor.)

Nuclear Power Reactors

Inoperable Safety Injection Pumps. On December 28, 1984, Consolidated Edison Company of New York declared all three safety injection (SI) pumps inoperable at Indian Point Unit 2. Because the reactor was in operation at the time, the licensee manually inserted the control rods to shut it down and take the plant to a "cold shutdown" condition. With all three SI pumps inoperable for a period of up to nine days, the plant's capacity to deal automatically with a design basis accident (e.g., steam line break) was significantly degraded. The three "intermediate" head SI pumps are part of the plant's emergency core cooling system (ECCS). The pumps have a nominal 1,700 psig discharge pressure, and there is no installed ECCS capability at full system pressure. In this plant, the boron injection tank (BIT), containing 20,000 ppm boron, discharge line is aligned to the suction of the SI pumps.

Investigation showed that two of the SI pumps had been degraded by a partial blockage of their suction path by solidified boric acid, together with gas entrapment, causing the pumps to bind. For the third SI pump, total blockage of its suction line had occurred because of solidified boric acid. The source of the boric acid is believed to be the BIT. The BIT was leaking past the two closure valves which isolate it from the SI pump suction. The BIT solution apparently precipitated and solidiied because the SI pumps are not heat raced. Incomplete flushing of the SI pumps following SI actuation could also result in the BIT contents reaching the SI pumps.

While venting the third pump, the licensee took a gas sample for analysis and found that the major constituent was nitrogen (97 percent). There are several potential sources for nitrogen gas, e.g., the isolation valve seal water system, which injects nitrogen between some SI valves in order to seal them and provide improved containment isolation following an accident; the nitrogen cover gas in the ECCS accumulators; or the nitrogen cover gas in the BIT.

As corrective actions, the licensee undertook to monitor the boric acid concentration in the BIT discharge line daily and to flush the line upon detection of increasing boric acid. The SI pumps are vented daily and monitored for gas. The emergency procedure for recovery from a spurious safety injection has been clarified to provide for adequate flushing of the BIT discharge line. The licensee also prepared a Technical Specification Amendment request to allow removal of the BIT. Several other Westinghouse plants have requested, and received NRC approval, to either remove the BIT, or reduce the boron concentration to the levels used in the SI piping and refueling water storage tank. (AO 85-5.)

Significant Deficiencies in Reactor Operator Training and Material False Statements. By letter of June 3, 1985, the NRC issued to Mississippi Power and Light Company, licensee for the Grand Gulf facility, a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $500,000 for identified deficiencies in their reactor operator training program and for making material false statements to the NRC. Applications for reactor operator licenses containing apparently false information were submitted to the NRC in September 1981, March 1982, and May 1982.

The June 3, 1985 NRC letter identified serious failures to comply with NRC regulatory requirements. Most of the violations pertained to the reactor operator (RO) and senior reactor operator (SRO) training program, including (1) inadequate procedures, instructions, and procedural controls; (2) training certifications which contained material false statements; and (3) failure to correct false submittals once the licensee became aware of them. The violations, classified as high as Severity Level I, were documented by special inspections by the NRC Region II (Atlanta) staff and by investigations by the NRC Office of Investigations. These were serious violations and positive corrective actions were not taken until the NRC became involved. The cause of these occurrences was failure to exercise management control; licensee management relied heavily on unmonitored contractors to train the operators and certify completion of training.

The licensee agreed to conduct a review of the previous training of all its licensed operators, shift technical advisors, and on-shift operations advisors. Certain operators were removed from licensed duties until they could be retrained and retested. A change of management personnel in the plant training staff improved the overall training program administration and brought about better documentation of training, clearer standards of acceptable performance for both students and training staff, and improved adherence to regulations, procedures, and commitments.

At the close of the report period, the licensee's reply to the proposed enforcement action was still under consideration by the NRC. (AO 85-6.)

Loss of Main and Auxiliary Feedwater Systems. On June 9, 1985, the Davis-Besse (Ohio) nuclear power plant underwent a complete loss of main and auxiliary feedwater for about 12 minutes, during an event involving an automatic shutdown from operation at 90 percent power. The event involved several equipment malfunctions (including several common-mode failures) and extensive operator activity, both in and outside the control room.

A total loss of feedwater is a significant safety-related event. Unless prompt and effective recovery actions are taken, severe consequences could occur (e.g., fuel damage, breach of the primary system, significant release of radioactivity). In this event, compensatory efforts were complicated by the many equipment malfunctions and by various operator errors. Nevertheless, the operators were successful in bringing the plant to a stable shutdown and in preventing any abnormal releases of radioactivity and any major damage to the plant.

Because of the potential safety implications of the event which clearly warranted further study, a special NRC Incident Investigation Team (see description earlier in this chapter) was sent to the site to perform a detailed investigation. The Team, composed of four technical experts, was charged to (1) make a factual determination as to what happened; (2) identify the probable cause as to why it happened; and (3) make appropriate findings and conclusions on which to base possible follow-up measures.

The results of the Team's investigation, contained in NUREG-1154, included identification of issues specific to Davis-Besse and several possible generic issues. In addition, the Team reached the major conclusion that the underlying cause of the loss of main and auxiliary feedwater event was the licensee's lack of attention to detail in the care of plant equipment. The licensee has a history of carrying out maintenance and testing of equipment, and of evaluating operating experience related to equipment, in a superficial manner; as a result, the root causes of problems in this facility are not always found and corrected. The effort to address equipment problems through sound engineering analyses has frequently either not been made or has not been adequate. It also became clear from operator interviews that equipment problems were not being aggressively addressed and resolved.

The licensee undertook an extensive study, including testing programs, of the multiple failures associated with the event, in order to determine root causes and to take effective corrective actions to minimize the chances of recurrence. In addition, the licensee was directed to address such other relevant issues as operational, management, and procedural deficiencies, as well as inadequate or inappropriate equipment design, testing, and maintenance practices. The licensee submitted a course of action by which to deal with these numerous areas of concern identified by the NRC.

Upgrading the reliability of feedwater flow at Davis-Besse is an issue first raised by the NRC in 1979; nevertheless, at the time of the June 9, 1985 event, the matter remained unresolved. Shortly after this event, the licensee took steps to accelerate the installation of an electrically driven startup feedwater pump, which was subsequently installed and tested.

In addition, the licensee:

(1) Has completed its analysis and evaluation for the cause of the equipment that failed and has developed an action plan related to each item, to assure prevention of similar failures in the future.

(2) Has inspected 2,425 pipe hangers and supports for safety-related systems; the results were still being evaluated by the licensee at the close of the report period.

Table 4. Abnormal Occurrence Reports Issued During FY 1985

Occurrences at Nuclear Power Plants *Designation (AO#)*	*Subject*	*NUREG-0090* *Issue*
85-5	Inoperable Safety Injection Pumps	Vol. 8, No. 2 November 1985
85-6	Significant Deficiencies in Reactor Operator Training and Material False Statements	
85-7	Loss of Main and Auxiliary Feedwater Systems	
85-12	Management Control Deficiencies	Vol. 8, No. 3 February 1986
85-13	Inoperable Steam Generator Low Pressure Trip	
85-14	Management Deficiencies at Tennessee Valley Authority	
85-19	Inoperable Main Steam Isolation Valves	Vol. 8, No. 4 May 1986
85-20	Management Deficiencies at Fermi Nuclear Power Station	
86-1	Loss of Power and Water Hammer Event	Vol. 9, No. 1 September 1986
86-2	Loss of Integrated Control System Power and Overcooling Transient	

Occurrences at Fuel Cycle Facilities (Other than Nuclear Power Plants) *Designation (AO#)*	*Subject*	*Issue*
86-3	Rupture of a Uranium Hexafluoride Cylinder and Release of Gases	Vol. 9, No. 1 September 1986

Occurrences at Other NRC Licensees (Industrial Radiographers, Medical Institutions, etc.) *Designation (AO#)*	*Subject*	*Issue*
85-8	Diagnostic Medical Misadministration	Vol. 8, No. 2 November 1985
85-9	Diagnostic Medical Misadministration	
85-10	Breakdown in Management Controls	
85-11	Therapeutic Medical Misadministration	
85-15	Therapeutic Medical Misadministration	Vol. 8, No. 3 February 1986
85-16	Therapeutic Medical Misadministration	
85-17	Exposure of Radiographic Personnel Due to Management and Procedure Control Deficiencies	

85-18	Diagnostic Medical Misadministration	
85-21	Diagnostic Medical Misadministration	Vol. 8, No. 4 May 1986
85-22	Therapeutic Medical Misadministration	
85-23	Diagnostic Medical Misadministration	
86-4	Therapeutic Medical Misadministration	Vol. 9, No. 1 September 1986
86-5	Overexposure to a Member of the Public from an Industrial Gauge	
86-6	Breakdown of Management Controls at an Irradiator Facility	
86-7	Tritium Overexposure and Laboratory Contamination	

Occurrences at Agreement State Licensees

Designation (AO#)	*Subject*	*Issue*
AS85-5	Overexposures of a Radiographer and an Assistant Radiographer	Vol. 8, No. 2 November 1985
AS86-1	Radiation Injury of an Industrial Radiographer	Vol. 9, No. 1 September 1986
AS86-2	Contamination of a Scrap Steel Facility	
AS86-3	Radiation Injury of an Industrial Radiographer	
AS86-4	Radiation Injury of an Industrial Assistant Radiographer	

(3) Was conducting a System Review and Test Program in which 34 safety systems were to be extensively reviewed, including an evaluation of the system design requirements. Previous surveillance tests of the systems were being analyzed, and additional testing performed to demonstrate the operability of the systems.

On December 13, 1985, the NRC issued a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $900,000 to Toledo Edison Company for multiple violations associated with the June 9, 1985 loss of feedwater event.

As of the end of fiscal year 1986, the licensee had not yet completed all actions required to receive NRC approval for plant restart. (AO 85-7.)

Management Control Deficiencies. Because of continuing problems with the operation of the LaSalle (Ill.) nuclear power plant, the NRC Region III Office (Chicago) established a special Task Force in July 1985 to perform an in-depth review of the facility's operations. Among the problems which triggered creation of the task force were three instances in which errors in installation occurred during equipment modifications affecting the operability of emergency core coolant systems (ECCS) and the shutdown cooling systems. The Task Force identified a number of items indicative of poor management performance on the part of the licensee, the Commonwealth Edison Company.

Since LaSalle Units 1 and 2 received full power licenses in June 1982 and March 1984, respectively, the licensee has experienced numerous personnel, equipment, and regulatory problems, many of which can be attributed to deficiencies in management controls. These recurring problems have not individually been of major safety significance, but they represent a trend which is not acceptable over the long run at an operating nuclear power station.

The reasons for the NRC's concern included:

- A total of 172 violations of NRC requirements identified by NRC inspections from 1982 through July 1985. Three fines have been assessed, and a fourth one has been proposed.
- Twenty-four instances between October 1984 and July 1985 where personnel errors or other actions during

maintenance or modification work affected the operation of the plant.

- Repeated equipment problems—caused either by hardware failures or personnel errors—in a number of systems.
- The high volume of outstanding work requests for repairs or maintenance of control room equipment—running about 80 per unit in September 1985.
- The excessive backlog of equipment modifications. In September 1985, the number totaled 543, not including those in progress, and 270 of those were designated as priority modifications.

The NRC Region III forwarded a Confirmatory Action Letter to the licensee on July 22, 1985, documenting additional actions to be taken by the licensee prior to startup of either Unit 1 or Unit 2.

Over the past two years, the licensee has undertaken a company-wide Regulatory Improvement Program to improve the performance of its management in its nuclear power plants. The program has included issuance of policy directives, organization modifications, some personnel changes, increased management involvement in the day-to-day operations at the nuclear facilities (by both corporate and station management), training activities, and efforts to reduce the number of personnel errors and procedural violations. The licensee also retained a consultant to review its station operations.

In regard to equipment modification errors which occurred in June and July 1985, the NRC issued a notice of violation on September 27, 1985, and proposed imposition of civil penalties of $125,000. The violations cited in the NRC's September 27, 1985 letter indicate that more effective controls must be implemented to ensure that operability tests will be performed on safety-related systems after maintenance or modification, and before these systems are returned to service.

The licensee has met periodically with the NRC staff to review the status of its improvement program. Data collected by the licensee through April 1986 generally gives evidence that the improvement program is working to reduce such concerns as the number of automatic shutdowns (scrams) and unneeded actuations of emergency safety systems, the backlog of work requests, and the number of Licensee Event Reports attributed to personnel errors. Region III will continue to monitor the licensee's program in implementing its measures to improve regulatory performance. (AO 85-12.)

Inoperable Steam Generator Low Pressure Trip. On August 7, 1985, Maine Yankee Atomic Power Company found nine of the 12 pressure transmitters that monitor pressure of the three steam generators (SGs) at the Maine Yankee plant inoperable because of closed or partially closed root valves. These transmitters provide low steam pressure to the reactor protection system (RPS), the main steam isolation system, and the feedwater isolation system. The closed root valves caused three of the four low-SG-pressure logic channels of these systems to become inoperable. The significance of this condition is that in the event of a steam line rupture and subsequent reactor trip, main steam isolation and main feedwater isolation would not be initiated automatically on low steam pressure signals. The condition had existed since June 20, 1984.

The cause of the event was determined to be inadequate administrative controls in the area of testing and plant modifications, resulting in the SG pressure instrument root valves being left in the closed position.

Three possible consequences of the closed root valves were of primary concern: (1) the loss of a trip signal to the RPS which controls the insertion of reactor control rods; (2)

The Maine Yankee Atomic Power Company submitted a program to the NRC to correct and prevent the recurrence of deficiencies in administrative controls of its Wiscasset, Me., plant which had been found to permit development of unsafe conditions. NRC review of the program concluded that it was comprehensive, well conceived and properly implemented.

delayed isolation of the intact steam generators from the ruptured pipe, following a main steam line break accident; and (3) the prevention of automatic main feedwater pump trip. Following from these possible conditions, and of greatest concern, was the possibility of an extended steam blowdown outside the containment and of reactor overcooling and possible recriticality. In addition, for ruptures outside containment, there would be adverse environmental conditions for the equipment in the turbine building.

The licensee has developed a program to correct and prevent recurrence of the mispositioning of instrumentation root valves, as well as inadequate design change review. The program was formally transmitted in a letter to the NRC Region I Office (Philadelphia) on September 13, 1985, and it addressed the points cited above. A special inspection following the enforcement conference found that the licensee's corrective program was comprehensive, well conceived and properly implemented.

On October 29, 1985, the NRC Region I Office issued a Severity Level II violation and civil penalty in the amount of $80,000. The Regional Administrator emphasized that corrections were needed in the areas of improved administrative control of valves, control of design changes, preparation and implementation of temporary procedures, and control of the post-maintenance or post-modification testing process—including test design, test procedures, and the review thereof. (AO 85-13.)

Management Deficiencies at Tennessee Valley Authority. Because of its serious concerns regarding programmatic and management deficiencies at nuclear facilities of the Tennessee Valley Authority (TVA), the NRC issued a request on September 17, 1985, for information pursuant to 10 CFR § 50.54(f), to enable the NRC to determine whether or not the licenses for the Browns Ferry (Ala.) and Sequoyah (Tenn.) facilities should be modified or suspended for or the application for the Watts Bar (Tenn.) facility should be denied.

Operations at all three Browns Ferry units had been suspended by the licensee since March 1985, and operations at both Sequoyah units had been suspended by the licensee since August 1985. The two units at Watts Bar (Tenn.) are under construction, and fuel loading for Unit 1 had been projected to take place in January 1986. The licensee issued a stop-work order on all safety-related welding activities for both units during August 1985. This order was lifted in September 1985.

The numerous significant events that have occurred at Browns Ferry and Sequoyah since March 1, 1984, include the following:

At Browns Ferry:

(1) On August 14, 1984, overpressurization of the Unit 1 core spray system occurred while a core spray system logic surveillance test was being conducted with the reactor operating at 100 percent power.

(2) The numerous procedural and equipment deficiencies that occurred during a shutdown margin test on Unit 3 on October 22, 1984, following an extended shutdown for refueling, modifications, and maintenance, necessitated a re-evaluation which extended the outage another month.

(3) During a Unit 3 reactor startup on February 13, 1985, the licensee failed to satisfy technical specification requirements for reactor vessel water level operability.

(4) During a routine inspection on August 16, 1985, it was determined that major discrepancies existed in the design of cable tray supports at all three units.

(5) On September 24, 1985, TVA declared all eight emergency diesel generators associated with the standby a.c. power supply and distribution system inoperable.

For the Sequoyah facility:

(1) On April 19, 1984, a significant event at Unit 1 occurred involving damage to a compression fitting at the incore probe seal table.

(2) During 1985, following NRC's expression of concern regarding environmental qualification of electrical equipment issues, an independent contractor hired by TVA found that documentation in compliance with 10 CFR § 50.59 appeared to be inadequate at all TVA sites.

The TVA has taken corrective action in regard to specific problem areas and has initiated efforts to improve plant management staffs.

In order to assure high-level attention to the problems at TVA, an NRC Senior Management Team was formed consisting of the Executive Director for Operations, the Directors of the Offices of Nuclear Reactor Regulation, Inspection and Enforcement, and Investigations, the Regional Administrator of Region II, and senior managers from these four Offices. The Team meets regularly to discuss and implement corrective measures related to the facilities in question, such as the introduction of an augmented inspection program for Browns Ferry; a re-evaluation of the Browns Ferry Regulatory Performance Improvement Plan; augmented Systematic Appraisal of Licensee Performance of all TVA facilities; and an in-depth operational readiness inspection program. The Commission is kept informed of judgments and actions taken by the Team.

Besides those mentioned above, there are six areas in particular where there has been considerable TVA effort towards improvement, and which have also received a significant level of NRC attention. These are equipment qualification, operational readiness, employee concerns, welding, electrical design calculations, and simulator evaluations of Sequoyah licensed personnel. At the close of fiscal year 1986, these areas remained under in-depth licensee review. (AO 85-14.)

Inoperable Main Steam Isolation Valves. On September 27, 1985, during main steam isolation valve (MSIV) surveillance testing at Brunswick Unit 2 (N.C.), Carolina Power and Light Company discovered that three MSIVs would not fast close. At the time of the event, the plant was in cold shutdown following controlled shutdown from power on September 26, 1985, as a precaution adopted in advance of then approaching hurricane Gloria.

The safety objectives of the MSIVs in a direct cycle boiling water reactor are to: (1) prevent damage to the fuel barrier by limiting the loss of reactor coolant, in case of a major leak from the steam piping outside the primary containment; (2) limit release of radioactive materials by closing the nuclear system process barrier, in case of gross release of radioactive materials from the reactor fuel to the reactor cooling water and steam; and (3) limit release of radioactive materials by closing the primary containment barrier, in case of a major leak from the nuclear system inside the primary containment. Each main steam line contains two MSIVs in series to ensure that safety objectives are met. In this event, two MSIVs in one line failed to close fully, thus providing a release path.

Initial investigation disclosed that a problem existed within the double solenoid valve on the actuator of one MSIV. In addition, a subsequent fast closure test of a second MSIV revealed a closure time of approximately 45 seconds; a maximum closure time of five seconds has been prescribed to contain fission products and to ensure that the core is not uncovered following line breaks.

From visual inspection of the valves, it was concluded that one had failed to fast close because the exhaust port was blocked. Another valve failed to fast close because the solenoid valve disc had adhered to the valve seat, and the valve could not move. The third valve failed to close either because the solenoid valve disc was stuck to its seat or because disc material had broken off and plugged the exhaust port.

An evaluation by the valve vendor concluded that the elastomer on one valve had deteriorated because of contamination. The vendor did not identify the contaminant, but it was felt that it had not been introduced during manufacture or assembly of the solenoid valve assembly. Later, the licensee determined that the most likely failure mechanism of the elastomer material was a combination of temperature effects, hydrocarbon contamination, and internal geometry.

The MSIV solenoid valves were replaced with valves using "Viton" as the elastomer; Viton is impervious to the hydrocarbon contamination, and licensee tests have shown that it can withstand temperatures that will degrade ethylene propylene (EP), the elastomer used in the valves. Viton, however, is less resistant to radiation than EP by a factor of ten, so it is expected that the solenoid valves will be replaced at about three-year intervals.

A detailed inspection to review the licensee actions with respect to the solenoid failures was conducted by NRC Region II. Inspectors from NRC Region II and NRC headquarters reviewed test results at both the site and the utility's research facility. Inspection and Enforcement Information Notice 86-57, "Operating Problems with Solenoid Operated Valves at Nuclear Power Plants," regarding this event, was issued in July 1986. (AO 85-19.)

Management Deficiencies at Fermi Nuclear Power Station. On December 24, 1985, NRC Region III issued a letter under 10 CFR 50.54(f) seeking information from the Detroit Edison Company on plans to improve the operations at Fermi Unit 2 (Mich.) nuclear power plant. The letter identified a series of operational and equipment problems at Fermi Unit 2, beginning in July 1985, and attributed them to ineffective management systems.

On July 1, 1985, during startup of the reactor, a reactor operator failed to follow the procedure for withdrawing control rods from the reactor core. The operator withdrew 11 rods to the fully withdrawn position instead of to the next intermediate position. As a result, the reactor went critical earlier in the startup procedures than planned.

The NRC's inspection and review of this premature criticality event revealed nine apparent violations of NRC requirements associated with it. Because the plant had just achieved its initial criticality on June 21, 1985, and therefore had little residual radioactivity in the reactor core, this event was of minimal safety consequences.

But other problems subsequently emerged at the plant; some of these were caused by personnel errors, and some were strictly equipment failures. These problems included:

- A reactor feedwater pump turbine was damaged during testing on July 22, 1985; the pump turbine had a history of excessive vibration during pre-operational testing.
- On July 26, 1985, while a diesel generator was undergoing testing, a low flow of cooling water for the diesel was observed, and the diesel was shut off. The licensee's investigation disclosed that a cooling tower bypass valve had been closed.
- On September 2, 1985, the licensee discovered that a containment monitoring system valve was open and uncapped, thus providing a breach of the primary reactor containment; the valve had been left open following installation in June 1985.
- Since receiving an operating license in March 1985, Fermi Unit 2 has been the site of numerous personnel errors. Of 78 Licensee Event Reports submitted by the licensee between March and November 1985, 41 involved personnel errors.
- NRC inspectors, in a report covering the events listed above, as well as other problems at the Fermi Unit 2 facility, identified 26 items of apparent violation

of NRC requirements. Enforcement action on these items was pending at the close of the report period.

In addition to the problems cited above, significant problems were encountered with this facility's emergency diesel generators. This condition came to light during a unit shutdown on October 10, 1985, for the installation of a remote shutdown control panel, in order to meet an NRC requirement of the capability to shut the plant down safely from a point outside the main control room. During testing of a diesel generator in November 1985, excessive noise and vibration were observed. The diesel generator was shut down, and an examination showed evidence of bearing damage. Subsequent inspections of two additional diesel generators (the site has a total of four) showed similar damage. The plant had previously encountered bearing problems with two diesel generators in January 1985, and the equipment had been repaired at that time.

The licensee and the diesel vendor decided that the damage was attributable to an insufficient break-in period. Further testing of the diesels was undertaken, and the root cause of the bearing problems remained under review by the licensee, its consultants, and the NRC at the close of the report period.

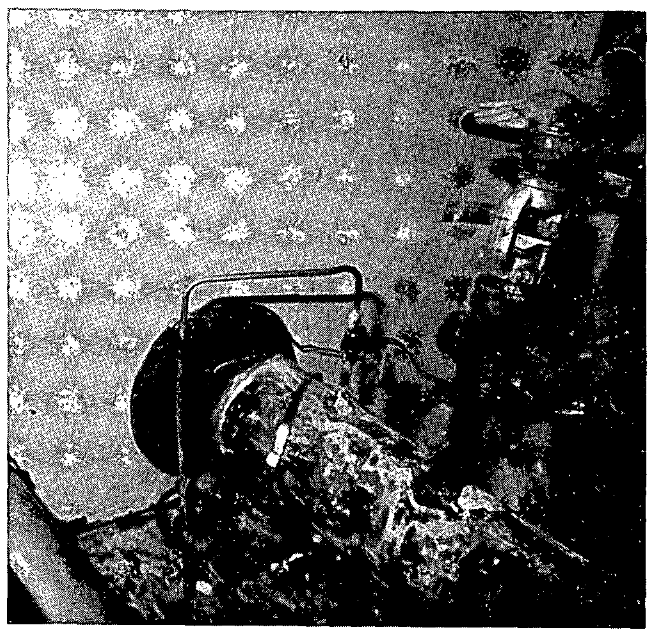
The licensee has instituted various measures to improve its regulatory performance, including retraining of personnel and revision of procedures. An independent overview committee of consultants has been formed by the licensee to review the management and operation of the Fermi facility.

Following the July 1, 1985 premature criticality incident, NRC Region III issued a Confirmatory Action Letter to Detroit Edison, confirming the licensee's agreement not to operate the unit above 5 percent power until the premature criticality incident was fully analyzed and corrective action taken. Operation above 5 percent power would not occur without authorization from NRC Region III. That restriction was not lifted prior to the October 10, 1985 outage. A December 24, 1985 request for information under 10 CFR 50.54(f) sought the licensee's response on the adequacy of management and management structures, on changes in controls needed to improve regulatory performance, and on actions planned to ensure readiness of the facility to resume operations and testing activities. As part of its response to this request, the licensee established an Independent Overview Committee to assess the management of the Fermi Unit 2 plant and to provide the licensee with recommendations for improvements. The Committee issued its preliminary evaluation report on January 30, 1986. Its recommendations included the hiring of individuals with previous experience at commercial nuclear power plants to fill key management positions at Fermi Unit 2.

The licensee has also prepared a Performance Improvement Program for its security program as a result of numerous violations of NRC requirements and inadequate implementation of the licensee's security plan. On May 21, 1986, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty ($50,000), for 13 violations of security requirements. The licensee has paid the fine. (AO 85-20.)

Loss of Power and Water Hammer Event. On November 21, 1985, the San Onofre Unit 1 (Cal.) nuclear power plant underwent a partial loss of in-plant a.c. electrical power while the plant was operating at 60 percent power. Following a manual reactor trip, the plant lost all in-plant a.c. power for four minutes and experienced a severe water hammer in the feedwater system which caused a leak, damaged plant equipment, and challenged the integrity of the plant's heat sink. The most significant aspect of the event involved the failure of five safety-related check valves in the feedwater system, without detection, which jeopardized the integrity of safety systems. The event involved a number of equipment malfunctions, operator errors, and procedural deficiencies.

The plant was operating at 60 percent power when a ground fault was detected by protective relays associated with the C transformer, which was supplying off-site power to one of the two safety-related 4160 V electrical buses. The resulting isolation of the transformer caused the safety-related bus to de-energize, which tripped all feedwater and condensate pumps on the east side of the plant. The east feedwater pump discharge check valve (FWS-436) failed to seat as the de-energized pump coasted down. This provided a path for the discharge of the still operating high pressure (1300 psig) west feedwater pump to the low pressure (350 psig) east condensate piping and components. Some



In November 1985, a four-minute loss of electric power at the San Onofre Unit 1 nuclear power plant, near San Clemente, Cal., led to severe water-hammer damage to the plant's feedwater piping system. The photo shows a damaged pipe and a circumferential crack in the turbine building wall where the pipe penetrates.

evaporator condenser tubes became overpressured, ruptured and overpressurized the evaporator shell, causing the shell to develop a fishmouth opening approximately 20 feet long and 2 feet wide, which relieved the pressure.

The operators, as required by emergency procedures dealing with electrical systems, shut down the reactor and turbine-generator. As a result, the plant underwent the first complete loss of steam generator feedwater and in-plant electrical power since it began operation. The manual trip of the main generator caused loss of a.c. power to the remaining in-plant loads. The subsequent four-minute loss of in-plant electric power started the emergency diesel generators (which by design did not load), de-energized all safety-related pumps and motors, significantly reduced the number of control room instrument indications available for operators to diagnose plant conditions, produced spurious indications of safety injection system actuation, and caused the the NRC Emergency Notification System (ENS) phone on the operator's desk to start ringing. Restoration of in-plant electric power was delayed by improper operation of an automatic sequencer.

The loss of the feedwater pumps, in combination with the failure of five feedwater check valves to close (one at the discharge of each feedwater pump and one in the feedwater line to each of the three steam generators), allowed loss of inventory from all three steam generators. Since the feedwater piping to the steam generators had drained because of the failed check valves, the pipes contained water and steam at high temperature and pressure from the steam generators. As the auxiliary feedwater system filled the piping with relatively cold water, an instability occurred at the steam/water interface, which created a water hammer which damaged the piping system.

Despite these problems, operators succeeded in recovering water level in the two steam generators not directly associated with the feedwater piping leak. With the reestablishment of steam generator levels, the operators safely brought the plant to a stable cold shutdown condition, without a significant release of radioactivity to the environment (a pre-existing primary-to-secondary leak was not aggravated) and without significant additional damage to plant equipment.

The most significant aspect of the event was, as noted, that five safety-related feedwater system check valves degraded to the point of inoperability without detection by the licensee, and that their failure jeopardized the integrity of safety-related feedwater piping. The root causes of the check valve failures were a combination of inadequate maintenance, inadequate inservice testing, inadequate design, and inadequate consideration of the effects of reduced power operations.

The licensee performed an extensive study, with testing, of the multiple failures associated with the event, in an effort to find the root causes and take effective corrective actions. The licensee concluded that the most likely cause of the cable failure which initiated the event was temperature-induced degradation, ascribable to the presence of local heat sources such as hot pipe flanges. The licensee also concluded that the failure of the five check valves was caused by (1) their proximity to turbulent flow, (2) the fact that they were not properly sized for design flow conditions and therefore did not remain fully open in normal operation, (3) the design by which the valve disc was fastened to the valve hinge, and (4) extended reduced flow operation at 90 percent power which exacerbated the effects of the design deficiencies.

The licensee carried out a number of corrective actions, among them such repairs and design changes as the redesign and replacement of the damaged feedwater lines, replacement of the failed check valve design with another design, and the addition of an extra check valve in each feedwater line. The licensee has also agreed to undertake substantial improvements in plant performance in the future.

On November 22, 1985, a special NRC Incident Investigation Team (IIT), in conformance with an NRC staff-proposed Incident Investigation Program, went to the site to investigate the event. The results of the Team's investigation (NUREG-1190) included identification of problems specific to Unit 1 and several possible generic issues. The Team concluded that the most significant aspect of the event was that five safety-related feedwater system check valves had deteriorated to the point of inoperability in less than year without detection, and that their failure had jeopardized the integrity of safety-related feedwater piping.

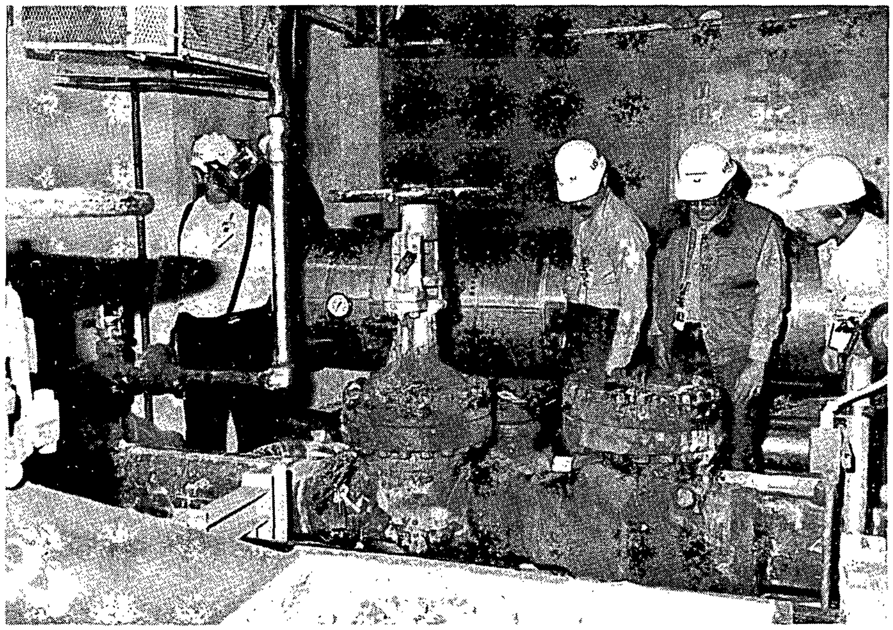
The licensee developed a Material Condition Review Program designed to define a suitable material standard for systems and components in an older plant and to ensure that the material condition of those items was maintained.

Based on a review of the numerous corrective actions taken by the licensee, the NRC agreed that the plant could be restarted. On July 15, 1986, the reactor was taken critical for low power testing and was connected to the grid on July 26, 1986. (AO 86-1.)

Loss of Integrated Control System Power and Overcooling Transient. On December 26, 1985, the Rancho Seco (Cal.) nuclear power plant underwent a loss of d.c. power within the integrated control system (ICS), while the plant was operating at 76 percent power. Following the loss of ICS d.c. power, the reactor shut down in automatic response to high reactor coolant system (RCS) pressure; this event was followed by a rapid overcooling transient and automatic initiation of the safety features actuation system on low RCS pressure. The overcooling transient continued until ICS d.c. power was restored, 26 minutes after it went out. The significance of the event is that a non-safety-related system failure initiated a plant transient which could have been more severe under other postulated circumstances.

During the first seven minutes of the incident, excessive steam and feedwater flows resulted in the rapid RCS cooldown, and the RCS depressurized to about 1,064 psig

Members of an NRC Incident Investigation Team examine Valve FWS-346 (right) during an investigation of the San Onofre (Cal.) incident. The valve shown is one of five safety-related check valves that had deteriorated to inoperability in less than a year at the facility. Team members are, left to right: Thomas Martin, William Kennedy, Anthony D'Angelo and Wayne Lanning.



and then began to repressurize. This repressurization caused the reactor coolant to enter the so-called "pressurized thermal shock (PTS)" region, in which damage to the reactor may be anticipated and must be guarded against. Operators were able to stabilize the plant and bring it to a cold shutdown condition without a significant release of radioactivity to the environment and without significant additional damage to plant equipment.

The overcooling incident did not seriously threaten the integrity of the Rancho Seco reactor vessel. However, the plant has had a number of such incidents in its 12-year operating history. Each time one occurs, the potential exists for additional operator errors and equipment failures that might compound the failure and threaten reactor vessel integrity. Thus, by various plausible chains of events, more serious consequences could follow than have heretofore.

The fundamental causes for this transient were determined to be design weaknesses and vulnerabilities in the ICS and in the equipment controlled by that system. These vulnerabilities were not adequately compensated for by other design features, plant procedures or operator training. The presence of the vulnerabilities was largely known to the licensee and the NRC staff by virtue of the number of precursor events and from related analyses and studies. Still, adequate plant modifications were not made so as to render this event improbable, or to assure that its course and consequences would be altered and controlled.

The licensee has undertaken extensive study—including controlled disassembly, examination, and testing of equipment—of the multiple failures associated with the event, to determine root causes and to take effective actions to prevent recurrence. Specific improvements were identified from these efforts, in such areas as plant modifications, training, maintenance programs and emergency procedures. These were to be implemented prior to plant startup.

On December 27, 1985, an NRC Augmented Inspection Team (AIT) was sent to the site to begin an investigation. On December 31, 1985, the responsibility for the incident investigation was expanded to a special NRC Incident Investigation Team, in conformance with an NRC staff-proposed Incident Investigation Program. As a result of the Team's investigation (reported in NUREG-1195), problems were identified that were specific to Rancho Seco, and also several possible generic issues were cited.

The NRC Executive Director for Operations also addressed this event in a January 24, 1986 letter to the Babcock & Wilcox Owners Group (B&WOG), which stated that the NRC staff would be reassessing the overall safety of B&W plants. Following the accident at Three Mile Island Unit 2, a B&W reactor facility, in 1979, there has been a growing realization among the NRC staff of the sensitivity of B&W plants to operational transients. A number of events during the report period reinforced their concerns regarding these designs and lead the staff to conclude that there is a need to re-examine the basic design requirements for B&W reactors. While they believe that this reassessment is needed, they also believe that B&W reactors can safely be permitted to operate in the interim. The B&WOG has committed itself to taking a leadership role in the reassessment.

Besides addressing those issues deriving directly from the December 26, 1985 cooldown transient, the NRC Region V Office (San Francisco) re-evaluated the status of earlier inspection findings at Rancho Seco to determine if there were other matters to be resolved before allowing restart of the plant. The licensee and the NRC included the assessment of these inspector findings in restart plans. Also, the

NRC staff encouraged the licensee to re-examine the status of all critical plant systems to assure readiness for operation and maximum reliability, so that operation of the plant could be continued with a low probability of disruption from internal causes. Some of these efforts were to be observed by NRC inspectors. In addition, the licensee initiated a performance improvement program which would address management, training, and maintenance issues. (AO 86-2.)

Fuel Cycle Facilities and Other NRC Licensees

Rupture of a Uranium Hexafluoride Cylinder. At 11:30 a.m. on January 4, 1986, a cylinder filled with uranium hexafluoride (UF-6) ruptured while it was being heated in a steam chest at the Sequoyah Fuels Corporation's facility near Gore, Okla. One worker died from pulmonary edema brought on by inhalation of hydrofluoric acid, a product of the reaction of UF-6 with airborne moisture. Much of the facility complex and some off-site areas to the south were contaminated with hydrofluoric acid, and a second reaction product, uranyl fluoride. The period of release was about 40 minutes. The licensee experienced another incident involving an overfilled uranium hexafluoride cylinder on March 13, 1986; however, in the latter incident, the overfilled cylinder was not heated and no damage to the cylinder occurred. (Uranium hexafluoride is a volatile compound of uranium and fluorine, created at a late stage in the fabrication of reactor fuel.)

The incident involving a fatality began at approximately 10:00 a.m. on January 3, 1986, when the filling of a 14-ton capacity cylinder with UF-6 was commenced. On January 4, a chemical operator was unable to add further material to the cylinder, even though the targeted load of 27,500 pounds had not been achieved. The cylinder and its attendant cart had been placed on a scale during the filling process in order to monitor the net weight of the cylinder. At this point, the scale indicated that the cylinder contained 26,400 pounds of product.

The operator inspected the cylinder and observed that the cart on which it rested had not been fully moved onto the scale platform. When the cart and cylinder were repositioned onto the scale platform, the scale dial indicator registered its maximum possible reading of approximately 29,500 pounds. It was thus apparent that the cylinder had been filled with a quantity of UF-6 in excess of the desired amount, in excess of an amount measurable by that scale, and in excess of the maximum shipping weight specification of the cylinder, which is 27,560 pounds.

The operator began to evacuate UF-6 from the cylinder back into the plant process vessels. When he was relieved by the day shift chemical operator, the evacuation process continued until the material began to solidify in the cylinder. This operator consulted with the assistant shift supervisor, who is the ranking production manager at the site, and he instructed the operator to move the cylinder to a steam chest located outside the process building. The steam chest was to be used to heat the cylinder to approximately 210xF, thus liquefying the UF-6 within. Although some material had been removed from the cylinder, the scale indicator still registered approximately 29,500 pounds before the cylinder was removed. Heating an overfilled cylinder, it was later recognized, is an action contrary to company procedures.

At about 11:30 a.m., the cylinder ruptured in the steam chest. Liquid UF-6 flowed from the four-foot lengthwise rupture and rapidly reacted with moisture in the air to form uranyl fluoride and hydrofluoric acid. The resulting vapor cloud was carried south-by-southeast by a wind gusting to 25 mph. The cloud enveloped the process building, and the acidic vapor fatally injured a chemical operator located within a structure approximately 70 feet southwest of the cylinder. Most of the approximately 40 workers at the site were in the plant lunch room and quickly evacuated the building. The airborne release continued for about 40 minutes, crossing an interstate highway one mile to the south and passing over private residences beyond.

The licensee immediately notified local, State and Federal officials. Four injured workers were transported to a local hospital. A private physician arrived at the site within one hour of the accident and examined plant workers. During the afternoon, residents living downwind of the site were personally notified to go to nearby hospitals and clinics for examinations.

An NRC Augmented Investigation Team was formed to investigate the incident. Their report and findings (NUREG-1179, Vol. 1) were published in February 1986. An assessment of the public health impact of the accident (NUREG-1189) was published in March 1986.

After the January accident, the licensee set out to drain the UF-6 remaining in plant vessels into 10-ton shipping cylinders, in order to facilitate needed modification at the plant; this work commenced on March 12, 1986. During the draining process, a scale malfunctioned causing UF-6 to be drained into a cylinder in excess of limits. Most of the excess material was immediately evacuated from the cylinder before the UF-6 solidified. The cylinder was not heated, and was undamaged. The results of the NRC investigation into this overfill event—together with a report of a detailed metallurgical examination performed on the cylinder damaged on January 4, 1986—were published in June 1986 (NUREG-1179, Vol. 2).

Investigation disclosed that the causes of the January 4, 1986 event were (1) the physical equipment and facilities used for filling and weighing UF-6 cylinders were inappropriate for safe use with 14-ton cylinders, and (2) the training of workers in operating procedures, and ensuring implementation of the procedures, were not carried out effectively.

The causes of the March 13, 1986 event were that (1) the scale used for weighing the cylinder being filled malfunctioned, apparently because of inadvertent damage incurred during decontamination after the first incident; (2) the procedures for draining did not include any provisions for ensuring proper scale function; and (3) the supervisor in charge of the operation did not recognize early indications of malfunction (an operator advised management of peculiar scale behavior during the filling of the cylinder).

The licensee agreed to keep the plant shut down until equipment modifications were made, plant personnel are retrained, plant procedures are rewritten, organization changes have been implemented, and the NRC approves plant restart.

An NRC Lessons Learned Task Group was formed to review regulatory practices regarding these kinds of fuel facilities in general. The Group interviewed appropriate members of the NRC staff, the licensee, and State and local authorities. A Lessons Learned Report was completed in May 1986 and published in June 1986 (NUREG-1198). A request to restart the facility was received by NRC in May 1986 and was under review at the close of the fiscal year. The NRC is monitoring licensee plant modification work; enforcement actions were pending.

Amidst the publicity surrounding the Sequoyah Fuels accident, NRC Region III (Chicago) received an inquiry concerning a newspaper report about an incident occurring December 7, 1984, at the Allied Chemical Company plant in Metropolis, Ill., involving overfilling and subsequent damage to a UF-6 cylinder. The licensee was asked about the incident and provided the requested information. The incident had not been previously reported to the NRC; the licensee stated that it had considered reporting it but concluded that it did not meet any NRC reporting requirements.

The overfill incident occurred on December 7, 1984, when a cylinder was overfilled and subsequently damaged when it was heated to remove the excess uranium hexafluoride. There was no release of any uranium hexafluoride to the environment as a result of the incident, and there were no injuries. The licensee later provided information to the NRC on overfill incidents at the Metropolis facility for the years 1981 through 1985. During this period, there were 41 overfills—of which three were greater than 1,000 pounds. With the exception of the December 7, 1984 event, none of the other overfill incidents involved damage to the cylinders. No releases of UF-6 occurred in any of the incidents. Another overfill incident occurred on March 23, 1986, attributed to an operator error, when a cylinder was overfilled by 1367 pounds; the excess was successfully removed without applying additional heat.

After the December 1984 incident, the licensee installed new load scales at each fill location to provide clearer, more reliable weight measurements in the control room. A scale was also added to the overhead crane used to lift the cylinders to allow weighing of the cylinders without transporting them more than 50 feet to another weighing location. After the January 1986 Sequoyah Fuels accident, the licensee installed a flow totalizer, which measures the flow rate of the liquid UF-6 and has an alarm and automatic shutdown function based on total flow and data from the load scale. The licensee also improved its training and retraining programs, procedures, and the level of supervision for cylinder filling activities.

In response to the January 1986 accident at Sequoyah Fuels, NRC Region III conducted a special investigation at the Metropolis facility on January 14-15, 1986 to observe the Allied Chemical Company cylinder handling procedures. Additional inspections were conducted to examine the circumstances of the December 7, 1984 incident, and the licensee's actions to preclude the occurrence of significant overfills.

Region III issued a Confirmatory Action Letter to the licensee on January 10, 1986, documenting the licensee's agreement that no overfilled cylinders would be heated without the review and concurrence of Region III. A second Confirmatory Action Letter was issued on March 24, 1986, documenting the licensee's planned actions in response to the March 23, 1986 overfill incident. These corrective measures included increased supervision of filling activities, the prohibiting of cylinder filling unless two independent methods are available to determine the amount of UF-6 in a cylinder, and completion of the installation of the new UF-6 flow readout and alarm functions by April 15, 1986.

Although no regulations exist for off-site emergency response, the licensee has taken the initiative to work with the appropriate off-site groups to establish a coordinated capability.

On June 27, 1986, the NRC forwarded to the Allied-Signal Corporation (parent company of Allied Chemical Company) a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $25,000. The violations included the failure to report the December 7, 1984 incident to the NRC, and three instances of failure to follow procedures during the March 23, 1986 overfill incident. (AO 86-3.)

Communications Breakdown Causes Overdose. On December 28, 1984, the NRC Region I Office (Philadelphia) received written notification, dated December 24, 1984, of a diagnostic misadministration that occurred at the Hospital of St. Raphael, New Haven, Conn., on August 7, 1984. A patient was administered a 10-millicurie (mCi) dose of iodine-131 instead of an intended 400 microcurie (uCi) dose of iodine-123.

The licensee stated that in December 1984 the patient's hormone values were in the borderline hypothyroid level and the hospital had recommended that the patient be placed on exogenous thyroid medication, possibly to be continued for life. The patient was placed on this recommended medication and is under a physician's care. The hospital determined that the patient received on the order of 2,000 rads to the thyroid from the iodine-131. The NRC medical consultant believed the dose to be closer to 6,000 rads. (A rad, an acronym for "radiation absorbed dose," is the basic unit for measuring absorption of ionizing radiation.)

Based on NRC inspection findings, the principal cause of this incident appeared to be inadequate communications between physicians and technologists. Other causal factors include the lack of written orders or schedules; the lack of review, approval, and scheduling of procedures through the radiologist; and the possible need for some clinical retraining of the technologists.

In regard to licensee corrective actions, clinical retraining has been provided for the technologist, procedures were altered, and upgraded supervision and concurrence measures have been provided.

The licensee was issued a Notice of Violation by NRC Region I for failing to report the misadministration to the NRC within the required time and for failing to require a written request from the physician performing the procedure. (AO 85-8.)

Mistaken Identity Leads to Medical Misadministration. On March 19, 1985, a patient at Mercy Hospital, Pittsburgh, Pa., received five millicuries of iodine-131 rather than 10 millicuries of technetium-99m, in a routine thyroid scan. The normal iodine-131 dose for thyroid uptake and scan can be from five to 100 microcuries.

Two outpatients were scheduled for nuclear scans at Mercy Hospital. One was scheduled for a routine thyroid scan using 10 millicuries of technetium-99m. This amount normally results in a dose to the thyroid of about 30 to 40 rads. The second patient was scheduled for an iodine-131 whole body scan. This study is less common and is normally done following the removal of a cancerous thyroid to detect metastases in other parts of the body. When the patient who was scheduled for the routine technetium-99m study arrived, the nuclear medicine technologist gave the patient the dose for the whole body scan without verifying the patient's identity. The mistake was recognized immediately, and potassium perchlorate was promptly given to reduce the thyroid uptake.

The result of the misadministration was that a patient received an unnecessary dose to the thyroid, estimated by the licensee to be about 1,000 rads. An NRC medical consultant agreed with the licensee's prompt action in administering potassium perchlorate. He estimated a dose of 45 rads to the thyroid, considerably less than the dose estimated by the licensee. The consultant concluded that the increase in the probability of developing cancer from this exposure was inconsequential, especially in light of the fact that this patient had subsequently undergone a partial thyroidectomy.

The licensee re-emphasized to its personnel the importance of correctly identifying patients before administration of doses. No violations of NRC regulations were associated with this incident. (AO 85-9.)

Breakdown in Management Controls. On May 24, 1985, the NRC issued an Order (effective immediately) to Pittsburgh Testing Laboratory (PTL), Pittsburgh, Pa., which required: (1) the removal of the District Manager and Radiation Safety Officer (DM/RSO) for the licensee's Cleveland, Ohio facility, and (2) the suspension of all licensed activities at the Cleveland, Ohio facility until certain conditions were implemented. This action was taken as a result of the licensee's assigning uncertified people to perform radiography, providing false information to the NRC, and falsifying training records.

NRC investigations showed that an uncertified individual was deliberately directed by the District Manager of the Cleveland, Ohio facility to perform the duties of a radiographer on seven occasions during February and March 1984. Later the manager deliberately directed, on two occasions, another uncertified individual to perform the duties of a radiographer. The investigation also showed that the manager had falsified the training records of the uncertified individual who had performed the licensed radiography activities during February and March 1984, so as to indicate that the individual had received the required training.

When questioned about the occurrences during the August inspection, the manager denied them. In a subsequent NRC investigation, he admitted the occurrences and admitted deliberate violation of NRC requirements.

There was no evidence that any overexposures occurred while the uncertified personnel performed radiography. However, the use of radiographic devices by these personnel constituted a significant hazard not only to themselves, but also to their fellow workers and to several members of the public who were working in the areas where the radiography was being performed.

The cause of the violations is a serious breakdown in management controls, both at the licensee's Cleveland facility and at the corporate headquarters in Pittsburgh.

This case was subsequently referred to the U.S. Department of Justice for prosecution. The NRC held additional enforcement action in abeyance pending any action by the Department. On February 19, 1986, the former Radiation Safety Officer of the Cleveland facility, and the President of Pittsburgh Testing Laboratory, acting for the Corporation, appeared in the U.S. Federal District Court for the Northern District of Ohio and pled guilty to violations of 18 USC 1001 and the Atomic Energy Act. In accordance with a plea bargaining agreement, the former Radiation Safety Officer was fined $2,500 and the Corporation was fined $15,000. The judge strongly admonished the defendants regarding the seriousness of their actions.

On April 7, 1986, the NRC issued to the licensee a Notice of Violation and Proposed Imposition of Civil Penalities in the amount of $58,000. This enforcement action was taken because of the deliberate violations of NRC requirements, the falsification of records by site management, and the lack of candor demonstrated by both site and corporate management in their dealings with the NRC. (AO 85-10.)

Iridium Inplant Delivers 14,000-Rad Dose. On June 17, 1985, Christ Hospital in Cincinnati, Ohio, reported to the NRC that a 57-year old patient had received a 14,000-rad dose to the left lung instead of the prescribed 5,000 rads. The patient was to be treated for lung cancer using implanted iridium-192 radiation sources designed to deliver a prescribed radiation dose of 5,000 rads to the left lung. The radiation physicist stated that she did not calculate the treatment time needed for delivering the prescribed dose, but the physician on the case stated that a 50-hour treatment period had been proposed by the physicist.

On June 17, 1985, following discussions between the physician and the radiation physicist, the radiation physicist realized that no calculations had been performed to determine the treatment period. The physicist then determined that, by that time, 14,000 rads had been delivered to the patient's left lung, and the iridium seeds were promptly removed.

The patient was monitored by the licensee through periodic examinations and bronchoscopy. No short term effects were observed. The nature of the radiation therapy with the iridium-192 seeds is such that the principal radiation effects would be very localized in the left lung area. An NRC medical consultant has reviewed the case and is satisfied that adequate medical followup is being provided by the licensee.

The misadministration was caused by the failure of the radiation physicist to calculate the treatment period for the iridium-192 seeds to remain implanted in the patient. The hospital had no quality assurance procedures designed to verify the accuracy of radiation treatment plans and assure that they would deliver the prescribed radiation dosage.

On June 19, 1985, the hospital implemented revised treatment procedures to require that written calculations be carried out in connection with any radiation therapy plan prior to the initiation of a treatment and that the calculations be reviewed by a second qualified individual.

The NRC conducted a special inspection of the licensee's radiation therapy program and the circumstances of the misadministration. The licensee's corrective actions were judged to be acceptable, and no violations of NRC requirements were identified. On September 11, 1985, the NRC issued a Confirmatory Order Modifying License requiring the licensee to immediately implement the revised treatment procedures. (AO 85-11.)

Palliative Therapy Miscalculated. From October 17, 1984 to November 1, 1984, a patient being treated on the cobalt-60 teletherapy unit at the University Health Center of Pittsburgh's Joint Radiation Oncology Center, Magee—Women's Hospital site, received a radiotherapy administration of 3,584 rads rather than the prescribed 2000 rads.

The patient was receiving the second of two courses of therapy to the ninth and 10th ribs using cobalt-60 external beam therapy. This region was being treated for palliative purposes, to relieve pain from metastatic disease. The first administration to the area, 2000 rads in five treatments, prescribed on September 13, 1984, had been completed without any problems. The prescription for the second course of treatment, prescribed on October 16, 1984, was 2000 rads to be delivered in 10 treatments. However, when the treatment dose for the second course was calculated, the dosimetrist assumed that the prescription was the same as the earlier one.

Rather than checking the prescription and preparing a new calculation, as required by the Joint Radiation Oncology procedures, the dosimetrist relied on a verbal communication and only "decay-corrected" the output from the first treatment. As a result, the patient began to receive treatment fractions that were twice those of the prescribed dose. This error was not discovered until the patient had received 3,584 rads in the second course of therapy, when one of the treatment technologists noticed that the delivered dose differed from the prescribed dose by more than 10 percent. Further treatment was stopped at that time.

The consequence of this incident was that the patient received an unprescribed dose to the ribs of 1,584 rads. The licensee reported that, although the patient received more radiation than was prescribed, the patient has not, and likely will not, suffer any ill-effects other than a modestly aggravated soft tissue reaction. The licensee reported that the actual dosage received is within a clinically acceptable range for the desired effect.

The cause of the incident was defined as failure to comply with established procedures and oversights by the responsible staff. The policies and procedures which should have prevented this occurrence are clearly set forth in a Physics Procedure Manual in the possession of each of the licensee's dosimetry staff. The policies were reviewed at inservice meetings held for the staff on July 2, 1984, and July 16, 1984, which dosimetrists and technologists attended.

Enforcement of these policies will receive the continued vigilance of management and supervisory staff. A general meeting of the entire physics and dosimetry staff was held on November 8, 1984, to discuss this issue and to reemphasize the importance and necessity of strictly following the procedures.

No violations of NRC regulations were associated with this incident. An NRC medical consultant reviewed the case and confirmed that the second course of treatment to metastases involving the ninth and 10th ribs was about 40 percent greater than that which had been planned. He concluded that the misadministration did not result in significant harm to this patient, and in fact might be suspected to have produced an excellent palliative therapeutic result. (AO 85-15.)

Mix-up of Bottles Causes Overdose. On October 25, 1984, NRC was notified that a patient of the Milton S. Hershey Medical Center in Hershey, Pa., received 15 millicuries of iodine-131 rather than the prescribed dose of 10 millicuries. A 10 millicurie dose of iodine-131 had been ordered for one patient for treatment for hyperthyroidism, and a five-millicurie dose had been ordered for a second patient as a whole body scanning dose. When the first patient arrived, the technologist opened the bottle containing capsules which had the first patient's name on it, dumped them in the patient's hand and gave the patient water to take them with. The technologist neither verified the activity nor the number of capsules. Later in the day when the second patient arrived, the bottle with this patient's name on it was found to be empty. Several days passed before the licensee was successful in contacting the first patient who remembered taking three capsules, rather than the appropriate two.

The capsules were provided in vials labelled with each patient's name by Nuclear Pharmacy, Inc. It appears that, although Nuclear Pharmacy accurately verified the activity of each capsule before dispensing, they dispensed three capsules in one bottle and none in the second. The prescriptive-information labels indicated two capsules of five millicuries each in the first bottle and one capsule of five millicuries in the second bottle.

The referring physicians decided that health effects of the error should be minimal and involve only an increased probability of the first patient ultimately developing hypothyroidism.

The incident was reviewed by an NRC medical consultant. The medical consultant estimated the delivered dose to the thyroid to be 12,000 rads rather than the intended 8,000 rads. The patient is clinically hypothyroid and is doing well on medication. The consultant concluded that hypothyroidism also would have resulted from the intended dose, and that many clinicians routinely choose the higher dose for initial treatment.

The cause of the error was twofold: failure on the part of a nuclear medicine technologist to verify the activity of the administered dose and failure on the part of Nuclear Pharmacy to properly dispense two patient doses. Nuclear Pharmacy, Inc. has significantly improved management control of its dispensing operations. Significant improvement was also made to recordkeeping and auditing procedures. (AO 85-16.)

Exposure of Radiographic Personnel Due to Management and Procedural Control Deficiencies. On August 5, 1985, Western Stress, a firm with offices located in Evanston, Wyo., and Houston, Tex., notified the NRC Region IV Office (Dallas) that radiographic personnel had received whole body radiation exposures in excess of NRC regulatory limits. Subsequent NRC inspections showed that the root causes of the over-exposures were serious management and procedural control deficiencies.

On August 1, 1985, a radiographer and his helper went to a site at Table Rock, Wyo., to perform radiography with a radiographic camera containing a 29 curie Ir-192 source. After radiographing some welds, the helper began developing the film while the radiographer disassembled the radiographic equipment. After being informed that one of the films did not receive the sufficient degree of exposure, the radiographer went to examine it. As he did, the helper reconnected the drive cable and radioactive source guide tube to the radiography camera. The source was cranked out, and a second exposure of the weld was made. Next, the radiographer developed the film as the helper disconnected the equipment. Neither realized that the radioactive source had not been connected to the drive cable and therefore could not be returned to the exposure device. Consequently, the radioactive source remained in an unshielded condition at the end of the guide tube. The radiography camera was placed near the rear of the truck, but the guide tube and cable were left lying on the ground approximately 20 feet from the weld. The two men then prepared to perform the job of stress relief around the weld. The process took about five hours. During this operation, they were either in the truck cab or watching the instrumentation near the weld.

Upon completion of the work, all equipment was placed in the truck, including the guide tube containing the source, and the men returned the truck and equipment to the Western Stress facility in Evanston. The following day, two other radiographers took the truck to a job site at Black Canyon, Wyo. Radiography was performed using the guide tube containing the 29-curie Ir-192 source attached to a radiography camera different from the camera used the previous day. Several exposures were made, and the developed film was found to have double images. The radiographer then became aware there was a problem and placed the exposure device and guide tube in a transport container and covered it with bricks. The radiographers contacted the company Radiation Safety Officer, and the truck and equipment were returned to the Western Stress facility in Evanston. The source was secured by the Radiation Safety Officer the following day.

The personnel dosimeters of the employees involved in the incident were evaluated and indicated whole body radiation doses of 22.1 rems, 7.4 rems, and 0.6 rems respectively to the original radiographer, to his helper, and to another employee. (A rem, an acronym for "roentgen equivalent man," is a unit expressing the dose of ionizing radiation that will have the same biological effect as one roentgen of X-ray or gamma-ray dosage.)

On August 6, 1985, NRC Region IV (Dallas) inspectors met with representatives of Western Stress in Evanston, Wyo., to discuss the incident and evaluate all available information relative to the event.

Subsequently, on August 12, 1985, an anonymous caller contacted NRC Headquarters concerning Western Stress and stated that there had been other work done involving the truck with the exposed source, by radiographic personnel who were unauthorized and who did not wear personnel monitoring devices. The job site at which such work was performed—after the Table Rock and before the Black Canyon jobs—was at Green River, Wyo.

On August 13, 1985, NRC Region IV was notified by Western Stress management that additional use of the truck containing the radioactive source while in the unshielded condition had not been reported to the NRC during the week of August 6.

An extensive inspection was launched by NRC Region IV personnel at Evanston, Wyo., on August 14, 1985. The inspection confirmed the information reported by the anonymous caller and later reported by company management. Interviews with radiographic personnel and reenactment of the events indicated that as many as six members of the general public may have received some exposure. Best estimates were that the exposures were very low.

Oak Ridge Associated Universities Medical and Health Science Division performed cytogenetic dosimetry evaluations on blood samples taken from the radiographic personnel involved. Results of these studies showed, at the 80 percent "confidence level" that the original radiographer's dose was not smaller than eight rads nor larger than 31 rads, and that his helper's dose was not greater than 15 rads.

The root cause of this problem was judged to be a serious breakdown in management controls and oversight of the licensed program. On August 9, 1985, Western Stress voluntarily agreed to suspend operations until management had made the necessary changes in the program to satisfy the NRC that they could meet the NRC's regulatory requirements. Permission to resume operation was given on October 3, 1985, after an additional NRC Region IV inspection confirmed that program improvements had in fact been made. A license amendment was subsequently issued on October 4, 1985, to Western Stress, which included procedural and management changes.

On August 21, 1985, an enforcement conference was held in the NRC Region IV Office with members of Western Stress management. Among the items discussed were the use of unauthorized radiographers to perform radiography, failure to wear personnel monitoring equipment, multiple failures to make the radiation surveys required by regulations or company procedures, and the general breakdown in management controls and oversight of the licensed program. The event was still under review by the NRC at the close of the report period. (AO 85-17.)

Calibrator Moves Decimal Point—Dose Ten Times Prescribed. On August 19, 1985, Riverside Methodist Hospital of Columbus, Ohio, reported to the NRC that a 78-year-old patient had received a radiation exposure from a diagnostic test that was 10 times greater than intended.

On August 17, 1985, the patient underwent a "blood pool imaging" study. This diagnostic test involves injecting a radioactive material (sodium pertechnetate-99m) into the patient and then recording the movement and location of the radioactive material with a scanning device. The diagnostic test called for use of 20 millicuries of the sodium pertechnetate-99m; but the patient received 200 millicuries.

A technologist prepared the material for the test, using a dose calibrating device to measure the amount of radioactivity. Measurements were made of the bulk supply of the sodium pertechnetate and of the single-dose syringe prepared by the technologist. The dose calibrator malfunctioned in both measurements, showing a measurement which was 1/10th of the actual amount. Thus the dose measured as 20 millicuries in the calibrator was actually 200 millicuries.

The misadministration was caused by malfunction of this dose calibrator, whose digital display misplaced the decimal point. The error was discovered when the scanning test was performed. The licensee calculated that the patient received a whole body radiation dose of 3.28 to 3.5 rads. This level is far below the point where any detectable medical effects would be anticipated.

The dose calibrator had previously malfunctioned in June 1985 and had been returned to the manufacturer for service. The malfunction by a factor of 10 resurfaced in August. The Chief Technologist was not informed of the problem, and no action was taken at that time. After the misadministration occurred, the licensee attempted to duplicate the instrument malfunction, but was unable to do so. It was placed back in service until August, when the malfunction recurred. The device was then returned to the manufacturer for repair.

A special NRC inspection was conducted on September 3, 1985, to review all circumstances of the misadministration. The licensee's handling of the incident and the corrective measures taken were found to be acceptable. No violations of NRC regulations were identified. (AO 85-18.)

Wrong Isotope and Wrong Dose Administered. On August 14, 1985, a patient at the Letterman Army Medical Center, Presidio of San Francisco, California, was inadvertently given the wrong dose of the wrong radiopharmaceutical for a scheduled thyroid uptake study and scan. This resulted in an administered dose which exceeded the prescribed dose by a factor of about 30.

The attending physician mistakenly prescribed a dose of 150 microcuries of iodine-131 instead of iodine-123. The radiopharmacy misinterpreted the prescribed dose as five millicuries of I-131. The five millicuries of I-131 were administered to the patient on August 14, 1985, when the patient returned to the hospital for the uptake study and scan.

The level of radiopharmaceutical administered is commonly given to patients for certain other diagnostic procedures, and, despite the mistake, the diagnostic scan and uptake desired for this patient were accomplished without administration of any additional radiation beyond the initial dose. Also, the patient had previously undergone a partial thyroidectomy and was taking thyroid hormones for thyroid gland suppression. The licensee states that, because of these circumstances, no adverse clinical symptoms were expected as a result of the misadministration.

The patient and the attending physicians were notified of the misadministration, and the licensee began an immediate investigation to determine what factors and circumstances may have contributed to the incident.

The misadministration was caused when the attending physician prescribed the wrong radiopharmaceutical, which was further misinterpreted by the radiopharmacy as a request for five millicuries of I-131. Effective August 16, 1985, the licensee instituted a new hospital procedure which would require that only the nuclear medicine staff would administer radiopharmaceuticals, and the radiopharmacist must authorize the release of radiopharmaceuticals from the pharmacy. All prescriptions will be in writing.

NRC discussed all aspects of the misadministration with the licensee. The licensee's corrective actions were determined to be acceptable. The NRC does not plan any further actions. (AO 85-21.)

Misadministration Treating Scar Tissue in Eye. On October 9, 1985, at the Queen's Medical Center, Honolulu, Hawaii, a patient was to receive 1,000 rads to the lateral limbal area of the right eye using a strontium-90 applicator; however, the medial limbal area of the right eye was treated. The attending physician realized the error when the patient returned for his second treatment on October 16, 1985. The patient and referring physician were informed of the error immediately. According to the attending physicians, there are no complications to be anticipated due to the incorrect treatment.

The misadministration was caused when the attending physician misinterpreted an area containing scar tissue in the medial limbal area of the right eye as the area to be treated. There were no written treatment instructions for the attending physician. Effective November 20, 1985, the hospital has made it mandatory that a written requisition be submitted by the referring physician prior to treatment. This requisition must include the patient's name and a description of the intended treatment area clearly identified. The physician must then sign and date the requisition. This requisition shall be kept in the patient's chart along with the treatment summary.

The misadministration incident was reviewed by members of the NRC Region V (San Francisco) management staff during a visit to the hospital on December 19, 1985. The licensee's corrective actions were determined to be acceptable. The NRC does not plan any further actions. (AO 85-22.)

Protocol Under Review Following Overdose. On December 9, 1985, a patient at Hospital Universitario in San Juan, Puerto Rico, received 4.98 millicuries of iodine-131 instead of the 10-to-15 microcuries dose usually given for a 24-hour thyroid uptake test. The patient arrived at the hospital's Nuclear Medicine Division on December 9, 1985, to receive iodine-131 for a 24-hour thyroid test. The test was part of the physician's plan to evaluate the patient for hyperthyroidism. Instead of the 10-to-15 microcuries usually given at this facility for such a procedure, the technologist mistakenly administered a dose of 4.98 millicuries, which is the customary dose for whole body scans with iodine-131.

The patient's referring physician was notified of the misadministration. Based on statements from the physician, the patient was a likely candidate for iodine-131 therapy for treatment of the hyperthyroid condition; therefore, the probable consequences for the patient would be consistent with the projected medical treatment.

The reason for the misadministration was an error by the technologist. The licensee re-examined the protocol used for hyperthyroid patients given radioiodine doses with the nuclear medicine staff. The NRC was to review the incident and the licensee's protocol during the next NRC routine inspection. (AO 85-23.)

Wrong Kidney Transplant Patient Irradiated. On February 7, 1986, a patient at the Washington Hospital Center, Washington, D.C., received a cobalt-60 teletherapy treatment of 150 rads to the abdomen which was intended for another patient. On February 6, 1986, an attending surgeon of the Renal Transplant Unit had ordered radiation therapy for one of his patients consisting of 150 rads per day, to be repeated every other day for a total of 600 rads. The treatment was intended to forestall rejection of the kidney implanted on the previous day. The Unit clerk, in entering the order for the treatment into the computer, for transmission to the Radiation Therapy Department for scheduling purposes, ordered the treatment for the wrong patient by a careless use of the computer light pen.

The wrong patient, who was also a kidney transplant recipient, was brought to the radiation therapy department on the morning of February 7. A radiation therapy physician checked her chart, noted that there was no order in the chart for radiation therapy, but, contrary to hospital policy, directed the technologist to administer the treatment anyway, since the computer schedule showed this patient's name. The mistake was discovered that afternoon and the correct patient was subsequently treated.

The consequence of this incident was that the patient received 150 rads to the abdomen contrary to the wishes of her physician. The physician stated later that, if in the future the patient showed signs of rejecting the kidney that had just been implanted, he would prescribe a course of radiation therapy similar to that which had been mistakenly employed. It was also to be noted that some physicians who perform renal implants routinely prescribe radiation therapy without waiting for evidence of rejection. The licensee's medical staff has concluded that the patient should experience no clinical complications.

The cause of the event was the failure of the radiation therapy physician to follow proper procedure. The physician should have investigated why a patient presented for radiation therapy did not have an order for such therapy written in her chart.

The licensee voluntarily suspended patient treatment pending the results of an internal investigation, and discussion of these results with NRC Region I (Philadelphia). Subsequently, the licensee agreed to take measures assuring that an authorized physician reviews every patient chart prior to the initiation of treatment and confirms that such treatment has been requested and is appropriate; the hospital will also require consultation between an authorized user and the referring physician prior to the initiation of treatment of any patient.

The licensee was inspected by an NRC Region I inspector on February 10-11, 1986, and the incident was reviewed by an NRC medical consultant. A Confirmatory Order Modifying License was issued on May 29, 1986, requiring that an authorized physician-user review every teletherapy patient chart to confirm that cobalt-60 teletherapy treatment has been requested and that the authorized physician-user consult with the referring physician or the Chief Resident prior to the initial treatment of each teletherapy patient. In their response to the Order, the Washington Hospital Center confirmed that the required procedures had been in place since February 18, 1986. The May 29, 1986 NRC letter also forwarded a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $5,000. (AO 86-4.)

Industrial Gauge Transferred from Authorized to Unauthorized User. On February 19, 1986, while checking a licensee which had apparently ceased operations, an NRC Region III inspector (Chicago) determined that an industrial gauge, containing a sealed source of cobalt-60, was in an unrestricted area of the former factory site. Subsequent inspection determined that at least two members of the public had received exposures to radiation as a result of the improper disposal of the gauge. The gauge had been licensed to C-E Glass, Inc., a Division of Combustion Engineering, Inc. The company operated a facility in St. Louis, Mo., until October 1981.

C-E Glass, Inc., was licensed in 1971 for the use of a level measurement gauge containing 2.5 curies of cobalt-60. The source was replaced in June 1978. In October 1981, the facility and equipment at C-E Glass's site was transferred (a move not authorized by the NRC) to Hordis Brothers, Inc., which continued operations until May 1982. The facility and equipment were later sold by Hordis Brothers to a salvage company. The gauge was placed near a scrap pile at the site, and a salvage company employee removed the gauge's shutter control in early December 1984. For the next two months, two employees of the salvage company handled the gauge and worked near it. It was later moved to a scrap pile where access by other individuals was limited.

Interviews with the two salvage company employees revealed that they frequently worked or took breaks in the vicinity of the gauge. Calculations based on the radiation level—with the shutter of the gauge open—led to the conclusion that one individual had received a radiation exposure to his buttocks of 0.6 to 1.7 rems and to his leg of 69 to 208 rems. A second individual received a significantly lower radiation dose. The first individual was examined by a physician and his blood count, bone marrow, and physical condition were found to be normal.

The uncontrolled use of the gauge and radiation exposure of at least two individuals were caused by the transfer of the gauge by the licensee to an unauthorized organization. The licensee is no longer in business and has no other gauges in its possession. An NRC medical consultant was retained to review the case and to provide assistance to the exposed individuals' physicians. On June 30, 1986, the NRC forwarded to the licensee a Notice of Violation and Proposed Imposition of Civil Penalty in the amount of $15,000, for violations associated with the handling of the gauge. (AO 86-5.)

Safety Systems Bypassed in Breakdown of Management Controls. On March 3, 1986, the NRC issued an Order Suspending License (effective immediately) to Radiation Technology, Inc. (RTI), of Rockaway, N.J. The Order was based on NRC inspections which identified a number of instances of bypassing safety interlock systems; these events indicated a significant breakdown in the licensee's management control system.

RTI has been licensed to operate a large irradiator near Rockaway since November 1970. The licensee's irradiator uses sealed cobalt-60 sources to produce high intensity gamma-ray fields for the sterilization of medical equipment and supplies, and also for various other industrial and scientific applications. At the time of the March 3, 1986 NRC Order, the President of the company was also the Chairman of the Board of Directors and the Radiation Safety Officer.

RTI has been the subject of several escalated enforcement actions in the past. Its license was temporarily suspended following an incident in 1977 when a plant worker was able to walk into the irradiation room because safety interlocks on the personnel access door had been rendered inoperable. The events giving rise to the most recent Suspension Order came to light during a routine NRC inspection in September 1984, when the inspector discovered that the licensee had been operating the irradiator since April 1984 with an inoperable safety interlock on one of the two conveyor openings used to transfer products into the irradiation room. On September 26, 1984, Region I issued a Confirmatory Action Letter that documented the licensee's commitment to operate the facility only if all safety interlocks were operable and to cease operations if any safety interlock failed to function as required. In November 1985, the interlock was replaced with a new design, without required NRC approval. During an inspection on February 26, 1986, the NRC staff determined that the licensee had been operating the facility for several days prior to the inspection in disregard of a malfunction of a radiation monitor which actuates the lock that assures that the personnel door to the irradiation room cannot be opened while the sources are exposed. Following this discovery, the staff requested that the licensee cease all operations until the monitor was repaired; daily inspections of the interlocks were then initiated.

Subsequently, the licensee requested lifting of the March 3, 1986 suspension by letters to the NRC dated March 4 and 5, 1986. After the Region I staff met with the licensee on March 6, a more complete submission was provided by the licensee on March 10. This latter submission proposed interim plant operations under the surveillance of an independent third party, reporting directly to a member of the RTI Board of Directors, who, along with the licensee, would be responsible for assuring that the facility would be operated safely and in compliance with all NRC requirements. Further, an independent fourth party would monitor the activities of the third party on a weekly basis. Both parties would provide uncensored reports directly to the NRC. With the licensee agreeing to certain additional provisos, the NRC staff concluded that temporary resumption of facility operations under these conditions would not endanger the health and safety of the public. Accordingly, a Conditional Rescinding of the Order Suspending License was issued on March 13, 1986.

Under a license amendment, an individual who had joined the company in March 1986 became the new Radiation Safety Officer. The person who had formerly held this position no longer has direct contact with, or responsibility for, this function. At a meeting of the Board of Directors, this same individual resigned as President but remained Chairman of the Board. The responsibilities of President were being shared among three Vice Presidents while a new President was sought.

Besides the actions described above, the NRC, on June 23, 1986, suspended the license again, based on investigative findings indicating repeated and intentional violations of NRC requirements and impedance of NRC inspections and investigations. The license will remain suspended until the NRC is assured that the licensee will abide by license conditions in the future. (AO 86-6).

Tritium Overexposure and Laboratory Contamination. During a routine inspection on March 12, 1986, at Ferris State College in Big Rapids, Mich., an NRC inspector determined that, based on a review of bioassay test results, a licensee researcher had received an overexposure to tritium (hydrogen-3) during experiments on August 3, 1985. The tritium exposure was calculated to be the equivalent of a whole body exposure of about 21 rems, an amount not normally expected to produce any medically observable effects. Continuing NRC inspections showed that two laboratories had been contaminated and there were numerous deficiencies in the licensee's use and control of byproduct radioactive material.

Having been notified of the initial NRC inspection findings, the licensee removed the researcher from any work involving radioactive material, restricted access to the laboratory areas, and began decontamination of the laboratory facility. Decontamination was subsequently completed, and the facility was released for normal use.

The NRC issued Confirmatory Action Letters to the licensee on March 19 and 21, 1986, documenting the licensee's agreement to remove the researcher from work involving radioactive materials, to restrict access to the laboratory areas, to undertake decontamination of the facility, and to stop all licensed activities except those associated with the nuclear medicine school.

NRC inspections, which began March 12, 1986 and continued through April 17, 1986, identified a total of 20 violations of NRC requirements. On April 28, 1986, the licensee's NRC license was amended, significantly restricting the scope of the authorized activities and providing that any new activities must be reviewed and approved by the NRC. On July 11, 1986, the NRC issued the licensee a Notice of Violation and Proposed Imposition of Civil Penalties in the amount of $10,500. (AO 86-7.)

Agreement State Licensees

Overexposures of a Radiographer and an Assistant Radiographer. On February 6, 1985, a radiographer and an assistant radiographer employed by World Technical Services in Deer Park, Tex., received 8.3 rems and 34.3 rems in whole body exposures, respectively. The exposures took place at the Amoco refinery in Texas City, Tex.

The overexposures occurred when a radiographer allowed his assistant to connect the source in the radiographic device to the drive cable without checking the connection. Contributory causes of the exposures include failure to survey the radiographic device and guide tube between radiographs, leaving the survey meter by the radiographic device while performing radiography, failure to check the survey meter when approaching the exposure device, and failure of the radiographer to follow the company's emergency procedures for a source disconnect. The personnel have been reinstructed by the licensee in emergency procedures, and the radiographers have been reinstructed to check the connections made by the assistant radiographers. The State Agency investigated the incident and cited the company for violations of procedures. (AO AS85-5.)

Unreported Radiation Injury Discovered by Subsequent Employer. On April 20, 1984, an individual employed by BF Inspection Services in Midland, Tex., received an exposure that resulted in a radiation burn, while performing radiography in Seminole, Tex. The licensee failed to notify the Texas Bureau of Radiation Control (Agency) of the incident after it knew the radiographer had received a radiation burn. The burn was reported by Permian Industrial X-ray, present employer of the individual, to the agency on November 8, 1985, when the radiographer had an apparent recurrence of the wound.

The radiographer and assistant arrived at the job site at 9:00 a.m. and used a 90-curie iridium-192 source to x-ray pipe in a pipe rack until 11:00 a.m. The radiographer worked on the pipe rack removing exposed film and placing new film on the weld to be x-rayed. The assistant worked the "crankout." The radiographer did not have a survey instrument with him. He depended on the assistant to tell him when the radiation levels decreased on the survey meter kept at the crankout.

At approximately 10:00 a.m., the radiographer took the exposed film to the central processing facility. When this film was developed, it was darker than it should have been and appeared to be exposed longer than necessary. At 11:00 a.m., the radiographer decided to stop work and go to lunch. At this time he locked the radiographic device, removed the guide tube and put the dust cover on the front of the device. No survey was performed on the radiographic device or guide tube at this time. The equipment was placed in a bucket used to transport equipment on the pipe rack and left there during lunch.

Before leaving the job site, the radiographers went to the job supervisor. The radiographer was informed that it appeared that he needed to decrease his exposure time; he was asked for any other exposed film. The radiographer told the supervisor that he would bring the remaining exposed film for developing after lunch.

When the crew returned from lunch, the radiographer decided to quit for the day, since he could not determine the cause of the film being overexposed. The equipment was moved from the bucket to the back of the truck. The radiographer carried the guide tube and crankout in his left hand and the radiographic device in his right hand. When the radiographer disconnected the crankout from the radiographic device, he discovered that the source was not connected to the drive cable. The radiographer then looked at his pocket dosimeter and found that it was discharged beyond its range. He then asked the assistant to look at his dosimeter and was informed that it was not discharged beyond its range. The assistant radiographer was instructed to check the radiation level, and he told the radiographer that he had a reading of 90. (The radiographer and the assistant did not remember what scale the survey was set on.)

The radiographer shook the guide tube and heard something rattling. He carried the guide tube, still rolled up, and the radiographic device to a large concrete slab about 60 feet from the truck. The guide tube was unrolled and when he shook it, the source fell out. The radiographic device was placed on top of the source and the radiographer went back to the truck. He then approached the device with the crankout and passed the drive cable through the radiographic device. The device was then moved behind the connector end of the source pigtail and the pigtail and the drive cable were connected. The source was returned to its shielded position and locked in place.

When the crew left the job site, the radiographer notified the job supervisor of the disconnect. The supervisor instructed the radiographer to notify his radiation safety officer (RSO) of the disconnect and to leave his film for processing. The supervisor stated that when this film was developed, it appeared that it may have been fogged.

On the following Monday, the RSO and the radiographer inspected the equipment. According to the radiographer, nothing was found to be wrong with the equipment. The radiographer stated that he turned in his and his assistant's film badges for immediate processing. At this time, the radiographer did not demonstrate any symptoms of a radiation injury.

Approximately five-to-seven days after the disconnect, the thumb, index and middle fingers of both the radiographer's hands became red and swollen. The radiographer was seen by a doctor and the three blood tests were performed, showing nothing abnormal. After a period of approximately two months, the radiographer's hands appeared to heal.

During the first week of November 1985, when the radiographer was working for another company, the middle and index fingers of his left hand became red and swollen. He again went to see a doctor. He notified his employer of the injury. The company RSO then notified the agency of the injury.

Based on statements by the radiographer, Agency investigators calculated that, as a result of his exposure from carrying the equipment to the truck and recovering the source, the radiographer may have received up to 29,000 rems to his left hand and about 47 rems whole body exposure.

The apparent cause of the disconnection of the source was that the source pigtail was not correctly connected to the drive cable when the equipment was set up. The exposure and subsequent burn resulted when the radiographer did not follow the licensee's Operating Procedures or the *Texas Regulations for Control of Radiation,* and failed to perform a survey of the radiographic device or guide tube between radiographs, when the equipment was secured for lunch, or at the end of the day. The radiographer also failed to follow the licensee's Emergency Procedures for a source disconnect.

The licensee's initial response to the agency's compliance letter was not deemed satisfactory as to what actions it had taken to prevent occurrence of this type of accident. The licensee's initial report of the incident does not address calculations of the radiographer's exposure, nor measures taken to prevent a recurrence. The agency cited the licensee for 14 items of non-compliance with the *Texas Regulations* *for Control of Radiation* and undertook additional investigation. (AO AS86-1.)

Contamination of a Scrap Steel Facility. On May 24, 1985, two 20-cubic-yard roll boxes being transported to the hazardous waste site at Kettlemen Hills, Cal., from the Tamco Steel Company set off the radiation alarms at the way station at Newhall, Cal. The trucks were turned back to the originator of the shipment, at the direction of the California Highway Patrol (CHP), on advice of the California Radiologic Health Branch (RHB).

The Tamco Steel Company processes scrap steel purchased from various suppliers throughout California, Nevada, and Arizona, into construction rebar. The scrap is segregated by metal type and sent directly to the melting furnace without inspection. Investigation disclosed cesium contamination of these shipments, and of various components at the facility, including the furnace. Evidently a device or source containing approximately 1.5 curies of cesium had been brought into the scrapyard undetected and sent to the furnace as part of the routine melt. Scrap metal dealers as a normal practice do not screen for radioactive material. Examinations did not detect any contamination of the workers at the company, where operations were suspended.

On May 25, 1985, Tamco had a contractor on site to begin a thorough survey and develop a clean up plan. The initial plan for decontamination was developed with the RHB. On August 1, 1985 the State Compliance Inspection Team completed its final survey. The RHB issued a departmental letter dated October 8, 1985, which released the facilities and equipment for unrestricted use.

Tamco Steel has installed low-level radiation monitors at the gate to check scrap coming into the facilities and product shipment leaving, and will also physically survey all scrap steel before it is placed in the furnace. The cognizant State Agencies monitored the decontamination of the facility and corrective actions taken there, and, after a final survey of the facility, released the facilities and equipment for unrestricted use. (AO AS86-2.)

State Considers Requiring Device with Built-in Warning System. On August 25, 1985, an industrial radiographer received a radiation injury of his left hand and a whole body overexposure. At the time of the incident, the employee (employed by Boothe-Twining, Inc.) was performing radiography at the company's field site in the Kern River oil field in Bakersfield, Cal. He was using a 46-curie iridium-192 source contained in a radiographic projector.

The radiographer (who had four years of radiography experience with Booth-Twining) encountered great resistance with the source crankout. He approached and manually adjusted the camera to reduce the kink in the guide tube. During this action his hands grasped the lock box and guide tube connector. At the completion of this readjustment, he moved away from the camera and observed that his 200 mr pocket dosimeter read off-scale. However, he did not report his dosimeter was off-scale, but reported a pocket dosimeter reading of 119 mr to his supervisor. His film badge was sent in for reading approximately seven days after the accident, after symptoms of his dose to the left hand were manifested and reported to management. He was seen by and remained under the care of a physician. Based on time and motion studies, preliminary estimates indicated a left hand dose of about 2,000 rads and a whole body dose of about six rads.

A State Investigative Panel was convened by an order of October 16, 1985, to determine the causes and extent of the radiation accident, and to recommend corrective action. Among its findings were: the radiographer had failed to adhere to established radiation safety and operating procedures; management had failed to communicate forcefully its intolerance of deviation from established safety procedures; instruction of radiographers, and specifically the overexposed radiographer, was found to be unacceptable; responsibility for the radiation safety program, although vested in the Radiation Safety Officer (RSO) of the licensee, was in fact exercised by the president of the company; management audits of the overexposed employee's work as a radiographer was not conducted as required by license condition, and records were not maintained; repeated and serious violations have occurred in the licensee's operations; and failures of the RSO to conduct on-site management audits at the frequencies promised and to provide comprehensive refresher training for radiographers were in violation of license conditions.

The immediate cause of the overexposure was the failure of the radiographer to adhere to established radiation safety and operating procedures. As discussed above, contributing causes are the serious breakdowns in management and procedural controls in the licensee's conduct of radiographic operations.

A Notice of Violation was issued to the licensee by the California Division of Occupational Safety and Health (Agency) on December 11, 1985. The testimony of company employees including management affirmed that the violations did in fact occur. The response also outlined corrective action to prevent recurrence of these violations.

The State Investigative Panel concluded that if the radiographer had been wearing a functional pocket radiation alarm, the radiographer would have had ample warning that the source was not in its proper shielded position. The Panel further agreed that the introduction of pocket radiation alarms into the practice of industrial radiographers is now imperative. The introduction of pocked radiation alarms is expected to reduce the frequency of excessive exposures and minimize the incident of injuries by giving radiographers timely warning of exposed sources.

California will consider adopting regulations that would require use of appropriate pocket radiation alarms for all radiographers and radiographers' assistants. This requirement would supplement and not in any way displace the present requirement for use of a survey meter in conducting required radiation protection surveys for industrial radiography.

California also will consider promulgating regulatory requirements and otherwise encouraging the development of a radiographic projector with an integral warning system built into the device to indicate in unambiguous fashion the safe, intermediate or unsafe position of the source. This may be done by announcing proposed legislative requirements to authorize only devices with this feature, starting in 1990. (AO AS86-3.)

Radiation Injury to an Untrained Assistant Radiographer. On November 9, 1985, an individual employed as an assistant radiographer by Basin Industrial X-Ray in Odessa, Tex., received a radiation burn of his left hand and an estimated 129-rem whole body exposure. The licensee failed to notify the Texas Bureau of Radiation Control (Agency) of the incident. Another licensee informed the Agency on November 26, 1985 that an incident had occurred involving Basin Industrial X-Ray.

The radiography crew was performing work at Fabricators Contractors, Inc., during the evening of November 9, 1985. The assistant radiographer was shooting the welds and the radiographer was developing the exposed film. At approximately 11:30 p.m., the assistant radiographer noticed that his survey meter, placed approximately two feet in front and to the right of the radiograhic device (a 76-curie iridium-192 radiography camera), was off-scale after the source was supposed to have been returned to its shielded position. He then checked his pocket dosimeter and found it was discharged beyond its range. He notified the radiographer, who was unsuccessful in his attempt to return the source to its shield using the crankout. The radiographer checked his pocket dosimeter and found that it was not discharged beyond its limit. He then notified the local supervisor, who was acting as the local radiation safety officer (RSO). The radiographer was instructed to isolate the area and wait for the supervisor.

On November 29, 1985, the assistant radiographer met with a State Agency representative. At this time, the individual's left hand had redness from the wrist to the base of the little finger. On December 2, 1985, the individual had a blister from the wrist to the base of the little finger on his left hand. When the assistant's film badge was processed, it indicated an exposure of 129 rems. When asked about the radiographer's exposure, the RSO stated that he did not have his badge processed with the assistant's. The Agency again instructed the RSO to have the radiographer's film badge immediately processed. The radiographer's exposure was determined to be 28 rems.

Based on statements made by the assistant radiographer and a re-enactment of the incident, Agency investigators calculated the exposure to the assistant to be about 129 rems whole body. The exposure to his left hand is uncertain; it may have been as high as 30,000 rems, or even considerably higher.

The Agency's investigation found that the individual had not received radiation safety training or formal training in industrial radiography from the licensee. It also appeared that the individual had falsified his application station that he had previous experience.

When asked why the licensee did not report the incident to the Agency, the RSO stated that he did not realize the severity of the incident, since he had not been provided the full details by the radiography crew. The RSO knew that the assistant radiographer's pocket dosimeter was discharged beyond its range but did not return his film badge for immediate processing. The licensee failed to perform a detailed investigation of the incident when it appeared that there could have been a serious radiation exposure. The RSO also informed the Agency that he did not know that the drive cable had been broken. When asked by the Agency investigators, the RSO stated that he could not locate the broken crankout cable.

The apparent cause of the exposure and burn appear to be that the licensee permitted an individual to perform the functions of a radiographer without providing the proper safety training, and also that the individual failed to perform surveys between radiographs.

The licensee has instituted tigher controls on its initial training program and hiring procedures. The Agency has cited the licensee for items of non-compliance with the *Texas Regulations for Control of Radiation*. In addition, a complaint has been issued to the licensee, notifying him that the Agency intends to revoke the license. The investigation of this incident was continuing at the close of the report period. (AO AS86-4.)

Nuclear Materials

CHAPTER 5

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

Fuel cycle and material safety encompasses licensing and other regulatory activities concerned with (1) conversion of uranium ore concentrates (after mining and milling) to uranium hexafluoride; (2) conversion of enriched uranium hexafluoride to ceramic uranium dioxide pellets and subsequent fabrication into light water reactor fuel; (3) production of naval reactor fuel; (4) storage of spent reactor fuel; (5) transportation of nuclear materials; and (6) production and use of reactor-produced radioisotopes ("byproduct material").

Highlights of activity undertaken during fiscal year 1986 include:

- Actions in response to the accident of January 4, 1986, at Kerr-McGee-Sequoyah Fuels Corporation in Gore, Okla.
- Completion of 32 major and 92 minor licensing activities dealing with fuel cycle plants and facilities.
- Completion of 114 design certification reviews for transportation packages.
- Completion of nearly 6,000 licensing actions on applications for new byproduct materials licenses and amendments and renewals of existing licenses. Over 5,500 of these actions were carried out by the five Regional Offices; the remainder were completed at Headquarters.

FUEL CYCLE ACTIONS

Accident at Kerr-McGee—Sequoyah Fuels Corporation

On Saturday, January 4, 1986, at about 11:30 a.m., a 14-ton cylinder ruptured at the Sequoyah Fuels Corporation (SFC) site resulting in a massive release of uranium hexafluoride (UF-6), lasting about 40 minutes. One SFC worker was killed and some workers were hospitalized.

The sequence of events leading up to the accident occurred as follows: A 14-ton, 12-foot-long cylinder was inadvertently overfilled with UF-6 and removal of the excess UF-6 could not be completed before it began to solidify. When the cylinder was subsequently heated, overpressurization resulted in a 52-inch split parallel to the axis of the cylinder forming an opening about eight inches wide at the midpoint of the split. Because of the high pressure in the cylinder, the size of the opening, and the fact that the split was on the lower side of the cylinder, much of the UF-6 rapidly escaped from the cylinder.

As noted, one individual was killed by exposure to airborne hydrofluoric acid. The maximum uranium intake among on-site workers was approximately 28 mg and, of the approximately 100 off-site individuals who submitted urine samples, seven were found to have received low-level uranium exposure, ranging from 0.1 to 0.9 mg uranium intake. (It is estimated that the lower range of transient kidney damage involves an intake of 1.8 to 12 mg of uranium.)

Post-Accident Activities

An Ad Hoc Interagency Public Health Assessment Task Force was assembled to investigate this accident, included members from the NRC, the U.S. Environmental Protection Agency (EPA), the U.S. Department of Health and Human Services, the U.S. Department of Agriculture, the Lawrence Livermore National Laboratory, the Oak Ridge National Laboratory and the University of Rochester.

The Task Force recommended that workers exposed during the accident be monitored carefully for at least one to two years for evidence of pulmonary or renal injury resulting from the acute exposure to uranium and hydrogen fluoride.

An important aspect of the NRC's assessment of the accident was the effort to identify actions that the NRC and its licensees might reasonably take to improve protection of the public from these kinds of events and their consequences. On February 20, 1986, the Acting Executive Director for Operations (EDO) formed a Lessons Learned Group to prepare a report based on experience gained from this accident, proposing licensing and inspection measures to prevent similar accidents, and also clarifying the NRC's regulatory role regarding facilities of this type. A further goal was to assess the adequacy of the NRC response and follow-up to the accident.

The observations and recommendations of the Lessons Learned Group were published as NUREG-1198, "Release of UF-6 From A Ruptured Model 48Y Cylinder At Sequoyah Fuels Corporation Facility: Lessons Learned Report," June 1986. The staff completed its consideration of the 58 recommendations made in this report, and a response to each was provided in NUREG-1198 Supplement No. 1, August 1986.

In responding to these recommendations, the staff considered actions that should be taken (1) for the restart of the Sequoyah Fuels facility, (2) to effect near-term improvements, and (3) to improve the regulatory framework. Efforts to identify further improvements to the licensing and inspection programs were continuing at the close of the report period. These improvements will be suggested by several continuing studies including an analysis prepared by the Office of the General Counsel on NRC's jurisdiction over chemical hazards. Recommendations in these issuances will be combined with other conclusions to prepare a complete picture of the adequacy of the existing licensing and inspection programs to fulfill NRC's legal requirements and to protect workers, the public health and safety, and the environment. From these recommendations the staff has adopted over 100; some of the proposed activities will be completed in fiscal year 1987, and others will take longer. The efforts will be followed to completion in the Material Safety Regulatory Improvements tracking system.

On October 16, 1986, the Commission voted to allow restart of the Sequoyah facility after improvements in equipment, retraining of personnel, establishment of quality assurance programs and rewriting of procedures. The restart activities were to be monitored 24-hours-a-day by a third party oversight group and by NRC inspectors.

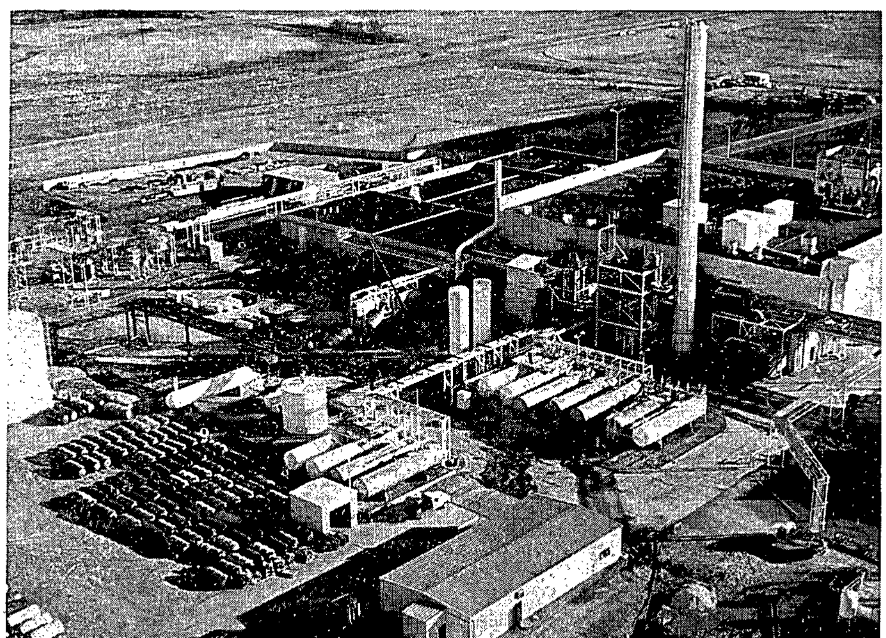
Hearings Involving Fuel Cycle Facilities

Sequoyah Fuels, Gore, Okla., UF-6 to UF-4 Production Plant. Kerr-McGee applied for permission to operate a UF-6 to UF-4 production plant. The accident of January 4, 1986, at the UF-6 production facility delayed the review process. The conference of parties to the informal hearing scheduled for August 11, 1986, was delayed and rescheduled for a week in January 1987.

West Chicago. In hearings involving the decommissioning of the Kerr-McGee Rare Earths Plant in West Chicago, Ill. (see the *1984 NRC Annual Report,* pp. 69, 70 and the *1985 NRC Annual Report,* pp. 79, 80) and the stabilization of the plant wastes, the Hearing Board directed the staff to supplement the Environmental Statement. The Draft Supplement was scheduled for issuance in December 1986, and the Final Supplement in July 1987. The Hearing Board has indicated that they may resume a portion of the hearing after the draft is issued.

Kress Creek. An Atomic Safety and Licensing Board (ASLB) decision dated June 19, 1986, dismissed the Order to Show Cause issued to Kerr-McGee Chemical Corporation on March 2, 1984, which would have required Kerr-McGee to prepare and execute a remedial action plan for the cleanup of radiologically contaminated areas in and along Kress Creek (near the West Chicago, Ill., facility) and the west branch of the Du Page River. On August 11, 1986, the staff filed an appeal with the ASLB.

Sequoyah Fuels Comprehensive Waste Disposal Plan. The hearing was held in abeyance pending submittal of a new plan, as part of the revision to the decommissioning plan required by Source Material License No. SUB-1010, Condition No. 21.



This aerial view of the Sequoyah Fuels Corporation plant at Gore, Okla., shows fuel casks (lower left) of the type that ruptured in an accident on January 4, 1986. An NRC task-group study of the accident and the investigative reports of an interagency task force resulted in publication of a "Lessons Learned Report" (NUREG-1198) in June 1986.

Incinerator Licensing

Babcock & Wilcox. Staff completed its Environmental Assessment in March and its Safety Evaluation Report in April 1986, related to Babcock and Wilcox's (B&W) proposed Volume Reduction Services Facility, comprising an incinerator and high-force compactor. In late September 1986, an informal hearing was held on this application near B&W's Parks Township, Pa., site. No decision was made by the Atomic Safety and Licensing Board Panel (ASLBP) Administrative Judge at that time.

Battelle Columbus. The staff also prepared an Environmental Assessment, in June 1986, on an application for the operation of an incinerator by Battelle Columbus Laboratories. This application was for a five-year demonstration of low-level radioactive waste volume reduction.

Decommissioning and Decontamination

West Lake Landfill, St. Louis County, Missouri. In 1973, approximately 7,700 metric tons (8,700 short tons) of leached barium sulfate residues were mixed with about 35,000 metric tons of soil and the entire volume emplaced in the West Lake Landfill in St. Louis County, Mo. This material derived from decontamination operations at the Cotter Corporation's plant where the material had been stored. Disposal in the West Lake Landfill was not authorized by the AEC/NRC and was at variance with the disposal location indicated in the NRC records. State officials were not notified of this disposal since the landfill was not regulated by the State at the time. An evaluation of environmental impacts of possible remedial actions to be taken on the site on a short-term basis (5-10 years) was conducted by the University of Missouri-Columbia (UMC). The UMC report, the 1986 radiological survey conducted by the Oak Ridge Associated Universities, and the NRC staff evaluation form the basis for documentation (NUREG-1222) scheduled to be published in the spring of 1987 which will specify options for the disposition of the radioactively contaminated residues presently in the landfill.

West Valley Demonstration Project. The Commission continued its safety oversight role for the West Valley (N.Y.) Demonstration Project (WVDP) in 1986. The primary purpose of this project is to demonstrate solidification and preparation of high-level radioactive waste for disposal in a Federal repository. The current schedule for WVDP indicates that the vitrification process for the high-level waste will start in West Valley by the end of 1988 and continue into the 1990's.

The Department of Energy (DOE) has published an Environmental Assessment for Disposal of Project Low-Level Waste and received Commission comments thereon. The DOE reported a "Finding of No Significant Impact" for the proposed disposal of project low-level waste.

The Safety Analysis Report (SAR) for WVDP is being prepared by DOE in separate sections, keyed to the development of discrete systems within the project. The Commission has received the following SAR sections for review and comment: Volume I-Project Overview and General Information, Volume III-Supernatant Treatment Systems, and Volume IV-Cement Solidification Systems. Subsections on Disposal Area Operations have also been submitted by DOE.

Interim Spent Fuel Storage

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

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

Two licenses for dry spent fuel storage were issued in 1986 under 10 CFR Part 72. A license was issued to the Virginia Electric Power Company (VEPCO) for its Surry nuclear power plant authorizing storage of spent fuel in General Nuclear Systems Castor V/21 casks. This cask is fabricated of nodular cast iron and has a capacity of 21 pressurized water reactor (PWR) assemblies aged at least five years since removal from the reactor. Loaded weight of the cask is about 117 metric tons. The VEPCO license covers use of up to 84 casks stored on concrete pads in a fenced, protected area on the site about one-half mile from the Surry reactors.

A second license for dry spent fuel storage was issued to the Carolina Power and Light Company (CP&L) authorizing the demonstration of a concrete module and stainless steel canister storage system at the Robinson nuclear power plant. Under the license, eight horizontal concrete silos may be constructed, each containing a steel canister with a capacity of seven PWR spent fuel assemblies aged at least five years from removal from the reactor. The loaded canisters are transferred from the reactor pool to the concrete modules by means of a modified spent fuel shipping cask. The eight storage modules for CP&L storage demonstration program will be located inside the fenced reactor area.

The concrete module and stainless steel canister system design was submitted to the staff in a topical report by

MATERIALS LICENSES ADMINISTERED BY NRC*
(AS OF SEPTEMBER 1985)

Licenses Administered By:

Headquarters	500
Region I	2800
Region II	1000
Region III	3600
Region IV	900
Region V	300
Total NRC Licenses	9100**

*In addition to the NRC licenses, some 14,000 licenses are administered by 27 states which have authority over certain materials under regulatory agreements with the NRC.

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

NUTECH, Inc., and received an NRC staff letter of approval in March 1986. The topical report, as approved, was referenced by CP&L in its license application.

Four topical reports for dry casks of varying designs have been submitted by Nuclear Assurance Corporation (NAC), Westinghouse, Transnuclear, and Combustion Engineering, and they are presently being considered by NRC staff. If found acceptable, these topical reports may be referenced by a utility in a license application or in an amendment to an existing Part 72 license to expedite the review of a proposed dry storage system or proposed modification to an existing system.

Monitored Retrievable Storage

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

The MRS proposal was submitted to the Commission on December 24, 1985, for review and consultation. Commission comments on the MRS proposal were forwarded to DOE on February 5, 1986.

The State of Tennessee filed a motion in the U.S. District Court for the Middle District of Tennessee requesting an injunction to prevent DOE from submitting its proposal and, therefore, the submission of Commission comments to the Congress. The Court granted the injunction and DOE appealed the decision.

In connection with its comments, the Commission staff issued NUREG-1168, "Staff Evaluation of U.S. Department of Energy Proposal for Monitored Retrievable Storage." The Commission also issued proposed revisions to 10 CFR Part 72, intended to provide a regulatory framework for licensing an MRS.

Licensing of Uranium Enrichment Facilities

Several meetings were held with industrial representatives interested in obtaining authorization for private uranium enriching activities. NRC staff has begun analysis of regulatory issues in anticipation of a license application.

MATERIALS LICENSING

The NRC currently administers approximately 9,000 licenses for the possession and use of nuclear materials in applications other than the generation of electricity or operation of a research reactor. Of these, about 100 are academic, 2,800 are medical and 6,100 are industrial licenses. The

NRC's licensing program is designed to ensure that activities involving such uses of radionuclides do not endanger the public health and safety. About 95 percent of these licenses are now administered by the NRC Regional Offices.

The agency took nearly 6,000 licensing actions during fiscal year 1986. Of these, 800 were on applications for new licenses, 4,100 concerned amendments, 1,000 were license renewals and 100 were sealed source reviews. In addition to these NRC activities, the 28 Agreement States administer approximately 14,000 licenses. These Agreement States have authority over such materials under regulatory agreements with the NRC (see Chapter 9).

Oversight Program

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

As the decentralized programs in each Region have matured, a concerted effort has been made to streamline the resources required to assess and upgrade the Regional programs and to improve Headquarters' ability to provide technical assistance. As a result of experience gained in previous years, increased reliance on preparatory analysis, and more focus on specialized reviews, fewer resources have been expended each successive year.

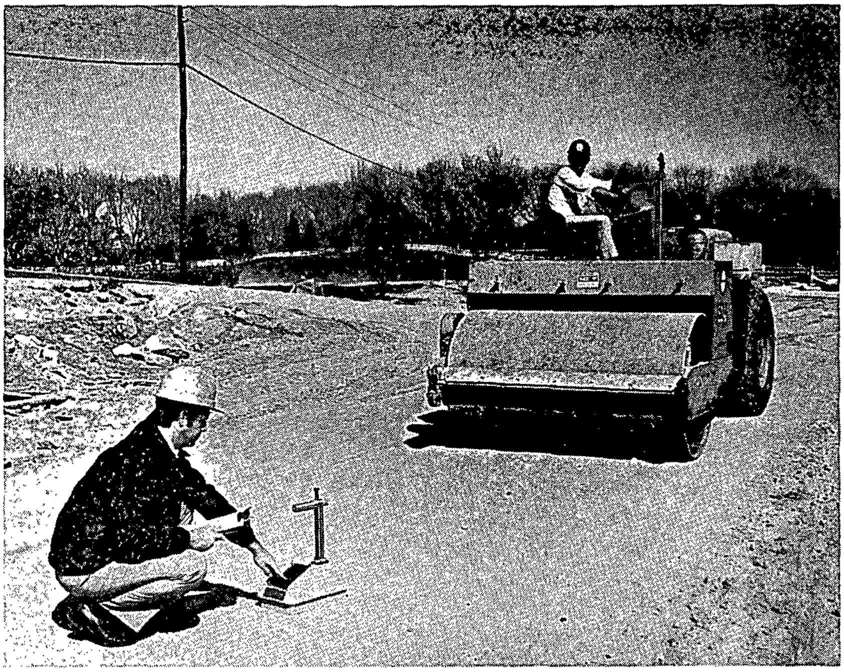
Industrial Licensing

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

Consolidation of Military Licenses. For several years the NRC staff has been reviewing the possibility of consolidating licenses covering military activities involving radioactive materials. The United States Air Force and the United States Navy expressed an interest in obtaining a consolidated license for their activities. The NRC issued a consolidated license to the Air Force's radioisotope program in June 1985. The consolidation was completed in October 1986 and Region IV (Dallas) was given the lead responsibility for the Air Force license. The Navy submitted an application for a consolidated license in August of 1986. This consolidated license would replace over 200 individual Navy licenses, and it is anticipated that substantial administrative resources and paperwork will be saved by consolidation.

General Licenses. There are two types of NRC licenses for byproduct, source, and special nuclear materials—specific and general. Specific licenses are issued only to named organizations following application and NRC review. General licenses take effect without the issuance of license documents to particular persons. However, the manufacturer of products to be distributed to general licensees must

As part of the materials licensing process, the NRC often performs licensee site visits. In this photo, Vandy Miller of the NRC's Region III (Chicago) office is being monitored before leaving a restricted area at Wright Patterson Air Force Base, near Dayton, Ohio.



Shown here is a typical construction use of a surface moisture density gauge, a device which contains byproduct material and requires a specific license for use. The gauge measures the moisture content and density of soils, cement and asphalt-treated bases, and asphalt paving.

apply to the NRC for a specific license. Before issuing this type of specific license for distribution, the NRC conducts a thorough safety analysis of the product. If it meets the criteria for a generally licensed product and the regulations contained in 10 CFR 32, 40, and 70, the applicant is granted a specific license for distribution of the product to general licensees.

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

The findings of the 1984 investigation of generally licensed gauge-users prompted an additional study to determine if similar problems existed with industrial devices other than gauges, used under the general license. Tentative findings are similar to the 1984 study. These include inadequate accountability and improper redistribution of devices. Users of the devices are often unaware of the regulations concerning transfers, disposals, and record-keeping, and labels on the devices often become unreadable because of corrosion and wear. Thus, the devices are susceptible to loss, improper transfers, or disposal.

The NRC is considering several additional changes to the current general licensing policy. A detailed assessment of potential radiation doses to the public resulting from devices used by general licensees is under way. To help alleviate some of the accountability problems and to keep users up to date on the regulations, the NRC is developing a computerized national registry to track all devices and users of the devices in the United States. The registry will allow the NRC to send periodic information notices to the users.

Source/Device Registration. The NRC and the Agreement States maintain a sealed source/device registration program which helps to expedite the licensing review process when new requests for sources or devices are received. During the report period, 150 safety evaluations were completed for radioactive sources and containment devices. A computerized registry system for approved sealed sources and devices is updated twice a year, issuing 400 reports to NRC Regional Offices and Agreement States. During the report period, approximately 70 special reports were produced for NRC and other government users. To augment the registration program, comprehensive draft regulatory guides were developed and distributed for comment. The NRC is now proceeding with a rulemaking which clearly states what radiation safety information about sources and devices is necessary for the safety review and the responsibilities of the registrant.

Medical and Academic Licensing

An estimated 10 million clinical procedures are performed each year in the United States using radioactive materials for the diagnosis or treatment of patients. Many of these procedures involve NRC-licensed materials, and may be conducted in hospitals or in physicians' offices. NRC-licensed materials are also used in universities, colleges, and other academic institutions in certain laboratory courses and in research programs. (See the *1982 NRC Annual Report*, pp. 67-68, for a more detailed description of these activities.)

Medical User's Qualifications. In May 1985, the NRC staff held a public meeting of the Advisory Committee on the Medical Uses of Isotopes (ACMUI, see Appendix 2) to consider NRC's training and experience criteria for physicians who use radiopharmaceuticals for diagnostic imaging procedures. (See the *1985 NRC Annual Report*, p. 83.) The NRC staff is preparing a proposed rule on training and experience criteria that will include a careful examination of the issues and NRC's role.

Part 35 Revision. The NMSS staff led the Task Force that prepared a revision of 10 CFR Part 35, "Medical Use of Byproduct Material," also noted in last year's annual report. The primary purpose of the revision was to consolidate the requirements that were spread throughout a variety of regulatory instruments, including regulations, regulatory guides and license conditions. Under the revision, licensees will be able to make minor changes in their radiation safety procedures that are not potentially important to safety without NRC review and approval. However, these changes will require approval by the licensee's Radiation Safety Officer, and at a hospital, by its Radiation Safety Committee.

Quality Assurance in Radiation Therapy. In response to occasional misadministrations of therapy amounts of radiation (see Chapter 4), the NRC staff is preparing rules which would require radiation therapy licensees to implement quality assurance programs with certain specified features. The staff will work with other government agencies and professional organizations in developing the rule.

Advisory Committee on the Medical Uses of Isotopes

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

TRANSPORTATION OF RADIOACTIVE MATERIALS

The Federal Government regulates safety in the transportation of radioactive materials primarily through the NRC and the Department of Transportation (DOT). These two agencies have delineated their respective regulatory responsibilities in this area through a Memorandum of Understanding. Shipments that occur within the United States also come under regulation by the States in certain circumstances. For international shipments, DOT is the designated U.S. Authority and is responsible for implementing International Atomic Energy Agency (IAEA) standards. The NRC advises DOT on technical matters.

The NRC staff worked on several tasks during fiscal year 1986 addressing transportation safety issues or to give greater clarity and stability to regulatory requirements regarding the transportation of radioactive materials. Discussion of a number of these efforts follows.

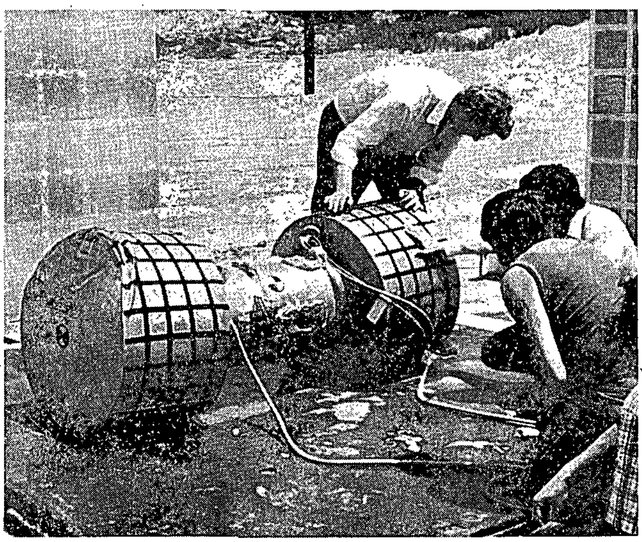
Spent Fuel Shipments

In connection with a reactor fuel reprocessing plant operated at West Valley, N.Y., in the late 1960's and early 1970's, a large quantity of spent nuclear fuel was stored at that site for future reprocessing. Subsequently, the reprocessing plant was permanently shut down and a decision was made to return the stored fuel to the nuclear power plants from which it had originally come. Return of the spent nuclear fuel to the nuclear power plants was started in 1983. The return shipments to three power plants have now been completed, with shipments to the fourth plant under way.

Irradiated Fuel Packaging

Three Mile Island (TMI) Cask. On April 11, 1986, the NRC issued a certificate of compliance (approval) for the Model No. 125-B rail cask. The two casks which have been constructed to the design are being used by DOE to transport reactor core material from Three Mile Island, Pa., to Idaho. The cask design was described on page 87 of the *1985 NRC Annual Report*. The approval was based on a demonstration that the design meets the requirements of 10 CFR Part 71. Applicable NRC regulatory guides and industry codes and standards were followed in the design and fabrication. A 1/4-scale model was subjected to impact and puncture tests.

Big Rock Point and Ginna Casks. Transnuclear, Inc., submitted to the NRC, on behalf of the DOE, a safety analysis report for Model No. TN-REG cask. This cask has a capacity of 40 fuel assemblies and is similar in design, construction and size to the Big Rock Point cask (Model No. TN-BRP) previously submitted by Transnuclear Inc., described on page 87 of the *1985 NRC Annual Report*. DOE plans to use these casks for the transport of fuel assemblies from the Nuclear Service Center at West Valley, N.Y., to Idaho. The review of the Model No. TN-REG (Ginna) cask is in abeyance pending resolutions of issues identified to DOE on February 12, 1986, for the Model No. TN-BRP cask.



Shown is a quarter-scale model of the Model No. 125-B rail cask. The quarter-scale model has been subjected to bottom end, top corner, and side free drops from 30 feet onto a flat, essentially unyielding surface. The sacrificial end-fitting foam-filled impact limiters have deformed and absorbed the energy from the drops.

PATRAM 1986

The Eighth International Symposium for Packaging and Transportation of Radioactive Materials (PATRAM 86) was held on June 16-20, 1986, in Davos, Switzerland. The PATRAM series of international symposiums is held about every three years and continues to be an important forum for exchange of experience, research, and development in packaging and transportation of radioactive materials.

PATRAM 86 was jointly sponsored by IAEA, the Government of Switzerland, and DOE. Nearly 500 participants from 36 countries and several international organizations were present. About 168 papers were presented during 21 sessions by people from 19 countries.

NMSS contributed to the PATRAM 86 by chairing one of the sessions, presenting two papers and supporting two contractors who each presented papers related to ongoing NRC activities in packaging and transportation of radioactive materials.

NRC/DOE Activities Under the Transportation Procedural Agreement

The NRC/DOE Transportation Procedural Agreement published in the *Federal Register* (48 FR 51875) on November 14, 1983, remains in force. This agreement focuses on the important task of exchanging information and identifying transportation packaging issues at the earliest opportunity to assist in DOE's new cask development program. In meetings of technical staff members on January 10, and September 23, 1986, with representatives of DOT also participating, NRC staff members reported on NRC-sponsored research activities and discussed package certification issues. DOE provided information on their plan and schedule for developing the new generation of shipping casks and the complete transportation system, including vehicles and equipment to load and unload the casks. The meetings included extensive discussion of methods of assuring that major public concerns are identified and addressed in the DOE development program. Future meetings will focus on various aspects of the development program for the new transportation casks.

Highlights of Transportation Safety Efforts

The NRC concluded the technical aspects of a major study of the safety provided by its design regulations for packages used to transport large quantities of radioactive material. This study, performed for the NRC by the Lawrence Livermore National Laboratory (LLNL), evaluated how well packages designed to meet NRC performance criteria will withstand the forces generated in severe accidents. The study considered data from severe non-nuclear accidents that have actually occurred, supplemented by data from various package test programs. Comparing the forces resulting from severe accidents with those the casks are designed to withstand gives a measure of the degree of protection afforded by casks that conform to regulatory requirements. Accidents which produced forces in excess of those the casks are designed to withstand were studied in more detail to assess the potential for release of radioactive material from the cask. Also, the probability of such an accident actually occurring was evaluated and the resulting risk to the public health and safety was compared with the risks previously calculated in the "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes" (NUREG-0170). From these evaluations, a determination can be made about the need for any changes to present performance criteria. The technical work was completed by LLNL in the spring of 1986.

Safeguards

CHAPTER 6

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

During fiscal year 1986, NRC safeguards requirements were applied to 101 power reactors, 68 non-power reactors, and 28 fuel cycle facilities. They were also applied to 178 shipments of spent fuel, 26 shipments of SNM involving more than one but less than five kilograms of highly enriched uranium, and four shipments of SNM involving five or more kilograms of highly enriched uranium.

STATUS OF SAFEGUARDS IN 1986

Reactor Safeguards

Power Reactors. The NRC continued the accelerated review of physical security plans received from applicants for licenses to operate power reactors. A total of five expanded comprehensive safeguards statements for the Safety Evaluation Reports was provided during fiscal year 1986.

In May 1986, the staff completed a study re-evaluating the bases and guidelines used to determine what equipment and areas need to be protected as "vital" in nuclear power reactors. The study was aimed at ensuring consistency and coordination from both the safeguards and safety perspectives. The recommendations of the study regarding vital areas and related assumptions are currently undergoing detailed regulatory analysis before being implemented.

Under the Regulatory Effectiveness Review (RER) program, evaluations continued of the effectiveness of safeguards and the validity of identifications of vital equipment at power reactors. These reviews are conducted independently of the NRC's regular inspection and enforcement activities and are intended to assure that safeguards programs, as implemented by licensees, are effective against the design basis threats defined in 10 CFR 73.1. During fiscal year 1986, reviews were conducted at 18 power reactors. RER's have led to the identification of both strengths and weaknesses in licensees' programs. Commonly noted strengths include effective routine access control features and good rapport and coordination with local law enforcement agencies. The most common problem areas identified in RER reports concern vital area barriers and intrusion detection and alarm assessment systems. Problems and issues raised in RER reports are resolved through voluntary actions of licensees or through licensing, inspection, enforcement, or rulemaking, as appropriate.

Inspection and Enforcement at Reactors. At operating power reactors, efforts continued to increase the Resident Inspectors' contributions to the safeguards program. (See Table 1.) Initiatives were also undertaken by the Region Inspectors to conduct exercises and drills, and to evaluate systematically the quality of guard force training and the ability of the force to respond to contingencies.

Fuel Cycle Facilities

The number of licensed fuel facilities subject to NRC safeguards requirements in fiscal year 1986 remained the same as in 1985. Specifically, there were 28 such facilities, 20 of which maintained both physical security and material control and accounting systems. Four of these 20 facilities had actual holdings of formula quantities of strategic special nuclear material (SSNM), requiring the implementation of extensive physical security and material accountability measures. The remaining eight facilities were required to implement a moderate level of physical security, but were not required to implement detailed material control and accounting systems. The activities at these 28 fuel facilities include full-scale production, pilot plant operations, decommissioning efforts and the storage of sealed items. The NRC received and completed action on approximately 120 licensing matters associated with these facilities in 1986.

Inspection and Enforcement at Fuel Cycle Facilities. During fiscal year 1986, the material control and accounting (MC&A) safeguards inspection procedures were revised to reflect additional guidance for inspecting records and measurement systems. The MC&A inspection procedures for low enriched uranium commercial fuel facilities were also updated to incorporate the inspection of those aspects of licensees' MC&A programs that are unique to fulfilling commitments made to the International Atomic Energy Agency (IAEA) under the US/IAEA Safeguards Agreement. (See Table 1.)

Transportation

Spent Fuel Shipments. During fiscal year 1986, the NRC approved 31 transportation routes with respect to acceptable protection against sabotage. One hundred seventy-eight spent fuel shipments went over these routes. To keep the public informed about spent fuel shipment routes, NRC publishes a "Public Information Circular for Shipments of Irradiated Reactor Fuel" (NUREG-0725) containing approved routes.

SSNM Shipments. Two export and two domestic shipments, each involving five or more kilograms of highly enriched uranium, were made during fiscal year 1986. However, the only commercial road carrier of formula quantities of SSNM notified the export licensee that it would no longer make these shipments. The licensee has arrangements with the Department of Energy under which the latter is providing transportation on an interim basis. There were also seven export, six in-transit, and 13 domestic shipments—each involving less than five but more than one kilogram of high enriched uranium—during the report period.

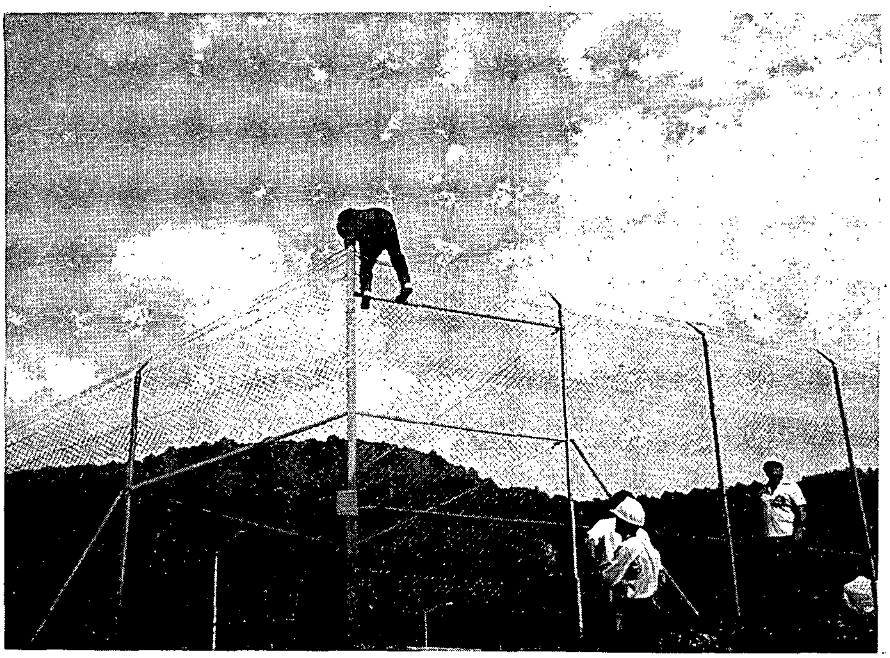
Shipment Route Surveys. In fiscal year 1986, NRC safeguards teams, each composed of two representatives from the Region concerned, worked with local law enforcement agencies to conduct field surveys of routes proposed for shipments of spent fuel or SSNM. Thirty-one routes were analyzed through 36 States, involving over 3,000 miles of travel. The NRC brochure entitled "Information Package on Spent Nuclear Fuel Shipments for Law Enforcement Agencies" was distributed to local officials and agencies during these surveys.

Transport Inspection and Enforcement. During fiscal year 1986, new inspection procedures were developed for transportation of nuclear material by road, rail, sea or air. In some cases, interim procedures were replaced to reflect new requirements in the regulations. The NRC continued to inspect selected domestic shipments and the domestic segments of import and export shipments of SSNM. These shipments were inspected at points of origin, in transit, during intermodal transfer and temporary storage, and at destinations. (See Table 1.)

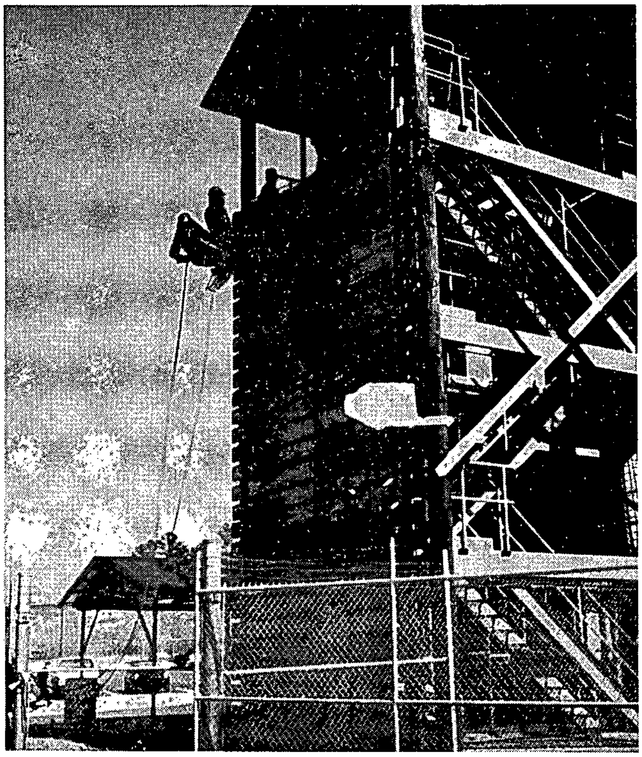
Contingency Planning and Threat Assessment

Safeguards contingency plans deal with threats, thefts, and sabotage related to licensed material and facilities. The NRC staff reviewed and revised its Headquarters' contingency plan and completed training in both response plan procedures and incident response center procedures in dealing with an incident involving a nuclear power facility. In August 1986, an incident response exercise was conducted that involved Headquarters, Region, and licensee personnel. Representatives of the Federal Bureau of Investigation observed and participated in that exercise.

The staff continued to review the domestic and foreign threat environments and their relationship to NRC's



Members of an NRC Safeguards Regulatory Effectiveness Review Team are shown testing the perimeter intrusion detection system at a nuclear power facility. Such reviews, carried out independently of the NRC's regular inspection program, were conducted at 18 power reactors in fiscal year 1986.



As part of its safeguards contingency planning against threats, thefts and sabotage, the NRC conducts periodic incident response exercises. These photos, taken during such an exercise in August 1986, show demonstrations of removal of an injured person by rappelling down the side of a building, and employment of live-firing techniques.

domestic safeguards regulations. Continuing liaison with other Federal agencies was maintained throughout 1986. Special attention was given to foreign terrorist group activities and to possible trends in their behavior. No significant change in the domestic threat environment was identified as a result of the staff's review and interaction with other agencies. As part of the Commission's on-going consideration of the NRC design basis threats, the views of other agencies were solicited regarding the domestic threat environment as it pertains to nuclear facilities.

The "Communicated Threat Credibility Project" continued to provide support in the form of guidance to the NRC, the Department of Energy, the Federal Bureau of Investigation and other concerned agencies for responding to written or recorded threats. Within the NRC, the Information Assessment Team, composed of NRC Headquarters and Regional personnel, continued to provide an on-call capability to furnish an initial assessment of all reported threats to NRC licensed material or facilities.

The staff has continued analysis of safeguards events data to identify trends, patterns, and anomalies. NRC's published compilation of safeguards events data, the "Safeguards Summary Event List" (NUREG-0525), was updated in January 1986 (Rev. 11). This document contains information about safeguards-related events involving licensed nuclear material and facilities.

NRC/IAEA Interaction. During 1986, the International Atomic Energy Agency (IAEA) carried out routine inspections of the Westinghouse low-enriched uranium (LEU) fuel fabrication plant in Columbia, S.C., the Salem Unit 1 reactor in New Jersey and the Turkey Point Unit 4 reactor in Florida. The NRC continued to submit accounting data on a monthly basis for these facilities, as well as for the LEU fuel fabrication plants of Babcock & Wilcox at Lynchburg, Va., of EXXON at Richland, Wash., of Combustion Engineering Corp. in Connecticut, and the General Electric plant at Wilmington, N.C.

In May 1986, representatives of the NRC and the IAEA met in Washington, D.C. to discuss IAEA safeguards implementation issues in the U.S. In September 1986, the NRC and the IAEA met in Vienna to complete agreements on the Facility Attachments for U.S. facilities subject to routine IAEA inspections. The NRC prepared an update of the eligible facility list for application of IAEA safeguards at licensed facilities to be submitted to the Executive Branch for review and transmittal to the IAEA.

Table 1. Summary of Safeguards Inspection Visits—FY 1986

Type	*Number of Licensees Inspected*[1]	*Inspection Visits*[2]	*Violations*	*Inspection Manhours*
FUEL FACILITIES				
Formula Quantity	4	55	12	2,386
Less than Formula Quantity	9	39	25	1,926
TOTAL	13	94	37	4,312
POWER REACTORS				
Operating	96	315	249	10,398
Pre-operating	15	45	14	1,978
TOTAL	111	360	264	12,376
NON-POWER REACTORS				
TOTAL	23	31	14	340
SHIPMENTS				
Formula Quantity	2	3	0	27
Less than Formula Quantity	6	9	0	143
Irradiated Fuel	1	4	1	22
TOTAL	9	16	1	192
GRAND TOTAL:	156	501	316	17,220

[1]A dual unit site is two licensees.
[2]Data based on inspection reports issued during period 10/1/85 to 9/30/86.

SAFEGUARDS REGULATORY ACTIVITIES AND ISSUES

Reactor Safeguards

Two of the three related rules comprising the Insider Safeguards Rules were published in final form on August 4, 1986. The Insider Safeguards Rules package was developed to assure continuing adequacy of protection against the "insider" threat at nuclear power reactors. The two published rules clarify and refine requirements for access to, and protection of, vital equipment within the plant, and requirements for contraband searches of employees and visitors prior to entry to the protected area of a nuclear power plant. The NRC plans to withdraw the third rule of the package, the Access Authorization Program (which was published as a proposed rule on August 1, 1984), in favor of a nuclear industry initiative to commit voluntarily to an industry-developed guideline on access authorization. This industry guideline, which is similar to the provisions of the proposed rule, calls for the screening of individuals granted unescorted access to a nuclear power plant through background investigations, psychological assessment, and continual behavioral observation. The NRC is working with representatives of the Nuclear Utility Management and Human Resource Committee (NUMARC) to develop a policy statement endorsing the NUMARC guideline on access authorization for review by the Commission during the first quarter of fiscal year 1987.

The NRC staff is also developing regulations to implement Public Law 99-399, the Omnibus Diplomatic Security and Antiterrorism Act of 1986. The Act requires that each individual granted access to Safeguards Information or unescorted access to a nuclear power plant be fingerprinted and a criminal history records check be made by the Attorney General. NRC's rule will provide for control of the data to prevent misuse, to limit re-dissemination, and to restrict use of certain arrest information.

Non-power Reactors. The NRC continued development of a proposed regulatory amendment to require improved physical security at non-power reactors using high enriched uranium. In a related matter, non-power reactor licensees removed excess unirradiated high enriched uranium fuel from their facilities, retaining the amount necessary to continue normal operation. In addition, during fiscal year 1986, the NRC issued a final rule requiring non-power reactor licensees using high enriched uranium to convert to low enriched uranium, contingent upon government funding availability. Based on the resulting reduction in high enriched uranium at non-power reactors from these two actions, the nature and type of the proposed regulatory amendment for improved physical security will be examined during fiscal year 1987.

Fuel Facilities Material Control and Accounting

A final rule that would revise material control and accounting (MC&A) requirements for certain fuel cycle facilities authorized to possess and use formula quantities of strategic special nuclear material was forwarded with a recommendation for approval and publication. The rule would refocus the emphasis of MC&A away from periodic physical inventories and toward the use of process monitoring information for safeguards. Timely detection of possible material losses and improved loss localization capabilities are the projected benefits. The final rule takes into account public comments on the proposed rule and information obtained during site-specific value-impact analyses. Reactors, waste disposal operations, and irradiated fuel reprocessing plants (if any should be licensed) would continue to be subject to the current MC&A requirements.

Transportation

Convention on Physical Protection. The United States is a signatory of the Convention which provides for the establishment and maintenance of adequate physical security for international shipments of significant quantities of source or special nuclear material. A final rule to bring NRC regulations into accord with the Convention was issued on March 28, 1985; however, the provisions of the rule will not become effective until 21 countries ratify the Convention. As of September 1986, 18 countries had ratified.

Protection of Spent Fuel Shipments. The development of a final rule for the protection of spent fuel shipments against sabotage continued. The results of an assessment of the relationship of the safety aspects of spent fuel transportation to the rule were incorporated. Additional technical information concerning the response of shipment cask materials to sabotage is being developed and evaluated.

SAFEGUARDS RESEARCH AND TECHNICAL ASSISTANCE

Approximately $4.3 million was spent in fiscal year 1986 on safeguards technical assistance and research contractual projects. Of this amount, approximately $3.8 million came from technical assistance funds, and the remaining $0.5 million from research funds. Some of the projects are described below.

- *Nuclear Materials Management and Safeguards System.* This continuing project, which is jointly funded with DOE, supports the national data base and information system for managing and safeguarding nuclear materials. It processes information needed by the government to regulate the receipt, manufacture, production, transfer, possession, use, import, and export of nuclear materials. It also fulfills international commitments derived from bilateral agreements, IAEA requirements for export/import reporting, and IAEA requirements under the US/IAEA Safeguards Agreement.
- *High Level Waste Storage and Spent Fuel Cask Certification.* The programmatic need for this work derives from the Nuclear Waste Policy Act of 1982, which directed the NRC to provide, through rulemaking, generic approvals for the use of dry spent fuel storage technologies at nuclear power reactor sites without, to the extent practicable, the need for site-specific approvals. This project will provide technical assistance to the NRC safeguards staff in assessing the public health and safety consequences of sabotage of dry spent fuel storage casks. The assessment of consequences will be one important determinant in framing of safeguards conditions to apply to spent fuel storage in dry casks.
- *Reactor Vital Equipment Determination Techniques.* This project identified and characterized existing reactor safety/engineering information in such a way as to provide a technical basis for modifying vital area determination techniques used by the staff.
- *Technical Assistance to Strengthen IAEA Safeguards.* This project provides technical assistance to the NRC staff in support of U.S. Government efforts to strengthen IAEA safeguards. Under the Nuclear Non-Proliferation Act of 1978, the United States is committed to continue strong support to strengthen IAEA safeguards. During fiscal year 1986, two studies were completed to identify design features which could be utilized to enhance the application of safeguards for away-from-reactor dry storage of spent fuel and for spent fuel rod consolidation. A new project was initiated for the preparation of a regulatory guide that could be utilized by NRC licensees to implement the US/IAEA Safeguards Agreement under 10 CFR Part 75.

Waste Management

CHAPTER 7

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

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

Highlights of 1986

In fiscal year 1986, NRC staff continued its work to assure that the milestones of the NWPA can be met. It is the NRC's policy that, absent any unresolved safety issues, the NRC will support DOE schedules for meeting NWPA requirements, as set forth in the DOE final Mission Plan and final Project Decision Schedule. During the year, the NRC defined its position on the implementation of DOE's Final Mission Plan and issued comments to DOE on both the draft Project Decision Schedule (PDS) and the proposal for the Monitored Retrievable Storage (MRS) facility. Other significant accomplishments include the development of NRC comments on each of DOE's five final environmental assessments, the issuance of 10 staff technical positions providing guidance to DOE on a variety of issues, and the publication of two rulemaking proceedings to amend 10 CFR Part 60, "Disposal of High-Level Radioactive Waste in Geologic Repositories," to conform with NWPA requirements. The NRC has also conducted meetings and workshops with DOE to resolve pre-licensing issues related to DOE's development of Site Characterization Plans (SCPs), and in the initial development of the Licensing Support System to streamline the licensing process. NRC staff provided extensive comments to DOE on the latter's draft Environmental Impact Statement for disposals of Defense Waste at the Hanford, Wash., facility.

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

The passage of the LLRWPAA on January 15, 1986, established an important NRC role in the national low-level waste (LLW) management effort. The agency met its first statutory mandate by publishing a Commission policy statement providing guidance for the filing of petitions for rulemaking to exempt specific waste streams as lying outside NRC regulatory concern. The LLRWPAA also directs the NRC to establish licensing review procedures and provide technical guidance for alternative disposal methods (which had been a part of NRC program efforts for some time). NRC published a draft branch technical position on the alternative methods which clarified the applicability of 10 CFR Part 61 and provided other general guidance. The NRC staff also published a draft Standard Format and Content Guide for a disposal facility license application and is continuing work on standard plans for review of an application.

With regard to uranium recovery activities, the staff continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive sites, as required by Title I of the Uranium Mill Tailings Radiation Control Act (UMTRCA) of 1978. The staff also continued work in conforming its 10 CFR Part 40 regulations to the final EPA standards for mill tailings. A final rule conforming Part 40 to requirements for radiological protection and long-term stabilization was published on October 16, 1985, while a proposed rule addressing groundwater protection was published July 8, 1986. The comment period closed on November 7, 1986.

The NRC also published an advance notice of proposed rulemaking on financial assurance requirements for funding the cleanup of accidental contamination at certain materials and waste management licensees.

HIGH-LEVEL WASTE PROGRAM

Regulatory Development

The staff continued its rulemaking proceedings to amend the procedural and technical provisions of 10 CFR Part 60 to bring them into conformity with NWPA requirements. A final rule was published on July 30, 1986, amending certain procedural aspects regarding site characterization and the participation of States and Indian Tribes. On June 19, 1986, a proposed rule was published amending Part 60 to conform with the EPA Standards for High-Level Waste Disposal. Public comments on the proposed rule have been received and a final rule is expected to be published in fiscal year 1987.

Two other rulemaking actions were initiated during the report year. Development of an Advanced Notice of Proposed Rulemaking (ANPR) to redefine "high-level waste" in light of the NWPA definition was initiated with publication expected by spring of 1987. Also, the staff initiated action to amend Parts 60 and 51 to conform National Environmental Policy Act-related requirements to NWPA requirements concerning NRC adoption of DOE's Environmental Impact Statement for the geologic repository. A proposed rule is expected to be published during fiscal year 1987.

Regulatory Guidance

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

The NRC staff continued to develop Generic Technical Positions (GTPs) and other guidance documents during the report period. The following GTPs were published during fiscal year 1986:

- Final GTP on In-Situ Testing
- Final GTP on Waste Package Reliability
- Final GTP on Design Information Needs in Site Characterization Plans
- Final GTP on Borehole and Shaft Seals
- Draft GTP on Sorption
- Draft GTP on Interpretation and Identification of the Disturbed Zone
- Draft GTP on Groundwater Travel Time
- Draft GTP on Items and Activities in the High-Level Waste Geologic Repository Program Subject to 10 CFR Part 60 Quality Assurance Requirements
- Draft GTP on Qualification of Existing Data for High-Level Nuclear Waste Repositories
- Draft GTP on Peer Review

Site Investigations

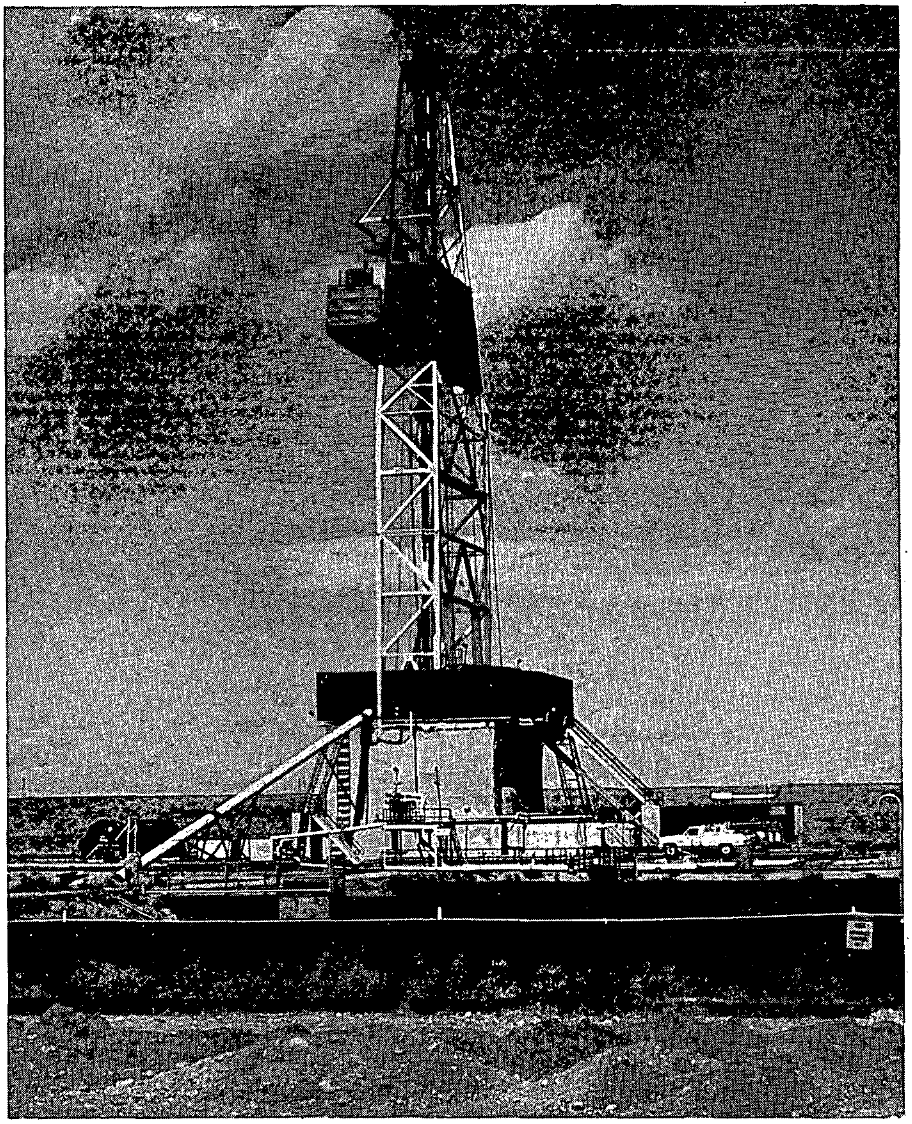
Section 112(b) of the NWPA requires DOE to recommend three sites to the President for characterization as the first repository, and to publish Environmental Assessments (EA's) for each of at least five nominated sites from which the recommended ones are to be chosen. In May 1986, DOE published final EA's for each of the five sites nominated as suitable for site characterization, and also recommended three sites for characterization. The President approved the DOE-recommended sites for characterization. The three sites are the Yucca Mountain site in Nevada, the Deaf Smith County site in Texas, and the Hanford site in Washington. The NRC staff has reviewed and developed coments on each of the five final EA's as part of its continuing pre-licensing arrangement with DOE to promote early identification and resolution of potential licensing issues.

Section 113(b) of the NWPA requires that, for each of the sites to be characterized, DOE must issue for NRC and State/Tribal comment a Site Characterization Plan (SCP), with a description of a proposed waste form and packaging and a conceptual repository design. While the SCP's were not scheduled for issuance until spring of 1987 or later, the NRC worked closely with DOE throughout the year to help ensure that the final products would be complete and of high quality. NRC activity has included reviewing available data and information on the sites from investigations to date, reviewing design documents and preliminary plans for site characterization, and working toward resolution of the significant concerns in open, documented technical meetings before the SCP's are issued.

Quality Assurance Activities

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

The staff prepared and issued for public comment three draft Generic Technical Positions which give guidance on specific QA issues for the repository program. The first of these identified repository-related items and activities subject to 10 CFR Part 60 quality assurance requirements and



This photo shows an exploratory-shaft drill rig situated at the Reference Repository Location at the Hanford candidate repository site in Washington State. Drilling of the exploratory shaft is scheduled to begin sometime after the U.S. Department of Energy (DOE) issues the Site Characterization Plans. DOE plans to drill a nine-foot diameter borehole to construct a six-foot diameter shaft to a depth of more than 3,300 feet through the Columbia River basalt.

gave detailed guidance on how to identify the scope of the QA program (the "Q-list"). The second GTP addressed the qualification of existing repository program data and discussed methods by which data collected prior to or outside of the Appendix B QA program can be determined to be of adequate quality for licensing. The third GTP gives guidance on peer review for work involving expert judgment. In addition to these GTP's, the staff also prepared letters giving QA guidance on other specific issues.

During 1986, the staff initiated a review of the first DOE-approved QA plans and procedures. DOE will provide a summary description of the QA program in the SCP, and the staff will assess the adequacy of the program in its review of the SCP. At the close of the report period, the staff was in the process of reviewing detailed QA plans and procedures for the program, as they became available. It is expected that issues associated with the plans will have been identified and resolved by the time the SCP's are issued. In October 1985, the staff provided comments on the QA plan of the DOE Headquarters (Office of Geologic Repositories). Several other reviews were initiated late in fiscal year 1986 and were scheduled to be sent to DOE in early fiscal year 1987.

The staff continued to provide limited oversight of the implementation of the DOE's QA program during 1986. The DOE program is still evolving and is not yet in full compliance with the Commission's requirements. As DOE upgrades its program, the staff is observing audits conducted by DOE and its contractors for assessing the implementation of the program. As a result of these audit observations, the staff has identified a number of improvements that could be made in the DOE audit program. As the program becomes qualified, the NRC staff will be conducting its own audits to assure satisfactory implementation.

DOE Mission Plan and Project Decision Schedule

Section 301(b)(10) of the NWPA requires that DOE submit to Congress a Mission Plan, delineating how the activities required by the NWPA will be implemented. Section 114(e) requires DOE to prepare and update, in cooperation with affected Federal agencies, a Project Decision Schedule (PDS) for those activities.

The Final Mission Plan was submitted to Congress on July 9, 1985. After staff review of the final document, the NRC testified before the Senate Subcommittee on Nuclear Regulation on October 30, 1985, and before the House Subcommittee on Energy Research and Production on November 6, 1985, concerning its view of the Mission Plan. NRC testimony focused on the time required for NRC review of DOE's license application, the timing of DOE's preliminary determination of repository site suitability, quality assurance, NRC-DOE pre-licensing consultations, and monitored retrievable storage.

On October 24, 1985, the staff provided comments to DOE on the draft Project Decision Schedule, which DOE had issued on July 18, 1985. The final PDS was issued by DOE on April 10, 1986, and NRC staff has been working with DOE staff to resolve specific concerns.

Mr. Russell Jim, Nuclear Waste Project Manager for the Yakima Indian Nation (right), introduced NRC Commissioner James K. Asselstine as the luncheon speaker at the Quarterly Meeting of potential host States and affected Indian Tribes (in the selection of a high-level nuclear waste repository site) in Portland, Ore., on August 13, 1986. Commissioner Asselstine's remarks concerned the implementation of the DOE's waste repository program.

State and Tribal Interactions

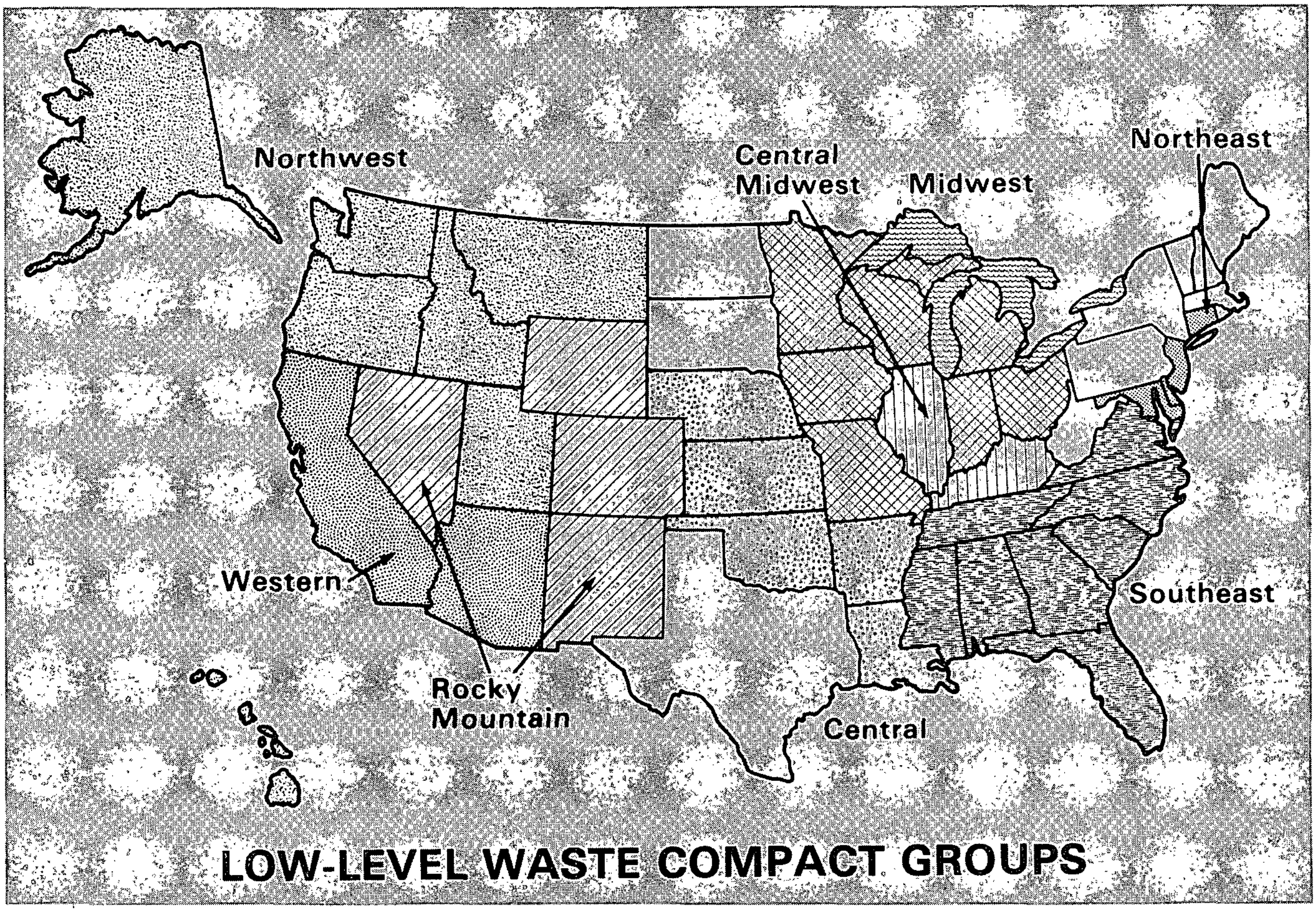
The NWPA contains provisions for State and Tribal participation in the repository program. It contains specific provisions for DOE's interaction with the States and Tribes and requires both NRC and DOE to provide "timely and complete" information to States and Tribes on all repository-related "determinations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

State and Tribal representatives met with the Commissioners on January 24, 1986, to discuss their views on the procedural amendment to 10 CFR Part 60 concerning site characterization and the participation of States and Tribes.

During the year, the staff gave presentations to the States and Tribes on the role of NRC in the NWPA process. Examples include briefing for the National Congress of American Indians, the Council of Energy Resource Tribes, and the National Conference of State Legislatures High-Level Waste Working Group. In addition, the staff met with States and Tribes to discuss NRC's efforts to streamline the licensing process, and to update the status of the Licensing Support System (LSS). Other presentations have covered topics such as the development of a negotiated rule on 10 CFR 2 for submittal of records, exploratory shaft design and construction, and seismo/tectonic investigations.

Other Activities

In October 1986, the Commission approved the establishment of an NRC-sponsored Federally Funded Research and Development Center (FFRDC) to provide long-term technical assistance and research related to NRC's regulatory program under the NWPA. An FFRDC is being proposed as a solution to the problems of contractor conflict-of-interest (with DOE and other parties to the high-level waste licensing proceeding) and to provide long-term continuity in NWPA-related technical assistance and research. The FFRDC will provide support to NRC in the following areas: (1) waste systems engineering and integration and overall program activities; (2) long-term performance of a geologic setting; (3) long-term performance of an engineered barrier system; (4) transportation, special project, and analytical studies; and (5) monitored retrievable storage (MRS) and repository design, construction and operation. A competitive solicitation for proposals to operate the Center and provide the necessary resources has been published, and it is expected that the Center will begin operations by fiscal year 1988.



During 1986, the staff reviewed DOE's draft Environmental Impact Statement entitled "Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes." NRC comments on this document were transmitted to DOE in September 1986.

The NRC initiated a pilot project during the report period to demonstrate two information management systems which are intended to facilitate license review activities and prelicensing guidance to DOE. A Licensing Support Pilot System will demonstrate the feasibility of full text electronic storage and retrieval of NRC high-level waste documents which are currently available in paper files in the docket control center and public document reading rooms. An efficient system is necessary to meet both the needs of the technical staff and the legal need for document discovery and responses to Freedom of Information Act requests. Information and experience gained from this Pilot Project is being shared with DOE as they prepare to develop the Licensing Support System.

The staff is also demonstrating a High-Level Waste Issue Tracking System, intended to identify and track the progress of licensing concerns. The system will identify by subject (e.g., waste package, hydrology) the major NRC licensing concerns and will be able to document progress toward the resolution of technical concerns. The system will assist the staff in focusing its efforts and resources on critical licensing concerns.

LOW-LEVEL WASTE PROGRAM

Regulation and Guidance

Throughout fiscal year 1986, NRC staff continued its efforts to develop guidance that will assist States in developing the disposal capacity required by 1993, by provision of the Low-Level Radioactive Waste Policy Amendments Act (LLRWPAA) of 1985. The Commission published a policy statement and accompanying staff implementation plan for expedited action on petitions for rulemaking to declare certain waste below NRC regulatory concern (Appendix B, 10 CFR Part 2). Other guidance published during 1986 includes a draft Standard Format and Content Guide for a disposal application and a draft technical position on alternative methods of land disposal. The staff is also developing draft regulatory guides on low-level waste classification and manifest reporting, acceptable waste forms, site selection, and technical positions on site closure and environmental monitoring. These guides and positions will be available soon either for public comment or as final products. Work is also underway in the areas of decommissioning wastes and the performance of concrete and steel as an engineered barrier.

The NRC is working to resolve regulatory jurisdictional issues with the Environmental Protection Agency, which regulates hazardous waste, and to develop options to deal with low-level radioactive wastes that also contain hazardous chemical constituents, known as "mixed wastes."

Status of Current Facilities

During fiscal year 1986, NRC staff worked on renewal of the licenses at all three of the active licensed disposal facilities in the United States: Barnwell, S.C., Hanford, Wash., and Beatty, Nev. Chem-Nuclear Systems, Inc., submitted a license renewal application for its special nuclear material (SNM) license at Barnwell, and the staff anticipates renewal will be completed by mid-1987. The staff continued reviewing U.S. Ecology's renewal application for its SNM license at Hanford. The renewed license will reflect the adoption of substantive operational requirements of 10 CFR Part 61. The Beatty, Nev., waste disposal site is expected to cease operation and close permanently in a few years. The NRC staff has continued to assist the State of Nevada to renew the Beatty SNM license for the years prior to closure and to develop an adequate closure plan.

During the past year, efforts have continued to close the Sheffield, Ill., low-level waste burial facility. With guidance from the NRC, the site owner (State of Illinois) and the site operator (U.S. Ecology), have been developing a mutually agreeable closure plan.

Assistance to States and State Compacts

The NRC is continuing an active outreach program as a means of providing guidance to States and State Compacts regarding development of new LLW disposal sites. Typical of such efforts were providing comments on Pennsylvania's screening and design criteria and exploring the implications of 10 CFR Part 61 with Maine and Illinois. The NRC also continued to provide technical assistance to the Agreement States of Nevada, New York, California, Washington, and Texas.

Work with Other Federal Agencies

The NRC and EPA staffs are working to resolve uncertainties imposed upon NRC-regulated activities by other Federal environmental laws and regulations, such as the Resource Conservation and Recovery Act, and planned EPA standards for low-level waste disposal.

The NRC staff consults with the DOE staff in three areas: coordinating management of the national low-level commercial waste program on such efforts as identifying alternative methods and developing data bases; reviewing the closure and disposition of waste at West Valley, N.Y., under the West Valley Demonstration Project Act (see Chapter 5); and implementing Section 151 of NWPA in the areas of financial assurance and criteria for the transfer of title and custody of closed commercial LLW sites to DOE.

URANIUM RECOVERY AND MILL TAILINGS

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

Regulatory Development

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

In the first step, completed in October 1985, NRC's regulations pertaining to radiological protection and long-term stabilization of mill tailings were modified to conform to the EPA standards. The second step remaining for the NRC staff is incorporation of the EPA groundwater standards. A proposed rule addressing groundwater protection was published July 8, 1986. The initial comment period expired, but was subsequently extended to November 7, 1986. The final rule is expected to be published in fiscal year 1987.

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

Licensing and Inspection Activities

During fiscal year 1986, the Denver Field Office (URFO) performed 33 inspections of uranium recovery facilities. The Office is reviewing a new commercial in-situ license application for an Everest Minerals site in Wyoming. In other regulatory actions, the URFO staff completed 3 license renewals, 28 major license amendments, and 79 minor amendments to licenses.

Of the 42 licensed uranium recovery facilities, 21 are uranium mills, 3 are heap leach/ore buying stations, 14 are research and development solution mining operations, and 4 are commercial in-situ facilities.

Only five of the 42 licensed facilities were in operation at the end of fiscal year 1986: three uranium mills and two research-and-development solution mining facilities. Given the economic state of the uranium industry, very little licensing of new facilities is expected. Over the next few years, much of the casework confronting the uranium recovery program will be in the areas of remedial activity and decommissioning, including remediation for groundwater contamination.

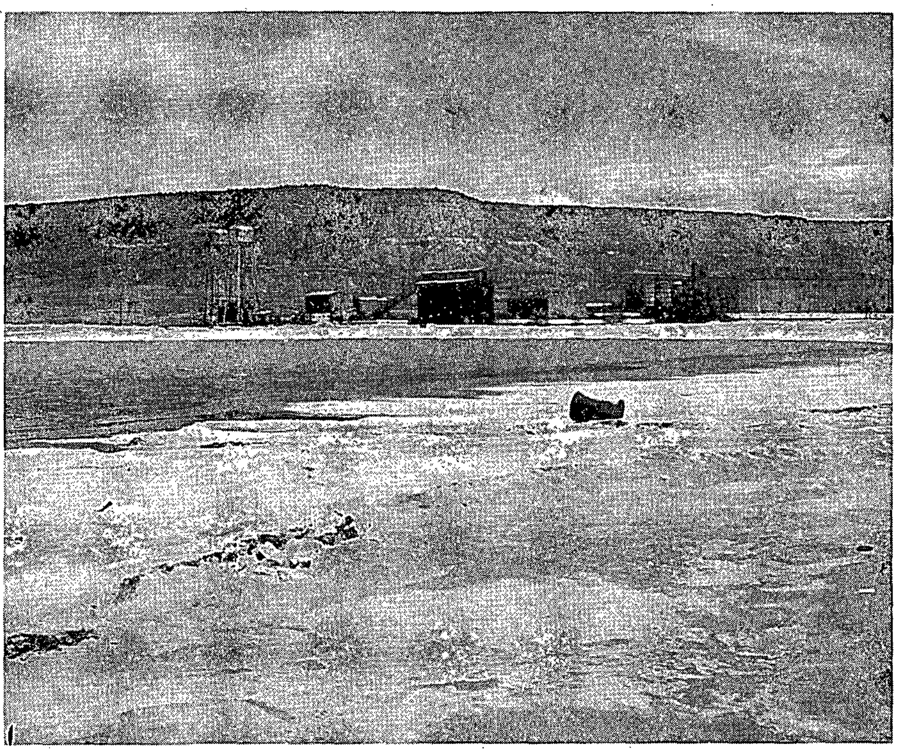
Technical Assistance to Agreement States on Uranium Recovery

Section 274 of the Atomic Energy Act of 1954, as amended, authorizes the Commission to enter into agreements with the Governor of any State providing for discontinuance of regulatory authority of the Commission with respect to source materials and byproduct materials associated with uranium recovery facilities. The NRC currently has such agreements with three states: Colorado, Texas, and Washington. (New Mexico returned its uranium recovery licensing program to NRC, effective June 1, 1986.)

The NRC conducts periodic reviews of the Agreement States' licensing and inspection programs to determine their compatibility with the NRC's programs, and provides training and technical assistance to the Agreement States to help them fulfill their regulatory responsibilities. During fiscal year 1986, the NRC reviewed the uranium recovery licensing programs of Colorado, Washington, and Texas, examining the States' programs for mills, commercial solution mining facilities, and research-and-development solution mining facilities. The NRC provided technical assistance to the Agreement States, on both generic issues and site-specific licensing issues, and conducted two generic and seven site-specific reviews.

Remedial Action at Inactive Sites

The NRC has continued its involvement in the Uranium Mill Tailings Remedial Action Program (UMTRAP) at inactive mill tailings sites, as required by Title I of UMTRCA. The NRC is a cooperating agency and is required by UMTRCA to concur in the remedial actions planned by DOE for inactive mill tailings properties.



This view of an abandoned uranium mill tailings site at Ambrosia Lake, N.M., with uranium mill tailings visible in the foreground. The DOE is conducting remedial activities at Ambrosia Lake under a project set up pursuant to the Uranium Mill Tailings Radiation Control Act of 1978. The NRC must concur in all remedial actions planned for inactive uranium mill tailings sites, as a cooperating agency.

NRC reviewed and commented on modifications to cooperative agreements with Utah, New Mexico, Idaho, and Oregon. Comparative Analysis of Disposal Site Alternatives Report (CASDAR) and NEPA documents reviewed by NRC included those for Ambrosia Lake, N.M.; Green River and Mexican Hat, Utah; Slick Rock, Maybell, Naturita, and Grand Junction, Colo.; and Tuba City and Monument Valley, Ariz. Remedial action plans and modifications reviewed by NRC in fiscal year 1986 included Durango and Grand Junction, Colo.; Canonsburg, Pa.; Lakeview, Ore.; Shiprock and Ambrosia Lake, N.M.; Salt Lake City and Mexican Hat, Utah; and Tuba City and Monument Valley, Ariz. Conditional concurrences in the selection of remedial action were provided for the Lakeview and Shiprock sites. The Canonsburg, Pa., draft certification of remedial action completion was reviewed and commented on by NRC. Construction design reviews and site inspections included Canonsburg, Lakeview, Shiprock, Salt Lake City, Tuba City, Grand Junction, Rifle, Colo., and the Burrell, Pa., Vicinity Property (VP).

Generic efforts included establishment of final Standard Review Plans, revision of the Memorandum of Understanding with DOE, institution of an NRC UMTRAP construction inspection guidance manual, and investigation of the feasibility of co-disposal of UMTRAP and active site uranium milling waste. The NRC also reviewed DOE generic documents, such as the Project Certification Plan, the Project Surveillance and Maintenance Plan, the Project Technical Approach Document, and the Interim Groundwater Protection Plan. NRC has continued to review VP Radiological and Engineering Assessment (REA) and completion reports associated with processing site cleanups. The VP Certification Plan and a VP Management and Implementation Manual modification were reviewed by NRC in fiscal year 1986.

Inspection, Enforcement, Quality Assurance and Emergency Preparedness

CHAPTER 8

In the inspection and enforcement sphere, fiscal year 1986 was marked by a shift in focus to plants which, based on problems and performance, called for special attention. For example, personnel of the NRC Office of Inspection and Enforcement (IE), both at Headquarters and in the Regional Offices, were integrally involved in the agency's effort to resolve the many problems with the Tennessee Valley Authority (TVA) reactors. Other plants were also identified for additional inspection effort, based on in-depth reviews of plant performance by senior NRC staff. Conversely, each Regional Administrator identified those facilities in the Region where very good performance on the part of the licensee justified a reduction in inspection effort. Substantial progress was made in developing "performance indicators" to track the changes in each plant's performance.

Safety System Functional Inspections and Safety System Outage Modification Inspections were more fully developed in fiscal year 1986 and have proven to be effective tools in assessing the readiness of key safety systems. Improvements were made in many other IE programs, as discussed throughout the chapter.

INSPECTION PROGRAMS

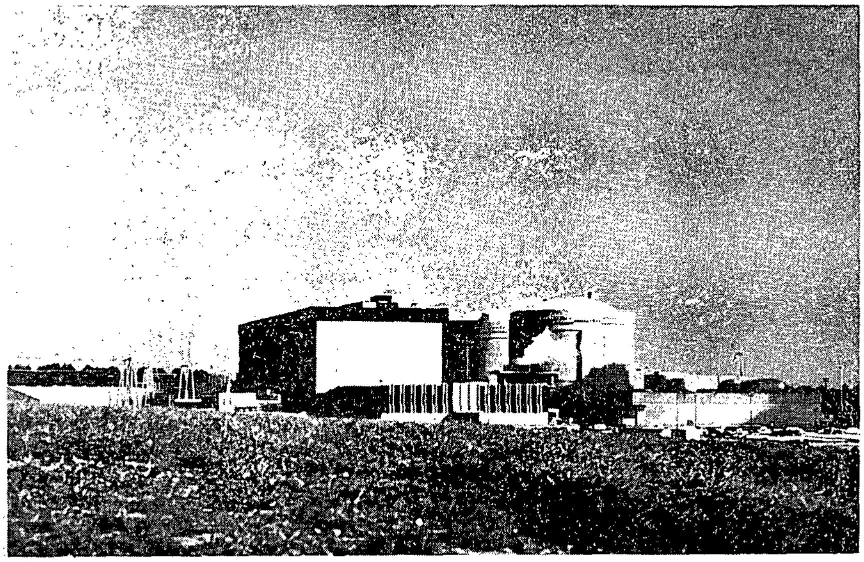
A basic element in the NRC regulatory program is the inspection of NRC licensees to determine whether they are in compliance with the provisions of their license and whether other conditions exist with safety implications significant enough to require correction. Primarily, the inspection programs of the NRC are carried out by the five NRC Regional Offices. As described later in this report, a limited number of inspection programs are conducted directly by NRC Headquarters. During fiscal year 1986 the Office of Inspection and Enforcement was responsible for developing inspection policies and procedures and for monitoring and assessing the effectiveness and uniformity of the programs carried out by the NRC Regional Offices. (Regional Offices are under the supervision of the NRC Executive Director for Operations.)

In addition to the routine, planned program of inspections for reactor, fuel cycle facility and materials licensees, the NRC conducts an aggressive program to deal with unsafe or potentially unsafe events or conditions at individual plant sites or involving licensed operations. In conducting these kinds of "reactive" inspections, the NRC seeks to determine the root cause of the event or condition, evaluates the licensee management's response to it, including action to prevent recurrence, and ascertains whether the problem could occur at other facilities.

Reactor Inspection Program

The operating reactor inspection program is conducted by both region-based and resident inspectors. In general, region-based inspectors are specialists, while resident inspectors are generalists. Resident inspectors provide the major on-site NRC presence for direct observation and verification of licensee activities. This work includes in-depth inspections of control room activities, maintenance and surveillance testing carried out by the licensee, periodic walk-down inspections to verify the correctness of system lineups for nuclear systems important to safe operation, and frequent plant tours to generally assess housekeeping, radiation control, security, equipment condition, and the like. The resident also acts as the primary on-site evaluator for the NRC inspection effort related to licensee event reports (LERs), events and incidents. Residents also serve as the NRC contact with local officials, the press, and the public. Region-based inspectors, on the other hand, perform detailed inspections in areas that include modifications, inservice inspection, fire protection, nondestructive testing, refueling, quality assurance, training, core physics testing, radiation protection, emergency planning, environmental protection, security/safeguards, and management systems.

Development and utilization of an innovative inspection approach to appraise the functionality of safety systems at operating plants continued in 1986. The new methodology, termed a Safety Systems Functional Inspection (SSFI), was employed at six plants, and refined in the process. Because of its proven usefulness in uncovering significant safety issues concerned with actual safety system availability and functionality, the SSFI inspection approach was adopted as the primary methodology to be employed by the Performance Appraisal Team in conducting its national inspection program for operating plants. The methodology was also included in the reactor inspection program for implementation by the Regions. Finally, the pilot program to test new Probabilistic Risk Assessment (PRA) inspection approaches continued in 1986. Four methodologies for using PRA to focus inspection activities were developed and implemented on a trial basis. A PRA team inspection procedure was developed and plant-specific PRA reference documents for resident and region-based inspector use were distributed as they became available. Two interactive computer programs



Alabama Power Company's Joseph M. Farley nuclear power plant, near Dothan, Ala., was one of the five top-performing reactor facilities and was put on the NRC's Special Minimum Inspection Program in 1986. The program, while initiated to allow more efficient use of NRC inspection resources, provides a number a precautionary measures, such as periodic regional performance reviews and special resident inspection schedules. The other plants qualifying for the special category in 1986 were St. Lucie (Fla.), Monticello (Minn.), Prairie Island (Minn.), and Kewaunee (Wis.).

now exist that can recalculate risk values based on existing plant conditions. A comparative evaluation of results from these trial efforts will be made during fiscal year 1987.

In 1986, IE made the following changes in the reactor inspection program to allow for more efficient use of limited inspection resources.

- The operating reactor inspection program is divided into three subprograms—Minimum, Basic, and Supplemental—to provide a priority for implementation. Over the past year, the inspection procedures in each subprogram were categorized into functional areas (operations, maintenance, surveillance, training and qualification, etc.). These functional areas are identical to those used to evaluate licensees in the Systematic Assessment of Licensee Performance (SALP) program. This scheme has allowed the allocation of inspection resources to a given plant to be based primarily on the results of the latest SALP evaluations, thereby concentrating those resources in precisely those functional areas in need of regulatory improvement.
- A program to reduce routine inspection effort at top-performing operating plants was implemented. This program, called the Special Minimum Program, contains precautionary features that (1) limit the number of plants in each Region that can simultaneously be placed on the program, (2) provide criteria for plant selection, (3) require periodic regional review of plant performance to justify continuation of the program at plants at which it is implemented, and (4) provide guidance regarding the scope of resident inspection to be in effect while the plant is subject to this program. During 1986, these plants were placed on the Special Minimum Program: Farley (Ala.), St. Lucie (Fla.), Monticello (Minn.), Prairie Island (Minn.), and Kewaunee (Wis.).
- The program begun in 1985 to place additional resident inspectors at single-unit operating reactor sites continued. As of the end of fiscal year 1986, 19 residents had been placed, consistent with the fiscal year 1986 staffing plan. The placement of the remaining 67 residents under this program will continue through fiscal year 1988. Placing additional residents at single unit operating reactor sites has allowed increased coverage for both routine and reactive on-site inspections and could lead to a decrease in the need for Regional inspector support.
- New procedures were added to the program for inspecting: (1) material control and accountability at reactors, (2) licensee corrective action systems, (3) the operational status of emergency preparedness programs, (4) licensees' equipment qualification programs (10 CFR 50.49), and (5) licensees' programs for writing, training on, verifying and validating emergency operating procedures in accordance with NRR-approved Procedures Generation Packages.
- Revisions were made to existing inspection procedures in the functional areas of Quality Assurance (QA) and fire protection, as well as in the inspection of licensees' inservice inspection programs and low-level radioactive waste storage facilities. An effort was also initiated to revise the technical guidance section of the IE Manual to ensure that the most current NRC guidance on technical issues is available to the Regions.
- Increased emphasis was placed on inspecting water chemistry controls at light water reactors.
- The development of an inspection program for decommissioning of reactors was initiated.

During 1986, the Commission issued a policy statement (51 FR 27921) regarding operating plant conditions, stating that the Commission expects the nuclear power industry to develop and implement programs to assure that the workplace is free of alcoholic beverages and illegal drugs and that persons within protected areas are not under the influence of any substance, legal or illegal, that adversely affects their ability to perform their safety-related duties in any way. The policy statement places considerable reliance on industry initiatives in developing and implementing these programs. The Commission will evaluate the effectiveness of these programs and reassess the possible need for further NRC action during fiscal year 1987 and the first half of fiscal year 1988.

In 1986, the evaluation of team inspection approaches was expanded to include Headquarters inspection teams in addition to the Regional teams. The Safety System Functional Inspection and the Safety System Outage Modifications Inspection team concepts were developed and implemented at selected plants in operating and outage conditions. The use of Regional teams, reinforced by IE Headquarters inspectors, continued at plants undertaking restart from extended outages, in those cases where augmented inspection coverage was considered appropriate.

Special post-fire shutdown team inspections continued during 1986. Approximately 25 reactor-site visits were conducted to verify licensee compliance with the regulatory requirements of Sections III. G., J., and O. of 10 CFR 50, Appendix R. The results of these inspections indicate, in general, steady improvement in compliance, since their inception in 1983. Approximately 30 reactor site-visits are planned for fiscal year 1987.

The Safety Systems Outage Modifications Inspection program (SSOMI) was developed further during the year. The SSOMI is an intensive, team-oriented approach to the inspection of design, procurement, installation and testing of plant modifications that are made during outages at operating nuclear power plants. In 1986, major team inspections were conducted at Fort Calhoun (Neb.) and Dresden Unit 3 (Ill.) to gain experience with the SSOMI concept. The number and significance of the inspection findings at both plants indicated the need for greater in-depth inspection during outages in which plant modifications are being made. The lessons learned from the two trial inspections will be incorporated into the detailed inspection procedures that are being developed for use in future SSOMI inspections.

At sites where reactor plants are under construction, program requirements were revised to focus greater attention on the early identification and prevention of problems at construction sites; increased depth of construction inspections; special emphasis on any plant area that was assigned a SALP 3 rating; earlier resolution of allegations; and the training, qualification and performance of construction workers and inspectors. Also, region-based specialist inspectors continued to supplement the inspections of resident inspectors continued to supplement the inspections of resident inspectors by utilizing their expertise in the fields of welding; nondestructive examination; and civil, mechanical, electrical and instrumental engineering.

The NRC's Mobile Nondestructive Examination Laboratory program reached a milestone in 1986 when it became a routine activity in all five NRC Regions. In the photo, inspector William Coley of Region II (Atlanta) examines a radiograph of a weld in the Region's mobile laboratory.

Nondestructive Examination Program. Since 1981, the NRC has operated a mobile nondestructive examination laboratory to conduct inspections at nuclear power plants throughout the county. The original purpose of the Nondestructive Examination Program (NDE) was to allow the NRC to make independent assessments of the quality of construction under way at power reactor sites by performing those examinations required by the relevant construction codes and comparing its results with those obtained by the licensees. The program was later expanded to include independent evaluation of pre-service and inservice inspection activities associated with major modifications and piping systems at operating nuclear facilities. The NDE van has also been used to provideindependent findings in connection with the investigation of various allegations registered with the NRC. The program is funded through the Office of Inspection and Enforcement and operated out of the Region I Office (Philadelphia).

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

In fiscal year 1986, a key program milestone was achieved when the mobile laboratory was engaged in activities in all five NRC Regions, dealing with normal construction inspections, followup on important plant modifications, and

Table 1. Inspections Conducted During FY 1986

Type	*Number of Licensees Inspected*	*Inspections*[1]
POWER REACTOR CONSTRUCTION[2a]	25	763
OPERATING POWER REACTORS[2b]	103	3,119
OTHER REACTORS[3]	45	75
FUEL FACILITIES[4]	42	238
MATERIALS[5]	2,457	2,551
VENDORS[6]	119	124
OTHERS, INCLUDING SHIPMENTS[7]	189	189

[1]Inspections recorded as complete on 766 Inspection Statistics and Text System as of 11/20/86.

[2a]Only those licensees that have, as of 09/30/86, not received even an initial OL. Note: no inspections were recorded for five licensees listed by license fee management staff (ADM "LFMS") as being in this group: Harris 2, Marble Hill 1, 2, Midland 1, 2.

[2b]All those licensees that have, as of 09/30/86, even an initial OL. Note: No inspections were recorded for 1 license listed by LFMS as being in this group: Manufacturing FNPI-8.

[3]All other 05000—dockets, that is all but those listed by LFMS as operating power for power reactor construction.

[4]All 030-, 040-, 070-, and 072- dockets listed by LFMS as being in this group. Note: LFMS lists 79 such licensees.

[5]All 030-, 040-, 070-, dockets that are not listed by LFMS as being fuel facilities.

[6]All 9990-, docket inspections.

[7]All 999-, 150-, and 110- docket inspections.

plant-specific problems. The laboratory was used at Comanche Peak (Tex.) to examine structural and support welding, during the latter part of September 1985. At the Byron (Ill.) facility in October, at Nine Mile Point Unit 2 (N.Y.) in December and at Beaver Valley Unit 2 (Pa.) in June, piping system and component welding was examined. In January 1986, reactor coolant piping replacement work at Vermont Yankee was inspected. At the Sequoyah (Tenn.) plant in February and at Watts Bar (Tenn.) in July, piping and support welds were evaluated, as part of the NRC's oversight of TVA's nuclear operations. In April 1986, the laboratory was employed at San Onofre Unit 1 (Cal.), in following up on the water hammer event at that facility in November 1985.

Overall, the Nondestructive Examination Program continues to prove a valuable asset to the NRC, affording the staff the means by which to obtain its own data for determining the adequacy and quality of licensees' construction and plant modification actions, while providing a method for independently assessing a licensee's quality assurance program.

Fuel Facilities and Materials Inspection Program

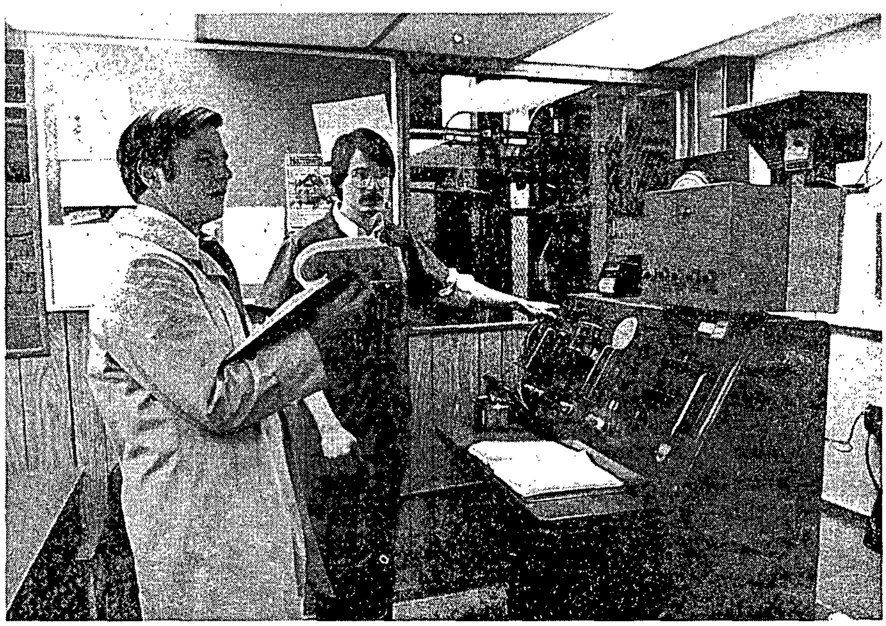
The fuel facilities inspection program covers all radiation safety- and safeguards-related activities at fuel production and research facilities and radiation safety at uranium mills and uranium conversion facilities. During the report period, special inspection procedures were developed and implemented for the construction and operation of large irradiators and systems for the dry storage of spent nuclear fuel. Radiological surveys of decontaminated sites to permit release for unrestricted use continued during the year.

Materials licensees also receive radiation safety inspections; included in this category are some 9,000 licensees whose activities involve nuclear medicine, radiography, industrial testing, well logging, research, production and distribution of radiation sources, and the handling and storage of radioactive wastes. Transporting, importing and exporting of materials which are subject to NRC licensing are also covered in this inspection program.

The NRC continues to give extensive attention and effort to identifying and correcting deficiencies in the handling of radioactive materials by licensees. During 1986, routine inspections of materials licensees were performed in accordance with the established frequencies, to identify violations and deficiencies requiring corrective action, and special inspections were conducted in connection with allegations against licensees and reported radiation incidents. A major special inspection effort was undertaken in connection with an immediately effective Order Suspending License to Radiation Technology, Inc., issued on March 3, 1986. This action was based on the fact that licensee personnel had repeatedly bypassed required safety interlocks on the service irradiator. Operation of a service irradiator with a safety interlock bypassed has the potential for serious adverse effects on individuals at the facility. For example, in 1977, when the personnel access door interlock system was rendered inoperable at this facility, a licensee employee entered the irradiator cell while the radioactive sources were exposed and received a radiation dose to the whole body of 150-300 rems. The licensee responded to the Order Suspending License in letters dated March 4 and 10, 1986. After careful consideration of the licensee's responses, the Director, Office of Inspection and Enforcement, determined that, subject to certain conditions, an adequate basis had been provided for conditionally lifting the suspension Order, including removal of the facility manager from all activities at the facility. This action was based on the licensee's plans and corrective actions to assure strict procedural compliance, to improve reliability, to clarify license conditions, and to hire independent radiological safety consultants to supervise the daily safety operations at the facility. These consultants had been given sufficient independent authority to make necessary safety decisions, including shutdown of the facility, when necessary and appropriate.

It has come to light from NRC inspections that certain licensees authorized to use industrial nuclear gauges containing sealed radioactive sources have discontinued operations without making adequate provision for the safe transfer of the gauges and without notifying the NRC of their intent to discontinue operations, so that the NRC could inspect the closeout procedures. An inspection at a licensee's facility completed on March 6, 1986, revealed how the failure to properly transfer a gauge and to properly notify the NRC resulted in loss of control of the device which contained a significant quantity of cobalt-60. The plant and all its equipment, including the gauge, was sold in 1981 to a non-licensee. Some time later, the non-licensee ceased operations and sold the plant and equipment to a salvage company. In late 1984, employees of the salvage company, unaware that the gauge contained a radioactive source, used a cutting torch on the gauge in an attempt to salvage some of the metal on the device. Fortunately, the sealed source itself was not damaged. However, the shutter mechanism was damaged and radiation levels at the open port of the gauge caused an employee to receive a significant radiation exposure.

Events involving medical facilities occasioned a number of materials inspections during the year. In one case, the University of Cincinnati Medical Center informed NRC Region III (Chicago) by telephone of the rupture of an



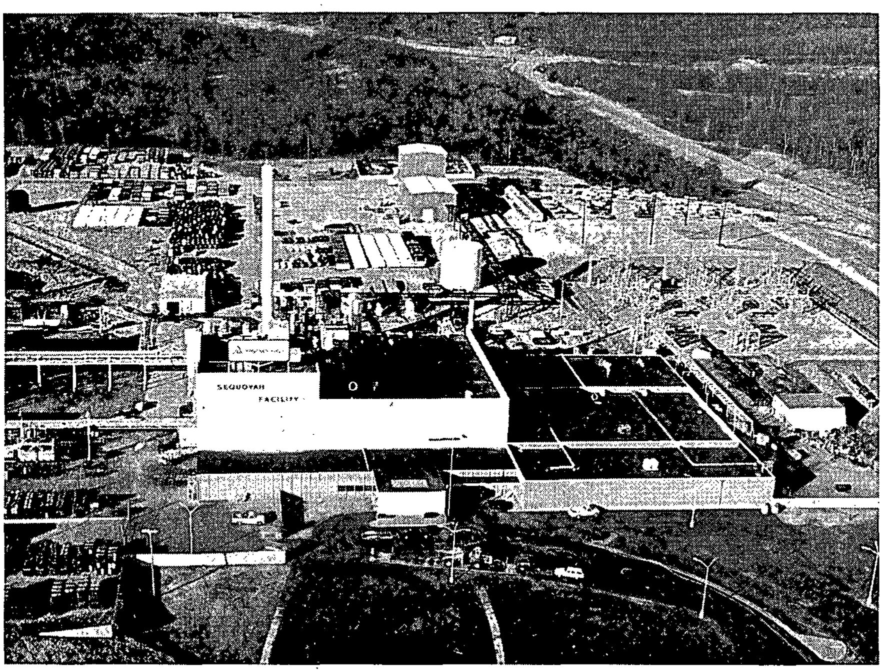
Francis Costello, Senior Radiation Specialist with NRC Region I (Philadelphia), reviews operating procedures with an employee of Radiation Technology, Inc., of Rockaway, N.J. The operations panel (at right) controls the raising and lowering of cobalt-60 radiation sources. The sources produce high intensity gamma radiation used to sterilize medical equipment and supplies and for other industrial applications. The NRC issued orders to shut the facility twice in 1986 for violations of safety requirements.

iodine-125 seed (nominal activity of 40 millicuries (the curie is the basic unit of radioactivity, representing 37 billion disintegrations per second)). The seed, which was one of eight being used by the University of Cincinnati Medical Center for brachytherapy treatment of brain tumors, was ruptured during removal of the seed from Heyer-Schulte coaxial catheters. The seeds contain iodine-125, adsorbed on anion exchange resin spheres within a .05 mm thick welded titanium capsule. As a result of the rupture, the patient received a thyroid burden and exposure of 557 microcuries and 2,087 rads respectively. (A rad, an acronym for "radiation absorbed dose," is the basic unit for measuring absorption of ionizing radiation.) All together, at least 60 hospital personnel, including personnel involved in the control and cleanup of the contamination, and also a friend of the patient, received thyroid uptake doses of .02 to 94.50 millirads. Because of the seed rupture event, the University of Cincinnati decided to terminate the use of high activity iodine-125 seeds until the safety and health physics aspects of such use were thoroughly explored and understood.

Six misadministrations in radiation therapy were reported by hospital licensees in which the patient received a radiation dose substantially greater than the prescribed dose. For example, in one case a patient received an estimated dose of 10,000 rads to the thyroid instead of the intended 0.9 rads. In another case, the licensee reported that a 32-year old patient undergoing brachytherapy treatment with a cesium-137 implant received a therapeutic radiation exposure of 8,015 rads rather than the prescribed 6,255 rads. An NRC medical consultant was retained to review the case. A patient that was misadministered phosphorus-32 as sodium phosphate instead of the prescribed phosphorus-32 as colloidal chronic phosphate received a significant unintended bone marrow dose. An NRC medical consultant believes the dose could have been as much as 700-800 rads to the patient's bone marrow with an increased chance of the patient contracting leukemia.

NRC-licensed fuel facilities are also subject to routine inspection. Special inspection procedures were mounted to deal with events at the Sequoyah Fuels Corporation's uranium hexafluoride conversion facility at Gore, Okla. At the Sequoyah Fuels facility, a cylinder overfilling and heating incident resulted in the release of the entire contents of a 14-ton cylinder of UF-6 to the plant environment and the death of a plant employee from hydrofluoric acid exposure.

As a result of lessons learned from the Sequoyah Fuels accident, a program of special team inspections was developed to be carried out at each major fuel cycle plant. The inspections dealt with a number of topics, including use of hazardous chemicals related to licensed use of radioactive materials, fire and explosion potentials affecting the NRC-regulated plant processes, conditions that might cause a failure of a component or a confinement barrier important to safe operation, procedures and training of employees in process operation, and other subjects related to potential accidents. Special team inspections were completed at the Combustion Engineering commercial fuel plant in Windsor, Conn., and at the United Nuclear navy fuel plant at Montville, Conn. The special inspections at other major plants were to be carried out early in fiscal year 1987. (See Chapter 4 for further discussion of this accident, and others cited.)



The accident of January 4, 1986, at the Sequoyah Fuels Corporation's plant in Gore, Okla., generated special NRC inspection procedures to be carried out at all major fuel-cycle plants. (See Chapters 4 and 5.) This view of the fuels facility shows, at top left, canisters of the type involved in the accident.

In 1986, the results of safeguards inspections performed at fuel facilities showed only minor violations of NRC requirements. Inventory differences at these facilities were either within regulatory limits or adequately explained by an investigation of the physical and book inventory data. (See Chapter 6.)

The report year saw a continuation of several of the spent fuel shipment series which began in late 1984. (Four earlier series, which began in 1983 and 1984, were completed by 1985; see p. 104 of the *1985 NRC Annual Report.*) During 1985, the continuing series of shipments that involved NRC licensees included two series by rail from power stations in Nebraska and Minnesota to the General Electric storage facility in Morris, Ill., and one set of highway shipments between the DOE West Valley, N.Y., facility and a power station in New York. Overall, since inspection of these shipments began in mid-1983, NRC has performed inspections with about a 45 percent frequency at point-of-origin and at the destination venues. These inspections are performed by Region-based inspectors from Regions I (Philadelphia) or III (Chicago), and, in some instances, by the NRC resident inspector at reactor sites.

Spent fuel shipment inspections continue to indicate that, with a few isolated exceptions, shipments have been in full compliance with NRC regulations. The exceptions involving "weeping cask" episodes continued to occur, though with much less frequency than before. (This phenomenon occurs when contaminated water residing in the pores of the external surface of the cask oozes out while the cask is being transported, slightly contaminating the cask surface.) NRC licensee efforts taken to accomplish better cask decontamination before shipment, combined with the use of higher efficiency smear-sampling techniques, have helped considerably in reducing these episodes. On June 10, 1985, the NRC Office of Inspection and Enforcement issued IE Information Notice No. 85-46, "Clarification of Several Aspects of Removable Radioactive Surface Contamination Limits for Transport Packages," which addresses the "weeping cask" phenomenon.

APPRAISAL PROGRAMS

Systematic Assessment of Licensee Performance

Under the NRC program for the Systematic Assessment of Licensee Performance (SALP), the performance of each licensee with a nuclear power facility under construction or in operation in the United States is evaluated by means of a periodic, comprehensive examination of all available data relevant to each facility.

The SALP process entails an integrated assessment based on manifold appraisals as to how licensee management directs, guides, and provides resources for the assurance of safety. The purpose of the SALP review is to direct both NRC and licensee attention toward those areas affecting nuclear safety that need improvement.

Part of the SALP assessment involves a review of the past year's Licensee Event Reports, inspection reports, enforcement history, and licensing issues. Also important are the evaluations by resident and region-based inspectors, licensing project managers, and senior regional managers, all of whom are to some degree familiar with the facility's performance. New data are not necessarily generated in the conduct of a SALP assessment. It ultimately consists of performance evaluations in a number of functional areas—including plant operations, maintenance, surveillance, emergency preparedness, security, licensing activities, training and qualification effectiveness, and so forth.

The SALP program supplements normal regulatory processes and is intended to be sufficiently diagnostic to provide meaningful guidance to utility management regarding NRC concerns about quality and safety in plant construction or plant operation. The results of the program comprise part of a data base for periodic reporting in the historical data summary, published in NUREG-1214. Computerized analysis of the data base for performance trends is currently under development. The SALP results are also used by regional managers as a primary determinant in allocating inspection resources.

Appraisal Teams

The Performance Appraisal Team (PAT) continued the comprehensive national inspection program for operating plants, principally by means of the Safety Systems Functional Inspection (SSFI), which combined programmatic and technical expertise on the same inspection team. Within the PAT framework, the SSFI assesses the operational readiness of key safety systems by examining system design, operation, maintenance, testing and modifications. The emphasis is on actual safety system availability and functionality, rather than on licensee preparation and implementation of the programs. The SSFI is an intense three-week inspection that focuses on the operational readiness and effectiveness of a selected safety system.

The PAT program advanced the SSFI initiative begun at Turkey Point (Fla.), conducting SSFIs at the Pilgrim (Mass.) plant, ANO-1 (Ark.), TMI-1 (Pa.), Oconee (S.C.), and Palisades (Mich.). One safety system at each plant was selected for inspection over a three-week period, and the resources of the team were directed into a detailed examination of the system's operational readiness. Another SSFI was conducted at the Trojan (Ore.) plant, where IE provided management and technical support to Region V (San Francisco) for the first such region-based inspection. At each plant, specific deficiencies in some aspect of safety system operation, maintenance, testing, or design were identified that probably would not have been found by the discrete programmatic and technical inspections. Because of the sound technical bases employed, the hardware orientation, and the

NRC's Executive Director for Operations, Victor Stello, Jr. (right rear), and former NRC Chairman Nunzio J. Palladino (center) visited the Tennessee Valley Authority's (TVA) Sequoyah nuclear power plant in 1986. The Commission staff continued its efforts during the year to resolve personnel and equipment problems at all TVA nuclear facilities.

genuine safety significance of SSFI findings, licensee acceptance of SSFI results and recognition of the problems has often been greater than for other approaches. Licensees have usually acted promptly to correct identified safety system deficiencies and have, in some cases, initiated their own inspection of other systems, using the SSFI approach.

In other activities over the report period, two team inspections took place at TMI-1 (Pa.) during the restart program for that plant (two PAT inspections, six months apart, had been ordered by the Commission as part of the TMI-1 restart) and there was a small-scale inspection at Diablo Canyon (Cal.). In addition, PAT inspectors accompanied region-based team inspectors at several plants, to give their assistance and expertise and to gain insight into regional inspection approaches and methods.

The Construction Appraisal Team (CAT) inspection program—similar in purpose to the PAT inspection program, but directed to reactors under construction—was discontinued after completion of inspections at South Texas Units 1 and 2 and at Palo Verde Unit 3 (Ariz.). As with previous CAT inspections at other facilities, these 1986 inspections evaluated design controls, construction practices and "as-built" conditions at the two plants. Examples of deficiencies found during the two CAT inspections include faulty electrical wiring of motor operated valves; failure to maintain material traceability, especially for fastener materials; unsatisfactory control of design and installation documents; mispositioned reinforcing bars in masonry walls adjacent to safety-related items; and vendor radiographs not of the quality required by the ASME code.

Previously budgeted resources for CAT inspections were reallocated to the inspection of the problems at TVA facilities (see below) and to the newly established Safety System Outage Modifications Inspection program (SSOMI).

TVA Problem Reactors

During fiscal year 1986, IE participated with other NRC Offices in the oversight, review and inspection of TVA efforts to resolve identified problems at all TVA nuclear power plants. IE was directly involved in the review of the TVA Nuclear Performance Plan, in the TVA employee concerns program, and in team inspections of the implementation of the TVA Plan for resolving the identified problems at its plants. IE had the lead responsibility for conducting inspections in the area of construction, equipment qualification and quality assurance. In 1986, teams made up of staff members from IE, NRC Region II (Atlanta) and NRR conducted two inspections at Watts Bar (Tenn.) and four inspections at Sequoyah (Tenn.) related to TVA's employee concerns program, and one inspection at TVA Headquarters (Knoxville, Tenn.) and at Watts Bar (Tenn.) regarding intimidation and harassment issues. An inspection of procurement activities was begun at Sequoyah but not yet completed by year's end. The six inspections related to employee concerns generally verified the adequacy of TVA procedures for dealing with these matters; however, additional inspection effort is required to complete the NRC's examination of how TVA is handling employee concerns. (See Chapter 2 for further discussion of NRC efforts to deal with problems in the TVA nuclear program).

THE ENFORCEMENT PROGRAM

The NRC's enforcement program has the objective of protecting public health and safety by ensuring that NRC licensees comply with regulatory requirements. The program is currently carried out under the revised enforcement policy (10 CFR Part 2, Appendix C, (1986)) which calls for strong enforcement measures to encourage full compliance and which will not permit operations by any licensees who fail to achieve adequate levels of protection.

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

While some Headquarters enforcement functions have been regionalized, the Regional Administrators have always been authorized to issue Notices of Violation not involving Civil Penalties. They are also authorized to issue proposed Civil Penalties, with the concurrence of the Director of the Office of Inspection and Enforcement. The Director of IE, however, remains responsible for all enforcement decisions and issues all Orders, including those imposing or proposing Civil Penalties.

Table 2 provides a listing and brief summary of the 96 Civil Penalty actions taken during fiscal year 1986. With some cases still pending and some of the penalties remitted or mitigated, a total of $3.095 million in penalties had been collected by the close of the report period. Some of these were Civil Penalties originally proposed in fiscal year 1985 or earlier.

Table 3 provides a description of the 10 enforcement Orders issued during fiscal year 1986.

QUALITY ASSURANCE

Quality Assurance Program Plan

In December 1985, the Commission transmitted to Congress its conclusions and exceptions regarding the Quality Assurance (QA) Report to Congress (NUREG-1055). That transmittal summarized actions under way or planned by the Commission as a direct consequence of the QA report. The staff moved to effect the plan intended to realize the recommendations of the QA report, as modified following public comments and ACRS and Commission guidance. Four major areas receiving staff emphasis are inspection programs, standards development, QA for waste management, and design inspection activities. Activities and accomplishments in these areas are described below.

Inspection Programs for Quality Assurance

Readiness Reviews. Readiness Reviews were identified as a topic for further analysis in the QA Report to Congress. A Readiness Review is a formal assessment of the licensee's readiness to construct or operate a nuclear power plant. It is a comprehensive evaluation of the licensee's program for design, construction, and pre-operational activities, and it examines the licensee's performance in meeting its commitments and regulatory requirements. It provides a systematic, structured mechanism for licensee self-assessments at key program milestones, so that issues and problems are identified at a stage when they can best be resolved.

Two such programs were initiated by utilities during fiscal year 1985 and continued during fiscal year 1986. Readiness Reviews are being performed at Georgia Power Company's Vogtle Unit 1 and Washington Public Power Supply System's WNP-3. The Vogtle Unit 1 plant was scheduled for low-power licensing in December 1986. The WNP-3 plant is in a deferred status. The Readiness Review Pilot Programs at these two facilities were testing the feasibility of the concept when applied to a plant nearing operational status and a deferred plant, respectively.

The Vogtle Readiness Review Program, which began in March 1985, constitutes a self-assessment undertaken by the licensee, the Georgia Power Company (GPC). The program was organized into 28 work packages or modules; GPC submitted its final module of the program to the NRC in July 1986. As of September 30, 1986, the NRC had completed 75 percent of its review of the GPC self-assessment.

The WNP-3 Readiness Review Program comprises two phases. The first phase, which began February 1986, entails an assessment of the quality of all work completed prior to work stoppage. This phase involves preservation engineering and construction assurance reviews. The current plans for this phase include a scheduled completion by mid-1989. The first three of nine engineering assurance review modules and two of 15 construction assurance review modules were initiated during fiscal year 1986. The construction modules will contain the review of the preservation program. Phase Two would begin after the restart of construction and is expected to be structured according to lessons learned from the Vogtle Readiness Review Program.

QA Inspection Procedures. Consistent with the recommendations of the QA report to Congress, the staff is reorienting the NRC QA inspection program for operating reactors to provide proper emphasis to QA program performance and effectiveness. The reorientation will sharpen the focus on confirming the implementation of QA programs and the quality of completed work. QA inspection procedures that emphasize program implementation and QA program effectiveness are being developed and, after field testing, will be incorporated into the NRC inspection program.

QA Standards Development

Efforts are under way to revise, consolidate where possible, or develop as necessary new NRC Regulatory Guides addressing quality assurance program requirements for nuclear power plant design, construction and operation. This effort is intended to reflect industry changes, the lessons described in the report to Congress, recent events, and NRC inspections. Revisions to the Regulatory Guide for QA Operations are under development. Similar activities incorporating lessons learned will be undertaken after completion of this central task.

Table 2. Civil Penalty Actions During FY 1986

Licensee	*Amount*	*Reason*
Mississippi Power & Light Company (Grand Gulf)	$125,000 proposed in FY85; withdrawn in FY 86	Violations involved five alleged material false statements regarding development of technical specifications. Civil penalty withdrawn after staff determined four of the five violations should be withdrawn or reduced in severity level.
Kansas Gas & Electric Co. (Wolf Creek) EA 84-87	$64,000 proposed in FY 84; paid in FY 86	Violation involved discrimination against a member of the Quality Assurance/Quality Control organization.
Duke Power Company (Catawba) EA 84-93	$64,000 proposed in FY 85; $20,000 imposed; and paid in FY 86	Violation involved discrimination against a Quality Control welding inspector.
Nuclear Fuel Services, Inc. Rockville, MD EA 84-128	$20,000 proposed in FY 85; $15,000 imposed; in FY 86	Violation involved the accumulation of uranium-bearing solids in process equipment above specified limits.
North American Inspection, Inc. Laurys Station, PA EA 85-01	$5,000 proposed and imposed in FY 85; withdrawn in FY 86	Violations involved inadequate management control and oversight of the radiological safety program.
Louisiana Power & Light Co. (Waterford) EA 85-10	$130,000 proposed in FY 85; $110,000 imposed and paid in FY 86	Violations involved quality control issues identified during inspections and investigations conducted to evaluate allegations received in 1983.
American Can Co. Greenwich, CT EA 85-47	$500 proposed in FY 85; $250 imposed and paid in FY 86	Violation involved the unauthorized removal of radioactive material from an unrestricted area.
Commonwealth Edison Co. (Byron) EA 85-53	$25,000 proposed in FY 85; imposed and paid in FY 86	Violation involved a failure to adequately implement compensatory measures to control access into a vital area.
Princeton University Princeton, NJ EA 85-70	$4,000 proposed in FY 85; $2,000 imposed and paid in FY 86	Violations associated with an individual receiving a skin exposure of 38 rems.
Florida Power & Light Co. (Turkey Point) EA 85-80	$100,000 proposed in FY 85; paid in FY 86	Violation involved the failure of the licensee's staff to determine whether a modification to the spent fuel pits piping created an unreviewed safety question.
Astrotech, Inc. Harrisburg, PA EA 85-86	$5,000 proposed in FY 85; imposed in FY 86; pending	Violations involved inadequate management oversight and control of licensed facilities.
Hurley Medical Center Flint, MI EA 85-89	$2,500 proposed in FY 85; imposed in FY 86; pending	Violations involved a breakdown in management oversight and control of the licensee's health physics program.
Met-Chem Engineering Laboratories Salt Lake City, UT EA 85-92	$5,000 proposed in FY 85; paid in FY 86	Violations involved the use of an unauthorized and unqualified individual to perform licensed activities and an overexposure to a radiographer.
Calumet Testing Services, Inc. Griffith, IN EA 85-93	$16,000 proposed in FY 85; $12,000 imposed and paid in FY 86	Violation involved an overexposure greater than 75 rems to the hand of a radiographer.

Licensee	*Amount*	*Reason*
American Electric Power Service Corp. (DC Cook) EA 85-94	$100,000 proposed in FY 85; paid in FY 86	Violation involved a failure to maintain adequate control over access to vital areas and a reporting failure.
Commonwealth Edison Co. (LaSalle) EA 85-95	$125,000 proposed in FY 85; paid in FY 86	Violations involved failure to ensure that modifications performed on safety-related systems were adequately controlled.
Metro Health Center Erie, PA EA 85-98	$3,750 proposed in FY 85; imposed and paid in FY 86	Violations involved a breakdown in management oversight and control of the radiation safety program.
Commonwealth Edison Co. (Byron) EA 85-99	$50,000 proposed in FY 85; $25,000 imposed and paid in FY 86	Violations involved the failure of management to follow radiation protection procedures, evaluate radiological conditions, or prepare adequate procedures.
Baltimore Gas & Electric (Calvert Cliffs) EA 85-102	$50,000 proposed in FY 85; paid in FY 86	Violation involved the licensee's failure to implement and maintain the post-accident sampling system.
Sacramento Municipal Utility District (Rancho Seco) EA 85-103	$50,000 proposed in FY 85 paid in FY 86	Violation involved design deficiencies which led to a non-isolable primary coolant system leak. Also, numerous piping supports were not installed in accordance with procedures or drawings.
VA Electric & Power Co. (North Anna) EA 85-68	$20,000 proposed and paid in FY 86	Violation involved the falsification of records regarding deficiencies in the application of unqualified protective coatings in the reactor containment buildings.
Arizona Public Service (Palo Verde) EA 85-87	$50,000 proposed and paid in FY 86	Violation involved the post-accident sampling system, that resulted from relocating a post-accident sampling point without evaluation to show specific requirements were met.
Vermont Yankee (Vermont Yankee) EA 85-105	$50,000 proposed, imposed and paid in FY 86	Violation involved an unplanned radiation exposure of approximately 1.3 rem to the whole body of a radiological survey technician.
Boston Edison Company (Pilgrim) EA 85-106	$50,000 proposed; $25,000 imposed and paid in FY 86	Violations involved multiple examples of openings in a vital area barrier.
Toledo Edison Company (Davis Besse) EA-107	$900,000 proposed in FY 86; pending	Violations involved the loss of main and auxiliary feedwater.
Maine Yankee Atomic (Maine Yankee) EA-108	$80,000 proposed, imposed and paid in FY 86	Violations which rendered inoperable all four channels of the reactor protective system for the low steam generator pressure trip function and three of four channels of the feedwater trip system for low steam generator pressure.
Commonwealth Edison (LaSalle) EA 85-114	$37,500 proposed and paid in FY 86	Violation involved the failure to maintain adequate control over security badge system.
Quality Assurance Test LaFox, Illinois EA 85-116	$500 proposed and paid in FY 86	Violations involved licensed material being used by personnel who had not completed training program, failure to use film badges or other dosimetry devices, and failure to leak test 2 moisture density gauge.

Table 2. Civil Penalty Actions During FY 1986
(continued)

Licensee	*Amount*	*Reason*
South Carolina Electric $ Gas (Summer) EA 85-121	$50,000 proposed and paid in FY 86	Violations involved the failure to meet technical specification requirements for low head safety injection and failure to meet operability requirements for overpressure delta temperature trap instrumentation circuits.
DelMonte Plymouth, In. EA 85-124	$500 proposed and paid in FY 86	Violation involved the unauthorized removal and improper disposal of licensed material.
Exam Company Tulsa, OD EA 85-125	$5,000 proposed and imposed in FY 86; pending	Violations involved an over-exposure, failure to perform surveys, and failure to follow operating and emergency procedures.
South California Edison Company (San Onofree) EA 85-128	$50,000 proposed and paid in FY 86	Violation involved an improper and unauthorized repair to the oil sight glass for the turbine-driven auxiliary feedwater pump, which resulted in one of two pumps not being able to perform its intended safety function when called upon.
Quality Assurance Labs South Portland, ME EA 85-129	$5,000 proposed and paid in FY 86	Violations involved failure to conduct adequate radiation surveys, failure to post a high-radiation area, failure to wear a film-badge during radiographic operations, and failure to maintain surveillance during radio-graphic operations.
E. L. Connell Bridgeport, PA EA 85-130	$500 proposed and paid in FY 86	Violations involved possession of licensed material at unauthorized locations, failure to secure licensed material in unrestricted area, failure to adhere to DOT requirements, failure to leak test sealed sources and failure to conduct a physical inventory every 6 months.
West Virginia University Morgantown, WV EA 85-134	$1,250 proposed and paid in FY 86	Violations involved failure to perform bioassays, inventories, surveys, and maintain records.
Chen and Associates, Inc. Casper, WY EA 85-135	$500 proposed and paid in FY 86	Violations involved failure to make required surveys, promptly evaluate apparent overexposures, properly block and brace transportation packages, use only properly trained personnel, conduct leak tests, and wear film badges as required.
Sacremento Municipal Utility Dist. (Rancho Seco) EA 85-137	$25,000 proposed and paid in FY 86	Violations involved inadequate access control of the protected area.
Pacific Power & Light Point of Rocks, WY EA 86-02	$500 proposed and paid in FY 86	Violations involved the use of an unsupervised individual to perform licensed activities and failure to follow established procedures.
Nebraska Public Power Dist. (Cooper) EA 86-03	$50,000 proposed and paid in FY 85	Violation involved failure to properly implement surveillance procedures which rendered the automatic start capabilities of both trains of the standby-gas treatment system inoperable and negated the capability of automatically isolating the reactor building ventilation system. Also, the facility operated with drywell snubbers inoperable.

Licensee	Amount	Reason
Pacific Gas & Electric (Diablo Canyon)	$50,000 proposed and paid in FY 86	Violation involved the inoperability of one channel of the automatic actuation logic for one main steam isolation valve.
Portland Gen. Electric (Trojan) EA 86-05	$50,000 proposed and paid in FY 86	Violations involved a breakdown in the quality assurance program for maintenance activities.
Texas Utilities Gen. Co. (Comanche Peak) EA 86-09	$250,000 proposed in FY 86; pending	Violations involved significant weaknesses in the implementation of the quality programs during construction and weaknesses in the cable tray reinspection program.
Commonwealth Edison Co. (Byron) EA 86-18	$50,000 proposed and paid in FY 86	Violation involved failure to control access to a vital area.
Commonwealth Edison Co. (LaSalle) EA 86-19	$50,000 proposed and paid in FY 86	Violation involved two Unit 2 ECCS divisions being inoperable at the same time.
Florida Power & Light Co. (Turkey Point) EA 86-20	$300,000 proposed and	Violations involved inadequate maintenance on the auxiliary feedwater system and supporting backup nitrogen system.
Duke Power Co. (Oconee) EA 86-21	$25,000 proposed and paid in FY 86	Violation involved the failure to maintain adequate barriers for a portion of the protected area.
Florida Power Corporation (Crystal River) EA 86-22	$100,000 proposed; $50,000 imposed and paid in FY 86	Violations included a breached vital area barrier.
American Electric Power Service Corp. (Indiana Michigan Electric D.C. Cook) EA 86-23	$100,000 proposed and paid in FY 86	Violations involved a failure of plant personnel to correctly align valves in the performance of a containment integrated leak rate test on Unit 1, an inoperable Unit 2 control room ventilation system resulting from a closed air intake damper, a failure perform airlock leak testing, and inadequate channel calibration and functional tests.
Professional Service Oak Brook, IL EA 86-24	$500 proposed and paid in FY 86	Violations involved the failure to ensure shipping papers accompanied licensed radioactive material and the improper storage of shipping papers with a gauge.
Florida Power & Light Co. (Turkey Point) EA 86-28	$50,000 proposed and paid in FY 86	Violations involved a material false statement regarding the installation and operability of two environmentally qualified nuclear instrumentation channels.
Abington Memorial Hospital Abington, PA EA 86-31	$2,500 proposed and paid in FY 86	Violations involved failure to hold quarterly meetings of the Radiation Safety Committee, perform daily constancy checks, perform molybdenum-99 contamination tests, secure licensed material, perform radiation surveys of incoming packages, and use a dosimetry system calibrated by an accredited laboratory for the annual calibration of the teletherapy unit.

Table 2. Civil Penalty Actions During FY 1986
(continued)

Licensee	*Amount*	*Reason*
Ambric Testing and Engineering, Alexandria, VA EA 86-36	$250 proposed and withdrawn in FY 86	Multiple violations included failure to evaluate and report thermoluminescent dosimetry badge readings, use of unsupervised individuals to perform licensed activities, and failure to comply with requirements for the testing, inventory, and transportation of licensed material. After considering the licensee's response, the civil penalty was fully mitigated.
Florida Power & Light Co. (Turkey Point) EA 86-38	$50,000 proposed and $25,000 imposed in FY 86; pending	Violations involved an unauthorized entry of a worker into a high radiation area.
Mercy Hospital Wilkes Barre, PA EA 86-40	$500 proposed in FY 86; pending	Violation involved the failure to report a misadministration and a material false statement.
Washington Hospital Center Washington, D.C. EA 86-43	$5,000 proposed and paid in FY 86	Violations involved operation of a teletherapy unit with an inoperable safety interlock.
Nebraska Public Power Dist. (Cooper) EA 86-44	$50,000 proposed in FY 86; pending	Violations involved an inadequate barrier to the intake structure.
South Carolina Electric & Gas (Summer) EA 86-45	$50,000 proposed and imposed in FY 86; pending	Violations involved incorrect breaker alignment which resulted in component cooling water and service water pumps being inoperable.
Commonwealth Edison Co. (Byron) EA 86-48	$100,000 proposed and $50,000 paid FY 86	Violations involved two degraded trains of an emergency core cooling system.
Commonwealth Edison Co. (Zion)	$25,000 proposed in FY 86; pending	Violation involved an inoperable auxiliary feedwater pump.
Louisiana Power & Light (Waterford) EA 86-50	$50,000 proposed, imposed, and paid in FY 86	Violation involved the changing of operational modes with train of the containment spray system inoperable.
Combustion Engineering Windsor, CT EA 86-51	$15,000 proposed in FY 86; pending	Violation involved the transfer of licensed material to an unauthorized person.
Duke Power Company (McGuire) EA 86-52	$50,000 proposed and imposed in FY 86; pending	Violation involved the restart of Unit 1 without satisfying the technical specification requirements for operable flow paths for the Chemical Volume Control portion of the ECCS.
Portland Gen. Electric (Trojan) EA 86-54	$80,000 proposed and paid in FY 86	Violations involved failure to maintain the control room emergency ventilation system in an operable status over an extended period of time.
TVA (Browns Ferry) EA 86-56	$150,000 proposed in FY 86; pending	Violations involved inadequate design control and corrective actions for safety-related cable tray supports.

Licensee	*Amount*	*Reason*
Philadelphia Electric Company (Peach Bottom) EA 86-59	$200,000 proposed in FY 86; pending	Violations involved the failure to implement required procedural controls in withdrawing a control rod out of sequence and in bypassing the Rod Sequence Control System.
Detroit Edison Company (Fermi) EA 86-61	$300,000 proposed and paid in FY 86	Violations involved operator failure to observe the rod pull procedure and management oversight of the control room.
Texas Utilities Gen. Company (Comanche Peak) EA 86-63	$120,000 proposed and $40,000 paid in FY 86; $80,000 pending	Violations involved intimidation of Quality Control/Quality Assurance personnel.
Arizona Public Service (Palo Verde) EA 86-65	$100,000 proposed in FY 86; pending	Violations involved the failure to implement and maintain the physical security plan.
Detroit Edison Company (Fermi) EA 86-66	$50,000 proposed and paid in FY 86	Violations involved security matters including the falsification of required records by a security guard.
Allied Corporation Merto, WK Metropolis, IL EA 86-67	$25,000 proposed and paid in FY 86	Violations involved the overfilling and pressurization of uranium hexafluoride cylinders.
Georgia Power & Light (Hatch) EA 86-69	$50,000 proposed and paid in FY 86	Violations involved the failure to implement approved compensatory measures following the loss of alarm annunciation capability in the central and secondary alarm stations.
Consumers Power Company (Palisades) EA 86-78	$50,000 proposed and paid in FY 86	Violation involved inadequate corrective actions when containment penetration leakage problems were identified.
Progressive Engineering Grand Rapids, MI EA 86-79	$500 proposed in FY 86; pending	Violation involved unauthorized use of licensed material and other radiological safety problems.
Harvard University Cambridge, MA EA 86-80	$2,500 proposed and paid in FY 86	Violations involved unsecured radioactive materials and improper transportation of radioactive materials.
Public Service Corp. of Colorado (Ft. St. Vrain) EA 86-81	$65,000 proposed in FY 86, pending	Violations involved potential unauthorized access to controlled areas.
Kansas Gas & Electric Co. (Wolf Creed) EA 86-84	$40,000 proposed in FY 86; pending	Violations involved protected and vital area barriers.
Virginia Electric & Power Company (North Anna) EA 86-86	$25,000 proposed and paid in FY 86	Violation involved unauthorized entry to controlled areas.
Commonwealth Edison Co. (Byron) EA 86-87	$25,000 proposed in FY 86; pending	Violation involved discrimination against a quality control inspector.
TVA EA 86-93	$150,000 proposed and paid in FY 86	Violations involved discrimination against three TVA QA/QC employees and a Watts Bar engineer.
Ferris State College Big Rapids, MI 86-96	$10,500 proposed in FY 86; pending	Violations involved an overexposure of 21 rems.

Table 2. Civil Penalty Actions During FY 1986
(continued)

Licensee	*Amount*	*Reason*
South Cal. Edison Company (San Onofre) EA 86-97	$180,000 proposed in FY 86; pending	Violations involved a loss of a.c. power and subsequent feedwater system water hammer event.
Public Service Corp. of Colorado (Ft. St. Vrain) EA 86-98	$75,000 proposed and paid in FY 86	Violations involved exceeding of authorized power limit.
Florida Power and Light Company (St. Lucie) EA 86-99	$25,000 proposed and paid in FY 86	Violations involved degraded vital area barrier.
Cook County Highway Dept. Chicago, IL EA 86-102	$500 proposed in FY 86; pending	Violations involved unauthorized storage of licensed material.
E. I. Dupont Boston, MA EA 86-103	12,500 proposed and paid in FY 86	Violations involved unsecured radioactive material and transportation matters.
H&G Inspection Company Houston, TX EA 86-104	$2,500 proposed and paid in FY 86	Violations involved failure to perform surveys and an overexposure.
Texas A&M College Station, TX EA 86-105	$1,250 proposed in FY 86; pending	Violations involved exceeding technical specification limitations on the reactivity worth of experiments.
Maryview Hospital Portsmouth, VA EA 86-107	$2,500 proposed and paid in FY 86	Violations involved a medical therapeutic misadministration.
South Cal. Edison Company (San Onofre) EA 86-108	$25,000 proposed and paid in FY 86	Violations involved failure to implement Safeguards Contingency Plan.
Gulf States Utility Company (River Bend) EA 86-109	$65,000 proposed and paid in FY 86	Violations involved failure to control access to vital areas.
Detroit Edison Company (Fermi) EA 86-112	$75,000 proposed and paid in FY 86	Violations involved a license condition and technical specifications regarding service water systems, containment integrity, and a room cooler.
Union Electric Company (Callaway) EA 86-119	$25,000 proposed in FY 86; pending	Violation involved failure of licensee personnel to recognize LCO requirements.
James River Corporation Pardment, MI EA 86-120	$500 proposed and paid in FY 86	Violation involved a missing gauge containing radioactive material.
Donohue and Associates, Inc. Sheboygen, WI EA 86-129	$250 proposed and paid in FY 86	Violations involved failure to maintain adequate surveillance and control over a gauge containing radioactive material.
Star-Jet Services, Inc. Oklahoma City, OK EA 86-134	$500 proposed in FY 86; pending	Violations involved radiation safety and failure to control access to licensed material.
South Carolina Electric and Gas (Summer) EA 86-126	$50,000 proposed in FY 86; pending	Violations involved failure to maintain the charging pumps operable.

Table 3. IE Orders Issued During FY 1986

Licensee	*Date*	*Reason*
Institute Medical Research Bennington, VT EA 85-110	October 25, 1985	Order Suspending License (Effective Immediately) Reason: Failure to meet requirements in license including unauthorized use of radioactive material.
Pesses Company Pulaski, PA EA 85-122	January 22, 1986	Order Modifying License (Effective Immediately) Reason: Failure to control licensed radioactive material.
Radiation Technology Rockaway, NJ EA 86-017	June 23, 1986	Orders Modifying and Suspending License (Effective Immediately) Reason: Defeating of interlock systems and providing false information to the NRC on the circumstances regarding changes to the conveyer interlock system.
Florida Power and Light Company (Turkey Point) EA 86-020	August 12, 1986	Confirmatory Order Reason: Failure to satisfy Technical Specification and 10 CFR 50.59 requirements.
Bloomington Hospital Bloomington, IL EA 86-039	April 22, 1986	Order To Show Cause Why License Should Not Be Suspended and Modified (Effective Immediately) Reason: Failure to report diagnostic misadministrations and impeding NRC's inspection effort.
Mercy Hospital Wilkes Barre, PA EA 86-040	June 17, 1986	Order To Show Cause Why License Should Not Be Modified Reason: Failure to report a misadministration and providing false information to the NRC.
Valley Radiology Association Kingston, PA EA 86-041	June 17, 1986	Order To Show Cause Why The License Should Not Be Modified Reason: Failure to report a misadministration and providing false information to the NRC.
Washington Hospital Center Washington, D.C. EA 86-043	May 29, 1986	Confirmatory Order Reason: A medical misadministration and operation of a teletherapy unit with an inoperable safety interlock.
Detroit Edison Company (Fermi) EA 86-061	July 3, 1986	Order Modifying License Reason: Inadequate supervision of licensed activities in the control room.
Maryview Hospital Portsmouth, VA EA 86-107	August 7, 1986	Confirmatory Order Modifying License Reason: A medical misadministration to a patient involving the improper chemical form of a radiopharmaceutical.

QA for Waste Management Activities

Federal regulations (10 CFR Part 60) require the U.S. Department of Energy (DOE) to execute a QA program to reinforce confidence in the data on which the NRC can base an authorization to construct—and eventually a license to operate—a permanent geological repository for high-level nuclear waste. In the next few years, DOE's "site characterization" activities for the three candidate sites—Yucca Mountain (Nev.), Hanford (Wash.), and Deaf Smith County (Tex.)—will generate much of the technical data in support of the DOE application for an NRC license. The NRC continued its effort to develop guidance on quality assurance programs for site characterization, as a supplement to the NRC regulatory requirements. A task force has prepared supplementary guidance, called Generic Technical Positions, to cover such specific waste management issues as peer review, QA for existing data, and determination of the items and actions to which the QA program applies. Guidance in other areas—such as configuration management for conceptual design, and computer software QA—is still under development. The staff has been reviewing DOE's site characterization QA plans and will participate in audits by the NRC Office of Nuclear Material Safety and Safeguards of the implementation of these QA plans, once site characterization begins.

The IE Quality Assurance Branch is also providing QA guidance or consultation to the other program offices of the NRC on such issues as decommissioning, independent spent fuel installations, and monitored retrievable storage facilities.

Design Assurance Activities

As part of the program to improve the assurance of design quality at nuclear power plants, the NRC has developed and effected several approaches toward providing a comprehensive examination of design development and design implementation for a selected safety-related system on a given reactor project. The results of the examinations are conveyed to the appropriate Regional and Headquarters Offices and are used as part of the overall NRC assessment of the plant prior to a decision on issuance of an operating license.

Direct NRC inspection of design activities was carried out at several facilities through an Integrated Design Inspection (IDI) program. Remaining "open items" from the last scheduled IDI (at the Seabrook plant (N.H.)) were completed in fiscal year 1986.

The Independent Design Verification Program (IDVP) addresses the same purposes as the IDI, except that an independent contractor is employed by the licensee to conduct the review. The NRC oversees the contractor's activities with some limited direct inspection of the design. The IDVP involves an examination of the design process, including a sample of design details. The IDVP also includes elements of on-site verification in selected areas. Since IE assumed responsibility for the IDVP program in 1984, IDVPs have been performed at the Byron (Ill.), Limerick (Pa.), Clinton (Ill.), and Hope Creek (N.J.) facilities. During fiscal year 1986, the last scheduled IDVP was completed for Hope Creek, and all issues developed by the IDVP were resolved. Also completed in fiscal year 1986 were the inspections of the design verification activities conducted by the licensee on the Byron Unit II and Braidwood Units I and II in Illinois, which are duplicates of Byron I. The licensee was asked by the staff to perform applicable corrective actions derived from inspections at Byron Unit I. An IDVP type of design review was completed for Vogtle Unit 1 (Ga.) in connection with a pilot readiness review program undertaken at this plant.

Self-directed licensee Engineering Assurance Programs (EAP) are also used to provide added confidence in the adequacy of design and in the design control process. This program is similar to an IDVP in that the technical reviewers of the design and design process have no part in the original design work. However, an EAP differs from an IDVP in that the former review may be conducted as an ongoing activity during the design and construction, rather than, as with the latter, taking place at or near the end of design and construction. It also differs in that the reviews are performed by technical personnel employed by the applicant as its principal plant architect-engineer, rather than by independent contractor personnel. EAPs have been initiated at the South Texas Project, Millstone Unit 3 (Conn.), Nine Mile Point Unit 2 (N.Y.), and Beaver Valley Unit 2 (Pa.). During fiscal year 1986, EAPs were completed for Millstone Unit 3 and Nine Mile Point Unit 2. For these two plants, all issues developed by the EAPs were resolved. The EAPs for South Texas and Beaver Valley Unit 2 were in progress at the close of the report period; an EAP type of design review was also being integrated into the WNP-3 (Wash.) readiness review.

Other design assurance activities are included in the Safety System Functional Inspection (SSFIs) and the Safety System Outage Modifications Inspections (SSOMIs), discussed at the outset of this chapter. These inspections are conducted at the licensee's office and the plant sites. The purpose of the design portions of these inspections is to ensure that the as-built design and installation matches the current design basis requirements, and that any system modifications implemented since initial licensing have not inadvertently infringed the original design basis for the plant.

VENDOR INSPECTION PROGRAM

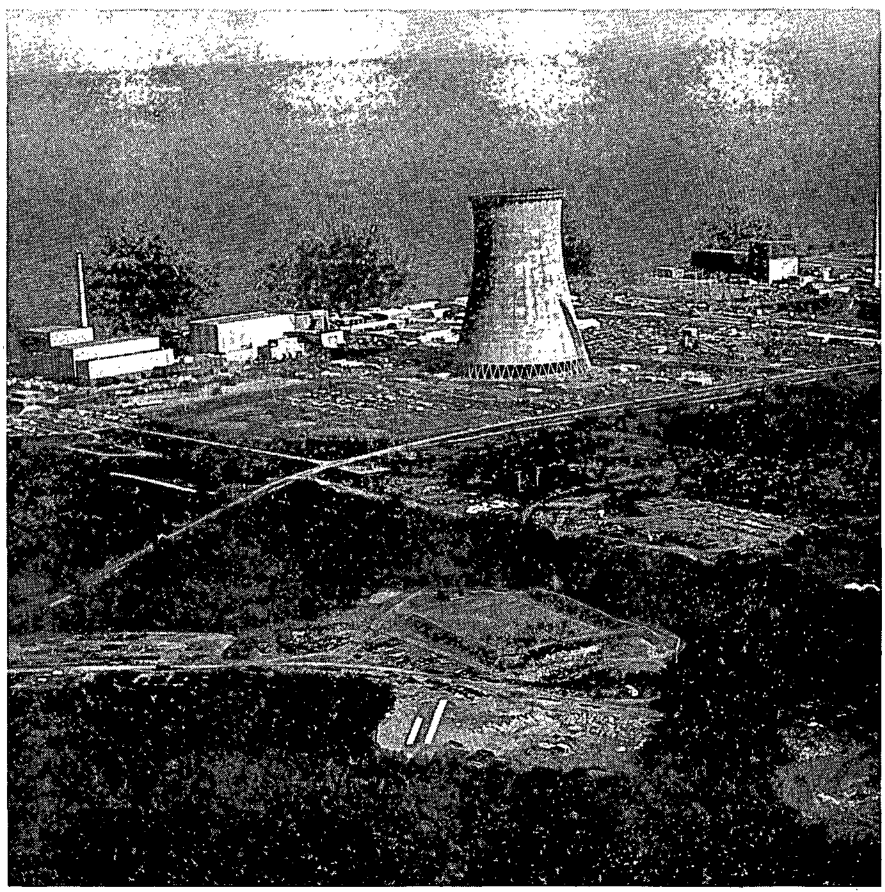
Vendor Program Branch (VPB) inspections are now focused on vendor activities associated with nuclear plant operation, maintenance, and modifications, since the number of plants in operation has increased substantially in recent years. Inspection emphasis is on the quality of the vendor products, licensee/vendor interfaces, environmental qualification of equipment, equipment problems found during operation, and corrective action in response to identified problems. Inspections of vendors and contractors are based on information from a variety of sources, including licensee construction deficiency and operating reactor event reports, vendor reports of product defects, reports of events from the NRC Regional Offices, allegations from members of the public pertaining to vendor activities, and vendor issues identified by the NRC through its inspection programs.

VPB Inspections

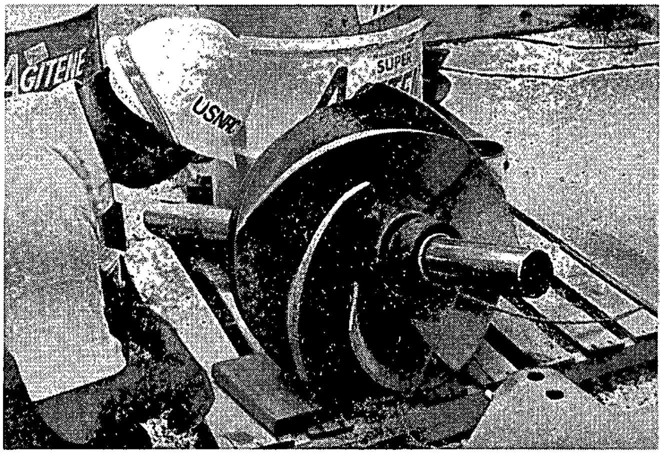
The VPB performed approximately 120 inspections during fiscal year 1986. The most frequently conducted inspections involved component manufacturers, equipment qualification at test facilities, and licensees' equipment qualification programs. Other inspections were directed toward design organizations, fuel fabricators, material suppliers, and licensees. Ten inspections were conducted in order to resolve allegations. In approximately one-half of the inspections, expert assistance was engaged from outside contractors, including the National Laboratories. The inspections generally result in the identification of nonconformances and/or violations that the vendor or contractor is expected to correct promptly.

Equipment qualification (EQ) inspections were performed to verify licensee compliance with requirements of the law (10 CFR 50.49) that all safety-related electrical equipment exposed to a harsh environment be qualified to operate during normal and "design-basis accident" environmental conditions. Eighteen EQ inspections of this type were performed in fiscal year 1986. A significant EQ problem was discovered concerning internal wiring for Limitorque valve operators. It was determined that some of the internal wiring was not included in the original qualification tests, and changes during valve operator assembly and installation resulted in the introduction of additional unqualified or unidentified wiring. Corrective action included replacement of all unqualified or unidentified wiring with qualified wiring. Information Notice 86-03 was issued to advise licensees of the potential problem.

Lead responsibility for the conduct of equipment qualification inspections at nuclear plants is being transferred to the Regional Offices. To date, Region III (Chicago) has been given the lead equipment qualification responsibility for nuclear plants in its jurisdiction. Regions I (Philadelphia) and II (Atlanta) are expected to assume the same task during fiscal year 1987. The VPB will continue to provide technical assistance and contractor support.



The licensee-directed Engineering Assurance Program—one of several types of design inspection and examination programs preceding licensing of a nuclear power plant—was completed during 1986 at the Millstone Unit 3 (Conn.) and Nine Mile Point Unit 2 (N.Y.) facilities. The latter is shown in the photo.



Some 120 NRC inspections in 1986 were conducted under the agency's Vendor Inspection Program. Most of these dealt with component manufacturers, test facilities and licensee equipment qualification programs. In these photos, NRC inspectors are shown checking a damaged intake cooling water pump impeller (left) at the St. Lucie (Fla.) nuclear power plant, and (right) as-built piping measurements at the Seabrook (N.H.) nuclear power plant.

VPB Information Notices

During fiscal year 1986, the VPB prepared and issued 15 Information Notices to licensees and their principal contractors. Many of the Notices addressed the generic implications of defects in various products including failed parts, lack of environmental qualification, deficient procedural information, failures resulting from corrosion, inadequate materials, and design and fabrication problems. One Information Notice emphasized the importance of the accuracy and completeness of safety-related information presented by contractors in the training of licensee employees.

Check Valve Problem

The VPB was instrumental in analyzing various aspects of check valve failures, following the November 1985 event at San Onofre Unit 1 (Cal.) which brought the problem to light. The feedwater system failure involved the undetected concurrent failure of five feedwater check valves, resulting in a condensation-induced water hammer event. Failure of the check valves was attributed to cyclical loading, itself the result of a marginal design application complicated by the non-integral disk/swing arm construction of the check valves. The VPB conducted inspections of vendors and design organizations regarding check valve design and fabrication practices, design application requirements, and system design practices; the staff witnessed tests of various check valve designs and system configurations to assure proper resolution of the problem. Further, the NRC has encouraged industry owners groups to assume the lead in resolving the check valve problems. The VPB was involved in assuring the full and complete dissemination of information regarding this event from the NRC and industry sources, so as to preclude a similar occurrence at other nuclear power plants.

Assistance in Investigations

The VPB has provided technical assistance in the investigation of defects and corrective action for safety-related components and materials in which potential wrongdoing or criminal behavior is indicated. VPB technical assistance was provided to investigative organizations including the NRC's Office of Investigation and the U.S. Department of Justice.

Change in Confirmation Practice

The NRC this year modified its practice and is no longer issuing confirming letters to principal contractors certifying acceptable implementation of quality assurance (QA) programs; the step was taken because future NRC vendor program inspections will focus on selected technical problems, rather than addressing the overall implementation of QA programs. An Information Notice (IN 86-36, May 16, 1986) was issued to affected organizations informing them of the change in practice. In the past, NRC issued confirming letters to the principal contractors to indicate that NRC inspections have confirmed satisfactory implementation of the QA programs. Licensees and applicants could, at their option, use the letters to fulfill their obligation under 10 CFR 50 Appendix B, Criterion VII, requiring them to perform initial source evaluation audits, and subsequent periodic audits, to verify QA program implementation. This change reflects a greater NRC emphasis on licensee responsibility for vendor product quality.

Foreign Involvement

The VPB contracted for a study to characterize the type and level of foreign vendor involvement in the commercial U.S. nuclear industry, so that the present vendor program may be effectively structured to deal with current needs. The results of the study indicated that there are currently 59 foreign vendors who supply materials such as pipe, fittings, fasteners, and engineered components to U.S. nuclear power facilities. The study also presented an estimate, based on utility procurement data, that foreign vendors presently supply approximately 5 percent of the safety-related material, components, and services utilized by the commercial U. S. nuclear industry.

Quarterly Reports

The VPB issued four quarterly reports (NUREG-0040), presenting the results of the inspections conducted during the report year of licensees, contractors and vendors. These reports include the vendor related Information Notices that resulted from the analysis of certain problems with generic impact. The reports are distributed to nuclear utilities, to the organizations that were subject to inspection, and to certain other outside organizations; they are available to the general public for purchase through the Government Printing Office.

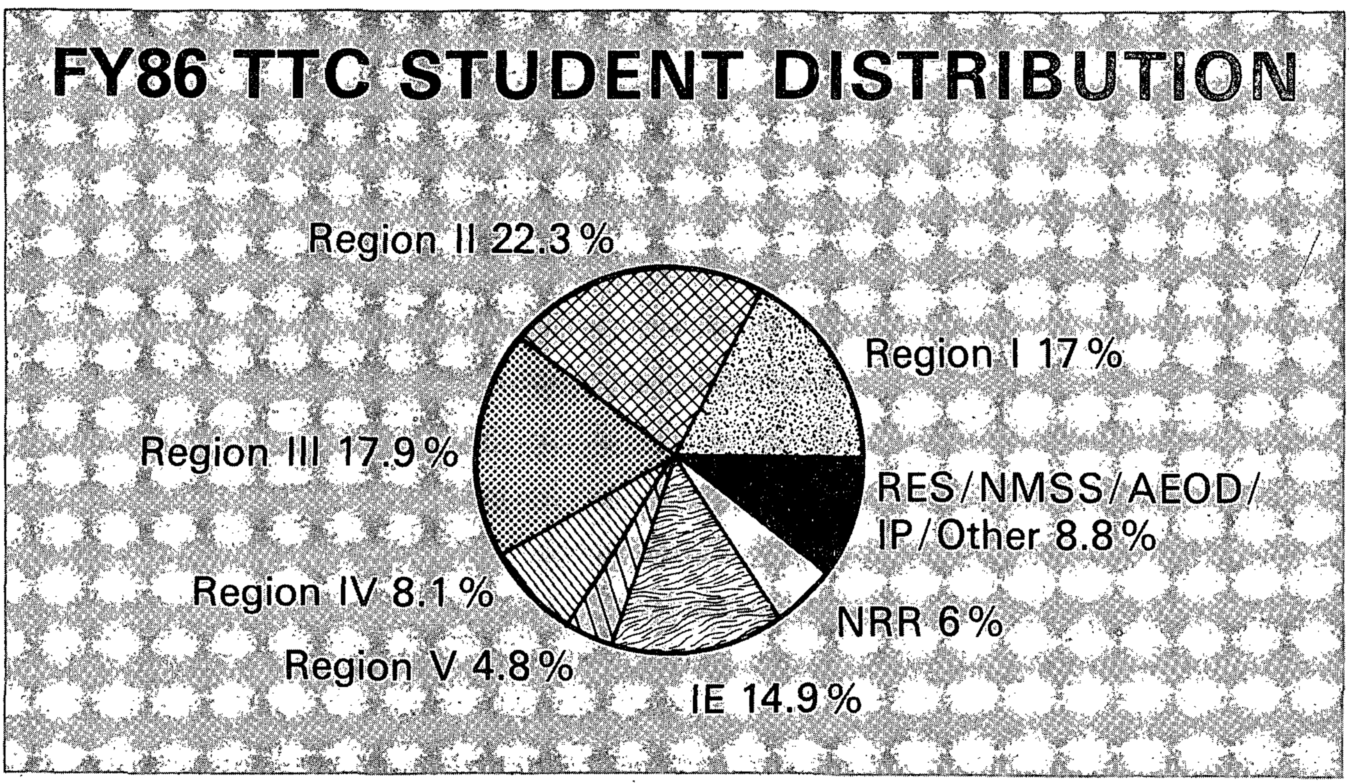
TECHNICAL TRAINING PROGRAM

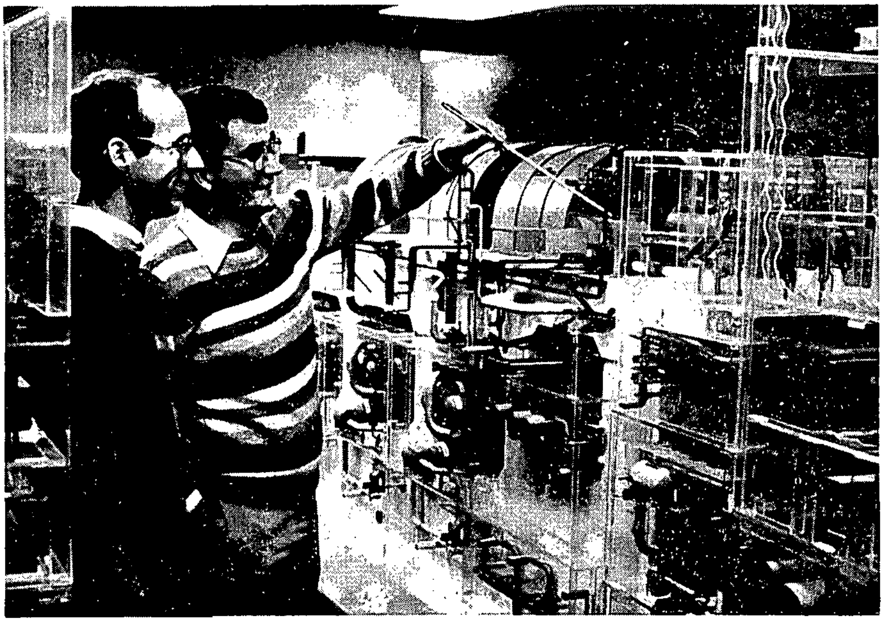
The NRC Technical Training Center (TTC), located in Chattanooga, Tenn., develops and carries out the policy and program for technical training of NRC inspectors, and also provides technical training for NRC personnel and other government agencies, as requested. The TTC provides training initially to teach and continually to sustain inspector and other NRC staff personnel at the level of knowledge of reactor technology needed to perform their assigned jobs. The training provided is in the area of BWR/PWR operation, system design, instrumentation and control, and other specialized technical training.

Although TTC courses were designed to provide specialized technical training to meet the specific job requirements of NRC inspectors, participants come from all NRC offices. TTC students include Resident Inspectors, region-based Project Inspectors, Engineering Support Inspectors, Specialized Inspectors (non-reactor), Operator License Examiners, Operations Center Duty Officers, Technical Staff from NRC Headquarters Offices, and Technical Managers. Representatives of other government agencies, NRC contractors, and NRC counterparts from foreign countries may also take training at the center, when priorities permit.

In fiscal year 1986, the TTC presented or coordinated attendance at 106 courses. A total of 1,088 students attended and 1,405 student-weeks of instruction were given. The TTC curriculum currently includes 69 different highly specialized technical training courses.

During the fiscal year, a contract was negotiated with the General Electric Company resulting in the move of the simulator (BWR/6 design) from the canceled Black Fox (Okla.) nuclear plant to the TTC facility in Chattanooga,





Students at the NRC Technical Training Center in Chattanooga, Tenn., use the scale model of the cancelled Hartsville plant to supplement their BWR systems training. The model was leased from the Tennessee Valley Authority and moved to the training center in 1985.

Tenn. The immediate access to the Black Fox simulator and the Hartsville nuclear plant engineering model (leased from the Tennessee Valley Authority) has permitted a BWR technology curriculum based on the design of these newer plant designs, with appropriate treatment of the differences among these and other BWR plant designs. This modification was completed during fiscal year 1986. The proximity of the simulator and model and availability of simulator time on day shift (never before possible) have allowed integration of both into the classroom phase of the BWR technology curriculum. It is now possible for TTC instructors to discuss a mechanical system in the classroom and have the students later go to the model and trace the system piping, locating major components which were discussed in the classroom. It is similarly possible for TTC instructors to discuss a control system in the classroom and have the students go to the simulator in small groups to observe a pre-planned simulator demonstration which illustrates the operation of the particular control system.

The TTC brought in-house the previously contracted simulator training associated with the Combustion Engineering PWR design during fiscal year 1986. The TTC staff now provides reactor systems and simulator training to the NRC staff for each of the four major U.S. reactor vendors (General Electric, Westinghouse, Babcock and Wilcox, and Combustion Engineering). Also during fiscal year 1986, the TTC staff developed and presented both Simulator Refresher Courses for Operator License Examiners and Technical Managers Courses in each of the four major U.S. reactor vendors.

A significant TTC effort during the report period was the development and presentation of a Power Plant Engineering (PPE) course. The PPE course evolved as one means of improving the level of basic power plant knowledge for new NRC employees. Intended to bridge the gap between theoretical and practical information, PPE course topics included electrical theory and equipment, diesel generators, fluid flow, heat transfer, plant chemistry, nuclear theory, piping and components, turbines, and plant instrumentation. This course is undergoing revision and will become part of the TTC curriculum.

The TTC continued efforts to meet the technical training needs of the agency in light of changing NRC roles and mission. A Training Advisory Group (TAG), consisting of senior managers from Regional and Headquarters Offices, has been established to advise the TTC on items of curriculum development and priorities for optimizing TTC training resources.

GENERIC COMMUNICATIONS

Information Notices

The NRC Office of Inspection and Enforcement issues Information Notices to licensees and construction permit holders to inform them of events or other matters that may have generic implications. Many of these issuances are based on events reported by licensees, NRC Inspectors, vendors or others, when a preliminary NRC evaluation indicates that the event may be of interest to or have generic safety implications for other licensees. A total of 112 NRC Information Notices were issued in fiscal year 1986, including six updates of previously issued information notices. (Table 4 lists all Information Notices issued in fiscal year 1986.) Information Notices provide information but do not require specific actions. They are rapid transmittals of information of which licensees should expeditiously be made aware, even though the data may not yet have been completely analyzed

Table 4. IE INFORMATION NOTICES—FY 1986

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
85-17 Sup. 1	Possible Sticking Of ASCO Solenoid Valves	10/1/85	All power reactor facilities holding an OL or CP
85-80	Timely Declaration Of An Emergency Class, Implementation Of An Emergency Plan, And Emergency Notications	10/15/85	All power reactor facilities holding an OL or CP
85-81	Problems Resulting In Erroneously High Reading With Panasonic 800 Series Thermoluminescent Dosimeters	10/17/85	All power reactor facilities holding an OL or CP; and certain materials and fuel cycle licensees
85-82	Diesel Generator Differential Protection Relay Not Seismically Qualified	10/18/85	All power reactor facilities holding an OL or CP
85-83	Potential Failures Of General Electric PK-2 Test Blocks	10/30/85	All power reactor facilities holding an OL or CP
85-84	Inadequate Inservice Testing Of Main Steam Isolation Valves	10/30/85	All power reactor facilities holding an OL or CP
85-85	Systems Interaction Event Resulting In Reactor System Safety Relief Valve Opening Following A Fire-Protection Deluge System Malfunction	10/31/85	All power reactor facilities holding an OL or CP
85-86	Lightning Strikes At Nuclear Power Generating Stations	11/5/85	All power reactor facilities holding an OL or CP
85-87	Hazards Of Inerting Atmospheres	11/18/85	All power reactor facilities holding an OL or CP
85-88	Licensee Control Of Contracted Services Providing Training	11/18/85	All power reactor facilities holding an OL or CP
85-89	Potential Loss Of Solid-State Instrumentation Following Failure of Control Room Cooling	11/19/85	All power reactor facilities holding an OL or CP
85-90	Use Of Sealing Compounds In An Operating System	11/19/85	All power reactor facilities holding an OL or CP
85-58 Sup. 1	Failure Of A General Electric Type AK-2-25 Reactor Trip Breaker	11/19/85	All B&W and CE power reactor facilities holding an OL or CP
85-91	Load Sequencers For Emergency Diesel Generators	11/27/85	All power reactor facilities holding an OL or CP
85-92	Surveys Of Wastes Before Disposal From Nuclear Reactor	12/2/85	All production and utilization facilities; power reactor and research and test reactors holding an OL or CP
85-93	Westinghouse Type DS Circuit Breakers, Potential Failure Of Electric Closing Feature Because Of Broken Spring Release Latch Lever	12/6/85	All power reactor facilities holding an OL or CP

Table 4. IE INFORMATION NOTICES—FY 1986
(continued)

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
85-94	Potential For Loss Of Minimum Flow Paths Leading To ECCS Pump Damage During A LOCA	12/13/85	All power reactor facilities holding an OL or CP
85-95	Leak Of Reactor Water To Reactor Building Caused By Scram Solenoid Valve Problem	12/23/85	All BWR facilities holding an OL or CP
85-96	Temporary Strainers Left Installed In Pump Suction Piping	12/23/85	All power reactor facilities holding an OL or CP
85-97	Jail Term For Former Contractor Employee Who Intentionally Falsified Welding Inspection Records	12/26/85	All power reactor facilities holding an OL or CP
85-98	Missing Jumpers From Westinghouse Reactor Protection System Cards For The Over-Power Delta Temperature Trip Function	12/26/85	All Westinghouse designed PWR facilities holding an OL or CP
85-99	Cracking In Boiling-Water-Reactor Mark I And Mark II Containments Caused By Failure Of The Inerting System	12/31/85	All BWR facilities holding an OL or CP
85-100	Rosemount Differential Pressure Transmitter Zero Point Shift	12/31/85	All power reactor facilities holding an OL or CP
85-101	Applicability of 10 CFR 21 To Consulting Firms Providing Training	12/31/85	All power reactor facilities holding an OL or CP
86-01	Failure Of Main Feedwater Check Valves Causes Loss Of Feedwater System Integrity And Water-Hammer Damage	1/3/86	All power reactor facilities holding an OL or CP
86-02	Failure Of Valve Operator Motor During Environmental Qualification Testing	1/3/86	All power reactor facilities holding an OL or CP
86-03	Potential Deficiencies In Environmental Qualification of Limitorque Motor Valve Operator Wiring	1/14/86	All power reactor facilities holding an OL or CP
86-04	Transient Due To Loss Of Power To Integrated Control System At A Pressurized Water Reactor Designed By Babcock & Wilcox	1/31/86	All power reactor facilities holding an OL or CP
86-05	Main Steam Safety Valve Test Failures And Ring Setting Adjustments	1/31/86	All PWR facilities holding an OL or CP
86-06	Failure Of Lifting Rig Attachment While Lifting The Upper Guide Structure At St. Lucie Unit 1	2/3/86	All power reactor facilities holding an OL or CP
86-07	Lack Of Detailed Instruction And Inadequate Observance Of Precautions During Maintenance And Testing Of Diesel Generator Woodward Governors	2/3/86	All power reactor facilities holding an OL or CP
86-08	Licensee Event Report (LER) Format Modification	2/3/86	All power reactor facilities holding an OL or CP
86-09	Failure Of Check And Stop Check Valves Subjected To Low Flow Conditions	2/3/86	All power reactor facilities holding an OL or CP

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
86-10	Safety Parameter Display Systems Malfunctions	2/13/86	All power reactor facilities holding an OL or CP
84-69 Sup. 1	Operation Of Emergency Diesel Generators	2/24/86	All power reactor facilities holding an OL or CP
86-11	Inadequate Service Water Protection Against Core Melt Frequency	2/25/86	All power reactor facilities holding an OL or CP
86-12	Target Rock Two-Stage SRV Setpoint Drift	2/25/86	All power reactor facilities holding an OL or CP
86-13	Standby Liquid Control System Squib Valves Failure To Fire	2/21/86	All BWR facilities holding an OL or CP
86-14	PWR Auxiliary Feedwater Pump Turbine Control Problems	3/10/86	All power reactor facilities holding an OL or CP
86-15	Loss Of Off-site Power Caused By Problems In Fiber Optics Systems	3/10/86	All power reactor facilities holding an OL or CP
86-16	Failures To Identify Containment Leakage Due To Inadequate Local Testing Of BWR Vacuum Relief System Valves	3/11/86	All power reactor facilities holding an OL or CP
86-17	Update Of Failure Of Automatic Sprinkler System Valves To Operate	3/24/86	All power reactor facilities holding an OL or CP
86-18	NRC On-Scene Response During A Major Emergency	3/26/86	All power reactor facilities holding an OL or CP
86-19	Reactor Coolant Pump Shaft Failure At Crystal River	3/21/86	All power reactor facilities holding an OL or CP
86-20	Low-Level Radioactive Waste Scaling Factors, 10 CFR Part 61	3/28/86	All power reactor facilities holding an OL or CP
86-21	Recognition Of American Society of Mechanical Engineers Accreditation Program For N Stamp Holders	3/31/86	All power reactor facilities holding an OL or CP and all recipients of NUREG-0040
86-22	Under-response Of Radiation Survey Instrument To High Radiation Fields	3/31/86	All power reactor facilities holding research and test reactors
86-23	Excessive Skin Exposures Due To Contamination With Hot Particles	4/9/86	All power reactor facilities holding an OL or CP
86-24	Respirator Users Notice: Increased Inspection Frequency For Certain Self-Contained Breathing Apparatus Air Cylinders	4/11/86	All power reactor facilities holding an OL or CP; research and test reactor facilities; fuel cycle licensees and Priority 1 material licensees
86-25	Traceability And Material Control Of Material And Equipment, Particularly	4/11/86	All power reactor facilities holding an OL or CP
86-26	Potential Problems In Generators Manufactured By Electrical Products Inc.	4/17/86	All power reactor facilities holding an OL or CP

Table 4. IE INFORMATION NOTICES—FY 1986
(continued)

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
86-27	Access Control At Nuclear Facilities	4/21/86	All power reactor facilities holding and OL or CP; research and nonpower reactor facilities; and fuel fabrication and processing facilities
86-28	Telephone Numbers To The NRC Operations Center And Regional Offices	4/24/86	All NRC licensees except power facilities and fuel facilities with an installed ENS
86-29	Effects Of Changing Valve Motor-Operator Switch Settings	4/25/86	All power reactor facilities holding an OL or CP
86-30	Design Limitations Of Gaseous Effluent Monitoring Systems	4/29/86	All power reactor facilities holding an OL or CP
86-31	Unauthorized Transfer And Loss Of Control Of Industrial Nuclear Gauges	5/5/86	All NRC licensees authorized to possess and use industrial nuclear gauges
86-32	Request For Collection Of Licensee Radioactivity Measurements Attributed To The Chernobyl Nuclear Plant Accident	5/2/86	All power reactor facilities holding an OL or CP
86-33	Information For Licensee Regarding The Chernobyl Nuclear Plant Accident	5/6/86	Fuel cycle licensees and Priority 1 material licensees
86-34	Improper Assembly, Material Selection, And Test Of Valves And Their Actuators	5/13/86	All power reactor facilities holding an OL or CP
86-44	Failure To Follow Procedures When Working In High Radiation Areas	6/10/86	All power reactor facilities holding an OL or CP; and research and test reactors
86-45	Potential Falsification Of Test Reports On Flanges Manufactured By Golden Gate Forge and Flange, Inc.	6/10/86	All power reactor facilities holding an OL or CP; and research and test reactors
86-46	Improper Cleaning And Decontamination Of Respiratory Protection Equipment	6/12/86	All power reactor facilities holding an OL or CP; and fuel fabrication facilities
86-47	Erratic Behavior Of Static "O" Ring Differential Pressure Switches	6/10/86	All BWR and PWR facilities holding an OL or CP
86-48	Inadequate Testing Of Boron Solution Concentration In The Standby Liquid Control System	6/13/86	All BWR facilities holding an OL or CP
86-49	Age/Environment Induced Electrical Cable Failures	6/16/86	All power reactor facilities holding an OL or CP
86-50	Inadequate Testing To Detect Failures Of Safety-Related Pneumatic Components Or Systems	6/18/86	All power reactor facilities holding an OL or CP
86-51	Excessive Pneumatic Leakage In The Automatic Depressurization Systems	6/18/86	All BWR facilities holding an OL or CP

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
86-52	Conductor Insulation Degradation On Foxboro Model E Controllers	6/26/86	All power reactor facilities holding an OL or CP
86-53	Improper Installation Of Heat Shrinkable Tubing	6/26/86	All power reactor facilities holding an OL or CP
86-54	Criminal Prosecution Of A Former Radiation Safety Officer Who Willfully Directed An Unqualified Individual To Perform Radiography	6/27/86	All holders of NRC license authorizing the possession of byproduct, source, or special nuclear material
86-55	Delayed Access To Safety-Related Areas And Equipment During Plant Emergencies	7/10/86	All power reactor facilities holding an OL or CP
86-56	Reliability Of Main Steam Safety Valves	7/10/86	All PWR facilities holding an OL or CP
86-57	Operating Problems With Solenoid Operated Valves At Nuclear Power Plants	7/11/86	All power reactor facilities holding an OL or CP
86-58	Dropped Fuel Assembly	7/11/86	All power reactor facilities holding an OL or CP
86-59	Increased Monitoring Of Certain Patients With Implanted Coratomic, Inc. Model C-100 and C-101 Nuclear-Powered Cardiac Pacemakers	7/14/86	All NRC licensees authorized to use nuclear-powered cardiac pacemakers
86-31 Sup. 1	Unauthorized Transfer And Loss Of Control Of Industrial Nuclear Gauges	7/14/86	All NRC general licensees that possess and use industrial nuclear gauges
86-60	Unanalyzed Post-LOCA Release Paths	7/28/86	All power reactor facilities holding an OL or CP
86-61	Failure Of Auxiliary Feedwater Manual Isolation Valve	7/28/86	All power reactor facilities holding an OL or CP
86-62	Potential Problems In Westinghouse Molded Case Circuit Breakers Equipped With A Shunt Trip	7/31/86	All power reactor facilities holding an OL or CP
86-13 Sup. 1	Standby Liquid Control Squib Valves Failure To Fire	8/5/86	All BWR facilities holding an OL or CP
86-63	Loss Of Safety Injection Capability	8/6/86	All PWR facilities holding an OL or CP
86-64	Deficiencies In Upgrade Programs For Plant Emergency Operating Procedures	8/14/86	All power reactor facilities holding an OL or CP
86-65	Malfunctions Of ITT Barton Model 580 Series Switches During Requalification Testing	8/14/86	All power reactor facilities holding an OL or CP
86-66	Potential For Failure Of Replacement AC Coils Supplied By The Westinghouse Electric Corporation For Use In Class 1E Motor Starters and Contractors	8/15/86	All power reactor facilities holding an OL or CP
86-67	Portable Moisture/Density Gauges: Recent Incidents And Common Violations Of Requirements For Use, Transportation, And Storage	8/15/86	All NRC licensees authorized to possess, use, transport, and store sealed sources in portable gauges

Table 4. IE INFORMATION NOTICES—FY 1986
(continued)

Information Notice No.	*Subject*	*Date of Issuance*	*Issued to*
86-68	Stuck Control Rod		All BWR facilities holding an OL or CP
86-69	Spurious System Isolations Caused By The Panalarm Model 86 Thermocouple Monitor	8/18/86	All GE BWR facilities holding an OL or CP
86-70	Potential Failure Of All Emergency Diesel Generators	8/18/86	All power reactor facilities holding an OL or CP
86-71	Recent Identified Problems With Limitorque Motor Operators	8/19/86	All power reactor facilities holding an OL or CP
86-72	Failure 17-7 PH Stainless Steel Springs In Valcor Valves Due To Hydrogen Embrittlement	8/19/86	All power reactor facilities holding an OL or CP
86-73	Recent Emergency Diesel Generator Problems	8/20/86	All power reactor facilities holding an OL or CP
86-74	Reduction Of Reactor Coolant Inventory Because Of Misalignment Of RHR Valves	8/20/86	All BWR facilities holding an OL or CP
86-75	Incorrect Maintenance Procedure On Traversing Incore Probe Lines	8/21/86	All power reactor facilities holding an OL or CP
86-76	Problems Noted In Control Room Emergency Ventilation Systems	8/28/86	All power reactor facilities holding an OL or CP
86-77	Computer Program Error Report Handling	8/28/86	All power reactor facilities holding an OL or CP and nuclear fuel manufacturing facilities
86-78	Scram Solenoid Pilot Valve (SSPV) Rebuild Kit Problems	9/2/86	All BWR facilities holding an OL or CP
86-79	Degradation Or Loss Of Charging Systems At PWR Nuclear Power Plants Using Swing-Pump Designs	9/2/86	All power reactor facilities holding an OL or CP
86-80	Unit Startup With Degraded High Pressure Safety Injection System	9/12/86	All power reactor facilities holding an OL or CP
86-81	Broken Inner-External Closure Springs On Atwood & Morrill Main Steam Isolation Valves	9/15/86	All power reactor facilities holding an OL or CP
86-82	Failures Of Scram Discharge Volume Vent And Drain Valves	9/16/86	All power reactor facilities holding an OL or CP
86-83	Underground Pathways Into Protected Areas, Vital Areas, Material Access Areas, And Controlled Access Areas	9/19/86	All power reactor facilities holding an OL or CP
86-84	Rupture Of A Nominal 40-Millicurie Iodine-125 Brachytherapy Seed Causing Significant Spread Of Radioactive Contamination	9/30/86	All NRC medical institution licensees

OL = Operating License
CP = Construction Permit

by the NRC. Licensees receiving an Information Notice are expected to review the information for applicability to their facilities, and consider taking action, if indicated, to prevent a similar problem from occurring at their facilities. The NRC then may follow up through inspections to ensure that licensees have an adequate system for reviewing and appropriately acting on Information Notices. Further, regional inspections are made on selected safety issues, including those addressed by some Information Notices.

Bulletins

The Office also issues bulletins, which both provide information about one or more similar events of safety significance and also require that licensees take specific actions. The licensee reports back on actions taken, or to be taken, and provides information the NRC may need to assess the need for further action. Prompt response by licensees is required; failure to respond will normally result in NRC enforcement action. Before issuing a bulletin, the NRC may seek comments from the nuclear industry. This technique has proven effective in generating faster and more informed responses from affected licensees. However, the nature of the problem and a need for timely action may sometimes limit prior consultation. Bulletins generally require one-time action and are not intended as substitutes for formally issued regulations or for imposed license amendments.

In fiscal year 1986, NRC issued five bulletins. The subject of each of the bulletins and the required licensee actions are summarized below.

(1) IE Bulletin 85-01, issued on October 29, 1985, informed nuclear power reactor licensees and construction permit holders of a potentially serious safety problem that had occurred at several operating facilities, involving the inoperability of auxiliary feedwater (AFW) pumps as a result of steam binding. Numerous events had been reported since 1981 where hot water leaked into AFW systems and flashed to steam, disabling the AFW pumps. In July, 1984, the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) issued a case study report entitled "Steam Binding of Auxiliary Feedwater Pumps." This study identified 22 events since 1981 (13 of these occurred in 1983). The case study indicated that the unavailability of the AFW system as a result of steam binding contributes significantly to the risk of core melt in PWRs. Based on operating experience, it appears that backleakage into AFW could occur in any pressurized water reactor. In December 1984, following the occurrence of several more events involving backflow of hot water into AFW systems, the NRC's Office of Nuclear Reactor Regulation (NRR) determined that steam binding of AFW pumps was a "generic issue" and assigned it a high priority (Generic Issue 93). To determine the extent of the safety issue and the need for short-term corrective actions, the NRC's Regional Offices conducted a survey in April and May of 1985. Although corrective actions had been taken at all units, many licensees had not incorporated these measures into procedures to detect or correct steam binding. Licensees for certain PWRs with operating licenses and all PWR applicants with construction permits were asked to: develop procedures for monitoring fluid conditions within the AFW system on a regular basis during times when the system is required to be operable; develop procedures for recognizing steam binding and for restoring the AFW system to operable status, should steam binding occur; maintain procedural controls until completion of the hardware modification to substantially reduce the likelihood of steam binding, or until superseded by action implemented as a result of resolution of Generic Issue 93.

(2) IE Bulletin 85-02, issued November 5, 1985, informed nuclear power reactor licensees and applicants of recent reactor trip breaker (RTB) reliability problems. On October 29, 1985, the D.C. Cook Unit 2 (Mich.) nuclear power plant tripped (shut down) from approximately 80 percent of full power. The reactor trip was triggered by a spurious indication of a "low flow" condition in loop two. Investigation into the event by the licensee and Westinghouse attributed the failure of the Train "A" reactor trip breaker to a faulty undervoltage trip attachment (UVTA) which did not provide enough lifting force to the breaker trip bar. The breaker was identified as the Westinghouse Type DB-50. Subsequent to the October 29, 1985 event, all RTBs at the D.C. Cook Station were tested. The UVTA trip force margin testing of the Unit 2 "B" breaker identified degradation of the margin below the 20-ounce specified margin. The seriousness of the degradation of the UVTA attachment of the "B" breaker, in conjunction with the failure of the "A" breaker, renewed concern for common mode failures that could potentially lead to an anticipated transient without scram (ATWS) event. As a result of previous events involving RTB/UVTAs, NRC Generic Letter 83-28 had been issued directing corrective actions by the licensees. These actions had two major aspects related to the RTB/UVTAs: the installation of a plant modification that provides for the automatic activation of the shunt trip coil of the RTB for any automatic reactor trip signal, and reliability improvements in the RTB/UVTA based upon UVTA testing by the vendor. The failures at D.C. Cook involved RTBs that had been refurbished by the vendor only four months previously, including the installation of new UTVAs. The bulletin required that licensees of operating Westinghouse PWR facilities who had not yet implemented the actuation of RTB shunt trip coil on all automatic reactor-trip signals

Among the top-priority issues addressed by the NRC staff during 1986 were the major technical and management problems arising at nuclear projects of the Tennessee Valley Authority (TVA) in Tennessee and Alabama. NRC Headquarters and Region II (Atlanta) personnel, including a Senior Management Team of office-level directors, provided concentrated oversight as the TVA worked to recover its nuclear program. In the photo, NRC Executive Director for Operations, Victor Stello, Jr. (right), and Ken Jenison, Region II Senior Resident Inspector, are shown during a January 1986 inspection of corrective actions taken at the Sequoyah nuclear power plant near Chattanooga, Tenn.

perform a test of the UVTA of each RTB in service to determine that adequate force margin existed; modify the monthly test procedure for the Reactor Protection System and add to it the conducting the UVTA force margin test, prior to any lubrication or adjustment of the UVTA, for each RTB in service; and finally to provide written instructions to the plant operating staff requiring each licensed operator to review the bulletin at the start of his or her next duty shift, to declare a reactor trip breaker inoperable if the UVTA either did not successfully pass the force margin test or otherwise might not be capable of performing its intended safety function, to permit plant operation with either RTB inoperable only as provided for in the plant Technical Specifications, and to notify the NRC within four hours of declaring any RTB inoperable.

(3) IE Bulletin 85-03, issued November 15, 1985, informed nuclear power reactor licensees and construction permit holders of several events during which motor-operated valves failed to respond, in a common mode, because of improper settings. Two events prompted issuance of the bulletin. On June 9, 1985, the Davis-Besse (Ohio) plant underwent a complete loss of main and auxiliary feedwater. Normally open, Limitorque motor-operated auxiliary feedwater (AFW) gate valves which had been inadvertently closed failed to reopen on either an automatic or manual signal from the main control room. Although other failures also occurred in the AFW system during this event, the failure of these two valves was itself enough to prevent AFW from reaching either steam generator. In recovery from this event, the valves were opened with the handwheels. The licensee's analysis of the event led to the conclusion that the setting for the torque switch bypass limit switch in each valve's control circuit had not been set to remain closed long enough to provide the necessary bypass function on the valve's opening with differential pressure conditions across the valve. On May 2, 1985, at the Sequoyah Unit 2 (Tenn.) nuclear plant, an event involving partial loss of main feedwater occurred while the facility was in startup and returning to power after a reactor trip. Feedwater was being supplied through the main feedwater (MFW) system isolation valve bypass lines. Operators attempted to open the MFW system isolation valves to supply water to the steam generators; however, two of the four MFW isolation valves would not open. The startup was discontinued and the unit was returned to hot shutdown. Upon investigation, the licensee discovered that both valve stems had sheared from their discs. The motor-operators on the valves have limit switches to control valve motion in the open direction. The limit switch setpoint was not correctly established and the disc impacted the backseat during opening. The failure mechanism of these valves was identified by the licensee to be impact loading of the stem on the opening strike, as a result of the disc impacting the backseat, combined with a stress failure of the remaining portion of the stem on the opening stroke. The bulletin also addressed previous instances where common mode failures to respond on demand of valves were identified. The bulletin required all holders of operating licenses or construction permits to develop and implement a program to ensure that for motor-operated valves in the high pressure coolant injection/core spray and emergency feedwater systems (RCIC for BWRs) that are required to be tested for operational readiness in accordance with 10 CFR 50.55a(g), valve operator switches are selected, set and maintained properly.

(4) IE Bulletin 86-01, issued May 23, 1986, informed all licensees for General Electric (GE) boiling water reactor facilities, with either an operating license or a construction permit, of a problem with the minimum flow logic, by which a single failure could disable all residual heat removal (RHR) pumps. During a review of an IE Information Notice, operators of the Pilgrim (Mass.) nuclear power plant discovered that a single failure under certain accident sequences could result in all RHR minimum flow bypass valves being signaled to close, while all other pump discharge valves are also closed. That situation could mean there would be no flow through the RHR pumps, and that condition could lead to the pumps running "dead headed," with the potential for pump damage in just a few minutes. If this single failure occurred in conjunction with an automatic start of the RHR system,

Table 5. IE Bulletins Closed Out and Status Reports Issued in FY 1986

Bulletin No.	*Subject*
79-07	Seismic Stress Analysis of Safety-Related Piping
80-01	Operability of Automatic Depressurization System (ADS) Valve Pneumatic Supply
80-04	Analysis of PWR Steam Line Break with Continued Feedwater Addition
80-16	Potential Misapplication of Rosemount Pressure Transmitters
80-20	Failures of Westinghouse Type W-2 Spring Return to Neutral Control Switches
Significant Status Reports Issued	
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts
79-14	Seismic Analysis for As-Built Safety-Related Piping Systems

RHR pump damage could occur if the failure went unrecognized by the operator. This event could disable RHR functions—including Low Pressure Coolant Injection (LPCI), head spray, drywell spray, shutdown cooling, torus spray and suppression pool cooling. As a result of the loss of suppression pool cooling over a long period of time, core spray pumps could ultimately lose net positive suction head and also be unavailable. Although the NRC staff has judged that the overall probability of a serious core damage accident resulting from this problem is low, such a single failure vulnerability is not in compliance with the regulatory requirement for independence in emergency core cooling systems, and it could compromise several important systems. The bulletin required GE BWR licensees: to determine promptly whether their facilities have this single failure vulnerability; to immediately instruct all operating shifts of the problem and measures to recognize and mitigate the problem; to identify the short-term modifications to plant operating procedures or hardware to be implemented in assuring safe plant operations; and to inform the NRC of the schedule for long-term resolution of problems that were identified as a result of this bulletin.

(5) IE Bulletin 86-02, issued July 18, 1986, informed all nuclear power reactor licensees and construction permit holders of the erratic behavior of Static "O" Ring (SOR) differential pressure switches during an event at LaSalle Unit 2 (Ill.) and during subsequent testing. On June 1, 1986, LaSalle Unit 2 underwent a feedwater transient that resulted in a low water level in the reactor vessel. One of four low level trip channels actuated, resulting in a "half scram." The operator recovered adequate water level and power operation was continued. However, subsequent reviews by the licensee raised concerns because the level apparently had gone below the scram setpoint, and a malfunction of the reactor scram system may have occurred. Based on this concern, the licensee declared an alert, shut the plant down, notified the NRC, and subsequently informed the manufacturer of the differential pressure switches of possible switch malfunctions. After recalibrating the level switches on June 1, the licensee tested their performance by lowering water level in the reactor and reading the levels indicated on level transmitters when each of the four level switches tripped. The test results were erratic. As of June 9, 1986, the licensee had tested differential pressure switches in the residual heat removal systems and the high pressure core spray system. One of the switches actuated within the range permitted by technical specifications; the others did not. On the basis of these test results, the licensee declared all emergency core cooling systems for Units 1 and 2 to be inoperable. The SOR Series 103 differential pressure switches involved were installed in La Salle Units 1 and 2 as part of an environmental qualification modification which was performed after initial operation of the unit. SOR Series 102 differential pressure switches have been identified as having important similarities to Series 103 switches. Both models became the subject of this bulletin. The bulletin required all licensees to report on the extent to which SOR Model 102 or 103 differential pressure switches were installed (or planned) as electrical equipment important to safety, as defined in 10 CFR 50.49(b). Licensees that had either model installed in systems that are subject to limiting conditions for operation in technical specifications were required to take positive action to assure that licensed reactor operators on duty were prepared for potential malfunctions of SOR switches, to conduct a special test of each SOR switch to determine if the switch and system functioned properly or if short term cor-

rective actions were necessary, to report failures in accordance with 10 CFR 50.72 and 10 CFR 50.73, to develop and implement an interim performance monitoring program to provide continuing assurance that the performance of the switches and plant systems remained acceptably reliable until long term corrective actions could be fully implemented, to describe the margin and basis for switch actuation and the long term corrective actions to be taken, to define the consequences of potential common mode failures, and to provide an analysis to demonstrate that the system involved will meet regulatory requirements and function reliably.

Bulletin Closeout. Since 1983, IE has formally closed out a number of bulletins issued in 1978 or later. The objectives of this procedure are:

(1) To determine whether further generic actions are needed to fully resolve the issue (examples of such actions are the revisions to the inspection program, modification of licensing basis or technical specifications, or development of new regulatory guides).

(2) To determine whether plant-specific regional action is needed.

Since the program was initiated, a total of 26 bulletins have been closed out and major status reports issued on two others. During fiscal year 1986, five bulletins were closed out and status reports were issued on two others (see Table 5). Of the 26 bulletins closed out, six have resulted in recommendations for generic actions and five others in some general recommendations and conclusions. Almost all of the closeout efforts have resulted in plant-specific regional follow-up at some plants.

PERFORMANCE INDICATORS

In March 1986, the Commission directed the staff to propose a "performance indicators" program, defining a set or sets of data that can be correlated with individual plant safety performance. The Commission intends for the performance indicators to be a contributor to the NRC management decision-making process in the determination of appropriate regulatory actions in response to poor or declining performance, as well as to good or improving performance.

In May 1986, the staff produced a Commission paper discussing the status of current and potential uses of performance indicators, and the work of the inter-office task group that had been established to review previous efforts in this area, to select performance indicators and conduct a trial program, and to provide a final recommendation to the Commission by the end of September 1986. The task group was chaired by IE, with representatives from the NRC Offices of Nuclear Reactor Regulation (NRR), Research (RES), and the Analysis and Evaluation of Operational Data (AEOD), as well as the NRC Regional Offices. The task group identified key elements of the performance-indicator development process and developed a set of assumptions on which to base their work. The task group also developed a set of ideal attributes for the initial selection of indicators to be considered during the trial program. Seventeen indicators and 50 plants at 30 sites were selected for the trial program.

Although the principal objective of the task group was to develop a "minimum" set of performance indicators, in effect, an "optimum" set was engendered. Eight indicators were selected which, in the group's overall judgment, represented the optimum set of indicators for implementation at this time. In addition, where the indicators were similar to indicators already being used by the industry's Institute of Nuclear Power Operations (INPO), definitions identical to INPO's were adopted to avoid confusion. The set of eight performance indicators includes the following: automatic scrams while critical; safety system actuations; significant events; safety system failures; forced outage rate; maintenance backlog; enforcement action index; and equipment forced outage-per-1000-critical-hours. Each of the eight performance indicators supports at least one of the elements of the plant safety logic model. Some of the indicators, such as "significant events," support more than one element of the logic model. Considering certain negative industry comments, the task group determined that the maintenance backlog indicator should not be included in the final set, pending the collection of additional data to determine the best means of monitoring maintenance effectiveness. To obtain additional insight into the plant/management behavior that resulted in changes in certain performance indicators, the task group recommended the further collection, development and monitoring of information in several areas.

Following up on the trial program encompassing 50 plants, the NRC Executive Director for Operations directed the staff to expand the scope of the program to include all operating reactors, in preparation for a senior management discussion of plant performance to be held in the fall of 1986. The staff was also directed to proceed with implementation the performance indicators program. That program has several salient implications. Data collection for performance indicators will better integrate already established data collection efforts of various offices. A computer system developed for use in the trial program will be used for data collection and analysis in the short term, while in the long term, the Corporate Data Network (CDN) will be used, enabling the performance indicator program to become an on-line system accessible to the entire agency, while improving the reliability of the data. Performance indicator data will be presented on a quarterly basis in a report including the latest and four-quarter profile of average values for each of the indicators, as well as the SALP ratings for all plants, and plant-specific trending information.

PERFORMANCE INDICATORS

- Automatic scrams while critical
- Safety system actuations
- Significant events
- Safety system failures
- Forced outage rate
- Maintenance backlog
- Enforcement action index
- Equipment forced outage per 1000 critical hours

The quarterly report will also contain a summary of the evaluations performed by the NRC staff. These evaluations will examine both the trends and values of indicators; they expected to identify plants that may require a closer look to see if their performance is in fact poor or declining, or for determining the underlying reasons for decline and appropriate NRC response. Additional evaluation will then be performed by both Regional and Headquarters Offices to supplement the report. These evaluations will be more comprehensive and will include consideration of previous regulatory or licensee actions, the results of recent inspections or evaluations, and the judgments of knowledgeable staff. The plants selected by senior management will then be discussed at the next senior management meeting with the EDO. Senior management will then determine the appropriate regulatory action in response to poor or declining performance.

INCIDENT RESPONSE

Events Analysis

The Nuclear Regulatory Commission maintains a 24-hour-a-day, 365-day-a year Operations Center in Bethesda, Md. The Operations Center is the NRC's center for direct communication, through dedicated telephone lines, with licensed nuclear power plants and certain fuel cycle facilities, in receiving reports of and in dealing with significant events at these facilities.

The staff at the Operations Center evaluates telephone notifications and, depending on the safety significance of the event, notifies appropriate NRC Headquarters personnel and other Federal agencies. In all cases, the NRC Regional Office in the region from which the facility is reporting the event is notified. Response to an event may vary from simply recording the circumstances of the event for later evaluation to immediately activating response organizations within Headquarters and the affected NRC Region. Upon activation, these response organizations monitor the event to ensure that appropriate actions are being taken to protect the health and safety of the public. The NRC recognizes that the agency's role is secondary to that of the licensee and off-site organizations, whose immediate response is defined ahead of time in their emergency planning.

Each event reported to the Operations Center by a licensee or Regional Office is evaluated to determine whether there are any generic implications for other facilities. Event reports are screened for this purpose early during the first working day after receipt. Followup of plant specific events is accomplished by the appropriate Region. Where an event indicates significant system interaction and raises questions as to plant safety, an augmented inspection team or an incident investigation team may be formed. Events that may be significant from a generic standpoint receive additional in-depth evaluation and, if appropriate, the NRC issues an Information Notice or bulletin to potentially affected licensees and construction permit holders.

Operations Center

During fiscal year 1986, a number of exercises involving various accident scenarios and engaging the Operations Center were conducted, in order to confirm and maintain the capabilities of the agency response personnel. In addition, response personnel participated in on-site full scale licensee emergency preparedness exercises with each NRC Region, as part of their annual assessment of the Regions' implementation of their emergency preparedness and incident response programs. These exercises were also supported through the Operations Center. Throughout the year, tours of the Operations Center were frequently provided for representatives of other NRC offices, industry, state and local governments, and foreign countries. The tours included detailed descriptions of the NRC response role and typical activities within the Center during an exercise or event.

During the report year, the Operations Center was involved in several real events which, while not requiring complete activation, necessitated the use of the Center's capabilities. The Center was staffed to monitor the event at the Kerr-McGee facility near Gore, Okla., that involved

Periodic emergency response exercises are carried out at every operating nuclear power plant, the NRC, through its Regional Offices, participates in and oversees the conduct of such drills. The photo shows J. Nelson Grace, NRC Region II (Atlanta) Administrator (right), in the Emergency Operations Facility at the South Carolina Electric Company's Virgil Summer nuclear power plant (S.C.). With Dr. Grace are Kenneth Clark, Region II Public Affairs Officer (left), and John A. Olshinski, Deputy Administrator.

a ruptured uranium hexafluoride cylinder. Additional staffing was also provided to follow the fire that occurred at the Perry (Ohio) plant in an offgas ventilation charcoal bed. Tracking of hurricanes "Juan" and "Charley" also resulted in augmented staffing of the Center. Following the event at Chernobyl, the Operations Center was used by the NRC Chernobyl Incident Tracking Team to coordinate the NRC's support to EPA (see Chapter 2). A major activity in this context was the receipt and recording of radiological measurements from all over the world. As the data were received, they were logged in and plotted by fractions of EPA Protective Action Guidelines, day-by-day. The results permitted tracking of the radiological consequences of the Soviet event in neighboring and other nearby countries, as well as in the United States. In August 1986, the Operations Center served as the agency clearinghouse for information flow from the International Atomic Energy Agency meeting on the Chernobyl event that was held in Vienna, Austria.

Emergency Response Data System

Development work on the Emergency Response Data System (ERDS), for use during emergencies at commercial nuclear power plants, continued during 1986. The ERDS concept provides for licensee-activated automatic transmission of pre-selected plant data from the licensee to a computer at the NRC Operations Center. The design phase of ERDS development included surveys of existing electronic data systems at operating and nearly completed nuclear power plants, as well as a determination of hardware and software requirements at licensee facilities. The survey was accomplished through visits to each licensee.

Shoreham Emergency Preparedness Exercise

The Federal Emergency Management Agency (FEMA) conducted a graded exercise at the Shoreham (N.Y.) nuclear power plant in February 1986. In addition to the normal NRC Region task of observing the licensee's on-site portion of the exercise, the NRC gave support to FEMA in conducting the exercise. The additional support was for 11 persons simulating representatives of State and local government and eight off-site controllers. The logs of these individuals were provided to FEMA shortly after the conclusion of the exercise to be used in preparing the FEMA evaluation report.

Regional Response Capability

The extent of regional office response to an incident is based on a pre-defined classification of events and associated NRC response modes. For a significant event, a Regional base team and a Regional site team are assembled. The base team monitors licensee performance and represents the NRC role, until a decision is made to dispatch a team to the site. If the decision is made, the site team goes to the site and is responsible for coordinating the NRC's on-site incident response activities. Each Region has its own supplement to the NRC Incident Response Plan providing specific implementation details. During the year, Headquarters and the Regions worked together to develop standardized portions of the Regional Supplements to upgrade the agency-wide response capability.

Regional response capabilities are assessed to ensure an adequate agency-wide response capability. The major evaluation in 1986 was of the Regions' performance in exercises, with emphasis on assessing response management and information flow. The assessment was accomplished through on-site observation of regional interaction during one licensee emergency preparedness exercise at each Region.

Emergency Response Training

Development of a standard response training program was continued during the year, with the assistance of the Regional Emergency Response Coordinators.

The NRC provided instruction to utility, State and local personnel on protective action decision-making, reactor systems, and the NRC response role at Federal Radiological Emergency Response Plan courses, held at FEMA's Emergency Management Institute.

Federal Response Capability

During the first quarter of fiscal year 1986, the NRC participated in the planning for and conduct of FEMA's Relocation Tabletop Exercise, held at the Emergency Management Institute in Emmitsburg, Md. Other participants included personnel from NRC Region I and the Beaver Valley (Pa.) nuclear power plant. The purpose of this exercise was to provide greater insight for all the participants into questions of relocation, recovery and re-entry, following a peacetime radiological accident.

Planning was begun during the report period for the second Federal Field Exercise of the Federal Radiological Emergency Response Plan. The exercise is scheduled for June 1987 at the Zion (Ill.) facility. Planning efforts have included site and time selection, scenario development and extensive coordination with the Commonwealth Edison Company (licensee for Zion), the States of Wisconsin and Illinois, and the other Federal agencies that will participate.

EMERGENCY PREPAREDNESS

Emergency Preparedness Task Group

During the report year, a task group composed of Headquarters and Regional personnel was formed to evaluate the emergency preparedness program for nuclear power reactors and make recommendations for program changes. The major activities of the task group were (1) to assess the degree of licensee implementation of the program, (2) to evaluate the relative safety benefit of each activity performed by Headquarters and the Regions, (3) to identify redundant activities and areas which are not covered, and (4) to consider changes in program emphasis and make recommendations thereon. The task group found that the post-TMI implementation of emergency preparedness provided substantial increases in quality and valuable accumulated experience in emergency preparedness. A change in emphasis from the review of emergency plans prior to plant licensing to the inspection of the routine emergency preparedness programs and observation of annual exercises for operating reactors was noted. A need for close integration of the licensing and inspection functions was recognized, and a need for more uniformity of Regional licensing activities was cited. The task group concluded that continued technical, administrative and managerial resources were needed, ranging from Regional guidance, support and assessment to coordination with FEMA to addressing generic emergency preparedness issues. These needs arose from the rapid implementation of post-TMI requirements, the evaluation of emergency preparedness experience, and the emergence of new issues from State and local governments, the courts and the public.

Off-site Emergency Medical Service

On September 17, 1986 the Commission published a policy statement in the *Federal Register* (51 FR 32904) entitled, "Emergency Planning—Medical Services." The Commission stated that its regulation required pre-accident arrangements for medical services for individuals who might be severely exposed to dangerous levels of off-site radiation following an accident at a nuclear power plant. In addition to the previously required list of medical facilities, such arrangements would include: (1) identification of the capacities, special capabilities or other unique characteristics of the listed medical facilities; (2) a "good faith" reasonable effort by licensees or local or State governments to facilitate or obtain written agreements with the listed medical facilities and transportation services; (3) provision for making available necessary training for emergency response personnel to identify, transport, and provide emergency first aid to severely exposed individuals; and (4) a good faith reasonable effort for licensees or State or local governments to see that appropriate drills and exercises are conducted which include simulations of severely-exposed individuals.

Support to Licensing Activities

During the report period, IE staff continued to evaluate the adequacy of applicant on-site plans to be included in the Safety Evaluation Report, and supplements thereto, for each plant in a near-term licensing status (designated NTOLs). The staff also took part in licensing hearings before Atomic Safety and Licensing Board panels and served on inspection teams appraising applicants' implementation of emergency preparedness programs and their full-participation exercises. NTOLs appraised during fiscal year 1986 included the Clinton (Ill.), Seabrook (N.H.), Vogtle (Ga.), Nine Mile Point Unit 2 (N.Y.), and Braidwood (Ill.) facilities. Pre-licensing activities also included an evaluation, by the NRC and FEMA, of a full participation emergency preparedness exercise at these sites.

Emergency Response Facilities

During fiscal year 1986, appraisal of emergency response facilities (ERFs) continued. The adequacy of these support facilities for nuclear power plants is assessed against requirements of Supplement 1 to NUREG-0737, as issued in generic letter 82-33. ERF appraisals were completed at the Fort Calhoun (Neb.), Palo Verde (Ariz.), San Onofre Unit 1 (Cal.), and Susquehanna (Pa.) plants. Evaluations will extend over the next several years, as ERFs are completed at each plant.

Cooperation with the States

CHAPTER 9

NRC contacts with regional, State and local agencies for purposes other than inspection and enforcement or emergency planning are administered through NRC's Office of State Programs. (Some elements of NRC's State programs are implemented by the Regional Offices under policies and procedures established by the Office of State Programs.) This chapter reports on activities in three major areas: the State Agreements Program; various liaison and cooperative programs; and financial protection and related concerns.

STATE AGREEMENTS PROGRAM

By formal agreement with the Nuclear Regulatory Commission, a total of 28 States have assumed regulatory responsibility over byproduct and source materials and small quantities of special nuclear material. The latest (28th) agreement, with Iowa, became effective on January 1, 1986. At the end of fiscal year 1986, there were about 14,000 radioactive material licenses in these Agreement States; they represent about 60 percent of all the radioactive materials licenses in the United States. (See map of Agreement States in this chapter). The NRC State Agreements Program is implemented by the NRC Regional Offices in accordance with policies and procedures established by the Office of State Programs.

In May 1986, the Governor of Maine notified the Commission of Maine's intention to prepare for Agreement State status and requested assistance in this endeavor. A proposal for Agreement is expected from Maine in 1987.

In October, 1986 the Governor of Illinois submitted a proposal for an Agreement covering byproduct, source and special nuclear material in less than critical mass quantities. The proposal also requested authority to regulate commercial low-level radioactive waste disposal. Approximately 800 licenses would be transferred to Illinois under the Agreement, including that for the existing commercial low-level waste disposal facility at Sheffield, Ill. The NRC staff prepared an assessment of the proposal which was published in the *Federal Register* for public comment. A decision was expected some time early in 1987.

Review of State Regulatory Programs

The NRC is required by the Atomic Energy Act of 1954 to periodically review Agreement State radiation control programs and confirm that they are adequate to protect public health and safety and are compatible with NRC programs. The reviews follow the guidelines contained in a Commission Policy Statement published in the *Federal Register* on December 4, 1981. Any problems identified in these reviews are brought to the attention of State authorities with recommendations for corrective action. Twenty-four routine program reviews and two follow-up reviews were conducted in 1986. As part of the program review, the NRC technical staff accompanied State inspectors to State-licensed facilities to evaluate inspector performance and reviewed selected license and compliance casework in detail. Follow-up reviews of the status of previously identified program deficiencies were conducted in New York City and Alabama in 1986.

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

NRC Technical Assistance to States

The NRC provided technical assistance to Agreement States during 1986 in the areas of licensing, inspection, enforcement and proposed statutes and regulations. For example, assistance was provided to New Hampshire in the review of a decommissioning plan for a thorium processor, to Kentucky in the review of an application for uses of isotopes in laboratory animals, to Mississippi in its review of an application for use of isotopes in off-shore waters, to North Carolina in its review of an application for a nuclear pharmacy license, to Tennessee in its review of an application for the use of tritium in a laboratory and to Texas during the loading of a large irradiator.

Training Offered by NRC

State radiation control personnel regularly attend NRC-sponsored courses to improve their technical and administrative skills and, thus, their ability to maintain high quality regulatory programs. In 1986, the NRC sponsored 12 short-term training courses, attended by 224 State personnel. Courses included health physics, industrial radiography safety, nuclear medicine procedures, introduction to licensing practices, inspection procedures, well logging, transportation of low-level radioactive waste and other

Dr. Sally DeNardo, member of the NRC's Advisory Commitee on the Medical Uses of Isotopes (see Appendix 2 for full membership), discusses new developments in nuclear medicine involving the use of monoclonal antibodies, in an address to the opening session of Agreement States radiation control program directors in October 1986, in San Francisco.

nuclear materials. On-the-job training in licensing and compliance was provided to individual staff members in New York City, Florida, Mississippi, Texas, Kansas and Oregon. On-the-job training in radiation control program management was provided to Iowa, Nevada, and Oregon.

Annual Agreement States Meeting

The annual meeting of Agreement State radiation control program directors, held in October 1986 in San Francisco, Cal., covered a wide range of regulatory issues being faced by State personnel, including low-level waste and decommissioning, materials licensing and compliance, revision of regulations, experience involving radioactive material in unauthorized places and financial assurance for licensed activities.

Regulation of Low-Level Waste

The NRC continues to provide technical assistance to States in their programs for regulating low-level radioactive waste. NRC provided technical assistance to Washington and Nevada in the renewals of U.S. Ecology licenses for low-level waste disposal sites. Assistance was provided to Florida in the renewal of a major waste processor license and Tennessee for a waste processor application. In Arizona, NRC provided technical assistance to the State in evaluation of the University of Arizona's on-site disposal. South Carolina and Washington continue to participate in the NRC review of several topical reports on high integrity containers, waste solidification processes and computer codes, used in implementing 10 CFR Part 61.

In June 1986, the NRC sponsored a special meeting with selected State and compact representatives (see below) and NRC staff on the implementation of the Low-level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA). There were approximately 100 attendees, representing the States and compacts and other interested parties. Both NRC and State representatives gave formal presentations on the LLRWPAA. The NRC covered those items requiring action by NRC and discussed its approach and progress to date. The NRC staff engaged in constructive interchange with representatives conversant with the relevant technical and institutional issues to clarify other areas the NRC might address.

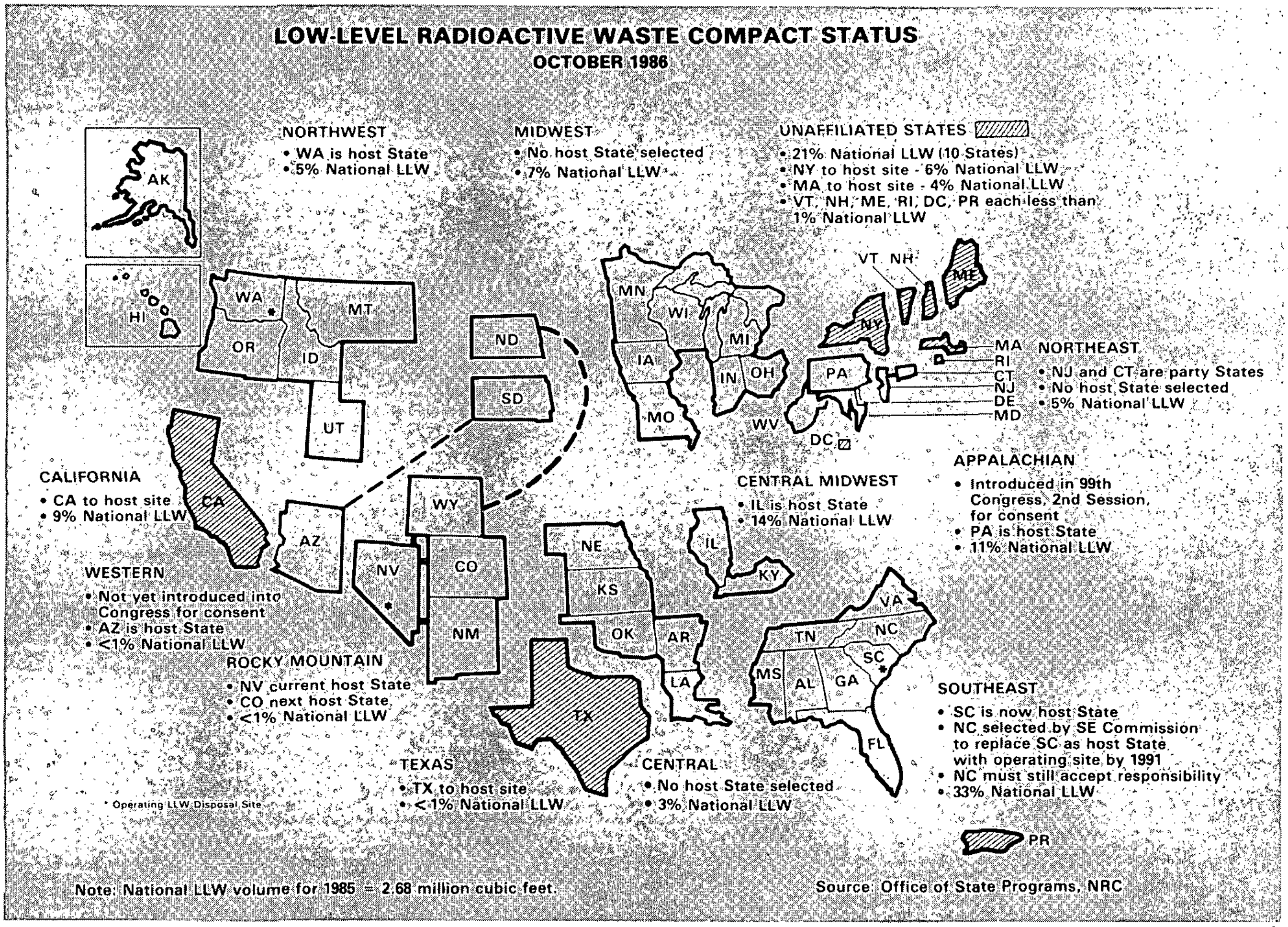
Regulation of Uranium Milling

The NRC is continuing to provide assistance to the Agreement States in their programs for regulating uranium milling. This assistance has included guidance on surety arrangements and in implementing the Environmental Protection Agency ground water requirements; direct technical assistance on specific cases to the States of Colorado, New Mexico, Texas, and Washington has also been arranged.

On March 18, 1986, the Governor of New Mexico requested that the NRC reassert its authority over uranium milling and mill tailings in that State primarily because of severe State budget constraints and the difficulty the State was having in implementing the EPA regulations, 40 CFR 192. The NRC acceded to this request and reasserted its authority in these areas effective June 1, 1986. This action did not affect the remaining portion of the New Mexico Agreement State program.

COOPERATIVE ACTIVITIES

A number of States have long sought an independent expertise in understanding the risks to public health and safety



from incidents at nuclear power reactors, so as to assure themselves that all reasonable steps are being taken to prevent an incident and to reduce the risk thereof. This interest goes beyond that addressed by the requirement that the NRC notify States of "no significant hazard," according to the so-called Sholly Amendment. The concern has grown greater in the post-TMI period and been reinforced by certain well publicized problems at nuclear power plants. The accident at Chernobyl in the Soviet Union has, of course, heightened the desire at the State level for a first-hand understanding of nuclear risks. Some State governments want to be in a position where they are not depending solely on the NRC for information on a reactor's status during or following an accident. Given these official concerns and the intense public interest in nuclear safety issues, it is to be expected that Governors and other officials would seek an arrangement by which they can routinely remain apprised of the status of specific facilities over which the NRC may have primary responsibility but which have obvious and serious safety implications for their citizens.

During the past several years, a number of States have taken steps to more closely monitor and, in some cases, to become directly involved in certain basic nuclear issues, including the regulation of nuclear power reactor operations. For example, the State of Oregon has had a State Resident Inspector, authorized by State law, at the Trojan nuclear facility since 1980. Other States have added nuclear engineers to their staffs with responsibility for monitoring specific power plants. Illinois recently became the first State to execute a Memorandum of Understanding (MOU) with the NRC to perform periodic inspections of the areas of low-level waste packaging and of transport activities at NRC-licensed facilities, including reactors, operating within the State. Similar MOUs are currently being negotiated with several other States. Some States have indicated a desire to be present at NRC inspections, "inspection exit" meetings, and enforcement conferences and to participate in other regulatory activities involving reactors. At the close of the report period, the NRC had taken these suggestions and indications under advisement.

State Involvement in Boiler and Pressure Vessel Safety

The majority of States have boiler and pressure vessel laws that cover both the nuclear and non-nuclear components at a nuclear power plant. The States recognize NRC's regulatory authority over the nuclear portions of the plant, and therefore generally focus their attention on the non-nuclear components. The NRC mandates the use of the American Society of Mechanical Engineer's Boiler and Pressure Vessel Code (the ASME Code) for construction and inservice inspection of these components.

During the last year, NRC has been negotiating an agreement with the Illinois Department of Nuclear Safety whereby Illinois can implement a State statute in accordance with the ASME Code requirements, as required by the NRC. The objective is to assure that Illinois rules are not disruptive of or inconsistent with Federal regulatory actions.

Low-Level Radioactive Waste Compacts

The Low-Level Radioactive Waste Policy Amendments Act of 1985, enacted January 15, 1986, ensures that currently operating disposal facilities will remain available until the end of 1992, subject to specified volume limitations and other requirements; establishes a system of incentives and penalties to promote steady progress toward new facility development; and, under Title II, grants consent to seven interstate low-level waste disposal compacts, covering 37 States (see "Waste Compact Status" map in this chapter). Two issues not resolved by the legislation and calling for continued attention are the regulatory uncertainties related to mixed waste disposal (i.e., waste that contains radioactive material regulated by NRC and also hazardous waste regulated by the Environmental Protection Agency) and naturally-occurring and accelerator-produced materials that are not under NRC jurisdiction.

The NRC has continued to assist the Compacts and the States by providing training in the regulation of low-level radioactive waste (10 CFR Part 61), developing criteria for the selection of an alternative disposal technology, and furnishing technical data. The NRC also comments on State enabling legislation related to the disposal of low-level radioactive waste.

State Liaison Officers

The NRC State Liaison Officer's (SLO) program was established over 10 years ago to provide a direct channel of communication between the States and the NRC. There is one Governor-appointed SLO for each of the 50 States and for the Commonwealth of Puerto Rico. The NRC looks to the SLO as the key person in the State for keeping the Governor completely informed on regulatory or emergency matters, keeping other State officials informed on such matters as are relevant to their responsibilities, and eliciting answers to questions posed by the NRC to the States on particular issues. Both regional and national SLO meetings are held periodically to keep the SLOs updated on major aspects of NRC programs. During the fiscal year, three regional meetings for the SLOs were held: in NRC's Region I (Philadelphia) in November, 1985, in Region II (Atlanta) in April, 1986, and in Region IV (Dallas) in June 1986. Subjects discussed at the meetings included, among others, the NRC's interactions with the States during unusual events, high- and low-level waste activities, transportation, emergency preparedness and uranium milling.

Workshops

In 1986, the Office of State Programs sponsored two workshops for the benefit of the States. The first was a one-day seminar for members of the National Conference of State Legislatures, held in Washington, D.C. The seminar focused on those issues legislators must understand in upcoming years to enact responsible legislation regarding low-level waste disposal, radioactive materials transportation (particularly regarding high-level waste and spent fuel), nuclear plant-life extension and plant decommissioning. Sixty legislators and interested individuals attended the seminar, which featured guest speakers from various State agencies, the NRC, the Department of Transportation and the Congress.

The second workshop, coordinated by NRC's Region IV office, brought together representatives of the Federal agencies with nuclear power plant licensees and representatives of the States in Region IV where nuclear plants are operating or are being built. Over 120 representatives of the States, utilities and other Federal agencies attended the workshop, which focused on developing and promoting a better understanding of the contacts and interactions among emergency response organizations necessary to effectively coordinate emergency preparedness efforts. The NRC hopes to sponsor and promote opportunities such as these in the future, for the mutual benefit of the States and the NRC.

Liaison with American Indian Tribes

The passage of the Nuclear Waste Policy Act of 1982—which provides for Federal agency consultation and cooperation with affected Indian Tribes, as well as with the States involved—has led to regular meetings between NRC staff and representatives of American Indian Tribes. During the report period, NRC staff attended and participated in a number of meetings, notably those held by the National Congress of American Indians (NCAI), an organization representing over 200 Tribes. At its Mid-Year Conference in Albuquerque, N.M. (June 18-20, 1986), and its 43rd Annual Convention in Phoenix, Ariz. (September 22-26, 1986), NCAI's National Indian Nuclear Waste Policy Committee discussed its concerns and views in the areas of waste regulation, notification of transportation of high-level radioactive waste, training and emergency preparedness. Also noteworthy was the participation of affected Indian Tribes in the negotiated rulemaking process regarding development of a computerized electronic documentation control system, to be known as the Licensing Support System (LSS). This system will be designed—with input from many sources, including the affected Tribes—to encompass all of the documentation dealing with the licensing of a high-level waste repository.

The NRC continues to be sensitive to concerns of American Indian Tribes and their representatives and is committed to providing assistance wherever possible.

INDEMNITY, FINANCIAL PROTECTION AND PROPERTY INSURANCE

The Price-Anderson System

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

The first layer requires all licensees of commercial nuclear power plants rated at 100 electrical megawatts or more to provide proof of financial protection in an amount equal to the maximum liability insurance available from private sources. Currently, this amount is $160 million.

The second layer provides for a retrospective premium payment mechanism whereby the utility industry would share liability for any damages resulting from a nuclear incident in excess of $160 million. In the event of such an incident, each licensee of a commercial reactor rated at 100 electrical megawatts or more would be assessed a prorated share of damages up to the statutory maximum of $5 million-per-reactor-per-incident. At present, the secondary financial protection layer is $510 million (a figure derived from 102 power reactors rated over 100 MW(e) licensed to operate at the close of the report period times $5 million-per-reactor).

The third layer—Government indemnity—had formerly been fixed as the difference between the $560 million limit of liability and the sum of the first and second layers. Government indemnity for reactors was phased out for large power reactors, however, on November 15, 1982, when the sum of the first and second layers reached $560 million. The limit of liability for a single nuclear incident now increases without limit in increments of $5 million for each new commercial reactor licensed.

Price-Anderson Renewal

New bills were introduced in the 99th Congress to modify and extend the Price-Anderson Act. Hearings were held on two of the bills—H.R. 3653, introduced by Congressman Udall, and S. 1225, introduced by Senators McClure and Simpson. The Commission testified on S. 1225 before the Senate Subcommittee on Nuclear Regulation on October 23, 1985, and on H.R. 3653 before the House Subcommittee on Energy Conservation and Power on July 17, 1986. S. 1225 was reported out by the Senate Energy and Natural Resources and Environment and Public Works Committees. H.R. 3653 was reported out by the House Interior and Insular Affairs, Energy and Commerce, and Science and Technology Committees. Neither bill was brought to the floor of the two chambers during the report period.

Indemnity Operations

As of September 30, 1986, 145 indemnity agreements with NRC were in effect. Indemnity fees collected by the NRC from October 1, 1985 through September 30, 1986 totalled $98,493. Fees collected since the inception of the program total $23,216,395. Future collections of indemnity fees will continue to be lower since the indemnity program has been phased out for commercial reactor licensees. No payments have been made under the NRC's indemnity agreements with licensees during the 29 years of the program's existence.

NRC Chairman Lando Zech (right) met with Willaim Doub, a former AEC Commissioner and now Federal Representative to the Southern States Energy Board (SSEB), and Ken Nemeth, SSEB Executive Director (center), in September 1986, to discuss board activities on nuclear issues. The board represents 17 southern Governors in matters of energy policy.

Insurance Premium Refunds

The two private nuclear energy liability insurance pools-American Nuclear Insurers and the Mutual Atomic Energy Liability Underwriters—paid policyholders the 20th annual refund of premium reserves under their Industry Credit Rating Plan. Under the plan, a portion of the annual premiums is set aside as a reserve either for payment of losses or for eventual refund to policyholders. The amount of the reserve available for refund is determined on the basis of loss experience of all policyholders over the preceding 10-year period.

Refunds paid in 1986 totalled $4,238,821, which is approximately 37.3 percent of all premiums paid on the nuclear liability insurance policies issued in 1976 and covers the period 1976-1988. The refunds represent 74.1 percent of the premiums placed in reserve in 1976.

Financial Qualifications Reviews Of Electric Utilities

The U.S. Court of Appeals for the D.C. Circuit issued a decision on July 11, 1986 in the financial qualifications rule case (*New England Coalition on Nuclear Pollution, et al.* v. *NRC*; D.C. Cir. Case No. 84-1514). The decision upholds the Commission's rule which exempts from staff review and adjudication the financial qualifications of regulated utility companies (and of utilities that set their own rates) applying for power reactor operating licenses. The three-judge appeals court panel voted unanimously to uphold the NRC's position in the case. (For background, see the *1985 NRC Annual Report,* p. 136.)

Incentive Regulation of Electric Utilities

Performance incentives established by State public utility commission (PUCs) are applicable to the construction or operation of about 41 reactors licensed for 27 investor-owned utilities in 16 States. (For background, see the *1985 NRC Annual Report,* p. 136.) The NRC staff continues to monitor development of the incentives and provides an updated report on all nuclear plant incentives to its Regional Offices twice a year. The staff maintains contact with the PUC staffs and the utilities responsible for implementing the incentives in order to obtain the updated information and to consider possible safety effects of the incentives.

Property Insurance

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

On November 8, 1984, the Commission published a proposed rule in the *Federal Register* (49 FR 44645) which would, if promulgated, increase the amount of on-site property damage insurance from the current minimum of $585 million to slightly over $1 billion. The main issues addressed in the proposed rule, in addition to the amount of insurance that should be required, were (1) whether the Federal Government can preempt State law that prohibits certain public utilities from buying insurance offered by mutual companies or insurance requiring payment of a retrospective premium, and (2) whether a priority in the payment of insurance proceeds for decontamination and cleanup should be imposed. Because of the complexity of these issues, promulgation of a final property insurance rule has been delayed because of unresolved policy questions. The final rule should be considered by the Commission in late 1986.

The fourth annual property insurance reports submitted by power reactor licensees indicate that of the 68 sites insured as of April 1, 1986, 42 carry the maximum available—$1.135 billion. Another 15 sites carry at least $1.0 billion in coverage. Four sites are exempt from NRC's full property insurance requirement. Most recently, on July 1, 1986, the Commission voted to approve Dairyland Power Cooperative's (Wis.) request to reduce the property insurance it is required to carry from $500 million to $180 million.

STATUS OF TMI-2 FACILITY

Financial Aspects of TMI-2 Cleanup

Funding by GPU. (For background, see the *1985 NRC Annual Report,* p. 137.) Revenues collected by General Public Utilities (GPU) Corporation's three operating subsidiaries in Pennsylvania and New Jersey continued to be expended on cleanup of the Three Mile Island Unit 2 facility during 1986. Customer funding of cleanup amounted to about $56 million in 1986, and is estimated to total approximately $250 million over the course of the cleanup. GPU continues to provide cash advances from internal sources to alleviate any cash flow problem related to cleanup. The total 1986 advance is estimated at $14 million. The GPU projections provided to NRC indicate a continuing GPU commitment to provide such cash advances as may be needed for the cleanup. Continued improvement in GPU's financial condition and cash flow position gives greater assurance that such cash advances will be made.

Cost Sharing Plan. During 1986, GPU continued to receive cash payments from all suggested contributors under the TMI-2 cleanup cost sharing plan proposed by Pennsylvania Governor Richard Thornburgh in July 1981. (For background, see the 1985 NRC Annual Report, p. 137.) The Edison Electric Institute's (EEI) industry cost-sharing program paid its committed $25 million annual contribution in 1986, the second year of industry contributions through the EEI program. The NRC will continue to monitor closely the funding of cleanup operations at TMI-2.

International Programs

CHAPTER 10

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

Highlights of Fiscal Year 1986

During the report period, the NRC:

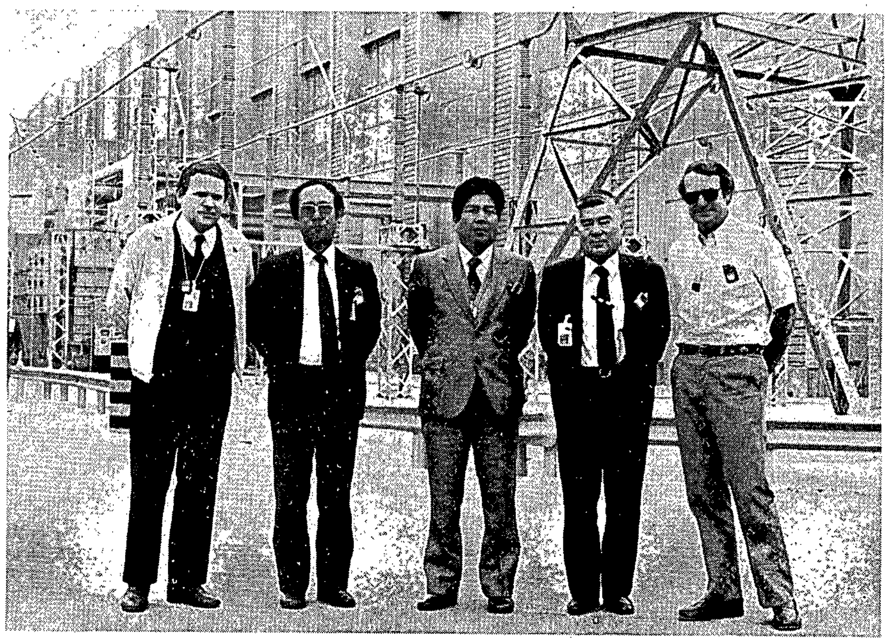
- Participated in the efforts of the U.S. Government Task Force to track recorded radiation levels in various countries following the Chernobyl accident (April 1986) in the Soviet Union.
- Participated in a major International Atomic Energy Agency (IAEA) meeting at which the Soviets met with the international community to discuss the Chernobyl accident in detail.
- Participated in the IAEA annual General Conference and the Special Session of the General Conference, which brought together Minister-level officials from around the world to discuss nuclear safety issues in the wake of the Chernobyl accident.
- Participated in the development of international conventions on early notification of a nuclear accident and on assistance in the case of a nuclear accident or radiological emergency.
- Renewed bilateral nuclear safety cooperation arrangements with China, West Germany, Japan, and the United Kingdom.
- Continued to expand its network of mutually beneficial agreements on nuclear safety research, including the signing of a seismic agreement with Japan.
- Published NUREG-1160, "International Cooperation During Radiological Emergencies," which addresses the technical cooperation NRC could provide to foreign regulatory agencies in the event of a radiological emergency.
- Arranged visits by representatives from governments and from public and private organizations overseas for discussions of nuclear safety issues.
- Sent experts to South Korea, Brazil, Mexico, Egypt, China and, for the first time, Indonesia in support of the technical assistance programs of the IAEA to provide safety advice in their nuclear programs.
- Participated in Operational Safety Analysis Review Team missions in Mexico, West Germany, and South Korea.
- Issued 138 export licenses and 17 amendments to existing licenses.
- Worked closely with the Executive Branch to assist the IAEA in strengthening international safeguards and physical security. The NRC sent experts to Japan, France, West Germany, the United Kingdom, the European Community, and Australia for discussions in this area.

INTERNATIONAL COOPERATION

Bilateral Information Exchange Arrangements

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

The NRC's bilateral information exchange program ensures that technical information on nuclear safety and research is shared among the 22 nations involved. In these photos, staff representatives of the French CEA and the NRC discuss the relatives merits of the two nations' pressurized water reactor designs (above), and a team from the Japanese Nuclear Power Research Center (at right) tour the Peach Bottom (Pa.) nuclear power plant to study the facility's maintenance program.



prompt and reciprocal notification of reactor safety problems that could affect both U.S. and foreign nuclear facilities, and to facilitate identification of possible "precursor events" that warrant further investigation. These arrangements also provide a framework for bilateral cooperation on nuclear safety, safeguards, waste management, and environmental protection, and they serve as the vehicle for the NRC to provide assistance in improving nuclear health and safety practices to developing countries operating power reactors supplied by the United States. The bilateral arrangements are effective for five years as written, but contain provisions for renewal by mutual agreement.

In 1986, the NRC's arrangements with China, West Germany, Japan, and the United Kingdom were renewed. Negotiations on the renewals of existing arrangements with Italy, South Korea, and Mexico were under way at the close of the report period.

Bilateral and Multilateral Safety Research Agreements

The NRC is currently involved in 55 agreements for research cooperation with 17 countries, under which it cooperates in ongoing nuclear safety research projects both in the U.S. and overseas. These research projects cover a wide range of research projects, including making direct contributions of data and analyses needed to confirm and assess computer codes used in the NRC licensing and regulatory process. (See the *1985 NRC Annual Report,* pp. 139, 140.)

In 1986, a significant agreement was concluded between the NRC and the Japanese Ministry of International Trade and Industry which provides for cooperation in high-level vibration tests of nuclear plant piping at the Tadotsu Engineering Laboratory in Japan. This agreement will enable NRC and the Electric Power Research Institute to have access to seismic test results not otherwise available in the U.S.

In addition, a general research arrangement was signed with Spain, and bilateral agreements on severe accident research were signed with Canada and the United Kingdom. Two agreements with West Germany were extended in the areas of pressure vessel testing and the SIMMER-II computer program, thermal-hydraulic research agreements were signed with Italy, South Korea, and the American Institute on Taiwan. An agreement on the CONTAIN, SIMMER, and Super Systems codes was also concluded with Japan during this reporting period. (See Chapter 11 for discussion of computer codes used in NRC regulatory research.)

Response to Chernobyl

Drawing on well established bilateral regulatory and safety contacts abroad, the NRC was able to obtain timely tabulations of radiation levels being recorded in various countries throughout the world as a result of the Chernobyl accident. The information was received by means of telefacsimile transmissions to the NRC Operations Center. This information, besides being factored into the tracking efforts of the U.S. Government Task Force, was provided to regulatory authorities in other countries, in order to aid in their evaluations and projections of the health effects of the accident.

International Emergency Preparedness Cooperation

In April 1986, NRC published NUREG-1160, "International Cooperation During Radiological Emergencies." This report defines the scope, application and limits of the technical cooperation which the NRC would be able to provide, upon request, to a foreign regulatory agency in the event of a radiological emergency. Although the report concentrates on the bilateral relationship between regulatory authorities which license similar kinds of technologies, the document also addresses the importance of regional cooperation during emergency situations. Under the regional cooperation portions of this report, and to some extent prompted by the Chernobyl event, the NRC is seeking to formalize existing communication ties and procedures with the Canadian Atomic Energy Control Board and other Canadian authorities, in order to ensure effective communication in the event of an emergency at a facility along the common border. Initial discussions and information exchanges have begun in this area and, in the year ahead, NRC and Canadian authorities plan to test communications and notification procedures.

Minister Walter Wallman, Federal Minister for Environment, Nature Conservation and Reactor Safety for the Federal Republic of Germany—who was active in organizing international meetings to review the Chernobyl accident—renewed an information exchange agreement when he visited the NRC in July 1986. The Minister is shown being welcomed by NRC Chairman Lando W. Zech, Jr.

Exchange of Information on Nuclear Waste Management with West German RSK

In May 1986, a delegation from West Germany's Reactor Safety Commission (RSK) visited the Advisory Committee on Reactor Safeguards to discuss nuclear waste management topics. The NRC's Office of Nuclear Material Safety and Safeguards also briefed the RSK on U.S. waste management practices, approaches, and regulatory procedures, and received information on waste management approaches pursued in West Germany.

Technical Safety Cooperation

In 1986, the NRC held policy and technical meetings with over 170 visitors from foreign countries and organizations. The Office of International Programs coordinated these visits in advance with the NRC staff to assure extensive and detailed discussion of topics of mutual interest and to promote a two-way flow of information. The Office also gathered and conveyed responses to more than 100 requests for technical and safety information.

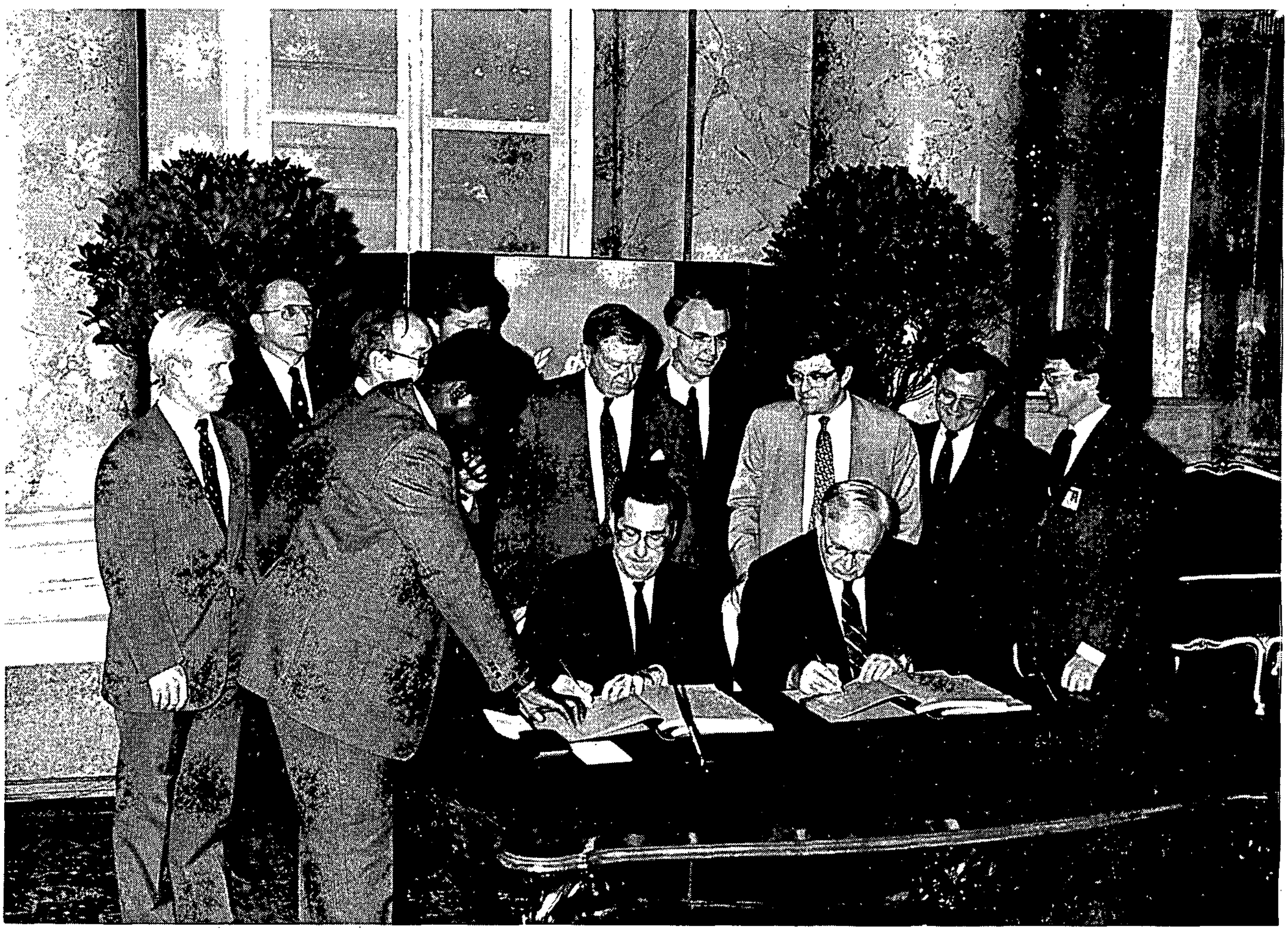
Foreign Assignees to the NRC Staff

Strong interest continued during fiscal year 1986 in the NRC's on-the-job training program for foreign nationals. This year the NRC hosted 23 foreign assignees from 12 different countries. Their primary interests lay in risk analysis, radiological protection, electrical instrumentation and control, human factors, reactor and containment systems, emergency planning and response, international cooperation, and legal aspects of regulation. The assignees served in the NRC's Offices of Nuclear Reactor Regulation, Inspection and Enforcement, International Programs, Analysis and Evaluation of Operational Data, and in several of the NRC Regional Offices. Many among them also took the opportunity during their long term assignments to attend technical training courses at the NRC's Technical Training Center in Chattanooga, Tenn.

PARTICIPATION IN INTERNATIONAL ORGANIZATIONS AND CONFERENCES

Post-Chernobyl Review Meeting

The International Atomic Energy Agency (IAEA) convened a meeting in Vienna on August 24-29, 1986, at which the Soviets met with the international community to share for the first time the detailed results of their analyses of the causes, course, and implications of the Chernobyl accident. The U.S. was represented by a major interagency delegation that included Harold R. Denton, Director of Nuclear Reactor Regulation, and other senior NRC staff members. The Soviets freely shared the data they had gathered and the assessments they had performed, both in the scheduled group sessions and in the bilateral exchanges in which they participated. (See Chapter 2 for discussion of the accident at Chernobyl.)



Two important international conventions were developed as a result of the Chernobyl accident and introduced as a Special Session of the General Conference of the IAEA in Vienna in September 1986. Shown signing for the United States is Ambassador Richard T. Kennedy (right), a former NRC Commissioner. Among the U.S. witnesses, standing behind the signatories, are NRC Chairman Lando W. Zech., Jr., and Secretary of Energy John Herrington.

IAEA General Conference and International Conventions on Emergency Notification and Assistance

NRC Chairman Lando W. Zech, Jr., International Programs Director James R. Shea, and David Humenansky of the Chairman's staff participated in a series of meetings in Vienna from September 24 to October 3, 1986, including the IAEA's Thirtieth General Conference, held September 29-October 3. Harold R. Denton, Nuclear Reactor Regulation Director, participated in special IAEA sessions on nuclear safety issues during the General Conference. Chairman Zech took the opportunity to visit a nuclear power plant in Czechoslovakia, renewed a nuclear safety protocol with National Nuclear Safety Administration of the People's Republic of China, and presented documentation on NRC's source term code package to the IAEA, with the complete package to follow shortly, in order to facilitate the ongoing international safety assessment of the Chernobyl accident.

At a Special Session of the General Conference on September 24-26, two separate Conventions were introduced—one on Early Notification of a Nuclear Accident and the other on Assistance in the Case of a Nuclear Accident or Radiological Emergency. The U.S. delegation, together with many participating countries, signed these two international conventions which were developed as a direct result of the Chernobyl accident. The first convention calls for prompt notification of the IAEA and affected States of a radiological emergency which has the potential for international impact. The second convention concerns the provision of technical assistance to Member States, either on a bilateral basis or through the IAEA, to aid them in their response to an event. The NRC was represented in the interagency and international meetings which led to the drafting of these two binding conventions and drew heavily from its bilateral experience in contributing to the framing of these international instruments. Procedures to fulfill the reporting requirements of the notification convention were drawn up and will be tested in the year ahead.

Technical Safety Assistance

During fiscal year 1986, the NRC—acting through the technical assistance program of the IAEA and also directly through its bilateral contacts with other countries—continued to provide nuclear safety advice and assistance to countries developing their nuclear power programs, as well as to several countries with established nuclear programs. A Region IV (Dallas) inspector visited South Korea on two occasions to advise on the fluid inspection/instrumentation and control systems at the KORI nuclear power reactors. A Region III (Chicago) inspector travelled to Brazil to advise the National Nuclear Energy Commission on the safety evaluation of the Angra I nuclear power plant. Three NRC inspectors visited Mexico on separate missions to advise on Mexico's radiation protection inspection program and the conduct of technical specification reviews. NRC personnel also conducted workshops in Egypt on nuclear safety, and two NRC staff members travelled to China to lecture on reactor risk analysis and pressurized water reactor coolant chemistry control. Also, for the first time, an NRC expert was sent to Indonesia to advise on the health physics organization in the Batan facilities.

In 1986, there was an increase in the number of Operational Safety Review Team (OSART) missions sponsored by the IAEA. Through these missions, a member state may request that an OSART team—made up of various IAEA and technical representatives from other countries—visit a nuclear plant site to review local operating practices and exchange views with plant managers on ways to enhance safety. The NRC participated in OSART missions in Mexico, West Germany, and South Korea this year.

Activities in the OECD/NEA

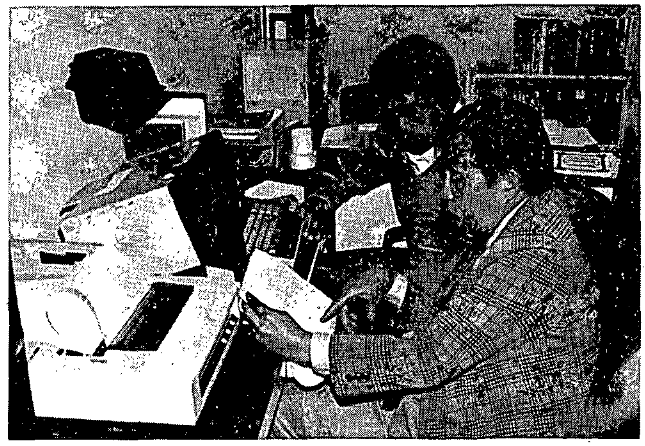
The NRC, through its continuing participation in the 24-nation Organization for Economic Cooperation and Development (OECD), took part in several post-Chernobyl meetings on improved reactor safety sponsored by the OECD Nuclear Energy Agency (NEA). Specific areas of interest included reactor operation, incident reporting, human factors, the management of accident situations, radiation protection, and third party liability. Much of this activity was carried out in coordination with the post-Chernobyl nuclear safety and radiation protection efforts of the IAEA, the Commission of the European Communities, and the World Health Organization. The OECD/NEA includes the industrialized countries of Western Europe plus Australia, Canada, Japan, and the United States.

EXPORT-IMPORT AND NON-PROLIFERATION ACTIONS

NRC Export License Summary

The NRC has responsibility under the Atomic Energy Act, as amended, for the licensing of the export of nuclear-related materials and equipment. This export authority extends to production and utilization facilities, to special nuclear and source material, to byproduct materials, and to certain nuclear-related components and other materials. In carrying out its responsibilities for approving exports, the NRC obtains the views and recommendations of other governmental agencies and departments as required.

In 1986, the NRC issued 138 export licenses and 17 amendments to existing licenses. Of these, 100 were "major" licenses or significant amendments in four categories: special nuclear material, source material, nuclear reactor materials (graphite and deuterium), and major reac-



The NRC's international information exchange programs are carried on at all levels of the agency's operations. These photos, taken during visits to the NRC Region II office in Atlanta, show: top left, Swedish representatives Leif Ericson, Olaf Hormander and Lars G. Larson, top right, Philippine Atomic Energy Commission member Alejandro J. Mateo, and, at left, directors of Spain's Consejo de Securidad Nuclear Luis E. Echavarri and Jose Villadoniga.

tor components. The majority of these major cases involved routine exports of low-enriched uranium intended for use in commercial light-water power reactors. Sixteen licenses involved exports of high-enriched uranium to research reactors (247.214 kilograms) and to the German Thorium High Temperature Reactor (218.545 kilograms). A total of 11 nations received shipments of special nuclear material under major export licenses during the year. As in the previous year, several major export licenses were issued for shipment of source material to the European Community for enrichment and subsequent power reactor use. The remaining 55 licenses and amendments included nine for exports of small quantities of special nuclear materials, eight for source material, nine for byproduct material, 12 for components and materials, and 17 for miscellaneous license amendments such as extensions of expiration dates.

NRC Consultations with the Executive Branch On Nuclear-Related Export Matters

During 1986, the NRC was consulted by the Executive Branch on 37 requests involving the retransfer of U.S.-origin nuclear material and heavy water, the export of U.S. nuclear-related technology, and a determination that safeguards could be applied effectively at the Tokai-Mura reprocessing facility in Japan. One significant request involved the retransfer of 6,500 kilograms of uranium and 280 kilograms of produced plutonium in spent mixed-oxide fuel to Sweden for storage and ultimate disposal in exchange for the transfer to West Germany of title to 57,000 kilograms of uranium and 428 kilograms of produced plutonium.

In 1986, revised interagency procedures were implemented to make the review of NRC export license requests more consistent with those for export-related licenses. Under the new procedures, certain retransfers of low-enriched material no longer require interagency review. Also, for requests involving enrichment abroad, separate interagency reviews are no longer required because these cases are now considered in the overall review of the NRC export license involved. These new procedures have substantially reduced the number of cases to be reviewed by the NRC and the Executive Branch.

The NRC was also consulted by the Department of Commerce on over 300 nuclear-related cases licensed by that agency.

SNEC—Interagency Review of Nuclear Exports. The NRC is an active participant in the Subgroup on Nuclear Export Coordination (SNEC), which was established in 1978 to provide a more effective framework to review nuclear-related license requests on an interagency basis. The cases include primarily Commerce-licensed requests for commodities that are controlled for nuclear non-proliferation reasons, but they also include NRC-licensed cases and U.S. nuclear-related technology transfers. Representatives from the Departments of State, Energy, and Defense, the Arms Control and Disarmament Agency, the NRC, and the intelligence community are all members. In 1986, SNEC members worked diligently to revise Commerce export regulations in an effort to simplify and clarify those entries covering nuclear-related commodities. A subgroup chaired by the NRC representative from the Office of International Programs is in the process of revising Commerce and NRC regulations on graphite exports. SNEC members reviewed over 200 cases in 1986. These cases covered dual-use commodities, primarily those intended for export to sensitive destinations such as Argentina, Brazil, India, Israel, South Africa, Iraq, and Pakistan.

International Safeguards and Physical Security

In all pending export cases to be licensed by the NRC, the staff reviews the implementation of the International Atomic Energy Agency (IAEA) safeguards and physical security arrangements to be applied to the exports in the receiving country. These reviews are performed in compliance with U.S. non-proliferation laws, to ensure that U.S. exports will be protected during transit and use in the importing country and that the exports will not be used for proscribed purposes, such as the making of nuclear explosives.

The NRC participates in U.S. Government efforts to assist the IAEA in improving its safeguards system. The U.S. Program of Technical Assistance to IAEA Safeguards and the U.S. Action Plan Working Group (APWG) are the primary programs in this area. Through the activities of these groups, the U.S. is able to participate in joint projects with other countries and the IAEA itself in support of the international safeguards system. Under the auspices of the APWG, the NRC participated in bilateral and multilateral discussions on safeguards experience with Japan, France, West Germany, the United Kingdom, the European Community, and Australia in 1986.

The NRC has substantial responsibility for implementing the U.S./IAEA Agreement to apply international safeguards to selected U.S. nuclear facilities. The NRC participates in negotiation of the arrangements for applying international safeguards on facilities it licenses. The NRC also assists the IAEA in scheduling and organizing its inspection activities at NRC-licensed plants and accompanies the inspectors during inspections. In 1986, three NRC-licensed facilities were subject to the application of international safeguards—two power reactors, Salem 1 in New Jersey and Turkey Point 4 in Florida, and the Westinghouse low-enriched uranium fuel fabrication plant in South Carolina. Four other NRC-licensed low-enriched uranium fuel fabrication plants are subject to limited international safeguards. They must report regularly to the IAEA on the amount of nuclear material in their inventory, and any changes in the amount since the previous report. These facilities are operated by Combustion Engineering, Exxon, General Electric, and Babcock & Wilcox.

In support of its review of physical security arrangements of U.S.-controlled materials in other countries, the NRC participates in Department of Energy-sponsored trips to the importing countries to discuss their physical security programs. In this regard, a U.S. delegation visited the Netherlands and West Germany during 1986.

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

Nuclear Regulatory Research

CHAPTER 11

The programs of the Office of Nuclear Regulatory Research (RES) are an essential and integral part of the regulatory process. Safety research supports nuclear regulation by providing defensible technical bases for regulatory action to ensure the protection of public health and safety. NRC research efforts emphasize early identification of potential problems with operating reactors and also the development of bases for reducing uncertainty in the licensing of facilities for the disposal of nuclear waste.

RES also has the responsibility for developing and coordinating NRC standards—the regulations and guides (see box on the following page) governing NRC-licensed activities. (NRC regulations, which are set forth in Title 10, Chapter I, of the *Code of Federal Regulations,* are published in the *Federal Register.* Regulations issued by the NRC in 1986 are listed in Appendix 4. Regulatory guides are described in Appendix 5, which includes a listing of those guides issued, revised, or withdrawn during fiscal year 1986.)

SETTING PRIORITIES

Facing a list of important problems to solve—with a limited budget and staff—crucial decisions must be made as to what major research issues shall be addressed in what order, and how the available resources shall best be employed in dealing with those major issues, as well as the subordinate issues subsumed under each major issue. Setting the right priorities, of course, is fundamental to the efficient management of human and fiscal resources and to the most effective allocation thereof, both in sustaining long term research efforts and in responding to unforeseen needs for research data as they arise.

A procedure for setting these priorities has been developed in the NRC which is sharply focused on the expeditious resolution of issues important to the NRC and the public, e.g., severe accidents, seismic events, thermal-hydraulic transients. Each major issue carries a set of risk and regulatory implications that provides the rationale for assigning a priority level to it. The major issues are divided into sub-issues involving one or more specific research projects, and priorities for these are also decided by the primary criterion of public protection as the regulatory imperative.

NUCLEAR POWER PLANT AGING

Reactor Pressure Vessels

The ability of the NRC staff to make decisions concerning the present and continuing safety of nuclear pressure vessels under both normal and abnormal operating conditions is dependent upon the existence of verified analysis methods and a solid background of applicable experimental data. It is the role of the pressure vessel research program to provide both the analytical methods and the experimental data needed. Further, it is the purpose of this program to determine the ways and the extent to which the light-water-reactor environment (including temperature, stress, coolant, and radiation) changes their operational life—in other words, to determine the aging effect of reactor pressure vessels upon the safe operation of these structures.

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

NRC-sponsored research efforts have been primarily conducted at the Oak Ridge National Laboratory (ORNL) under the Heavy Section Steel Technology (HSST) program, with supporting activities conducted at the Naval Research Laboratory (NRL) and at the Materials Engineering Associates, Inc. (MEA) laboratory. These activities have developed data that were instrumental in the early recognition and rapid resolution of the PTS problem. This resolution took the form of an embrittlement screening criterion to be applied to operating reactor vessels. This criterion, called the Reference Temperature or PTS, represents an embrittlement level beyond which utilities cannot operate

REGULATIONS AND GUIDES

NRC standards are primarily of two types:

- Regulations, setting forth requirements that must be met by NRC licensees in Title 10, Chapter I, of the *Code of Federal Regulations.*
- Regulatory guides, describing, primarily, methods acceptable to the NRC staff for implementing specific portions of NRC regulations.

When NRC proposes new or amended regulations, they are normally published in the *Federal Register* to allow interested persons time for comment before they are adopted. This is required by the Administrative Procedure Act. Following the public comment period, the regulations are revised, as appropriate, to reflect the comments received. Once adopted by the NRC, they are published in the *Federal Register* in final form with the date they become effective. After that publication, rules are codified and included annually in the *Code of Federal Regulations.*

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

The NRC encourages comments and suggestions for improvements in regulatory guides and, before staff review is completed, issues them for comment to many individuals and organizations along with the value/impact statements that indicate the objectives of each guide and its expected effectiveness and impact.

without the specific approval of the NRC. In 1985, an amendment to 10 CFR Part 50 (§ 50.61) established the screening criterion and stated that the NRC would issue a regulatory guide for utilities to follow as their vessels approach the screening criterion. This regulatory guide was issued for comment in 1986 and, at the close of the report period, was in the process of being issued as an active guide.

The research effort during 1986 was twofold: (1) the establishment through confirmatory research of the levels of conservatism built into the screening criterion, and (2) the development of support data on the degradation of pressure vessel toughness and on the fracture mechanics methods, both deterministic and probabilistic, used in the preparation of the regulatory guide on PTS. Part 2 is complete while part 1 is continuing. In part 1, a series of large-scale, model pressure vessel experiments was initiated at ORNL in 1983, as part of the HSST program, with the first pressurized thermal shock experiment (PTSE-1) being conducted in 1984. This was the first controlled experiment in which a pressure vessel with wall thickness approaching that of a full-scale reactor pressure vessel was flawed and subjected to combined thermal and pressure transients similar to those that could be encountered during an actual PTS event. The first test vessel was fabricated of present-practice, high-toughness steel that was heat treated to make its toughness equivalent to that of moderately irradiated, embrittled steel. This experiment confirmed the beneficial effects of the warm pre-stress phenomenon, which in some cases would prevent the initiation of cracking, and in other cases would limit the extent of cracking of a flawed pressure vessel subject to a PTS scenario. Further, this test demonstrated the accuracy of the theoretically predicted fracture behavior of the vessel, which constitutes the basis of the PTS screening criterion.

During 1986, procurement and fabrication of the vessel for the second pressurized thermal shock experiment (PTSE-2) was completed. This experiment, which will be conducted during December 1986, is similar to the first test except that the test section of the vessel is fabricated from a low-upper-shelf Charpy V-notch energy toughness material, which simulates some steels and welds found in some of our older reactors. This experiment is specifically designed to validate the applicability of the NRC's PTS screening criterion to all classes of reactor pressure vessel materials now in service.

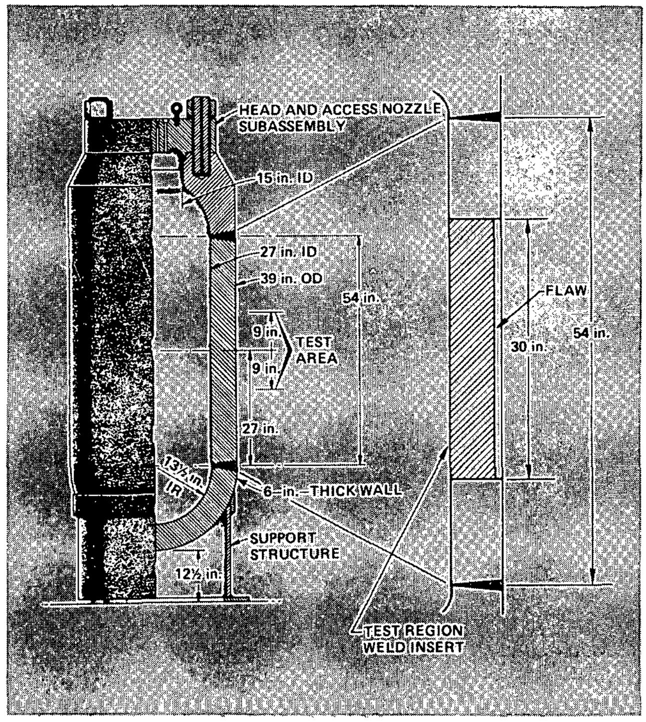
Significant supporting activities to the large-scale PTS experiments are being conducted at ORNL, the National Bureau of Standards (NBS), and MEA, with extensive subcontracted work by the University of Maryland and Southwest Research Institute. The experiments involve wide-plate crack arrest tests, dynamic effects in crack propagation and arrest studies, and the study of the effect that the stainless steel cladding, which lines all commercial reactor pressure vessels, has upon the initiation or mitigation of cracking of flawed vessels when subjected to a PTS transient. A cooperative effort between the NRC, through the HSST program, and the NBS involves a series of 18 wide-plate crack arrest experiments. These tests are intended to develop data to extend the application of fracture mechanics to wider ranges of materials and loading conditions. They involve the analysis of large-plate specimens that are deliberately flawed, subjected to severe thermal gradients, and pulled to produce long crack propagations and arrest. These tests require the use of the largest available loading machine in the country, which is at NBS. The first test was conducted in September 1984 and during 1985 and 1986 an additional seven experiments were carried out, bringing the total of tests completed to eight out of the 18 planned. The first six experimental specimens were fabricated from present-practice, high-toughness steel, similar to that material used in the first PTS experiment (PTSE-1) and in the majority of operating U.S. nuclear reactor vessels. The last three tests were conducted using a material that simulates the low-upper-shelf, Charpy V-notch energy toughness material discussed above, again simulating the material found in some of our older reactor vessels. The last six tests will involve the use of the actual low-upper-shelf toughness material taken from cancelled plant reactor vessels.

Data from the PTS experiments and the wide-plate tests represent the state of the art in fracture experimentation and have evoked interest in all the countries of western Europe and Japan. These data are being used to develop improved analytical models for a better understanding of the fracture process and of the margins that current design criteria provide against fracture. To implement a wide dissemination of the information gained from this program, an annual NRC Crack Arrest Workshop was instituted at the NBS in April 1985. The second workshop was held in April 1986 with broad attendance from U.S. and European researchers. The third workshop is planned for May 1987 with more emphasis on European and Asiatic attendance.

Though work on the effect of cladding on crack initiation, crack propagation, and crack arrest is not complete, work previously completed at MEA and ORNL during 1986 gave preliminary results that indicate that reactor pressure vessel cladding has no significant mitigating effect. This tends to confirm the NRC's original position in the development of the Reference Temperature-PTS screening criterion, wherein such mitigating effects were not considered. During 1987, irradiated stainless steel clad specimens, representing clad vessels at the end of pressure vessel life, will be tested. The program on cladding effects is planned for completion in 1988.

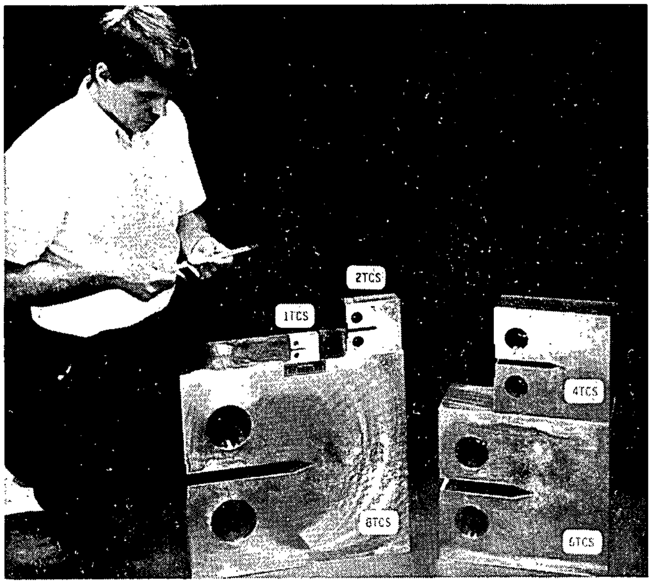
Reactor pressure vessels deteriorate with age because of three identifiable phenomena: (1) fatigue, (2) the effect of the chemical (coolant) and temperature environments upon crack initiation and growth, and (3) the effect of irradiation upon the material properties of the vessel steels. Much is known about fatigue in general, as incorporated into the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (particularly in Section III, "Nuclear Components") which forms the basis of U.S. nuclear pressure vessel design. What is not known is the effect of the LWR environment (high temperature, coolant water) upon the fatigue life of a pressure vessel. This lack of knowledge is clearly conceded in Section III, directing the designer to take cognizance of the effects of the environment, but yet not supplying the designer with any specific guidance. From the time of the development of Section III to the present, it was believed that the safety factors built into the Code would account for these unknown environmental factors. Work conducted at MEA during 1986 has produced evidence that the environment, particularly temperature, plays a much more significant role in fatigue than previously believed, thus reducing the expected margins of safety. Continued effort is planned in this area for 1987, with the main body of results to be completed in 1988.

Radiation Embrittlement. Normal operation of reactors produces excess neutrons that impinge upon the reactor pressure vessel walls. This action causes the steel of these walls to lose their fracture toughness to varying degrees,



Oak Ridge National Laboratory in Tennessee was the site of a substantial amount of NRC-sponsored research in 1986, under its Heavy Section Steel Technology (HSST) program. In December, the second of a series of eight pressurized thermal shock experiments was conducted on a type of vessel steel that simulates steels and welds in older reactors. The configuration shown here shows the vessel, test material insert and flaw dimensions addressed in that experiment.

dependent upon the chemical constituents of the steel. This problem has been studied for many years, and research here, particularly at the NRL and MEA, and abroad has identified certain alloying and/or residual elements of pressure vessel steels and weldments that, when struck by the excess neutrons from the core, embrittles the vessel wall over time. Based on this work, chemical composition standards for reactor pressure steels have been developed to effectively minimize this embrittlement. All the newer or present-practice reactor pressure vessels now in service were fabricated to these standards. In 1985, the HSST Fourth Irradiation Series was completed and the results analyzed in 1986. This study, which involves sufficient samples to allow statistical analysis of the ensuing data (missing in previous studies), confirmed the efficacy of the control of chemical constituents such as copper, nickel and phosphorus in the maintenance of vessel fracture toughness through the projected operational life of reactor pressure vessels. In 1986, work continued on the HSST Fifth and Sixth Irradiation Series. All irradiations of the specimens in the Fifth Series were completed and testing of the specimens will be completed in 1987. This series is designed to validate Code-designated trend properties for the irradiation-induced changes in fracture properties used to evaluate pressure vessel structural adequacy under both normal and accident conditions. The HSST Sixth Irradiation Series begun in 1985, using the same material as in the Fifth Series, will allow the examination of the effects of irradiation on the crack arrest properties



The radiation embrittlement of pressure vessel walls and welds, studied under NRC auspices for many years, has led to extensive testing of various steels, irradiated and unirradiated. This technician at Oak Ridge National Laboratory is examining fracture specimens under the NRC HSST program.

of vessel welds, again in order to confirm the Code-designated trend curves. In 1986, all testing of the unirradiated control specimens from this series was completed. Irradiations and all testing of the Sixth Series will be completed in 1987.

The study of the effect of neutron irradiation on fracture toughness degradation of pressure vessel steels and weldments has developed over the years in a somewhat empirical manner. In 1985, a research program was initiated at MEA to develop a mechanistic model for the irradiation-induced embrittlement of reactor pressure vessel steels and weldments. Successful completion of this effort will allow the NRC to place less reliance upon empirically developed data bases and will significantly aid in the development of a research basis for regulatory judgments on plant-life extension.

Surveillance Dosimetry. An important aspect of the surveillance program to determine the degree of embrittlement in the pressure vessel of an operating nuclear power plant is the prediction of the amount of neutron radiation exposure (neutron fluence) of the vessel. Fluence determinations are made by a combination of calculations to compute the fluence, dosimetry measurements at key surveillance locations, and a consolidation of the measurements and calculations to reduce uncertainties of predictions at critical locations of the vessel. It is necessary that these predictions be reasonably accurate to ensure that the plant is operating in conformance with NRC safety regulations.

The experimental parts of this program are designed to produce validated, standard methods for the prediction, calculation, and measurement of neutron flux and fluence parameters and the correlation of the parameters to embrittlement and degradation of mechanical properties of pressure vessel steel in LWRs. The result is achieved by establishing a series of benchmarks for use by vendors, laboratories, utilities, or others producing fluence and embrittlement values, and by the NRC for review of submittals and for preparation of regulatory positions, such as Regulatory Guide 1.99, on radiation effects on reactor vessel materials and the PTS rule. The program contains the elements of benchmarks, test and power reactor validation irradiation, embrittlement-damage analysis, and standards. These benchmarks derive from a very carefully controlled series of experiments representing reactor configurations, the results of which provide a calibration mark for the proof of accuracy of future predictions and analyses. In each case, theoretical calculations are made of the configuration (frequently by several laboratories using different methods and data bases) followed by extensive measurements of dosimetry, both during the irradiation and after the experiment is shut down. Further, determination of changes in metallurgical properties are often included. Discrepancies between calculations and measurements are resolved, and the entire experiment is documented for future use. The benchmarks that have been or are being produced are as follows:

- PCA (Pool Critical Assembly). A physics-dosimetry mockup of an LWR thermal shield/water gap/vessel wall that allows for reactor physics calculations to predict neutron flux and spectrum in this environment and extensive dosimetry measurements to verify the calculation. This work is completed.
- PSF (Pool-Side Facility). A mockup that duplicates the PCA except for the placement of the surveillance capsules and the use of fracture specimens simulating the through-thickness of a pressure vessel wall. Dosimetry and embrittlement measurements are completed, and a blind test of embrittlement predictions by individuals both here and abroad is essentially completed.
- SDMF (Surveillance Dosimetry Measurement Facility). A series of mockups to determine the effect of the presence of the surveillance capsule itself upon the accuracy of fluence measurements. The study of typical Westinghouse capsules is completed, and the study of the Babcock & Wilcox and the Combustion Engineering capsules are nearing completion.
- VENUS. A program undertaken by the Belgian Nuclear Laboratory at Mol, Belgium, with support provided by the NRC, to predict and measure the flux spectrum and intensity from the source in the last row of fuel elements completely out into the vessel wall. The experimental work is essentially completed. A follow-on experiment will simulate a low-leakage core being applied by certain utilities to minimize peaks in flux profiles, thus extending the operating life of the vessel.

- NESDIP (Nestor Dosimetry Improvement Program). A cooperative program with the UKAEA, Winfirth Laboratory in the United Kingdom, to establish benchmarks for dosimetry measurements in the cavity between the reactor vessel wall and the biological shielding. This element of the program will be completed in 1987.

Besides developing the benchmarks, this program is supplying dosimetry and performing calculations for use at several power reactors (H.B. Robinson (S.C.), Crystal River Unit 3 (Fla.), Turkey Point Unit 3 (Fla.), and Maine Yankee) that typify the various kinds of commercial reactors. Measurements and calculation procedures developed through use of the more simplified benchmarks will be further verified in these real environments. All elements of the surveillance dosimetry program will be completed in 1988.

Steam Generators

The Steam Generator Group Project at Battelle-Pacific Northwest Laboratories (PNL) works with a steam generator retired from service in an actual PWR facility as its test bed for research on a number of licensing, safety, and reliability issues.

In 1985, repair procedures for degraded tubing and antivibration bar removal and replacement procedures were demonstrated. During 1984 and 1985, the program concentrated on a series of nondestructive examinations (NDEs) of generator tubes and data analyses to provide input for evaluating both the reliability and repeatability of examinations for detecting and sizing various kinds of flaws and also to ascertain the best current methods available for inservice inspections. A large variability in the detection and sizing of various flaws has been noted between inspection teams and analysts.

Work in 1986 concentrated on developing procedures and the removal of approximately 550 tube samples from locations throughout the generator. This was a complex, time-consuming task, since the sample tubes from many locations in a severely degraded, radioactive steam generator had to be removed without physically changing their nature. The tubes are to be used for final validation testing of burst pressure capability (predictive models developed in an earlier phase of the program), for validation of NDE through metallographic examination, and comparison of results with the in-situ NDE conducted in 1984 and 1985. In 1986, some of the factors that contribute to variability and unreliability of NDE results were identified and recommendations developed for improving personnel training and qualification, data analyses procedures, and test methods. During 1987, the recommendations will be incorporated in revisions of the ASME Code requirements for inspection of steam generator tubes. The program will be completed in 1987, including metallographic validation work and burst- and leak-rate testing of degraded tubes. The information generated in this program will form the basis for developing, in 1987, revisions to two regulatory guides; one on inservice inspection requirements for steam generator tubing (sampling plans, number of tubes, period of inspection, and reliability required) and the other on tube-plugging criteria.

Piping

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

NRC research in this area has been directed toward developing the capability to predict stress corrosion cracking in BWRs and to verify the acceptability of proposed fixes. Proposed solutions to the problem include procedures that produce a more favorable residual stress state on the inner surface of the weldment, materials that are more resistant to stress corrosion cracking, and changes in the reactor coolant environment that decrease the susceptibility to cracking. In the past year, the research has included (1) the evaluation of stress improvement techniques, (2) studies of the stress corrosion cracking susceptibility of alternative piping materials, (3) the development of crack growth data for alternative materials and for pipe with weld overlay repair, and (4) studies on the effect of environment and temperature on stress corrosion cracking.

Analyses have been performed on weldments treated by the mechanical stress improvement process developed to eliminate tensile stresses at the inside surface of pipe. The weldments were evaluated nondestructively and metallographically, and measurements of residual stresses were made. The residual stress measurements indicate that the axial stress on the inner surface of the pipe was very strongly compressive after stress-improvement treatment. Some minor cracking observed in the weldment was attributed to chloride-induced stress corrosion cracking in the vicinity of localized tensile stresses. A 28-inch-diameter pipe weldment that had been treated with the process but not exposed to chlorides is being examined to resolve the issue of cracking in the previous weldments.

The stress corrosion susceptibility of the base metal and weldments of type 347 stainless steel was investigated in constant extension rate tensile experiments in high-temperature water containing dissolved oxygen and sulfates at low concentrations. Sulfate was added as an impurity that could potentially exist in a BWR primary coolant. The stress corrosion cracking susceptibility of type 347 stainless steel appears very similar to that of type 316 NG stainless steel, i.e., the material is susceptible to transgranular stress corrosion cracking in impurity environments at low strain rates.

Additional data have been analyzed in terms of the phenomenological model for stress corrosion cracking, which is based on a simplified elastic-plastic fracture mechanics analysis and a slip-dislocation mechanism of crack advance. The capability of the model to correlate crack growth results for various steels of interest has been demonstrated.

The process of crack growth in weld-overlay repairs of cracked pipe has been studied in simulated BWR environments and at low strain rates. The test specimens were fabricated, using standard industrial practice, in such a manner that the crack propagates through the original sensitized pipe material into the weld clad overlay. The results of the experiment indicate that the crack growth rate in the weld overlay is virtually the same as in the underlying pipe material, although the results have as yet not been verified metallographically.

In a program to evaluate the effects of welding and weld repairs of stainless steel piping, data have been developed through 1986 on temperature, strain, sensitization, and cold work accumulated on the inside surfaces of piping as a result of welding. These data are to be used to evaluate susceptibility to cracking of welded stainless steel piping. Since sensitization is a main factor in the susceptibility to cracking of stainless steel piping in reactor environments, a model was developed and validated in 1986 for predicting the level of sensitization in welded stainless steel pipe as a function of material composition, processing history, and welding parameters, including thermal cycles and accumulated strain. This information can be used both for setting criteria for acceptable welding and for optimizing welding procedures to minimize sensitization.

A thermal aging program was initiated in 1982 to evaluate the long-term effects on degradation of toughness in cast stainless steel as a function of time and temperature of exposure and the material composition. Through 1986, results have been accumulating to allow a quantitative evaluation of the degree and significance of toughness loss at reactor operating temperatures and operational times. Also, the mechanisms responsible for the toughness loss are being identified by evaluating both laboratory-exposed specimens and specimens removed from actual components in nuclear power plants. During 1986, a possible heat treatment was discovered for the recovery of toughness loss.

Piping Fracture Mechanics. The NRC's research activities in the piping fracture mechanics area have approached the topic from a broad perspective, with three laboratories contributing to the effort during the report period—David Taylor Naval Ship Research and Development Center (DTNSRDC), at Annapolis, Md.; Battelle Columbus Laboratories in Columbus, Ohio; and the Materials Engineering Associates, Inc. (MEA), in Lanham, Md. The research has contributed to further evaluation of piping analysis procedures, identifying areas where the intended margins are not being achieved, and has contributed to the technical basis for implementation of the General Design Criterion 4 (GDC-4) modification to accept leak before break.

The DTNSRDC program has resulted in the publication of several reports describing their pipe fracture testing and the development of laboratory specimen test methods that are suitable for standardization. This work has also identified a slight discrepancy in the ASME Boiler and Pressure Code, Section XI, paragraph IWB-3640, technical basis, in that one of the material parameters used in that technical basis is not fully supported by the actual material property data. This information was relayed to the pertinent Code group for their consideration. The focus of the program has shifted from the behavior of stainless steel pipe to the fracture behavior of welds in ferritic steel pipe. This work is expected to contribute to the technical basis for revising paragraph IWB-3620 of Section XI.

The MEA effort in preparing a material data base for use in piping fracture mechanics analyses has moved forward significantly in the last year. The computerized data base is now accessible, and the quantity of data included in the data base is being increased with the addition of data from other NRC contractors. The data base will be useful to the NRC and licensees alike in the preparation of justification for leak-before-break applications where archival materials are not available for testing.

The Battelle Columbus effort continues to be the mainstay of the piping fracture mechanics effort. The Degraded Piping Program-Phase II has provided a reasonably comprehensive data base of pipe fracture experimental results for quasi-static loading. These data have been used in evaluating the margin inherent in the IWB-3640 procedures for flaw evaluation. That margin has been shown to be less than anticipated in certain circumstances. This result has been identified to the pertinent Code groups and is currently being reviewed. The program results for ferritic steels and weldments are contributing to the revision of IWB-3620. Further, the program results have identified several areas where the criteria for accepting leak-before-break applications needed to be strengthened to ensure a uniformly conservative implementation of the rule change.

Battelle has also been instrumental in the formation of the International Piping Integrity Research Group. This group is composed of the NRC, EPRI, and several other governments and industrial organizations. The formation of this group was motivated by the clear need for, and international consensus on, leak-before-break regulations. The group provides a funding mechanism for the necessary research, as well as a forum for developing the needed consensus. The initial research is directed toward evaluating the effects of seismic/dynamic loading on the fracture behavior of nuclear piping. The work also addresses the accuracy and ranges of validity of the leak-rate estimation models.

The overall piping research activity has contributed to the evaluation of rule changes, such as the GDC-4 modification, and to the evaluation of revisions to the standard review plans associated with rule changes. Further, the NRC's effort has fostered a better understanding of international activities in establishing leak-before-break regulations. For example, the NRC hosted an international seminar addressing leak-before-break policies and supporting research. The seminar was attended by over 100 persons representing 13 nations. The interchanges among the participants led to a clearer picture of international regulations and pointed up a definite need for an international consensus on the technical basis for such regulations.

Electrical and Mechanical Components

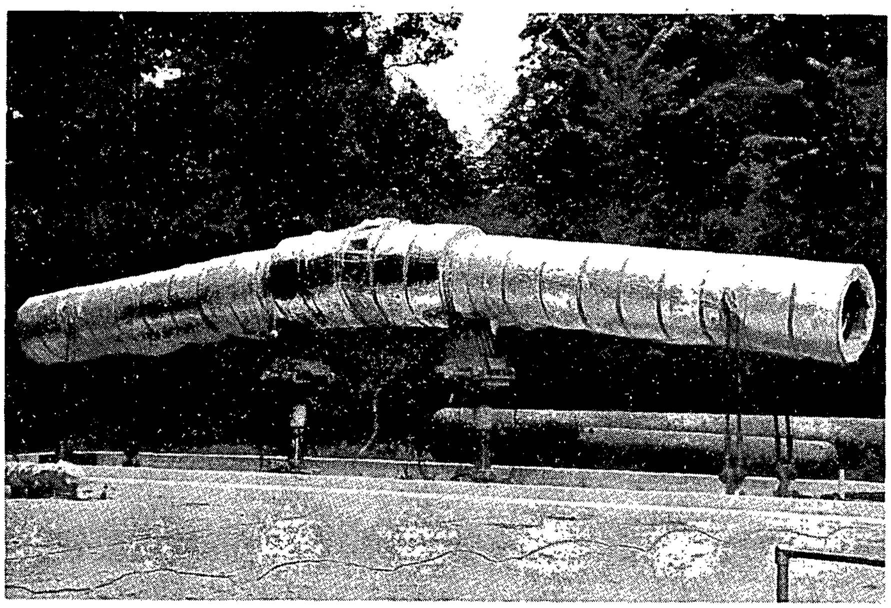
The purpose of this research is to identify and resolve safety issues related to the effects of plant aging (time-dependent degradation) and the anticipated requests for operating license renewals.

Studies were completed in 1986 on specific components, in order to (1) identify various failure mechanisms resulting from aging and service wear; (2) recommend maintenance, testing, and surveillance practices to ensure operational readiness; and (3) establish degradation patterns for use in detecting incipient failures. Based on information derived from operating experience records, nuclear industry reports, and manufacturers' supplied information, the initial engineering studies were completed, and reports were issued for check valves (NUREG/CR-4302), auxiliary feedwater pumps (NUREG/CR-4597), emergency diesel generators (NUREG/CR-4590), and battery chargers and inverters (NUREG/CR-4564).

A study was completed during the report period on the inspection, surveillance, and monitoring of pressure transmitters (NUREG/CR-4257). This report describes the stresses that cause age-related defects in pressure transmitters and makes recommendations for detecting and evaluating such defects. Criteria for determining the capability of pressure transmitters to withstand design basis accident conditions are also discussed in the report. It is shown in the report that a combination of operability monitoring and condition monitoring may improve the probability for withstanding design basis accident conditions.

A technical report (NUREG/CR-4740) has been prepared to describe the results of a review of a reactor trip system (RTS) and an engineered safety feature actuating system (ESFAS). Analyzed in the report are the operating experiences reported in licensee event reports, the Nuclear Power Experience data base, the Nuclear Plant Reliability Data System, and selected plant records. The report identifies materials susceptible to aging, stresses, environmental factors, and failure modes for the RTS and ESFAS, as typical systems. Functional indicators of degradation are listed, testing requirements are evaluated, and safety issues are discussed.

A preliminary assessment of the aging of selected major light-water-reactor components and structures was completed (NUREG/CR-4731, draft). The information developed will be used for resolving safety issues in consideration of license renewals. The kinds of stresses, possible degradation sites and mechanisms, potential failure modes, and currently used and emerging inservice inspection (ISI) and life assessment methods are discussed for pressure vessels, PWR containment structures, PWR coolant piping, PWR steam generators, BWR recirculation piping, and reactor pressure vessel supports. Major technical issues, including recommendations for advanced ISI and life assessment methods, are also discussed.



NRC-sponsored pipe fracture tests on typical reactor primary loop piping were performed during 1986 at Battelle Columbus laboratories in Columbus, Ohio. The specimen shown here was fabricated from cold leg piping obtained from a canceled nuclear power plant. The pipe (37 inches in diameter and 3.5 inches thick) was intentionally flawed, heated to typical operating temperature, and subjected to bending loads to produce crack extension. The resulting crack growth has raised questions about the role played by the microstructure of ferritic steel, as contrasted from stainless steel.

An inter-office working group, the Technical Integration Review Group on Aging and Life Extension (TIRGALEX), was established to facilitate planning and integration of agency activities related to plant aging and license renewal beyond the designated 40-year term. The objectives of TIRGALEX are to identify safety and regulatory issues related to plant aging and license renewal, and to develop a plan to integrate agency and external activities to address safety and regulatory issues.

Nondestructive Examination

This program includes studies of improved methods for the detection and sizing of flaws during inservice inspection of carbon steel, wrought and cast stainless steel piping, and pressure vessels. It also includes studies of on-line continuous monitoring techniques (using acoustic emission) for crack growth and leak detection.

Flaw Inspection by Ultrasonic Methods. A vastly improved method for the detection and sizing of flaws in BWR stainless steel piping and other reactor primary components—called SAFT-UT (Synthetic Aperture Focusing Technique for Ultrasonic Testing)—is a computer-based testing procedure that produces high-resolution, three-dimensional images of cracks and other material flaws. PNL is adapting this method for field inspections from earlier work done at the University of Michigan, where the technique was demonstrated in the laboratory. In 1985, a field system was assembled with its own computer and taken to a field site for a successful demonstration. Also a technique was developed along with the SAFT processing for optimum detection and sizing of intergranular stress corrosion cracking in stainless steel piping. In 1986, a special purpose SAFT processor prototype was created which will allow real-time processing of data and imaging of flaws on the spot as the inspection is being conducted in the field. This will allow for decisions to be made on the presence and nature of flaws in components while the inspection is being conducted. Also in 1986, a cooperative agreement was developed with Combustion Engineering for their technical and financial participation in the program and for commercialization and field implementation of the technology. Finally, extensive laboratory work was conducted to determine, as well as to evaluate, different ultrasonic parameters for optimization of procedures for calibration, inspection and data analysis, and interpretation for field application.

Flaw Inspection of Centrifugally Cast Stainless Steel. Although the ASME Code requires that cast stainless steel piping be inspected, the current inspection techniques have not been fully satisfactory. In 1985, an evaluation of various inspection techniques used worldwide was undertaken, in cooperation with the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development and the Commission of the European Communities (CEC), to identify the most promising ones for the inspection of cast stainless steels. Data analysis from these inspections of flawed cast stainless steel specimens was conducted in 1986 and showed that, although the techniques for inspection of cast stainless steel are not very reliable and produce a high rate of false calls, some techniques have promise for improving inspection if the false call rate can be minimized. Also in 1986, an extensive research and evaluation plan was prepared for developing improved techniques and more thoroughly evaluating the inservice reliability of piping materials in general, including cast stainless steels, under NEA/CEC cooperation.

Inservice Inspection System Qualification. Research work and field experience over the last several years have indicated that inservice inspection, as currently practiced, is not always reliable. NRC research results have indicated a need for qualification of the entire inservice inspection process, including the personnel, procedures and equipment. Research has been conducted and criteria developed for proper qualification of the system. Subjects of greatest relevance for qualification are the education, experience, and examination requirements of inspection personnel; procedure requirements; equipment measurements; and evaluation and requirements for actual performance testing of the total personnel-procedure-equipment aggregation, using actual components and realistic flaws, as a pre-requisite to conducting an inspection on reactors. Based on the research conducted at PNL, criteria were prepared and reviewed by the NRC and the industry in 1985. These criteria formed the basis for NRC cooperation with the ASME in developing and implementing inservice inspection system qualification requirements in the ASME Code. Accordingly, in 1985 and 1986, the NRC worked with designated Code committees to develop documents for incorporation into the Code of recommended qualification requirements. At the close of the report period, two Code Cases were being assessed by the appropriate committees for acceptance into the Code.

Continuous Monitoring for Crack Growth and Leak Detection. Research has been under way at PNL using acoustic emission (AE) for the continuous on-line monitoring of reactors to detect and locate crack growth and to estimate the severity of the cracking from the AE signals. Up to 1986, a large body of laboratory and some field data have been developed to establish feasibility and methodology for inservice monitoring of reactors and for evaluation of data. In 1985 and 1986, a great deal of data from an intermediate-scale test using a pressure vessel—which was conducted over a one-year period in the Federal Republic of Germany and which produced crack growth under simulated reactor operating conditions—was thoroughly evaluated to upgrade and validate existing models and technology. The evaluation established that the continuous monitoring of reactors for crack growth is feasible. In 1986, arrangements were completed with the Philadelphia Electric Company for the actual monitoring of stainless steel piping at the Peach Bottom Unit 3 (Pa.) reactor. This work will start in late 1986 or in January 1987. Also in 1986, an agreement was concluded with the Tennessee Valley Authority (TVA) for their technical and financial contribution to the research program to allow for the final field validation and commercial implementation of the technology. TVA will allow use of the Watts Bar Unit 1 (Tenn.) reactor for actual inservice monitoring of the reactor during power operation. Licensing of the reactor and start of the AE monitoring is expected in 1987. The availability and proper use of this technology will mean that reactors can be continuously monitored and that any cracks that develop can be continuously detected and evaluated. In this way, proper and timely action can be taken to avoid extensive crack growth, so that leakage or failure would never occur.

Decommissioning

The NRC continued to develop an information base for decommissioning light-water reactors and other nuclear facilities with four reports published during the year. They cover (1) classification of nuclear fuel cycle facility decommissioning waste (NUREG/CR-4579), (2) residual radionuclide contamination within and around commercial nuclear power plants (NUREG/CR-4289), (3) evaluation of *nuclear facility decommissioning projects* (NUREG/CR-4316 and -4315, Vols. 1, 2, 3, and 9) covering respectively the various relevant aspects of Humboldt Bay (Cal.) and TMI-2 (Pa.) decontamination, and (4) identification and evaluation of facilitation techniques for decommissioning light-water reactors (NUREG/CR-3587). The proceedings of the International Nuclear Reactor Planning Conference were published as NUREG/CP-0068.

Reports are in preparation on the decommissioning activities at the Humboldt Bay and Shippingport (Pa.) reactors. A collection of person-hour, radioactive dose, and waste quantity data is included in the reports.

Comment letters were received from 143 organizations and individuals on the proposed amendments to the regulations setting forth technical and financial criteria for decommissioning licensed facilities. In 1986, these letters were categorized and analyzed and, based on this analysis, a draft of the final rule amendments, the supplementary information thereto, the Final Generic Environmental Impact Statement, and the regulatory analysis were being prepared for division review at the end of the fiscal year.

Proposed amendments to 10 CFR Parts 30, 40, 50, 70, and 72 were published in June 1986 setting forth requirements that licensees notify the NRC in the event they are involved in bankruptcy proceedings. The purpose of this rulemaking is to have rules in place that require prompt notification of NRC of licensee bankruptcy, thus alerting NRC in a timely manner. The NRC can then take necessary actions to deal with potential hazards to public health and safety that may be posed by a licensee that does not have the financial resources to properly handle licensed radioactive material or to clean up possible contamination.

NUCLEAR POWER PLANT SAFETY

Earth Sciences

Risk studies on nuclear power plants have indicated that earthquakes are to be considered among the principal potential causes of accidents at nuclear power plants. For that reason, it is important that the NRC be able to define the potential for an earthquake at nuclear plant sites and to determine just what effects an earthquake, should it occur, would have on the plant and its safety systems.

A major focus of the NRC research program in geology and seismology continues to be identifying and defining potential earthquake sources or source zones in the Eastern United States and using that information in assessing seismic hazards with respect to nuclear power plants. Many unknowns exist regarding these issues, including seismic zonation, source mechanisms, characteristics of ground motions, and site-specific response. The NRC is addressing these uncertainties through research that encompasses sustained seismic monitoring, neotectonic investigations, exploring the earth's crust at hypocentral depths, and conducting ground motion studies.

In the western United States, the NRC is funding research with regard to defining the characteristics of ground motions.

Because of reductions in the Federal budget, the NRC has negotiated an inter-agency agreement with the U.S. Geological Survey (USGS) to jointly support the operation of the seismic networks in future years.

Northeastern Neotectonics. One of the NRC's primary goals in its seismic research program is to reduce uncertainty in the estimation of seismic hazard in the eastern United States. Toward that end, the NRC has contracted with Columbia University to investigate seismically active regions in the northeast for evidence of Quaternary surface or near-surface movements. The researchers are using several approaches in addition to classical field techniques. They are adapting newer techniques developed for neotectonic investigations in California. These include terrace correlation studies, paleoseismic evidence in sediment deposits, and ground-penetrating radar. They are also developing methods for determining crustal deformation in areas of low strain rates along the coastline, such as comparing tidal data and prehistoric sea level data from fossil salt marshes. The areas being investigated include the Lancaster, Pa., seismic zone, the Goodnow, New York-Adirondack seismic zone, the Lower Hudson Valley-Eastern Newark Basin seismic zone, and the New Jersey Coastal Plain. Whether or not Quaternary crustal movements are detected, the paleoseismic investigation, which was applied successfully in the Charleston, S.C., area in providing isotopic dates of large prehistoric seismic events, has the potential of providing deterministic guidance for return periods of large earthquakes in Northeastern United States. This would be a big step toward the goal of reducing uncertainty in seismic hazard assessments in the Eastern United States.

Paleoliquefaction Investigations. The NRC has funded over the past few years studies by the USGS and the University of South Carolina of soil deformed by liquefaction during the 1886 earthquake and of similar, but older, features (paleoliquefaction features) that were apparently formed by prehistoric earthquakes of about the same size. These investigations suggest recurrence intervals between 1,000 and 2,000 years for earthquakes of the same size as, or greater than, the Charleston event. To support the NRC position that the Charleston seismic area is unique, the NRC has encouraged expanding the area of investigation to find paleoliquefaction features elsewhere on the Atlantic Coastal Plain. The USGS has identified paleoliquefaction features beyond the immediate Charleston earthquake area and postulates that either a much larger earthquake occurred in the Charleston epicentral area, or earthquakes of similar size occurred prehistorically at other locations along the southern Atlantic coast. In 1986, the NRC awarded a research contract to Ebasco Services Incorporated to look for paleoliquefaction structures throughout the Atlantic Coastal Plain. The purpose is to either confirm that Charleston is tectonically unique or to obtain data to serve as a basis for reassessing the earthquake design of eastern nuclear plants if necessary.

Charleston Studies. A special study of the Charleston, S.C., area, being performed by Law Engineering Testing Co., is continuing with data gathering and development of computer models describing stress, strain, and rheology of the earth's crust in the area. A magnetotelluric survey conducted for this project found indications of crustal layering that can be correlated with COCORP (Continental Coastal Reflection Program) seismic reflection data. The survey indicates a conductive upper layer corresponding to coastal sediments, a 5-to-18-km thick resistive layer corresponding to a non-reflection zone on COCORP data, and a conductor at depth corresponding to reflections on COCORP data that are interpreted as sediments. A marked thinning of the resistive layer to the north may indicate the boundary of the southern Georgia rift.

A study of historical seismicity in the southeastern United States by Lamont-Doherty Geological Observatory leads to the conclusion that the level of seismicity at Charleston before the 1886 earthquake may have been low. In parallel with the findings on the Meers fault, the conclusion raises questions about the time stationariness of seismicity.

Northeastern Ohio Earthquake of January 31, 1986. The ongoing NRC interest in developing an understanding of the causes, sources, and characteristics of earthquakes in the Eastern United States was given further impetus when a magnitude 5 event occurred on January 31, 1986, in northeastern Ohio, about 18 kilometers south of the Perry nuclear power plant. Seismological data were obtained from the mainshock, and portable seismographic networks subsequently deployed by the utility's consultants and the USGS recorded 13 aftershocks. The NRC sponsored an indepth investigation by the USGS, prompted by three unusual features of this event:

(1) The mainshock was the largest magnitude earthquake known to have occurred in northeastern Ohio.

(2) Unusually high frequencies of ground motion were recorded for both the mainshock and aftershocks at considerable distances from the epicenter.

(3) The presence of three deep fluid injection wells within 15 kilometers of the epicenter suggested the possibility of their involvement in inducing the seismicity.

In a report of this study, published as USGS Open-File Report 86-331, the investigators concluded that the mainshock resulted from right-lateral strike-slip motion at depths of two-to-seven kilometers in response to regional stresses oriented northeast-southwest; that the high-frequency ground motion of 20 Hz resulted from some combination of source mechanism and path and site effects; that the injection wells are unlikely to have induced the seismicity; and that the magnitude, depth, and maximum compressive stress direction were consistent with other events that have occurred in the eastern United States.

Meers Fault Studies. Continuing investigations of the Meers fault in Oklahoma have shown, using several lines of evidence, that the fault has definitely been active in geologically very recent times. Carbon-14 dating has bracketed the date of the last movement between 600 and 1,600 years before the present, with a probable age of 1,100-1,200 years. Additional trenching and test pit excavation have shown that recent movement on the fault not only includes vertical displacement of up to five meters but also a left-lateral strike-slip component several times as large.

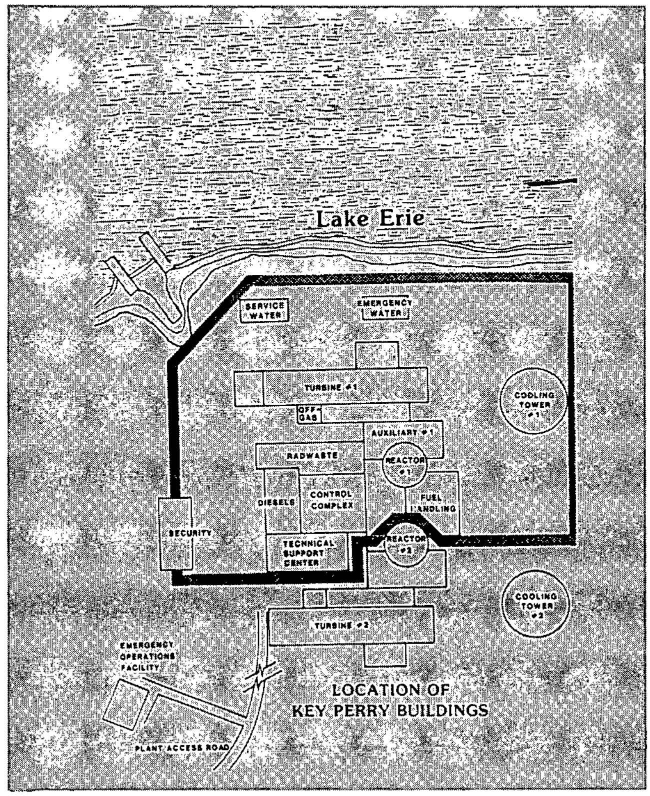
Up to this time, no seismic activity has been found associated with the fault. This may indicate that historical records of seismicity have not adequately described the seismic potential of a given area.

Analyses of the tectonic framework in which the Meers fault is located reveal that, even though indications of recent movements are limited to a length of 25 kilometers, the fault is located on a major northwest trending zone, which may be called the Amarillo-Wichita-Arbuckle uplift. Along the strike of this zone, additional indications of possible recent movement have been found in the Wichita Valley of Arkansas, along the Arbuckle uplift. The Amarillo-Wichita-Arbuckle uplift zone, as large as it is, may be further connected with the Rio Grande rift to the west and with the Ouachita thrust belt and possibly with the Reelfoot rift in the east. Tectonic activity along this whole zone seems to have begun with Eocambrian rifting. It continued in the Paleozoic era with strike-slip movement and continued with reactivation of faulting in recent times, as shown by the Meers fault.

Seismic Margins

Seismic Design Margins. The NRC has sponsored the development of review procedures to assess the capability of nuclear power plants to withstand earthquakes greater than the design bases. This effort addresses both the Charleston earthquake issue and the implementation of the Severe Accident Policy. The seismic design margins approach outlined in NUREG/CR-4334 and NUREG/CR-4482 makes use of the concept of "high confidence of low probability of failure" to provide lower bound (licensable) statements about the strengths of components and systems. A systems and components screening approach is used to focus the basis review on the safety functions really essential to mitigating core melt scenarios for earthquakes with peak ground accelerations up to half of gravity. A trial plant review of Maine Yankee Atomic Power was initiated in 1986 to demonstrate the use of these procedures and to provide a basis for improving them as a result of this review experience. This study should be completed in February 1987. In 1987, a trial seismic margins review of a BWR will also be initiated.

BWR Risk Assessment. The research program to assess the seismic risk of the LaSalle Unit 2 (Ill.) nuclear power plant continued. During 1986, calculations to determine the seismic risk using simplified and detailed logic models were performed. Logic models consist of event trees and fault trees that describe the ways by which a system can fail and its consequences, e.g., radioactive release. The detailed logic models had been developed as part of the Risk Methods Integration and Evaluation Program (see *1985 NRC Annual Report,* p. 157) and modified to include earthquake-initiated events. The final report, to be issued during the second quarter of next year, will compare the two calculational methods.



The magnitude-5 earthquake that occurred in January 1986 in Ohio (see Chapter 2) resulted in NRC sponsorship of an investigation by the U.S. Geological Survey into the precise nature of the quake, and a series of meetings between NRC safety experts and officials of the Perry nuclear power plant, some 18 miles north of the earthquake epicenter. Shown is a diagram of the Perry plant layout.

Fragilities and Responses

Soil Response to Earthquakes. The research program to validate dynamic effective stress models that would be capable of predicting soil settlement resulting from seismically induced liquefaction continues. The objective of the research, being conducted by the Army Corps of Engineers, is to evaluate various seismic settlement models identified in a previous phase of this project and reported in NUREG/CR-3380. During 1986, two two-dimensional plain-strain centrifuge experiments, simulating massive embedded structures such as nuclear power plant buildings, were conducted at Cambridge University in England. The purpose of these experiments was to determine differences in the dynamic effects predicted by two- and three-dimensional soil-structure interaction models and the effects of dry and saturated foundation soil conditions. Initial test results indicate that there is a zone around the structure where foundation soil loosening occurs, probably because of differences in the response of the soil and structure during the earthquake. The results also show deficiencies in the widely used FLUSH code for predicting dynamic soil settlement and motions transmitted through the soil and structure, and this is attributed mainly to pore-water pressure effects. Analyses of the experimental data are continuing, and expansion of the research project to consider modeling three-dimensional effects is planned.

Seismic Category I Structures Program. The dynamic testing of the first large, reinforced concrete model representing a portion of a nuclear power plant building (i.e., shear wall and floor segment) was performed this year. This series of static and dynamic tests began with small models (1-inch-thick shear wall) in 1985 and will continue through 1987 with large models (up to six-inch-thick wall). Different size models are used to demonstrate the applicability of extrapolating model test results to actual nuclear power plant non-containment buildings. The purpose of this test series is to investigate the large differences observed when analytical predictions of building responses are compared with experimental data. Preliminary evaluation of the data obtained in 1986 continues to support previous experimental observations, that is, analytically derived stiffness could be off by a factor of four and frequency off by a factor of two. The overall goal of this program is to assess (1) the ability of Category I structures other than the containment to sustain earthquake motions in excess of their original design bases, and (2) the effect that the changed building response has on the criteria used in the design of piping and equipment.

Seismic Component Fragilities. Based primarily on extensive evaluations of test reports from several independent sources, it has been demonstrated that seismic resistance of electrical components at nuclear power plants is greater than previously estimated. This conclusion is presently limited to motor control centers and switchgear that are properly anchored and maintained in conformance with technical guidelines. However, this effort is being expanded to include other electrical and mechanical components. The use of these results in future seismic margin studies and probabilistic risk assessments could diminish the significance of the seismic threat in the NRC's regulatory practices and licensing decision-making. In addition, new seismological insights that lead to increased seismic design bases, particularly in the eastern United States, need not require extensive backfitting for electrical components.

Standard Problems for Structural Computer Codes. A major portion of the work performed during 1986 concentrated on the soil-structure interaction (SSI) response of Category I structures under seismic loads. During the year, three volumes of a report (NUREG/CR-4588, Volumes 1, 2, and 3)—highlighting the effects of foundation liftoff, high ground-water table, and soil layering on SSI analytical methods—were published. Also, an SSI workshop was held in which information presented will be used to serve as a basis for resolution of Unresolved Safety Issue A-40, "Seismic Design Criteria." Proceedings of the workshop will be published in NUREG/CP-0054.

Finally, a review and evaluation of structural test data for Category I structures was initiated and will continue in future years. The purpose of this task is to define structural characteristics that are important to an evaluation of the peak seismic loading that can be sustained by a Category I structure (i.e., inherent seismic margin).

Design of Piping for Dynamic Loads. The design of piping for the inertial loading of seismic and other dynamic loads is a major factor in both the industry and regulatory costs associated with nuclear power plant design. The NRC Piping Review Committee expressed concerns in NUREG-1061 about the over-conservatisms of current design for inertial loading. In 1986, two major NRC-sponsored research programs addressed ways to reduce these overconservatisms.

(1) Pipe Damping Studies. The NRC-sponsored Pipe Damping Study Project at the Idaho National Engineering Laboratory (INEL) has been the major research contributor to date in the area of pipe damping. INEL testing and data evaluation has supported the development and endorsement of ASME Code Case N-411 by Revision 24 to Regulatory Guide 1.84 in 1986. Also in 1986, INEL completed testing and data evaluation of high-frequency and high-level pipe damping.

(2) *EPRI/NRC Piping and Fitting Dynamic Reliability Program.* Inertial loadings from dynamic events such as earthquakes are time-varying and have limited durations and energy content. Current ASME Code requirements evaluate these in the same manner as gravity and other sustained piping loads. Cross-sectional plastic collapse is assumed to be the dominant mode. However, an increasing amount of analytic and test data has shown that piping inertial loads behave differently from static loads. The margins-to-failure are greater than are predicted by current theoretical limit-load estimations, and ratcheting and fatigue appear to be the principal failure modes. If these new insights into dynamic piping failure can be demonstrated systematically and conclusively then significant changes can be made with regard to how the ASME Code sets limits on inertial stresses. This would dramatically change the nature of piping system design and could in turn reduce the number of snubbers used in nuclear power plants.

To address this issue, the NRC and EPRI are cooperating jointly in the Piping and Fitting Reliability Research Program. The joint EPRI/NRC program began in the spring of 1985 and will take approximately three years to complete. Tests of components, including elbows, tees, reducers, support connections, nozzles, and lugs, will be completed in late 1987, with the NRC and EPRI each funding 20 tests.

Based on the 12 tests completed as of October 1986, some preliminary conclusions can be drawn:

- Failure levels are much higher than previously believed—15 to 20 times higher than Level D allowable limits and measured movements are higher than theoretical collapse movements.
- Ratcheting and fatigue, not collapse, are the observed failure mechanisms for *all* cases that failed.
- Piping components bulged and cracked. They did not collapse and crimp. "Loss-of-flow" functionality failure did not occur.
- Equivalent linear damping values were computed to be much higher than design values (on the order of 30 percent of critical damping).

A separate but related NRC-sponsored pipe system test was conducted in 1986. The objectives of this test were to demonstrate the feasibility of failing a representative piping system under a high earthquakelike load and to provide information and insights needed in the test planning of the main EPRI/NRC program. The six-inch piping system withstood a 30g seismic input without rupturing the piping, although ratcheting was observed. This input was 15-to-20 times the Level D design limit. The system was eventually failed by applying sine wave inputs at the system's resonance frequence for a limited number of cycles.

Strain concentration and ratcheting were the cause of rupture.

Validation

Validation of Seismic Calculation Methods. Seismic probabilistic risk assessment (PRA) methods have been employed to clarify safety issues for nuclear power plants. The randomness of the seismic hazard, the uncertainties and variety of the data needed, and the inexactitude of the methodology raise questions of credibility with respect to the results of seismic PRAs. The objective of validation research is to obtain information that the NRC can use to develop criteria for judging predictions of the behavior of nuclear power plants subjected to large earthquakes and thereby improve the regulatory process. The predictive methods to be validated are used in both probabilistic and deterministic predictions.

The strategy is to engage in cooperative research programs in order to stretch available resources. The NRC is participating in the following three efforts:

(1) A soil-structure interaction experiment being performed in Taiwan by EPRI. Construction of a test structure and low-level tests with a mechanical shaker were completed in December 1985. Measurements of response during earthquakes are being taken and will be compared with predictions.

(2) The Phase II experiments being performed at the Heissdampfreaktor (HDR) facility in Kahl, West Germany, by Kernforschungszentrum Karlsruhe (KfK). In July 1986, the containment building was excited with a large mechanical shaker. Responses of a piping loop were measured for different support configurations. The measured responses will be compared with predictions.

(3) Tests of a 1/2-scale model of a PWR piping loop to be performed on the large shaker table in Tadotsu, Japan, by the Nuclear Power Engineering Test Center (NUPEC). In 1986, an agreement was completed to modify the Japanese piping loop so that it can be excited well into the inelastic range.

Other External Hazard Research

Severe Weather. Severe weather research in 1986 was primarily concerned with damage surveys of areas struck by tornadoes. The purpose of these damage surveys is to obtain information on the length and width of the tornado damage path and the severity of the tornado. Compilations of these data are needed to better define the risk to nuclear facilities posed by tornadoes.

Surface-Water Hydrology. The collection of meteorological and wave-surge measurements related to hurricanes by means of an instrument network along the Florida coast continues. This information will help improve models that predict flood levels during hurricanes so as to ensure that nuclear power plants in coastal areas are designed to withstand hurricane-induced flooding.

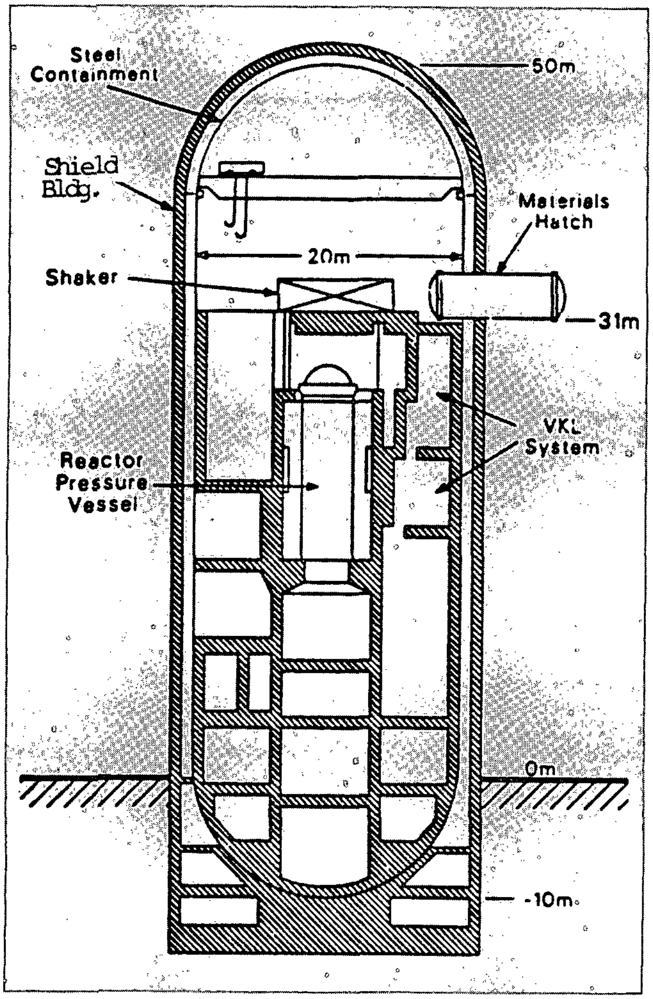
The development of a stochastic model for simulating long-return period floods, for use in nuclear power plant risk studies, continued in a Phase II validation study using watershed data. NUREG/CR-4496, issued in February 1986, documented the Phase I study on the development of a synthetic rainfall simulator used in a watershed simulation model that was calibrated by using stream flow data from the Russian River in California.

Equipment Qualification

Qualification of Electrical Equipment for Harsh Environment. The vulnerability of electrical penetration assemblies (EPAs) to failure during severe accidents was evaluated in 1986. An artificially 40-year pre-aged Westinghouse EPA (which uses epoxy electrical seals) was tested in a BWR Mark III containment drywell with a severe accident-simulated environment of 400xF and 75 psia for 10 days. In the second test, an artificially 40-year pre-aged CONAX EPA (which uses polysulfone electrical seals) was tested in a simulated BWR Mark I containment drywell with a severe accident-simulated environment of 700xF and 135 psia for nine days. Both EPAs had zero gas or steam leakage during and after exposure to the above severe accident environments, thus demonstrating containment integrity. However, the cables and connections to the EPAs inside the simulated containment test vessel were found to have degraded electrical resistance after about 18 hours into the BWR Mark III simulation test and eight hours into the BWR Mark I simulation test. Thus, radiation monitoring signals from within the containment and operator actions to mitigate the accident involving electrical equipment inside the containment (and hence requiring electrical control and power signals passing through the EPAs) would be uncertain after the above periods of time into a severe accident.

Environmental Qualification of Mechanical Equipment. This part of the equipment qualification program deals with the qualification of mechanical equipment subjected to loads resulting from temperature, pressure, humidity and radiation. Research completed during 1986 identified conditions under which reactor coolant pump face seals can suddenly open, and a leakage path leading to a small loss-of-coolant accident would result. With the absence of power to provide coolant to the seals and to the water makeup system pumps, there is a potential for uncovering the core as a result of this leakage. In addition to this seal stability effort, research was completed on determining the capability of 0-ring seals, which are also included in the reactor coolant pump seal assembly, to retain their sealing function.

This research is providing information to the resolution of the safety issue on reactor coolant pump seal failures. The



The NRC's continuing effort to develop criteria for judging predictions of nuclear power plant responses to seismic events has resulted in numerous cooperative research efforts within the U.S. and with foreign agencies. In July 1986, at the Heissdampfreacktor (HDR) facility in Kahl, West Germany, tests were run on a containment building shaken by mechanical means, as piping loops were measured in various support configurations. The figure above is a cross section of the HDR showing the shaker, experimental piping loop system (VKL), and the reactor vessel.

research results are also providing the NRC staff with the technical basis for evaluating the integrity of mechanical components under accident conditions.

Dynamic Qualification of Equipment. This part of the equipment qualification program deals with the dynamic (including seismic) qualification of mechanical and electrical equipment. Tests conducted in 1986 have demonstrated to the NRC staff that the operability and leak integrity of typical containment penetration valves are unaffected by large seismic loads. In addition, other tests have demonstrated that the combination of large seismic vibrations and flow-induced vibrations do not affect small gate valve operability. These results are contributing to the data base for identifying potential containment leakage paths during earthquakes and for determining multi-load characteristics that may be required for qualifying specific valves, respectively.

Another phase of the equipment qualification effort has resulted in identifying the population and associated qualification experience of isolation valves in specific high-energy piping of BWRs. This information is being used to determine whether these valves have been qualified to the appropriate levels to ensure that they will close against high flows in the event of a downstream pipe break. Research continues to focus on resolving this safety issue.

During 1986, effort was also devoted to working with the ASME Committee on Valve Qualification to incorporate the results of NRC research on butterfly valves into the standard on valve qualification. The specific change deals with demonstrating actuator torque capability for these kinds of valves.

The above effort is providing the NRC staff with a basis for evaluating the integrity of valves when subjected to various kinds of dynamic loads.

Spent Fuel Storage

Issued for public comment was the revision to Part 72 that incorporates (1) the effect of experience in using Part 72 in licensing Independent Spent Fuel Storage Installations and (2) the rule changes needed to extend the rule provisions to cover the licensing criteria for both short- and long-term storage of spent fuel and high-level radioactive waste in Monitored Retrievable Storage facilities to be constructed and operated by the Department of Energy.

NUCLEAR POWER PLANT RISK AND RELIABILITY

Risk and Reliability Applications

Guidance for Inspections. The Plant Risk Status Information Management (PRISIM) system was initiated in 1983 to develop a method for presenting probabilistic risk assessment (PRA) information in a form that could be used by Inspection and Enforcement staff to assist in setting priorities and planning their activities. After considerable study of NRC's inspection program—through discussions at the Regional Offices and extended field trips to witness resident inspector activities—a method was developed and demonstrated to the point where a field test was warranted. A PRISIM system, operable on an IBM personal computer, was constructed for the Arkansas Nuclear One Unit 1 (ANO-1) nuclear power plant and late in fiscal year 1986 was provided to both the site resident inspectors and Region IV for test and evaluation. Since considerable effort was made to ensure that the PRISIM system reflects the current

ANO-1 design and operational characteristics, it can provide the inspectors with the capability to evaluate the plant's actual risk status at any moment in time as a result of the existing plant status and then to determine what specific activities to pursue in order to address the specific change in plant risk existing at that time.

Development is nearly completed on a licensing version of PRISIM (designated N-PRISIM) for NRR project managers. N-PRISIM will provide project managers with immediate details of system layouts, the risk importance of each system and its individual components, and the importance of technical specifications for any set of plant conditions. In particular, N-PRISIM will provide the capability to vary parameters such as Allowed Outage Times (AOTs), Surveillance Test Intervals (STIs), and component reliability. An N-PRISIM system for ANO-1 will undergo user test and evaluation during 1987. Current plans include developing additional PRISIM-type systems (Peach Bottom Unit 2 (Pa.) is under way) and a related information base for use by NRC management in their long-range planning decisions.

Technical Specifications Evaluation. Technical specifications of nuclear power plants contain the requirements of STIs of various standby components and establish the AOT for on-line maintenance and repair of components detected as inoperable during testing. The purpose of technical specifications is to ensure that standby safety systems and components are available for accident mitigation conditions. Based on Executive Director for Operations (EDO) task force recommendations for enhancing plant safety through technical specifications (NUREG-1024), the Procedures for Evaluating Technical Specifications (PETS) program was initiated in 1984 to develop and demonstrate methodologies to use reliability and risk techniques in evaluating the scope, detailed requirements, and safety impact of plant technical specifications.

An early product of this effort was a detailed computer code for assessing system unavailability. This code (FRANTIC) has received considerable attention, both foreign and domestic, and has been used by industry in submitting technical specification-related analyses to the NRC. During 1986, the primary focus was to develop technical guidance for NRR to ensure uniformity in licensee submittals of technical specifications. A three-volume draft report discussed the regulatory decision process in AOT and STI modifications, as well as a methodology for AOT and STI risk evaluations. Other draft reports addressed the use of cumulative downtime for regulatory implementation in resolving Generic Issues B-56 (Diesel Reliability) and B-61 (Allowable ECCS Equipment Outage Periods) and an examination of diesel testing strategies involved in B-56. This program has maintained close coordination with industry through the LWR Owners Groups, Atomic Industrial Forum, and Electric Power Research Institute.

Generic Safety Issues Evaluation. The Committee to Review Generic Requirements has stated that there is a need for improved integration and accounting of generic safety issues evaluation, in order to re-establish the plant risk base in light of previous regulatory activities so that value-impact analyses will provide an accurate basis for proper resolution of generic issues. The System Analysis and Risk Assessment (SARA) program was initiated in 1985 to meet the need to take into account the fact that the risk base has changed over time with the imposition of many generic safety issues, and that the cumulative backfitting burden on licensees is quite large. Since this requirement covers a broad spectrum of PRA activities, SARA is also being designed as a flexible tool to support different levels of users who require risk and reliability information for decision-making and regulatory analyses. The system operates on an IBM personal computer and demonstrations of its potential usefulness have been made to various NRC offices. Improvements are planned, and current efforts are directed toward replacing the data base with recent plant data from the Reactor Risk Reference Document (NUREG-1150).

Accident Management. This research project is aimed at (1) developing methods for evaluating the effectiveness of operating plant personnel actions and plant equipment and emergency operating procedure guidelines, and (2) identifying the kinds of modifications thereto that could help mitigate the effects of a severe accident. Research carried out in 1986 was a feasibility test at a BWR Mark I for evaluation of containment venting under two severe accident sequences, i.e., anticipated transient without scram and station blackout. Four specific scenarios were employed, based on information about the plant's emergency procedures and equipment, and code calculations of release paths and values were determined. A human reliability analysis also was included in estimates of the probability of successful venting for each of the four scenarios. Results were incorporated in the Reactor Risk Reference Document (NUREG-1150). A separate (draft) report was prepared as NUREG/CR-4696 to document the venting analysis. Late in the year, the analytical effort was extended to other BWR Mark II and III plants, and plans were made for other accident management applications to PWRs.

Risk Methods and Data Development

The Risk Methods Integration and Evaluation Program (RMIEP) was started in 1984 to develop improved assessment methods to support probabilistic risk assessments of nuclear power plants. Initial integrated logic models of the LaSalle Unit 2 (Ill.) were completed during 1986, and the internal events screening analysis was completed. The location analysis for fire and internal floods was started. This analysis will yield the fire zones and component failures that are critical to plant safety and will be completed in 1987.

Work on a joint dependent failure analysis procedures guide with the Electric Power Research Institute in this country and the Safety and Reliability Directorate of the United Kingdom Atomic Energy Authority was started in late 1985, and a final draft is nearing completion. The development of simplified external events methods for seismic, fire, and internal floods was started in 1986. These methods for analyzing external events will be used in selected reference plant studies in support of the Reactor Risk Reference Document (NUREG-1150).

The initial version of the Integrated Reliability and Risk Analysis System (IRRAS) suitable for use on a personal computer was completed. A new fault-tree-processing algorithm developed at Lawrence Livermore National Laboratory made it possible to have a program efficient enough for implementation on a personal computer.

Operational Safety Reliability Research

Research on operational safety reliability arose from concerns that, during a plant's operating lifetime, the reliability of systems and activities important to safety might degrade from the designed-in reliability that was assumed during the licensing process. Despite the prescriptive changes made after the TMI accident in 1979 and the Salem automatic-trip failure in 1983, abnormal events continue to occur. Therefore, as a possible improvement, reliability technology from the nuclear industry and from other industries was integrated into a closed-loop reliability program applicable to reactor operational safety, and its potential effectiveness was assessed.

In 1986, Brookhaven National Laboratory completed a technical assessment of this reliability technology. The results indicate that reliability technology can help licensees maintain an acceptable level of safety throughout a plant's operating lifetime and can help the NRC staff to identify plants with deteriorating safety performance. Some of the initial results of this research are, therefore, being used to improve NRC's program. For example, reliability concepts for monitoring plant safety are incorporated qualitatively into NRC's selection in 1986 of variables to be monitored as plant performance indicators. Ways to quantitatively relate performance indicators to plant safety performance targets will be considered as technology is developed and validated.

Severe Accident Risk Assessment

As one part of the implementation of the NRC's Severe Accident Policy Statement, a reappraisal of severe accident risk was completed in 1986. This reappraisal focused on the risk of internally initiated severe accidents for a set of five commercial nuclear power plants of diverse design characteristics: Surry (Va.), Peach Bottom (Pa.) Sequoyah (Tenn.), Grand Gulf (Miss.), and Zion (Ill.). In addition to the risk reassessments, studies were made on the benefits and costs of accident prevention and mitigation features, such as improved on-site electric power and filtered-vent containment systems.

These risk assessment results constitute a substantial part of the technical base supporting the Commission's Severe Accident Policy Statement and Safety Goal Policy Statement. They provide measures of conformance with the safety goals for the studied plants, a data base for supporting the assessment of risks in the entire population of plants (as called for in the Severe Accident Policy Statement), and a measure of the effectiveness of possible plant modifications, where indicated, in reducing risk. Results will also be used in considering other possible regulatory changes (e.g., in emergency planning).

The results of these analyses have been documented in three levels of detail. At the first level, NRC contractors at Sandia National Laboratories, Battelle Columbus Laboratories, and Brookhaven National Laboratory have performed and documented the detailed risk analyses (as NUREG/CR-4624, NUREG/CR-4550, and NUREG/CR-4551). Supported by these reports, the NRC staff and contractors have developed summary reports on the plant results and related technical subjects, documented as appendices to the Reactor Risk Reference Document (NUREG-1150). The main body of this last report has been written to provide a more widely understandable discussion of the risk results, perspectives on these results, and their implications to a set of specific regulatory issues.

During 1987, this information will be subject to public review and comment. In addition, further plant studies will be performed (e.g., at the LaSalle facility in Illinois), the study of external events risk (e.g., earthquakes) initiated for a set of plants, and additional specific analysis. It is intended that the final version of NUREG-1150 and supporting documents will be released near the end of 1987.

NUCLEAR POWER PLANT THERMAL-HYDRAULIC TRANSIENTS

The principal purpose of the NRC's thermal-hydraulic program is to improve our understanding of, and ability to predict, plant behavior during accidents and transients. This capability is needed to provide an assessment of the adequacy of LWR design and operations to ensure that transients will not lead to more serious accidents and to modify NRC's regulation as required to ensure safe operation of nuclear power plants. The program continues to be based on both experiments and analysis methods. Experiments are needed to assess this ability of the codes to calculate complex plant transients. The codes are required because experimental data from scaled integral or separate-effect experiments are generally not directly applicable to the wide diversity of reactor designs. Current experimental facilities, e.g., Semiscale and LOFT, have been shut down or will reach the end of their experimental programs in the near future as their limits of useful data are reached. Future facilities will center around Babcock & Wilcox (B&W) reactor problems and will include integral and separate-effect experiments investigating the unique designs of B&W plants. Code improvements will continue in critical areas using these new data, and applicability of the codes to LWR-transient analyses will be demonstrated using data from throughout the free world through specific NRC-sponsored programs as well as in cooperation with the International Code Assessment Program (ICAP).

Technical Integration of Thermal Hydraulics

In response to an October 1985 request from the EDO, staff from RES, NRR, and IE developed and submitted a "Plan for Integrating Technical Activities within the U.S. Nuclear Regulatory Commission and Its Contractors in the Area of Thermal Hydraulics," dated January 23, 1986. The plan makes specific recommendations to improve and accelerate the integration of research results into the regulatory process that include establishment of a Regulatory Research Review Group (RRRG) and preparation of summary reports on completed research. The RRRG was established and became operational during 1986.

The plan was further implemented by establishing a Thermal-Hydraulic Technical Integration Center (TIC) at the Idaho National Engineering Laboratory (INEL). The principal purpose of the TIC is to ensure the continuing availability of the experience and depth of expertise needed to provide a response capability for priority issues or studies, as well as to perform ongoing work needed by the NRC. Code applicability is a form of code assessment that goes beyond the ICAP approach through a quantitative comparison of code predictions versus experimental results for every major transient of interest in each variety of operational LWR, so that the applicability of that code can be judged with respect to its ability to predict reactor behavior in the context of these major transients. It is in effect a summary report placed in a format most usable to the NRC. Priority studies would use staff from the other two major program areas in TIC, and other disciplines as needed, to resolve regulatory issues; in 1986 these priorities were (1) support for the revision of the emergency core cooling system (ECCS) rule and (2) development of methods by which the NRC staff can independently evaluate B&W Owners Group recommendations for safety improvements in B&W reactors. Both are described in more detail below.

Emergency Core Cooling System Rule Revision

Section 50.46 of 10 CFR Part 50, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) in Light Water Reactors," and Appendix K of Part 50, "ECCS Evaluation Models," currently specify how the ECCS is to be designed for nuclear power plants in the United States. In the decade since this ECCS rule was written, the NRC and AEC have spent seven hundred million dollars in ECCS research. This research has shown that Appendix K methods are highly conservative and that actual fuel rod temperatures during a loss-of-coolant accident (LOCA) would be much lower than those predicted by the Appendix K methods. Research has also provided more realistic methods of evaluating ECCS performance. As a result, the NRC staff is proposing a revision to the ECCS rule to permit licensees to perform realistic evaluations of their ECCS based on the body of research currently available. Such calculations would be coupled with an evaluation of the modeling uncertainties. Furthermore, the proposed rule would provide clear guidance with respect to the reporting of errors or changes appearing in evaluation models. This proposed rule revision is expected to provide more realistic and technically defensible regulation and has the potential to provide significant economic benefits.

In order to simplify implementation of the proposed rule, two supporting documents are being prepared: (1) a regulatory guide to provide guidance on how to perform a best-estimate calculation and what would be expected in the uncertainty evaluation, and (2) a document to summarize the large amount of research generally applicable to best estimates of ECCS performance and to describe NRC's efforts to quantify the uncertainty in ECCS performance calculations.

Scaling Relationships for Future Integral Facilities

By the end of 1987, all major integral thermal-hydraulic test facilities in the United States are scheduled to be shut down. This move will affect the NRC's ability to provide timely resolution of future unforeseen safety issues with a high level of technical confidence. The NRC is, therefore, evaluating available options for maintaining a continuing experimental capability for light-water reactors. To provide the most comprehensive and cost-effective set of options, a scaling study was begun in 1985 to evaluate capabilities and costs of several alternative scaling approaches to test facility design. The scaling approaches used included those of current facilities, plus newer approaches that could emphasize more realistic simulations of reactor operational transients. The methods and preliminary results were reviewed by a group of experts in early 1986 and again in late 1986. They recommended more emphasis on code assessment needs and on simulation of multi-dimensional behavior. Current work is involved with estimating the uncertainties in the phenomena that the facility is expected to model and with performing cost tradeoffs of size, power level, and pressure.

Babcock & Wilcox Reactor Safety Issues

The accident at the Three Mile Island reactor in March 1979 was a small-break LOCA in a B&W reactor that resulted in core damage. Because of this accident, the NRC reviewed the analytical predictions of small-break LOCAs for the purpose of ensuring the continued safe operation of all reactors. It was determined that additional experimental data were required to verify the analytical prediction for B&W plants because much of the available data from facilities such as LOFT and Semiscale are applicable to Westinghouse and Combustion Engineering plants, but not to B&W plants. The NRC initiated the Integral System Test (IST) program in 1983 in cooperation with the B&W Owners, B&W, and the Electric Power Research Institute to investigate the small-break LOCA in integral facilities representative of B&W plants. Small-break LOCA tests under the IST program have been conducted in the Once-Through Integral Systems (OTIS) and Multiloop Integral Systems Test (MIST) facilities and also at the University of Maryland to investigate the influence of test facility design and scaling.

More recent plant transients, such as the loss-of-feedwater event at the Davis-Besse plant (Ohio) in June 1985 and the loss-of-power event at the Rancho Seco plant (Cal.) in December 1985, have reinforced the conclusion that B&W plants are unusually sensitive to transients and pointed up the need to obtain experimental data covering a wider range of transients in addition to the small-break LOCA. The NRC has developed a comprehensive plan to obtain the required experimental data to verify analytical techniques over a wide range of B&W plant transients. The plan includes extended tests of transients other than the small-break LOCA in the MIST facility, constructing a new separate-effect facility to investigate the thermal-hydraulic response of the unique B&W once-through steam generators (OTSGs), and continued integral test capability as discussed above. During 1986, detailed test planning for the extended MIST program was completed, and initial design studies for the OTSG were started.

The experimental programs discussed above are longer-term efforts designed to verify and improve analytical techniques used to evaluate the performance of B&W plants during transients and accidents. In the interim, there is a need to apply existing technology to ensure continued safe operation of B&W plants and to evaluate the improvements that will be recommended by a study of B&W plant operation currently being conducted by the B&W Owners Group.

One of the major TIC priority studies is the development of integrated methods to evaluate B&W plant safety and any improvements resulting from the B&W Owners Group study. This task integrates the methods of thermal-hydraulic transient analyses (computer codes), probabilistic risk assessment (PRA) of accident sequences, and human reliability evaluations of plant operator responses. The goal of this work, to be completed in 1987, is to provide the best available tools to evaluate the significance of transients occurring in B&W plants and the effectiveness of steps taken to reduce the transient frequency and risk.

Semiscale Test Series

The final test series in Semiscale consisted of several small-break LOCAs in which either 0.5 or 2.1 percent break tests were performed. High-pressure coolant injection was purposely defeated in this test series. This scenario is a large contributor to core melt frequency. Operators used secondary system feed-and-bleed and primary coolant pump operation to reduce the primary system pressure to the low-pressure injection setpoint. The time at which these operations were initiated was one of the key experimental variables.

BWR Plant Analyzer and Accident Simulation Studies

The major activities involving BWRs were improvement, maintenance, and support for computer codes used by the NRC. The BWR plant analyzer developed at the Brookhaven National Laboratory was used in support of emergency response drills. The fast-running plant analyzer is able to simulate a plant accident in real time and thus provide information to improve the realism of the drill. Work continued on the improvement of TRAC-BWR, which is NRC's major computer code to analyze accidents and transients in BWRs. During the year, work was completed on a faster-running version of the code, and the capability to calculate reactor power during an anticipated transient without scram (ATWS) was improved. The NRC is currently leasing (with an option to buy) the control room simulator from the canceled Black Fox (Okla.) nuclear plant. This BWR control room is currently being used at NRC's training center in Chattanooga, Tenn. A study is being conducted to investigate how to improve the realism of this simulator using NRC's plant analyzer technology. A decision on the benefit of such an effort will be made next year, based on the study completed in 1986.

Fluid Mixing for Pressurized Thermal Shock Concerns

Thermal fluid mixing experiments ranging from 1/5-scale to 1/2-scale have been performed at Purdue University, Creare Inc., Imatran Voima Oy of Finland, and HDR of Germany. These experiments have shown that the cold emergency core coolant injected into the cold leg mixes well with the hot primary coolant so that the effects of thermal stratification are greatly minimized in the downcomer region. This mixing phenomenon significantly reduces the likelihood of pressurized thermal shock.

To confirm this result at a full-scale test facility, a fluid mixing test was performed at the German Upper Plenum Test Facility (UPTF) under the 2D/3D program. The UPTF test results were consistent with the small-scale test results. These favorable experimental results helped close thermal-hydraulic aspects of the pressurized thermal shock issue.

Water Penetration into Reactor Core In Upper Plenum Injection Plant

In certain two-loop PWR plants, emergency core coolant (ECC) is injected into the upper plenum as well as into the cold legs. Such plants are known as upper plenum injection (UPI) plants. In these plants there is a concern that the reactor core may not be uniformly cooled by ECC because of a possibility that the downflowing ECC liquid may be blocked by the rising steam in some areas. To address this concern, several tests were performed under the 2D/3D program in the 1/21-volume scale Cylindrical Core Test Facility (CCTF) located at the Japan Atomic Energy Research Institute. Results from this large-scale test facility showed that the electrically heated, scaled reactor core is effectively cooled by UPI even though water flows down through the core preferentially in certain regions of the core. The test core was divided more or less evenly between water downflow regions and steam upflow regions. The core cooling in the steam upflow region is also effective because of significant entrainment of liquid by the rising steam. The region of downflow seemed to vary from test to test, suggesting that the distribution of the downflow region is random. The effectiveness of core cooling in the UPI tests compared favorably with that of comparable cold leg injection tests in the CCTF.

Steam Binding Effect

Large-scale tests were conducted in the CCTF by the Japan Atomic Energy Research Institute under the 2D/3D program to investigate the steam binding effect. Steam binding occurs when pressure buildup in the steam generators, caused by evaporating droplets, prevents ECC from completely filling the core. The test results show that the electrically heated, scaled reactor core is effectively cooled, even though there is a considerable degree of steam binding. It is estimated that without the steam binding effect the core flooding rate in the CCTF would increase by about 20 percent. The core was quenched within three to 10 minutes from the start of the reflood, and the maximum temperature of the hottest rod was 1,280xK for tests with conservative test conditions. These results showed a significant safety margin to the licensing limit of 1,478xK.

Operational Change in the Ringhals Reactor in Sweden

In the winter of 1983-1984, a control rod trip occurred in the Ringhals reactor in Sweden, caused by overspeeding of the turbine. This led to imbalances in the network and caused an electrical blackout in a large portion of the country. The Swedish authorities are changing the setpoints for the turbine so that the reactor will not trip immediately. The delay in the trip will provide the operators about 20 minutes to correct the problem. The Swedish authorities are using the NRC code, TRAC-PF1/MOD1, to examine safety consequences of this operational change, which will improve the reactor availability in that country. It is clear that NRC codes can also be used in other countries to improve operations or availability of reactors.

Quantification of Code Uncertainty

The proposed ECCS rule requires that thermal-hydraulic codes be capable of predicting the peak clad temperature with 95 percent probability when a LOCA occurs. The quantification of the uncertainty of code predictions for LOCA is necessary for confirmation of this safety margin. The accuracy of code predictions can only be ascertained by comparing the predictions of the code with measurements in thermal-hydraulic experimental test facilities where LOCAs have been simulated.

Preliminary code uncertainty analyses using about 15 test cases and an older version of the code (TRAC-PD2) have already been made. The analyses showed that the code uncertainty is on the order of 150xF for calculation of the peak clad temperature in a LOCA. Appraisals of the probabilities have not been made. In 1986, a more detailed code assessment matrix containing about 30-35 cases was developed. Increasing the number of tests permits the NRC to make some probability statements. These tests will be calculated using the most recent version of the code (TRAC-PF1/MOD1) by the participants of the International Code Assessment Program and, where practical, by industry.

During the process of quantifying code uncertainty, some questions on the scalability of uncertainty have been raised. The codes have been assessed against data obtained from smaller test facilities, and they have not been tested against a full-scale large-break LOCA in a reactor. In order to answer these questions, another test matrix, which is mainly composed of separate-effect tests, has been formulated. A comprehensive methodology to integrate these results has also been formulated and reviewed by an independent peer group. This will permit extension of the uncertainty and probability statements from smaller facilities to reactor-scale facilities.

Plant Analyzer

The plant analyzer includes calculational tools designed to easily and accurately analyze plant transients and make rapid regulatory decisions on the safety of operations.

There are four primary purposes served by the nuclear plant analyzer (NPA):

(1) To reduce the person-hours required to perform analyses and prepare plant analysis models.

(2) To provide on-line interactive computer capability to simulate reactor operator actions during analyses

using NRC's best computer codes (TRAC and RELAP).

(3) To expedite TRAC and RELAP5 analyses, provide a timely technical basis for licensing decisions, and cut costs.

(4) To provide a colorgraphic representation of computer output showing the instantaneous thermodynamic state of the fluid and vapor throughout the primary or secondary loop in order to assist prompt comprehension of complex thermodynamic behavior during plant accidents.

During 1986, NPA capabilities were expanded to permit analysts to stop a computation in progress, generate X-Y plots of input or output data, and resume the computation without leaving the NPA program. The NPA program was also expanded to function with the TRAC-PF1/MOD1 code on the Los Alamos National Laboratory (LANL) CRAY-1S computer. A dedicated communication line between INEL and LANL was used to successfully demonstrate NPA access on the LANL computer from an INEL terminal workstation.

The minimum plant design and operational data required to generate plant models for TRAC-code analyses with the NPA have been defined, and the software program that will use these data to automatically generate TRAC plant models for four-loop Westinghouse plants was started, with completion scheduled in 1987.

Davis-Besse Uncertainty Study

A loss-of-feedwater transient occurred at the Davis-Besse (Ohio) nuclear power plant on June 9, 1985. Davis-Besse Unit 1 is owned and operated by the Toledo Edison Company. The plant is a PWR of the raised-loop B&W design with a rated core power of 2,772 MWt. This transient, which was initiated at 92 percent power, resulted in a temporary but total loss of main and auxiliary feedwater. Auxiliary feedwater was eventually restored, and the plant was taken to a safe and stable condition.

Because of the potential severity of the event—and because of previous feedwater transients in B&W reactors, including TMI-2—the NRC immediately began an intensive analysis of the Davis-Besse loss-of-feedwater transient. The analysis was to include parametric variations that were primarily related to the use of feed-and-bleed cooling. Feed-and-bleed cooling, which involves starting the makeup and high-pressure-injection pumps and opening the pilot-operated relief valve located on the top of the pressurizer, would have been used to remove decay heat from the core at Davis-Besse, had the auxiliary feedwater not been restored. The NRC pursued a two-pronged thermal-hydraulic analysis of the Davis-Besse event. First came an in-house analysis of the event, through NRR, using the nuclear plant analyzer; second, an independent analysis was performed at LANL using the TRAC-PF1/MOD1 computer code.

Although the NRR and LANL calculations indicated that feed-and-bleed could successfully cool the core if initiated early enough, there were several uncertainties in the calculations. These uncertainties were related to the thermal-hydraulic computer codes used in the calculations, the code input models, and the initial and boundary conditions. The NRC asked INEL to assess the uncertainties in the NRR calculations. INEL developed a detailed, quality-assured RELAP5/MOD2 model of Davis-Besse, which was then used to repeat the NRR calculation of the Davis-Besse loss-of-feedwater transient, as well as to repeat a loss-of-feedwater transient from 100 percent power with feed-and-bleed initiated 20 minutes after the start of the transient. The evaluation of uncertainty was based on comparisons between calculations and data, comparisons between the NRR and INEL calculations of the same transient, sensitivity calculations, and the propagation of the estimated uncertainty in initial and boundary conditions to the final calculated results.

The INEL calculation, in agreement with the NRR and LANL results, indicated that feed-and-bleed would successfully remove core decay heat in the event of a total loss-of-feedwater transient at Davis-Besse. Some significant differences were observed between the INEL calculation and the previous calculations that affected the course, but not the ultimate outcome, of the transient. These differences were attributed to differences in the boundary conditions. The uncertainty in the INEL and NRR calculations was estimated. The uncertainty did not alter the above conclusion relative to the effectiveness of feed-and-bleed cooling. The initial and boundary conditions were found to be the largest contributors to the uncertainty in the calculated results.

Analysis of Black Fox Simulator

Best-estimate methodology and computer codes in simulators are highly desirable when training operators and inspectors of nuclear reactors. Personnel from INEL visited the Black Fox simulator at Chattanooga, Tenn., to discuss simulator limitations with the NRC staff and General Electric personnel. Several limitations were identified. The mathematical modeling and numerical methods implemented on the Black Fox simulator were formulated with the intent of allowing steady state and operational transient simulation. Therefore, the models are mostly parametric in nature, relying on known empirical or pre-calculated data. For transient phenomena outside the design envelope (near steady state), numerical instability should be expected and the fidelity is in serious question. The training center staff pointed out that the simulator has demonstrated weaknesses in ATWS simulations and small-break LOCAs with core spray. The observed instabilities of the numerical scheme indicated that future multi-phase model add-ons alone are insufficient for a Black Fox upgrade. After the discussion with the training center staff and analysis of the hardware and technology available, several recommendations for improving the performance of the Black Fox simulator were made. The NRC will decide on the cost effectiveness of these recommendations during the upcoming fiscal year.

Core Level Depression During Liquid Holdup

A certain combination of reactor design characteristics may produce a core liquid level depression during a small-break LOCA and, consequently, a core heatup. These phenomena were first observed in Semiscale experiments in 1981 and 1985. Because the core heatup phenomenon has direct safety and licensing implications, the ROSA-IV program in Japan was requested to study the influence of scaling by duplicating the Semiscale experiments. The ROSA-IV facility is the largest test facility representative of Westinghouse PWRs. Its scale is 1/48, while the scale of the Semiscale facility is 1/1700.

During 1986, five experiments were conducted in the ROSA-IV facility to investigate the core level depression phenomenon. All experiments were 5 percent cold leg small-break LOCAs. The experiments confirmed that the core liquid level depression phenomena were not peculiar to the Semiscale facility. They also showed that the phenomena in these experiments may not be a safety problem in typical PWRs, because, in every case, the core heatup was not very high and the heater rod temperatures immediately turned around when the break was uncovered, allowing the steam to discharge from the primary system.

Further analysis of the phenomena in PWRs with best-estimate safety codes will be conducted in 1987, in order to permit NRC to assess the regulatory significance of the core liquid level depression.

Instrument Tube Line Rupture

Westinghouse reported that one or more in-core instrument tube lines installed in the bottom of a pressure vessel may rupture during a seismic event. During 1986, the NRC analyzed the consequences of instrument tube ruptures using the RELAP5 code. Results showed that the plant ECCS will prevent core uncovery when up to 11 tubes are ruptured. In order to assess RELAP5 calculational uncertainties, the NRC recommended instrument rupture tests be conducted in the ROSA-IV facility in 1987.

NUCLEAR POWER PLANT SEVERE ACCIDENTS

Source Terms

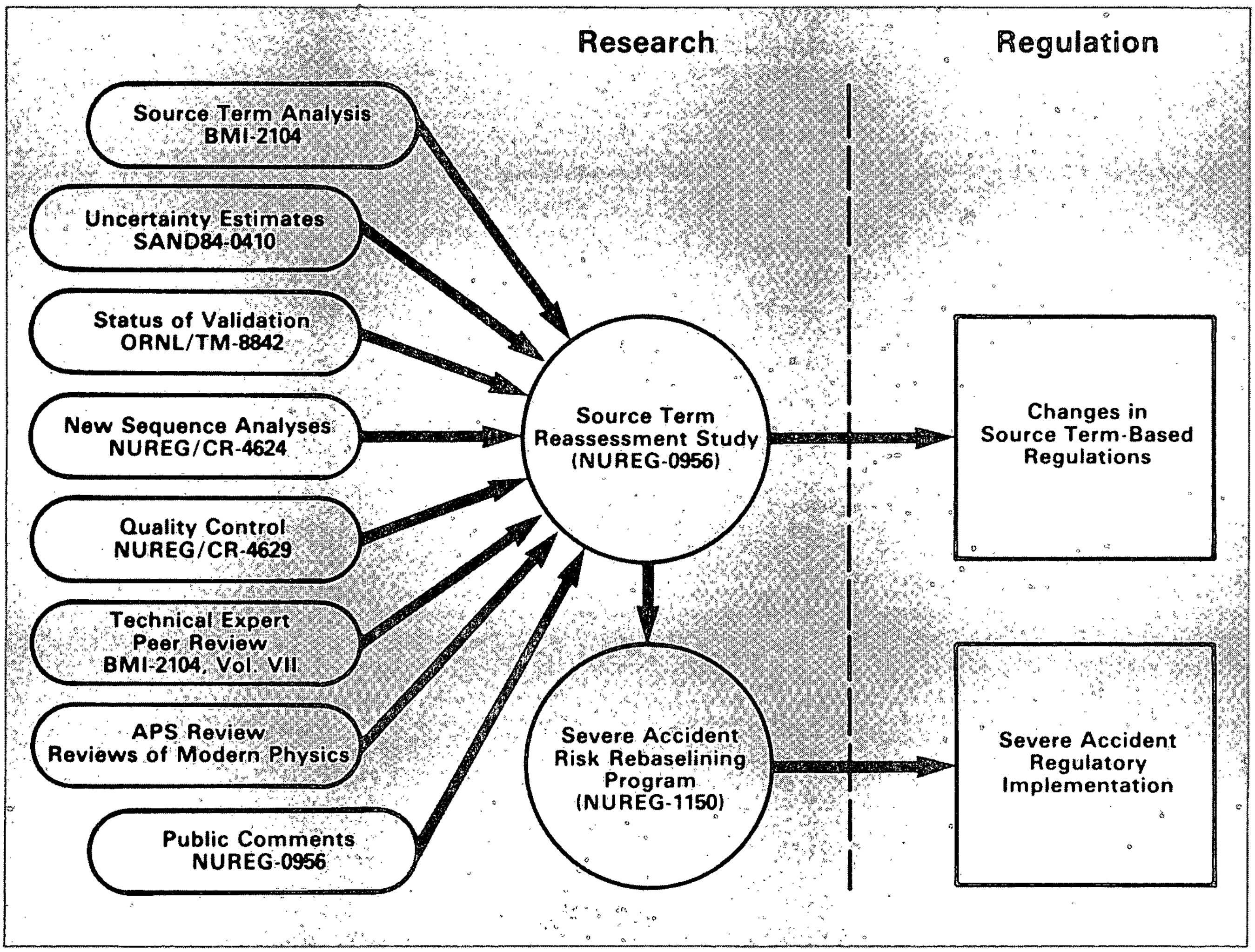
A source term is defined as the quantity, timing, and characteristics of the release of radioactive material to the environment following a postulated severe reactor accident. Source term technology is employed for a variety of regulatory applications, including plant siting evaluation, emergency planning, evaluation of performance of engineered safety features such as containment isolation and containment spray additives, qualification of safety-related electrical equipment for performance under accident conditions, environmental impact statements, post-accident monitoring requirements, and criteria for re-entry of a plant after an accident. In addition, an understanding and quantitative assessment of source terms is necessary for conducting probabilistic risk assessments, which are emerging as a significant contributor to the regulatory decision process. New information and insights on radioactive source terms may have an impact on rules, guides, and other regulatory practices in the aforementioned areas through implementation of the NRC Severe Accident Policy Statement.

Early research on severe accidents has pointed up the need to integrate the analysis of complex severe accident phenomena to obtain realistic estimates of source terms. Research programs initiated following the accident at Three Mile Island have covered all the important phases of severe accidents—from the period immediately following the initiation event, through the in-vessel and ex-vessel phases, and most importantly in the behavior of the containment. In particular, large first-of-a-kind programs in the Power Burst Facility and the LOFT facility have investigated core melt progression and in-vessel fission product release to an extent never attempted before. Major programs in the Sandia Large Melt Facility and the German BETA facility have studied the interaction between molten core debris and concrete as would occur in the reactor cavity beneath the reactor vessel. Other large programs both here (in the EPRI-sponsored LACE program at Hanford and in the Nuclear Safety Pilot Plant facility at Oak Ridge) and abroad (at Marviken in Sweden and in DEMONA in Germany) have investigated the behavior of aerosols in reactor coolant systems and in containments.

In light of this emerging severe accident technology and its expanding data base, the NRC undertook a reassessment of the technical bases for estimating source terms. The purpose of the reassessment was to evaluate the data base for validation of source term codes, to calculate source terms for selected plants and sequences, to conduct a broad-based peer review, and to appraise plant risk and the regulatory significance of the reassessed source terms.

When this effort was initiated, there was an expectation among many in the nuclear community that a correct treatment of the physical and chemical behavior of fission product release and transport would show a reduction of several orders of magnitude in calculated source terms, except for noble gases, compared with the Reactor Safety Study (WASH-1400) of 1975. Such a result would have made it easy to develop new generic source terms, and such reduced source terms would have translated directly into reductions in estimated risk without the need to re-evaluate other areas of risk assessment (event frequencies, containment performance, and off-site consequences). Such clear-cut reductions in source terms have not been found, however, and the self-consistent method of evaluating source terms has demonstrated a high degree of plant-specific variation. Nonetheless, profound results have been achieved, and data from the severe accident experimental program mentioned above have advanced considerably the understanding of source term phenomena. A reviewed and tested analytical tool, the NRC's Source Term Code Package, has emerged from this study, and this code package is capable of dealing with plant-specific variations in a realistic way. The Source Term Code Package has been used in a major new risk study to be incorporated into the Reactor Risk Reference Document (NUREG-1150).

A major document describing the advances in source term technology and the staff's technology assessment was published in July 1986 and is entitled "Reassessment of the Technical Bases for Estimating Source Terms" (NUREG-0956). The improved technology described in that report consists of (1) an extensive data base from severe accident research programs initiated following the TMI accident, (2) a set of coupled and integrated computer codes (the Source Term Code Package), which models key aspects of radionuclide behavior under severe conditions, and (3) a number of detailed mechanistic codes that bridge the gap between the data base and the Source Term Code Package. The improved understanding of severe accident phenomena has also allowed an identification of significant sources of uncertainty, which should be considered in estimating source terms. These sources of uncertainty are also described in the document. The current technology provides a significant improvement in evaluating source terms over that available at the time of the Reactor Safety Study and, because of this significance, the NRC staff is recommending its use. Key elements of the source term reassessment program are shown in the figure.



In July 1986, the NRC published a reassessment, developed by the staff, of the technical bases for estimating source terms (the quantity, timing and characteristics of a postulated release of radioactive material to the environment following a hypothetical reactor accident). The diagram shows key elements of the source term reassessment program and their relationship to regulation implementation.

Severe Accident Sequence Analysis

Analyses of dominant severe accidents for the Browns Ferry (Ala.) plant continued. The importance of a good multi-compartment representation of containment was identified, and the necessary modifications to the CONTAIN code were completed with the cooperation of Sandia. The study will extend into 1987. The multi-compartment capability was also used in a study of the effectiveness of the secondary containment for BWR Mark I reactors. Significant reductions in release—factors of 10 or more—were found for nongaseous fission products. A venting study for BWRs was completed that enumerated various detrimental effects from venting without plant modifications.

The integrated SDCAP/RELAP5 code was used to analyze a series of small-break loss-of-coolant accidents for the Bellefonte (Ala.) nuclear plant as part of the Severe Accident Sequence Analysis. The transients were all initiated by a two-inch-diameter break in a reactor coolant pump discharge line. These calculations showed that the availability of a single high-pressure injection pump was sufficient to prevent core damage for small breaks. To determine whether core damage could be prevented without high-pressure injection, operator actions were taken into account. It was shown that even without high-pressure injection, simple operator actions can effectively mitigate the transient and bring the plant to a stable condition, removing the decay heat.

Analyses of the Browns Ferry Unit 1 BWR using the SCDAP/RELAP5 code to determine reactor system and core response were also completed. During an anticipated transient without scram in a BWR, high-pressure emergency core cooling can be lost. Without mitigating actions by the operator, the situation could degrade to a high-pressure boiloff. A simulation of a postulated high-pressure boiloff was performed using the SCDAP/RELAP5 code. The purpose for using the SCDAP/RELAP5 code was to study the effects of coupling the reactor core behavior with the system thermal-hydraulic behavior.

Four parallel core regions were modeled to study multichannel effects on hydrogen production and flow distribution. Calculated damage to the core consisted of cladding failure, heavy oxidation, and relocation of core material. Oxidation, melting, and relocation of molten material were calculated for the inner three-fourths of the core, but very little oxidation and no melting were calculated on the outside of the core.

The study to identify local areas of potential hydrogen concentration that could detonate during severe accidents was completed. The potential for hydrogen stratification was shown to be negligible with the mixing and geometry of the Bellefonte plant. In addition, the limited areas that could collect detonable concentrations of hydrogen were shown to produce shock waves of sufficiently low intensity that neither missiles nor containment liner strains would be likely to cause containment failure. In addition, a parametric study of the effects of direct containment heating were completed in support of the experimental program at Sandia.

Core Melt Progression

Major phenomena that affect the course of an accident (such as natural circulation) and the challenge to the containment (such as direct containment heating, core-concrete interactions, hydrogen combustion, etc.) are governed by events that take place in the vessel during the initial core meltdown. Uncertainties regarding these phenomena are often primarily related to the state of the core debris (temperature, material composition, melt fraction, etc.), the rate and mode of debris removal from the vessel, the degree of oxidation of the debris, and the concomitant amount of hydrogen generated during oxidation. To reduce uncertainties in analyzing these major phenomena, it is necessary to understand and model the underlying core melt progression as the accident evolves.

Damaged Fuel Experiments. These experiments provide the information for modeling the state of the damaged core during the early phase of core melt progression in core uncovery accidents, before the gross fuel melting occurs. The state of the core determines the amount of hydrogen generated and the fission product release in this stage of the accident. Damaged fuel experiment DF-3 on the effect of PWR stainless-steel-clad silver-indium-cadmium control rod materials on fuel damage and on in-vessel core melt progression was successfully performed in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories in October 1985. Along with the results of post-irradiation examination of the final severe fuel damage test in the Power Burst Facility (PBF) test reactor at INEL in 1985 (which had high-burnup fuel and PWR control rods), the results from DF-3 have provided important information on control rod effects. The control rods fail before the fuel rods, the low-melting silver runs down the core to freeze in colder locations, the low-boiling-point cadmium deposits on colder surfaces above the core, and the stainless steel alloys with and liquefies the Zircaloy fuel cladding, increasing the fuel damage and the potential for blockage formation. A BWR control rod experiment will be performed in the ACRR in early fiscal year 1987.

Coolant Boil-Away and Damage Progression Tests. The Fuel-Length High-Temperature Test 4 (FLHT-4) was successfully performed in the National Research Universal (NRU) reactor in Canada by Battelle Pacific Northwest Laboratories in August 1986. A peak temperature of 2,500 Kelvin was reached. The test was then held at the same power for an additional 30 minutes, and continued hydrogen evolution was monitored during possible cladding relocation. These tests are reducing uncertainties associated with length and power distribution scaling factors, and they are enabling the interpretation of the results from small-scale separate-effect experiments.

Severe Accident Analysis Code Development Program. An integrated MELPROG/TRAC code is being developed to analyze severe accident progression in the reactor coolant system of a nuclear power plant from accident initiation through core melt to vessel failure. The code is being used as the major analytical tool in analyzing reactor coolant system natural circulation in a PWR station blackout accident. Validation of the code will be performed by comparison with the TMI-2 plant data, Westinghouse natural circulation experiments, and other severe core damage experiments. The current version analyzes PWR accidents with a future version to analyze BWR accidents.

An integrated SCDAP/RELAP5 code has been successfully developed to analyze recovered accidents in a PWR. The code was validated against severe core damage experiments in PBF, LOFT FP-2, and NRU. It will be used to provide analytical support to the KfK CORA experiments under an agreement between the NRC and the Federal Republic of Germany. The code will also be used to provide comparative calculations to check the MELPROG/TRAC results on reactor coolant system natural circulation.

Hydrogen Generation and Control

This program assesses both the consequences and methods used to control or mitigate deflagrations, diffusion flames, and detonations that might be caused by hydrogen burns in LWR plants. The HECTR Version 1.5 users' manual, NUREG/CR-4507, was issued during this period. The HECTR computer code was developed at Sandia National Laboratories and is used in the analysis of nuclear reactor accidents involving the transport and combustion of hydrogen. The assessment of HECTR is ongoing, and it includes extensive use of the data from EPRI/NRC large-scale hydrogen combustion experiments performed at the Nevada Test Site (see NUREG/CR-4138). A complete flammability map for the hydrogen-air-steam system in both turbulent and quiescent pre-ignition conditions was determined, and these data were used to develop a correlation for HECTR to describe the entire flammability region (see NUREG/CR-3468). The HMS-Burn Code (NUREG/CR-4020) was developed at Los Alamos and is being used to track HECTR and to provide more detailed hydrogen transport and mixing calculations. HECTR was used during this period to assess the hydrogen combustion models in the industry MAAP code in support of the resolution of NRC-industry issues. The HMS-Burn and HECTR codes were used topage assess the potential for local detonation at the Bellefonte (Ala.) nuclear power plant, a facility, under construction, with a large dry containment. Calculations by both codes suggest that locally detonable mixtures are likely only in the area near the break. HECTR calculations were performed for selected large, dry and subatmospheric containments. For some small containments of this kind, the ratio of calculated to design pressure exceeded 2.5 (see NUREG/CR-4599).

Debris Interactions with Concrete

When core debris melts through the reactor pressure vessel and comes into contact with structural concrete beneath the vessel, the concrete decomposes under this thermal attack and releases copious amounts of gases. Different types of concrete behave differently under these conditions, with the U.S. limestone types of concrete producing larger amounts of decomposition gases. The gases are non-condensable, and some of them are flammable; they threaten the integrity of the containment because of overheating and pressurization. At the same time, the decomposition gases bubble through the hot debris, fomenting the release of radioactive fission products to the containment atmosphere. To characterize the threat to containment integrity and the nature of the extra-vessel releases, these core-concrete interactions must be understood and mathematically modeled. A summary of some research accomplishments in this area follows.

A program at Sandia National Laboratory to study concrete erosion by high-temperature debris (both metals and oxides) with sustained heating was begun in 1986. Unlike the KfK BETA experiments, mentioned below, these experiments incorporate simulants of representative fission product materials in the melt, so that the release fractions needed for quantification of the radiological source term can be measured. It is known that during the later stages of a core melt accident, when the debris has solidified, concrete erosion continues. The experiments that were completed in 1986 provided data on concrete-erosion rates and gas generation rates and showed that aerosol production continues after debris solidification.

Preparation for an additional series of special-effect tests at Sandia on aerosol release from metallic and oxidic metals was completed. In these tests, a controlled flow of gas through the melt will be maintained to simulate concrete decomposition. There are two known aerosol generation mechanisms: (1) condensation of trapped vapors when gas bubbles break at the pool surface, and (2) mechanical aerosolization of bulk pool material, when the film formed by an emerging bubble fragments at the pool surface. Some tests are intended to study the condensation mechanism, while other tests address the mechanical formation process.

Under an arrangement between the NRC and the KfK laboratory in Karlsruhe, Federal Republic of Germany, materials were shipped to Germany to construct three crucibles made of prototypic U.S. reactor containment concrete. Two were made of limestone/common-sand aggregate, and one of pure limestone concrete. Three tests with these crucibles were conducted at the KfK BETA facility during 1986. This program yielded valuable information needed for improved modeling and validation of the CORCON MOD2 computer program used to analyze molten debris attack on concrete. The test with limestone concrete showed greatly enhanced aerosol production. These data are still under analysis.

The CORCON code is the NRC's computational tool for the prediction of containment loading and radiological source term parameters that depend on the interactions between molten core debris and the concrete basemat after primary vessel failure during a severe accident. During 1986, as a result of guidance derived from the KfK BETA experiments, a significantly improved heat transfer model was developed and incorporated in CORCON.

The VANESA code is the NRC computational tool used to predict the magnitude and character of aerosols generated by concrete-decomposition gases sparging through molten core debris. Characterization of the ex-vessel source term is necessary before containment response and atmospheric aerosol behavior can be analyzed to compute the ultimate radiological source term. VANESA also considers possible aerosol decontamination by water overlying the debris, if it is present. The technical reference and users' manuals were completed and released in 1986.

Both the CORCON and VANESA codes have been incorporated into the NRC Source Term Code Package that is being used to provide base-case calculations for NRC risk rebaselining studies. CORCON, along with VANESA, has also been incorporated into the CONTAIN detailed mechanistic code and the MELCOR integrated code. A number of code updates were prepared and distributed as a part of the ongoing user support program. CORCON has also been sent to the United Kingdom and a number of other countries where it is being used in reactor safety research programs.

Containment Loads Analysis

The CONTAIN computer code analyzes the complex physical, chemical, and thermal phenomena that characterize the interactions associated with severe accident conditions inside a multi-compartmented reactor containment system. No in-vessel processes are considered. CONTAIN input is derived from other codes dealing with earlier phases of the accident. Pressure and thermal loading are computed, and evolution of the radiological source term is tracked. This information is made available for the computation of environmental consequences in the event of containment failure. Since its release in August 1984, 24 copies of CONTAIN have been distributed to Government, private, and industrial laboratories throughout the United States and in five foreign countries. During 1986, modeling of suppression pools for BWRs and ice condensers for PWRs was completed. Models for calculating the effects of direct containment heating from high-pressure ejection of molten fuel into the containment during a severe accident were developed. Extensive analyses of the high-pressure ejection event were made in the program for quantitative evaluation of containment loading.

Fission Product Release and Transport

Fission Product Experiments. Work has been initiated to develop thermodynamically based fission product chemistry models. The models are intended for predicting fission product chemical forms during transport in the reactor coolant system and the containment. The chemical forms of fission products affect their transport characteristics and the extent of their retention or revaporization in the reactor coolant system and containment. The models will be capable of predicting chemical forms as a function of environmental conditions—a capability that is absent in existing codes. The effort is part of the work conducted to reduce the uncertainties in the areas of fission product revaporization and iodine chemical form. The latter are two of the eight uncertainty areas identified in NUREG-0956 and are some of the technical issues affecting the risk study (to be published in NUREG-1150).

Aerosol Experiments. The NRC is participating in an internationally sponsored project called LWR Aerosol Containment Experiments (LACE), being conducted by the Westinghouse Hanford Company. The six experiments, four of which were successfully conducted, are being performed to investigate inherent aerosol retention behavior in the containment or auxiliary buildings for postulated high-consequence accident conditions. These experiments will also provide a data base for validating containment aerosol and related thermal-hydraulic computer codes. Several NRC-sponsored contractors are participating in the pre-test and post-test computer code calculations in support of the LACE experiments.

Containment Failure Mode

Activity has continued on a set of programs whose objectives are to provide the data base required for the qualification of methods for predicting the response of LWR containment buildings during severe accidents (those beyond design basis events) and extreme earthquakes. This set of programs is examining the modes of containment failure that would result in the release of radioactive materials beyond the containment boundary. These modes include structural failure of the containment building, leakage through or past the penetrations (electrical or mechanical), failure of containment isolation systems, or failure of the basemat by the molten reactor core.

A 1/6-scale model of a reinforced concrete containment was completed in June 1986. The model, which will be tested by Sandia in the spring of 1987, consists of a seven-inch-thick hemispherical dome atop a circular cylinder that is 9-3/4 inches thick and is supported on a flat base mat. The inside diameter is 22 feet, and the overall height is about 37 feet. The steel liner is 1/16-inch-thick for the cylinder and basemat and 1/12-inch-thick for the dome. The liner is anchored to the concrete with closely spaced, headed studs. Reinforcing patterns that are representative of those in actual containments are included. In the cylinder are eight layers of closely spaced reinforcing bars, including four layers of hoop reinforcement, two layers of meridional reinforcement, and two layers of seismic reinforcement. Additional reinforcing steel is provided in some of the discontinuity regions, such as the base of the cylinder and regions of the cylinder near large openings. Six-millimeter deformed reinforcing bars are used for shear reinforcement. Mechanical splices provide reinforcement continuity. Personnel airlocks and operable equipment hatches are included in the model. A constrained penetration exists which will simulate the effect of the restraint of large bore piping on a penetration as the containment shell grows under pressurization. A variety of small mechanical and electrical penetrations are also included.

The containment model is currently being instrumented and will be tested to failure in 1987. Measurements taken during the test will be used to validate methods for predicting containment performance in postulated severe accident scenarios.

A personnel airlock, originally intended for use at a nuclear power plant that has been cancelled, was procured in 1986. The personnel airlock consists of a cylindrical sleeve or barrel with identical bulkhead/door assemblies located near each end of the barrel. The barrel is 10 feet in diameter and almost 20 feet in length. The doors are rectangular (42 inches in width by 80 inches in height) and are mounted on a pressure seating type double dog-ear seal. The bulkhead and door are flat plates that are stiffened with channel and 'T' sections. The lock was fabricated from a medium carbon steel.

The airlock will be tested to failure in 1987 under conditions representative of those postulated for severe accidents in BWR containments. It is anticipated that leakage will not occur unless relative deformations between the sealing surfaces are developed and performance of the seal material is compromised. The sealing surfaces could separate because of a mismatch in the out-of-plane displacements of the door and bulkhead, which resist internal pressure through bending action. The performance of the seal material may be compromised in two ways: (1) a loss of resiliency associated with thermal or radiation aging, and (2) degradation associated with exposure to very high temperatures.

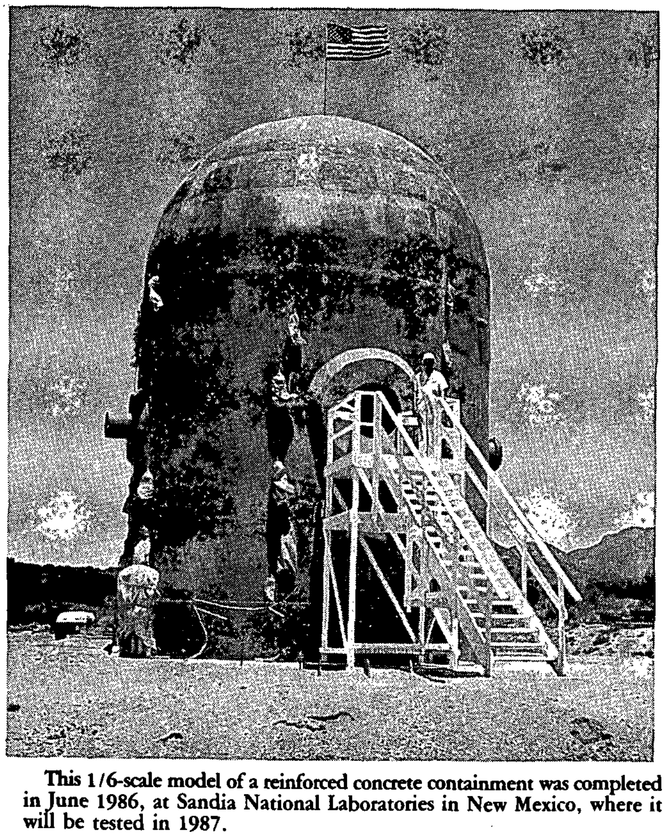
A finite-element analysis will be used to determine the sealing surface deformations and the stresses in the personnel airlock subject to internal pressure at elevated temperature. The analysis will be conducted with the MARC general purpose finite-element code. The bulkhead and door are represented using ¼ symmetry, and the double dog-ear seal is modeled using non-linear springs to allow direct calculation of separation of the sealing surfaces. Nonlinear material properties and large displacement effects will be taken into account.

Fission Product Control

Engineered-safety-feature (ESF) systems such as suppression pools, ice condensers, water sprays, containment coolers, and filters are likely to be operational during postulated accidents substantially more severe than current design basis accidents. However, there may be a substantial variation in the effectiveness of fission product removal by ESF systems under severe accident conditions. A program is in progress to facilitate review and evaluation of ESF system behavior under severe accident conditions. During 1986, computer codes for estimating the extent of fission product retention in BWR suppression pools and the ice condensers of PWR ice condenser containment systems were extended by adding sub-routines for estimating the extent of scrubbing of molecular iodine and methyl iodide. The development of a computer code to estimate aerosol removal rates by containment air coolers during severe accidents was completed. The design of an engineered-scale unit-cell test facility for use in obtaining data for the ice condenser code validation efforts was also completed. Facility construction and installation were about 75 percent completed.

WASTE MANAGEMENT

The NRC's waste management research seeks to develop and verify methods for predicting and assessing the performance of waste disposal facilities; evaluate and confirm the data bases used in such performance assessments; provide



This 1/6-scale model of a reinforced concrete containment was completed in June 1986, at Sandia National Laboratories in New Mexico, where it will be tested in 1987.

technical support to the licensing staff in their interactions with the Department of Energy (DOE) and the States (see Chapter 7); and develop regulatory standards to support the licensing of facilities and methods for the disposal and management of high-level and low-level radioactive wastes.

High-Level Waste

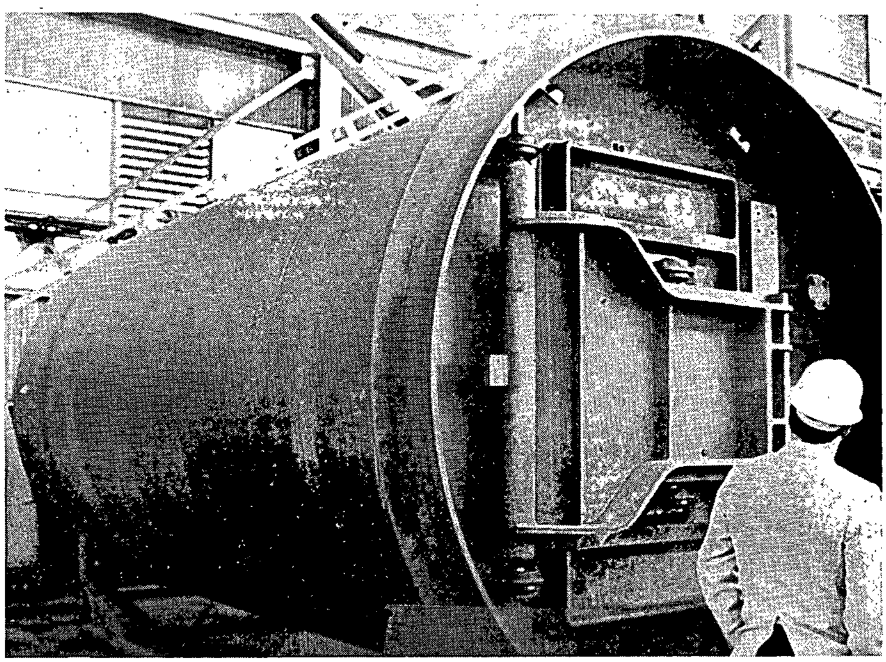
The NRC has active research programs in hydrology, geology, materials science, geochemistry, and several other disciplines related to the management of high-level waste (HLW). The research combines theoretical study with laboratory and field experiments to identify the physical processes that control and determine repository performance in the types of geologic media found at sites currently under consideration by DOE. The ultimate goal of the NRC's waste management research is to provide the technical basis for the licensing staff to make its own independent judgment as to the appropriateness and adequacy of DOE's demonstration of compliance with 10 CFR Part 60 and the EPA's HLW standard.

Geohydrology. Since transport by ground water is the most likely path by which radioactive nuclides from the disposed waste can reach the environment, the NRC is actively studying the movement of ground water in the types of media being considered by DOE. Experimental sites have been located in fractured rock, both above and below the water table, and field testing is being conducted by the University of Arizona and In Situ, Inc., to determine what type of measurements are needed to characterize the hydrology of fractured media and how measurement data should be analyzed to model ground-water flow. A field study in saturated fractured rock was initiated in September 1985 to test the relationships between field measurement of parameters and model scales derived from earlier work. The importance of large natural anomalous hydrologic features, appropriateness of continuum versus discrete fracture models, measurement of effective porosity, theories of spatially projecting dispersivity measurements, and distinctions between matrix diffusion, dispersion, and sorption are among the questions that are being addressed in this study.

The initial phases of a similar study examining unsaturated rock was completed at the University of Arizona in August 1986. This work focused on assessing techniques and methodologies for fracture characterization, infiltration and percolation studies, rock and matrix permeability testing, vapor phase flow and transport assessment, and numerical simulations of flow and transport in partially saturated media. Results are being published as NUREG/CR-4655, Volumes 1 and 2. In addition, the NRC, University of Arizona, and Sandia National Laboratories jointly sponsored a special workshop in January 1986 on unsaturated flow and contaminant transport related to high-level radioactive waste disposal. Processes and field studies dealing with unsaturated flow and solute transport phenomena were discussed, and laboratory and field data were presented.

In this past year, Sandia National Laboratories issued a data input guide (NUREG/CR-3162) and a theory and implementation document (NUREG/CR-3328) associated with SWIFT II, a computer program that contains mathematical models of the flow of ground water, transport of brine, transport of heat, and transport of radionuclides in saturated fractured rocks. The development of SWIFT II was supported by NRC's waste management research program.

Waste Package Performance. Investigating the performance that can be expected from the waste form and



Plans were completed in 1986 for this personnel airlock to be tested to failure under the severe accident conditions assumed for BWR containments. The airlock, originally intended for use at a nuclear power plant, was procured and moved to the Sandia Base test site during the year; it will be tested in 1987.

waste package is essential if NRC is to be able to independently evaluate DOE's demonstration that the waste form and waste package comply with the containment and controlled release requirements of 10 CFR Part 60. During 1986, NRC-sponsored corrosion research investigators applied statistical methods of experimental design, combined with cyclic voltammetry, to assess the potential for failure of HLW metal overpacks by stress corrosion cracking and localized corrosion. As a result of this work, a significant new understanding of localized corrosion in carbon steel, of particular importance to geologic disposal of HLW, was realized.

Work on the properties of spent reactor fuel as an HLW waste form was begun this year, and the Japan Atomic Energy Research Institute (JAERI) began a series of experiments of waste package and HLW glass waste form performance in high radiation environments, as part of the NRC-JAERI research information exchange agreement.

Geochemistry. The NRC has an active research program in the vital field of geochemistry related to the management of HLW. During this past year, researchers at Lawrence Berkeley Laboratory under NRC sponsorship demonstrated that radionuclide solubilities at 60xC, which were predicted using thermodynamic data measured at 25xC, do not compare well with solubilities measured at 60xC (NUREG/CR-4582). New solid phases for which 25xC data were unavailable appeared to be controlling solubilities at 60xC. These results show that thermodynamic data need to be developed by DOE over a realistic range of temperatures to understand repository radionuclide release.

In NRC studies at the University of California at Berkeley of bentonite packing and backfill materials for use in HLW repositories, it was found that surface diffusion increased transport through bentonite significantly over what had been expected. As a consequence of this work, the bulk diffusion coefficient used in modeling the performance of bentonite packing and backfill material will need to be increased.

A mathematical model of radionuclide transport using laboratory-based measurements was used to predict the distributions of radioisotopes being transported away from a uranium ore body. The predicted movement agreed very well with actual field observations. This work was conducted in cooperation with the Australian Atomic Energy Commission at a uranium ore site in the Northwest Territory of Australia. Further work at the site is being proposed for an international transport model validation project.

Borehole and Shaft Sealing. NRC's research program to assess experimentally the performance of existing technology for sealing boreholes at HLW repositories is continuing. During 1986, sealing research in crystalline host rock composed of granite and basalt was concluded. Some of the significant findings, reported in NUREG/CR-4642, were as follows:

- Sealing horizontal boreholes in the field with a swelling cement appears quite feasible.
- The hydraulic conductivity of cement plugs installed in a borehole increases about two orders of magnitude when the temperature is raised from about ambient (22 °C) to about 90-95 °C.
- The size of a cementitious borehole plug affects its sealing capability.
- Cement grout distribution in rock fractures was found to be uneven. However, the grout reduced the hydraulic conductivity of the fracture.
- The performance of bentonite/crushed basalt borehole plugs is dependent on the size of the crushed basalt and the ratio of bentonite to crushed basalt, by weight.

Thermohydraulic Studies. At Colorado State University, the NRC is supporting work on laboratory simulations of thermohydrologic interactions that may occur in HLW repositories. These experiments, designed to simulate realistic repository conditions by making use of the principle of dynamic similarity, employ laboratory models to observe heat and fluid flow within and around a repository and compare model predictions with the laboratory observations.

Rulemaking. In July 1986, the NRC published final procedural amendments to 10 CFR Part 60, dealing with site characterization and the participation of States and Indian tribes in the licensing process for an HLW repository. These amendments were needed to bring the procedures in Part 60 into conformity with those established by the Nuclear Waste Policy Act of 1982.

In June 1986, proposed amendments to Part 60 to eliminate inconsistencies between the EPA HLW standards and Part 60 were published for comment. NRC has received a wide variety of public comments, which are being considered in the preparation of the final amendments.

Low-Level Waste

NRC research in support of licensing activities for low-level radioactive waste (LLW) disposal facilities is focused on (1) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial (SLB) for LLW disposal, (2) evaluation of the overall performance of disposal systems, (3) water entry into disposal units, (4) performance of waste packages, (5) characterization of the LLW source term, and (6) mechanisms for transport of radionuclides from the disposal units. This research will be useful not only to the NRC licensing staff but also to States regulating LLW disposal. In order to make research results available to the States, meetings with research contractors are planned. The Department of Energy also sponsors an annual meeting at which DOE and NRC research results are presented; these meetings are well attended by State representatives.

Engineered Enhancements and Alternatives to Shallow Burial. There is great interest on the part of States and State compacts in alternatives to shallow land burial as it is currently practiced. Work begun in 1985 to identify and assess the importance of the key engineering design and safety features of a number of alternatives being considered by States and industry was completed. This work identified some engineered features common to many alternatives, as well as the relative contribution to safe disposal made by various features within a given alternative. The result of this work is available as NUREG/CR-4701.

Transport Modeling. An NRC-sponsored cooperative project between Atomic Energy of Canada Ltd. (AECL) and the Battelle Pacific Northwest Laboratories (PNL) has been using data collected from 40 years of LLW waste disposal at AECL's Chalk River facility to assess techniques for modeling LLW site performance. PNL is approaching the problem as though dealing with a pristine site, prior to waste disposal. They have been going through various stages of site analysis and site review, using those portions of the data base required to resolve licensing issues. The full data base from over 120 wells is being used to assess the validity of the model predictions and analyses based on a small data set typical of what a license applicant would be expected to develop during site characterization. Results to date indicate that such a limited data set (16 wells in this case) might be expected to conservatively bound the transport plume of radionuclides. Further, predictions based on the complete data set of 120 existing wells at the Nitrate Disposal Area did not offer significantly improved results. This exercise lends confidence to the practicality of modeling a site using a well chosen data set collected during site characterization. The project is now proceeding to look at a more complex site at Chalk River to confirm the results of this earlier work at a relatively simple site. This project is providing important insights into the design of data evaluation programs for future LLW disposal and the reliability of predictions based on the data.

LLW Waste Forms. In May 1983, the NRC issued a technical position paper that specified minimum performance standards for LLW waste forms. Current waste forms in commercial use are being tested by the Idaho National Engineering Laboratory to ensure that leaching characteristics and compressive strength of the waste forms are consistent with the standards specified in the technical position. Various decontamination wastes from actual power plants using commercial solidification processes such as Lomi, Candecon, NS-1, and Citrox are being investigated. In 1986, Brookhaven National Laboratory began an NRC research project to study the use of concrete and high-density polyethylene for LLW containers and engineered barrier materials. Representative samples of each material will be subjected to various environments expected in the waste forms and the surroundings—e.g., sulfates, acids, gamma fields—in order to study the failure and degradation mechanisms and, if possible, develop methods for predicting the performance of the materials over a period of 300-to-500 years.

Infiltration of Water. The University of California in cooperation with the University of Maryland is field testing at Beltsville, Md., a system of enhanced runoff and bioengineering to control water entry through trench covers. By artificially enhancing runoff and using vegetation to remove water through "evapotranspiration" (plant transpiration plus evaporation), water entry through disposal unit covers can be reduced to negligible amounts. Waste package degradation will therefore be reduced and the performance of the waste disposal system improved. The results of this work will be applicable to any disposal scheme employing earthen covers.

Radionuclide Mobilization by Plants. PNL completed an investigation of the role played by vegetation in radionuclide migration and found that plant roots exude agents that mobilize radionuclides to a degree greater than previously anticipated. The results of this research were factored into geochemical/hydrologic transport models for predicting the performance of an LLW disposal site.

Hydrology and Containment Transport. In August 1986, a report (NUREG/CR-4622) was issued which was jointly authored by the NRC research staff and PNL, Massachusetts Institute of Technology (MIT), and New Mexico State University researchers. The report presents the laboratory and modeling results that were used in the design of a field experiment to obtain data for validating the stochastic flow and transport models. Initially, the field work will focus on stochastic models developed by MIT. Ultimately, the study results will be used to provide guidance to the States and licensees for LLW site characterization and performance assessments.

RES has maintained technical supervision of a contract with the National Research Council to evaluate techniques for estimating probabilities of extreme floods. A panel of experts assembled by the Water Science and Technology Board of the National Research Council has held three meetings since January 1986 to review methods of determining extreme flood probabilities and make recommendations to the NRC. A manuscript covering major portions of their report summarizing the state of the science—with recommendations for interim methods and further research—has been prepared. Completion of the final report is anticipated in early 1987.

HEALTH PHYSICS

Radiation Protection and Health Effects

The NRC maintains a program of research and standards development in radiation protection intended to ensure continued protection of workers and the public from radiation and radioactive materials in connection with licensed activities. The program is currently focused on improvements in health physics measurement and the review of dose reduction research performed by other Federal agencies and industry. A goal is to provide acceptable performance standards for the many measurements required of licensees. The program also contributes to monitoring licensee performance in areas such as occupational dose and use of new dose reduction techniques.

The health effects research program has a goal of reducing the uncertainty associated with estimating health effects from exposure to radiation. Currently the staff reviews research funded by other agencies, such as the Department of Energy and the Department of Health and Human Services, and attempts to improve understanding of this critical area. Improved risk estimations are needed for assessing severe accident consequences and for establishing agency safety goals.

Brookhaven National Laboratory (BNL) ALARA Center. The BNL ALARA Center, funded by NRC, continued its work on surveillance of DOE and industry dose reduction and ALARA research. In 1986 BNL published Volume 2 of NUREG/CR-4409, which describes 140 national and international projects that contribute to dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination.

The center is recognized by the nuclear industry and others as a major source of information on new and effective dose reduction techniques and its publications are standard references for ALARA planning. The BNL staff is available through the center to the entire NRC organization for information and advice on all aspects of radiation protection.

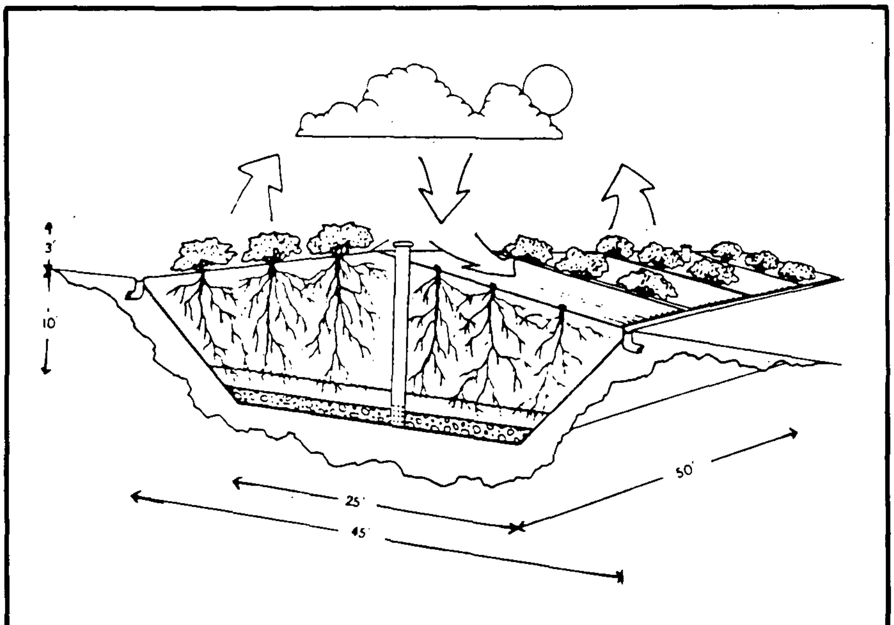
BNL reported in 1986 that dose reduction research is beginning to affect occupational radiation exposure. A clear reduction in exposures is observable in countries with dose reduction research programs—such as Japan, West Germany, Canada, Sweden, France, and the United States

Robotics. The Small Business Innovative Research contract for development of a surveillance robot is near completion. Remote Technology Corporation (REMOTEC) of Oak Ridge completed fabrication and testing of its robot and placed it in use at the Browns Ferry (Ala.) nuclear power plant. Preliminary analysis of the cost-benefit data collected at Browns Ferry indicates that cost savings far exceed projections, especially in reduced occupational exposure.

The robot is capable of radiation measurements, air sampling, contamination sampling, high-resolution color TV viewing, two-way voice communication with work crews, and many other modular measuring systems. It is programmable and can be moved from room to room in a contaminated enclosure.

The final REMOTEC report, a cost-benefit analysis and description of SURBOT's performance, will be published in early fiscal year 1987.

Proposed Revision to Radiation Protection Regulations. At the beginning of 1986, the NRC published for public comment a proposed revision of its basic radiation protection standards, 10 CFR Part 20. This issuance of a proposed rule was the culmination of efforts begun in 1980 when the



Part of NRC research on low-level radioactive waste disposal addresses the problem of water entering disposal units. A cooperative effort by the Universities of California and of Maryland to demonstrate the combined uses of engineering and vegetation to control water infiltration included the construction at Beltsville, Md., of four large lysimeters, which measure the percolation of water through soils, as part of a bioengineered system. The system can be used in waste storage structures both above and below ground.

NRC announced its intention to update its regulations for radiation protection. The update of 10 CFR Part 20 was intended to conform to the latest recommendations issued by various national and international radiation protection organizations and to revise the intake limits for radionuclides in air and water to reflect new dosimetry concepts and the more recent biological and metabolic data that have become available since the last complete revision of Part 20 was published.

By the end of fiscal year 1986, over 780 public comments had been received on the proposed rule. These comments, together with those on the Part 20 draft Backfit Analysis, will be analyzed early in fiscal year 1987 so that a recommendation can be made to the Commission on continuation of this rulemaking.

Bioassay. On December 3-4, 1985, public meetings on ultrasensitive bioassay techniques and nephrotoxicity of uranium were held by the staff. Approximately 60 people attended the bioassay meeting, while 30 were present for the nephrotoxicity meeting. Based on the results of animal experiments, it appears that the limit used for uranium in the kidney may be too high, possibly by a factor of five, for soluble uranium compounds. It may thus be necessary to lower the NRC limit on the intake of soluble uranium compounds. If this limit were to be reduced, information gathered at the meeting indicates that a number of practical techniques exist for the measurement of urinary uranium concentrations at sufficiently low levels for verifying compliance with lower intake standards.

Bioassay at Uranium Mills. An internal dosimetry model based on ICRP 30 developed for use in estimating the urinary natural uranium concentration at various times after inhalation of yellowcake or ore dust is described in a document (NUREG-0874) prepared by the staff for issuance in July 1986. New experimental data that illustrate the dependence of lung solubility on the thermal history of the inhaled yellowcake are incorporated in this model. This publication will permit followup guidance to licensees on bioassay methods to be developed by the staff.

Metabolism and Internal Dosimetry. An interim report (NUREG/CR-4355, Vol. 1) on development of metabolic models for alkaline earth and actinide radionuclides was published in September 1985. This report presented an overview of the metabolism of plutonium (IV) in monkeys. Detailed data on distribution in soft tissues and the skeleton as well as retention and elimination curves are provided. Differences from the International Commission on Radiological Protection (ICRP) models were evident, especially for trabecular and compact bone surfaces. Therefore, calculated doses to bone surfaces will be different from those that would be calculated with the ICRP models.

A report (NUREG/CR-3955) describing methodology for age- and sex-specific calculation of thyroid dose from administered iodine-131 was published in September 1986. It provides updated values using current dosimetric models.

Improvement of Health Models. Work continues on improving the health effects models described in NUREG/CR-4214, published for comment in 1985. These models are applied to such major subject areas as emergency planning, trial use of NRC's draft safety goals, risk estimate uncertainty analysis, staff environmental reports, and hearing testimony.

The experimental phase of studies of the early effects of radionuclides (alone and combined with external irradiation), using rats and dogs, has been completed. When the data analysis is completed, the results will be published, in fiscal year 1987. This work is intended to improve estimates of early lung effects following severe reactor accidents.

Continuing Projects. Continuing projects include a long-term epidemiological study of thorium workers (males) and radon dial painters (female). A large-scale study in mice of the radiobiological effectiveness of neutrons is in its final year. The results to date tend to support the recent ICRP recommendation to increase the neutron quality factor from 10 to 20.

Chemical Decontamination. The NRC continued to develop an information base for assessing the safety and effectiveness of decontamination alternatives for reducing occupational dose in nuclear power plants and for assessing the impact of decontaminations on nuclear plant solidification systems. Observations and measurements were made of selected chemical decontamination activities at the Millstone Unit 1 (Conn.) nuclear power plant and at Quad Cities Unit 1 (Ill.). A report analyzing these results and similar earlier measurements conducted at other nuclear power plants was published in 1986 (NUREG/CR-4445). NUREG/CR-3444, also published in 1986, describes the impact of light-water-reactor decontaminations on solidification, waste disposal, and associated occupational exposure.

Occupational Exposure Data System. In 1969, the Atomic Energy Commission began requiring certain licensees to submit reports on occupational radiation doses received by workers. These data are collected and computerized in an NRC system called REIRS (Radiation Exposure Information Reporting System). The system provides a permanent record of the data and permits expeditious analyses of the two kinds of reports required (annual statistical summaries and individual termination reports). Exposures received as a result of medical procedures are not required to be reported.

Summaries of the annual statistical reports for 1984 revealed that the seven categories of licensees required to report monitored about 193,000 individuals, of whom about 56 percent received measurable doses. The workers received a collective dose of 59,000 person-rems, or an average annual dose of 0.5 rem per worker among those receiving a measurable dose (0.3 rem per monitored person when the entire monitored population is considered). Of the persons monitored, 88 percent worked in nuclear power plants, and they incurred about 93 percent of the total annual collective dose. The average annual measurable dose received by individual nuclear power plant workers remained about 0.6 rem. However, compilations of the exposure data reported by nuclear power plants for calendar year 1985 indicate that the annual collective dose decreased by 20 percent, falling to its lowest value in five years. The average annual dose also decreased to 0.5 rem, which is less than 10 percent of the applicable dose limit.

A second kind of exposure report required of certain NRC licensees provides identification and dose data each time a monitored individual terminates work at the licensed facility. Such information is now maintained for some 360,000 persons, most of whom worked at nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories and to analyze the data for trends. The data also help ensure that transient workers moving from plant to plant do not receive doses in excess of regulatory limits. For example, analysis of the data reported for 67,400 persons terminating employment during 1984 revealed that 6,000 of them worked at two or more nuclear power facilities, and that none of them received doses in excess of the regulatory limits as a result of their multiple employment.

REGULATION DEVELOPMENT

Regulatory Analysis and Support

The NRC continued its efforts in 1986 to develop improved methods for performing value-impact assessments on reactor and non-reactor regulatory issues and plant-specific backfits; to identify, analyze, and, where warranted, relax regulatory requirements that may be of marginal importance to safety; and to develop a technical basis—employing results of new source term research—for protective strategies related to emergency planning for accidents at nuclear power plants. Contractual technical assistance from DOE laboratories was provided to the staff to support these major activities. In addition, RES continued assisting the EDO-directed management of rulemaking, including the RES independent review of new and ongoing rulemaking efforts.

The Office of Nuclear Regulatory Research has among its prime concerns the development and implementation of systematic methods that facilitate NRC decision-making. With the formation of the Committee to Review Generic Requirements in 1981, the recent issuance of a revised backfitting rule (§ 50.109 of 10 CFR Part 50), the endorsement of safety goals for nuclear power plants, and new source term research data, the need for regulatory analysis as a means to fostering a more disciplined regulatory process will continue. During the report period, the Commission initiated and completed several safety-related regulatory analyses using the methods prescribed in the value-impact handbook (NUREG/CR-3568), in both reactor and non-reactor applications. The methods and procedures of the handbook have been incorporated, by reference, in the revised regulatory analysis guidelines (NUREG/BR-0058) and Manual Chapter 0514 (Management of Plant Specific Backfitting of Nuclear Power Plants) for use by the NRC staff and industry in evaluating the need for and effectiveness of a variety of regulatory actions—including major rulemaking, standards development, and backfitting safety improvements on nuclear power plants. Workshops were held in the Regions to demonstrate to the staff the use of existing cost-benefit methods and resources available to assist in performing high-quality regulatory analyses, which as a matter of policy must accompany proposed rules, guides, and plant-specific backfits.

A program was initiated in 1985 and continued in 1986 to investigate existing regulatory requirements in terms of their risk effectiveness and to eliminate or modify those requirements that have a marginal safety importance. Two volumes of a research report were published (NUREG/CR-4330, Volumes 1 and 2). The first volume summarized the results of a survey conducted to identify regulatory requirements that may have marginal importance to safety. The second provided detailed technical evaluation of requirements in three regulatory areas: (1) reactor containment leakage, (2) leakage control system of main steam isolation valves, and (3) licensing safety review of fuel design. The NRC staff is currently considering these research results and will recommend whether to eliminate or modify related requirements that have marginal safety importance. In 1986, technical evaluation was initiated in four additional regulatory areas: (1) post-accident sampling system, (2) impregnated charcoal filters, (3) recombiners in BWR Mark I and Mark II, and (4) turbine missiles.

Management of NRC Rulemaking

Control of Rulemaking. In February 1984, the NRC Executive Director for Operations (EDO) directed that all offices reporting to the EDO and responsible for rulemaking must obtain the EDO's approval to begin and continue a proposed rulemaking action. The directive was aimed at ensuring that rulemaking activity was necessary and would be effective, efficient, timely, and of high quality.

RES was given the task of independently reviewing prospective rulemakings and making recommendations to the EDO as to whether to proceed with them. Late in 1985, the RES role was expanded to also conduct annual independent reviews and make similar recommendations to the EDO concerning ongoing rulemakings.

During fiscal year 1986, there were 68 rulemakings calling for independent review by RES and EDO approval for initiation or continuation. The status of these reviews, as of September 30, 1986, is provided in Table 1.

It is estimated that in fiscal year 1987 there will be approximately 47 rulemakings that will require RES independent review and EDO approval for initiation or continuation.

Timeliness of Rulemaking. RES also established a tracking and feedback system to help the EDO ensure the timeliness of approved rulemaking actions. To accomplish this, existing quarterly updating of rulemaking entries in the NRC Regulatory Agenda (NUREG-0936) was modified to require a timetable for each ongoing rulemaking sponsored by an office reporting to the EDO. In addition, RES reviews these timetables and provides the EDO with a summary report on the timeliness of the schedules of each rulemaking.

Table 1. Rulemakings Reviewed in FY 1986

Reviews completed in FY 1986		57
Approved by EDO for continuation	41	
Approved by EDO for termination	15	
Presently under review by EDO	1	
Reviews for initiation of rulemakings under way		3
Annual independent reviews under way		0
Reviews scheduled for FY 1986 but delayed until FY 1987		8
	TOTAL	68

Emergency Preparedness

In 1984, the Commission started a generic rulemaking entitled "Consideration of Earthquakes in the Context of Emergency Preparedness" (Part 50) intended "to address whether the concern for potential seismic impact on emergency planning is significant enough in large portions of the nation to warrant the amendment of the regulations to specifically consider these impacts." In 1986, the D.C. Court of Appeals agreed with the Commission's proposed rule that emergency plans need not consider the effects of earthquakes. The Commission approved withdrawal of the proposed rule and let the court decision stand.

Work is also under way to develop a more effective strategy for protective action decision-making for responding to accidents at nuclear power plants based on the new source term information.

TRANSPORTATION SAFETY RESEARCH

Efforts were completed in 1986 to document the extent of protection provided by spent fuel casks designed to existing regulatory standards when those casks are subjected to severe transportation accident conditions. A report of the study was submitted for an independent peer review, the results of which will be incorporated in the final report. The results of the study will be used as a basis for correcting deficiencies and to demonstrate the degree of protection provided for shipments of spent nuclear fuel.

MATERIALS SAFETY

In April 1985, the NRC proposed new regulations for well logging using radioactive sealed sources and tracers. Well logging is the measurement of the characteristics of the rock strata around a well by means of various instruments that are lowered down into the well. Approximately 100 public comments were received on the proposed regulations by the end of the comment period. A final rule is expected to be published by late 1986.

NATIONAL STANDARDS PROGRAM

The national standards program is conducted by the American National Standards Institute (ANSI). ANSI acts as a clearinghouse to coordinate the work of standards development in the private sector.

The NRC staff is active in the national standards program, particularly with respect to setting priorities so that regulatory views are known regarding the standards that can be most useful in protecting the public health and safety. NRC participation is based on the need for national standards to define acceptable ways of implementing the NRC's basic safety regulations.

Approximately 196 NRC staff members serve on working groups organized by technical and professional societies.

Proceedings and Litigation

CHAPTER 12

This chapter comprises two major treatments: (1) a description of the operations and report on select proceedings involving the NRC's Atomic Safety and Licensing Board Panel and the Atomic Safety and Licensing Appeal Panel; and (2) a review of noteworthy legal action and litigation involving the Commission, covering cases pending and closed.

ATOMIC SAFETY AND LICENSING BOARD PANEL

In fiscal year 1986, the Atomic Safety and Licensing Boards authorized operating licenses for six new nuclear power plant units and completed a total of 23 extensive proceedings. At the same time, the boards addressed a rapidly expanding number of smaller cases related to previously licensed facilities, and also began preparations for dealing with the extremely complex issues associated with the proposed high-level nuclear waste repository.

The Atomic Energy Act of 1954 requires that a public hearing be held on every application for a construction permit for a nuclear power plant or related facility. In certain circumstances, hearings are also held in connection with operating licenses, license amendments, antitrust issues, enforcement and civil penalty cases, and other matters as directed by the Commission. (See "The Licensing Process," in Chapter 2.) Boards composed of three administrative judges drawn from the Atomic Safety and Licensing Board Panel (ASLBP) perform the Commission's hearing function and render initial decisions in licensing cases; single administrative judges and administrative law judges also hear and decide other matters. These hearings are the Commission's principal public forum in which individuals and organizations can voice their interest in a particular licensing, enforcement or other matter of public concern and have those interests adjudicated by an independent tribunal.

As of September 30, 1986, the panel included 21 permanent and 25 part-time administrative judges drawn from various professions. There were 16 lawyers, 15 environmental scientists, 7 engineers, 5 physicists, 1 medical doctor, 1 economist and 1 chemist. (See Appendix 2 for the names of panel members.) The Commission appoints administrative judges to the panel based upon recognized experience, achievement and independence in the appointee's field. Judges are assigned to three-member Licensing Boards in cases in which their professional expertise will help to resolve the issues litigated. Generally, Licensing Boards consist of a lawyer as chairman, a nuclear engineer or reactor physicist, and an environmental scientist.

The hearing on a particular application for a nuclear facility license may be divided into several phases, each focusing on a particular licensing concern, for example: (1) health, safety, or the common defense and security aspects of the application, as required by the Atomic Energy Act; (2) environmental considerations, as required by the National Environmental Policy Act (NEPA); and (3) emergency planning requirements. These matters, as well as especially complex technical issues, are frequently the subject of separate initial decisions by the Licensing Boards.

Administration

As cases have become more intensely and actively litigated and the issues to be decided have grown increasingly complex, the effective management of the logistics of the hearing process has become especially important. In this effort, the boards have in the past been supported by a staff which included management personnel, a legal counsel, law clerks, a librarian, legal secretaries, and docket, computer, and information specialists. During fiscal year 1986, however, the panel's last two law clerks completed their terms and, because of required staffing reductions, were not replaced. To compensate for these actual and for anticipated restrictions in support personnel, the panel has aggressively pursued the automation of hearing functions. Administrative support through automation has been achieved through the use of such equipment as personal computers, word processors, a joint Licensing Panel and Appeal Panel library, the LEXIS automated legal research system, and other computerized systems. An internal computerized Hearing Status Report has a virtually complete data base and can rapidly generate valuable case management information.

Of particular note, the panel's Computer Assistance Project (CAP) to expedite large cases is well under way. Computerization of the Indian Point (N.Y.) record in 1983 proved that substantial time and labor can be saved by using a full text word search transcript. Building on that experience, the panel obtained advice and recommendations from consultants with both legal and computer expertise. They recommended a system combining personal computers with off-the-shelf software to establish a full text word search computerized record. In place at the outset of a large case, the system would permit electronic filing, expedited discovery, computerized transcripts, greatly expedited record searches, and faster and more complete decision-making and decision-writing. By using resources for the most part already in place, the cost-benefit ratio for large cases should be substantial. The transcripts of two cases—the Braidwood (Ill.) facility operating license and the Three Mile Island (Pa.) leak rate inquiry—were computerized during fiscal year 1986.

The Caseload

During the fiscal year ending September 30, 1986, Licensing Boards conducted 58 proceedings involving nuclear power plants and other nuclear facilities, with a construction value well in excess of $70 billion. Forty percent of these proceedings were completed. Some 172 days of hearing were held, comprising 111 days of trial and 61 days of pre-hearing conferences. Twenty-three proceedings were closed, while twenty-six new cases were opened. The operation of six nuclear power plant units was authorized.

More than twice as many new cases were opened in fiscal year 1986 as in the preceding fiscal year. The increase in new cases filed reflects a shift toward numerous smaller cases of greater diversity, a shift which is expected to continue over the next five years.

At the same time, however, the panel has begun the extensive advance planning process required to handle what will be the largest case in its history, and perhaps the largest Federal administrative proceeding ever. By early in the next decade, the U.S. Department of Energy is expected to file its application for construction of a high-level nuclear waste repository. Dozens of well-funded intervening parties are prepared to participate in a case that may involve more than 16 million documents. The panel's efforts to expand its ability to utilize sophisticated computer systems for document and hearing management will be essential to its role in this massive proceeding.

Hearing Procedure

The heavy ASLBP caseload, combined with increasing public awareness and involvement in the licensing process, has made effective hearing management critical to the timely completion of licensing decisions. Using the procedural tools available under Commission regulations, Licensing Boards have more sharply focused efforts to assure that issues for hearing are soundly based and well-defined. Pre-hearing conferences are utilized extensively for the purposes of reviewing and refining proposed contentions, defining the scope of relevant discovery, and developing realistic hearing schedules. The discovery process itself is closely monitored, in order to eliminate unnecessary or duplicate efforts and to assure the early resolution of potentially time-consuming disputes. As a result of this kind of active management, almost 90 per cent of the contentions filed in operating license proceedings were resolved prior to hearing. Most importantly, however, these efficiencies have been achieved through hearing management practices that ensure the fundamental fairness to all parties mandated by law.

Cases of Note

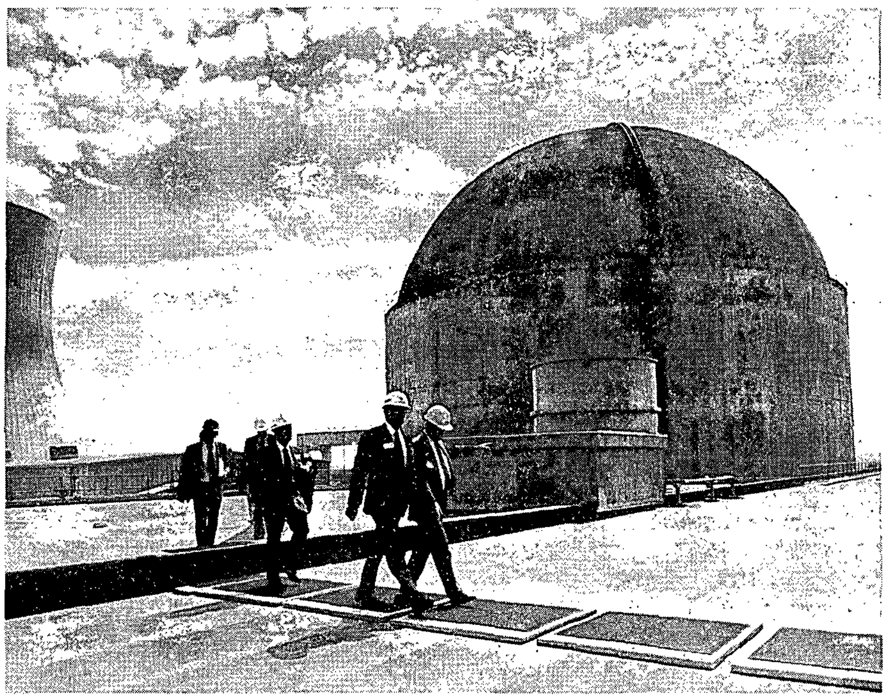
Operating Licenses. On April 28, 1986, the Licensing Board in *Shearon Harris* (N.C.) issued a final decision resolving two remaining contentions in favor of the applicant and authorizing the issuance of fuel loading and operating licenses. The board found that alleged drug use by workers at the plant had not been widespread, and that there was no evidence that drug use had resulted in any specific deficiency in work at the plant or in any other significant safety concern. The board also considered the issue of notification of area residents in the event of an emergency, in light of the Commission's requirement of "essentially 100%" notification. It found that a combination of sirens and tone-alert radios would result in notification of 90 percent of the residents within five miles of the plant within 15 minutes, and that the addition of mobile alerting would raise the coverage to 100 percent within 45 minutes.

In the *Limerick* (Pa.) proceeding, the Licensing Board on September 5, 1986, resolved in favor of the applicant a remanded issue concerning reasonable assurance of the availability of an adequate number of bus drivers to evacuate students in the event of a radiological emergency.

On August 29, 1986, the Licensing Board in *South Texas* (Tex.) issued a third and final Partial Initial Decision which resolved all remaining issues. The board granted summary disposition of an issue concerning the design of safety structures to withstand hurricane-generated missiles. The board found that the probability of damage to portions of three safety structures which had not been designed to withstand such missiles was so low that they need not be redesigned to withstand missiles. The board also dismissed another issue and completed its review of uncontested questions, accepting the applicants' proposal to modify their emergency plan as an adequate basis for finding emergency notification satisfactory. The board authorized licenses for fuel loading, low-power operations and full-power operations. This case was completed fully 10 months ahead of schedule.

Show Cause. On March 2, 1984, the NRC Staff issued an Order to Show Cause to Kerr-McGee Chemical Corporation to require the cleanup of thorium from Kress Creek (Ill.), alleging that the thorium had escaped from Kerr-McGee's West Chicago Rare Earths facility prior to its closing in 1973. Staff relied on the Environmental Protection Agency's "radium-in-soil" standard which was adopted pursuant to the Uranium Mill Tailings and Radiation Control Act of 1978, but conceded that this standard could not be retroactively applied and thus could be viewed only as guidance.

Following an evidentiary hearing, the board concluded that the "radium-in-soil" standard was not appropriate guidance in the context of the hazard posed by Kress Creek, On April 28, 1986, the NRC's Atomic Safety and Licensing Board Panel issued a final decision in a case involving the Shearon Harris (N.C.) nuclear power plant in favor of the plant operators and authorizing licenses to be issued for fuel loading and operation. Shown here are NRC Chairman Lando W. Zech and J. L. Willis, General Manager for the facility, during a tour of the plant in September 1986. Immediately behind is NRC Region II Administrator J. N. Grace.



that the dose limitation standards contained in Part 20 were both legally applicable and appropriate, and that the record did not show that the Part 20 standards were violated. The board rejected the "radium-in-soil" standard because the hazard regulated by that standard—thoron emissions—does not create a threat to health in the context of Kress Creek. (Thoron is a gaseous radioactive element formed from thorium and isotopic with radon.) The board dismissed the Order to Show Cause on June 19, 1986.

Civil Penalty . In *Reich Geo-Physical,* the presiding administrative law judge imposed a civil penalty of $1,600 on the licensee, finding that it was proper for the NRC staff to have treated a number of related violations as constituting, in the aggregate, a more serious Severity Level II violation. In light of the licensee's financial circumstances, the judge also granted its request to pay the penalty in installments.

Special Proceedings. On May 2, 1986, an *ad hoc* review group presented a major report on an incident at the Davis-Besse (Ohio) nuclear power plant and the NRC's response to that incident. The five-member group had been established by the Commission, which appointed two panel judges to it, one to serve as Chairman.

Four areas were identified by the Commission for examination: (1) pre-event interactions between the licensee and the NRC concerning the reliability of the auxiliary feedwater system and associated systems; (2) pre-event probabilistic assessments of the reliability of plant safety systems, NRC's review of them, and their use in regulatory decision-making; (3) licensee management, operation and maintenance programs, as they may have contributed to equipment failures, and NRC oversight of such programs; and (4) the mandate, the capabilities of the members, the operation, and results of the NRC Davis-Besse Incident Investigation Team (IIT), and the use to which its report was put by the regulatory staff.

In fulfilling its charter, the Group interviewed over 50 individuals from Headquarters and Regional staff, Toledo Edison (the licensee for Davis-Besse), Babcock & Wilcox (maker of the reactor), the Institute of Nuclear Power Operations, and management officials from Commonwealth Edison, Duke Power, Florida Power and the Sacramento Municipal Utility District. It also interviewed the Davis-Besse Incident Investigation Team and the leaders of the San Onofre (Cal.) and Sacramento IITs.

The Group found that the staff's Incident Investigation Program in general and the performance of the Davis-Besse IIT in particular were adequate. Recommendations for improvements included changes already being implemented by the staff. The Group also found, however, that Toledo Edison and NRC staff actions prior to the incident were not adequate to prevent the incident. The Group recommended improving communications between the staff and licensees, improving communications among the various staff organizational units, and encouraging prompt NRC decision-making in the resolution of plant-specific problems. The Group also endorsed the Commission's 1986 Policy and Planning Guidance directives to conduct a comprehensive review of NRC regulations and to seek a reduction in the number and prescriptiveness of both regulations and Technical Specifications.

ATOMIC SAFETY AND LICENSING APPEAL BOARDS

Atomic Safety and Licensing Appeal Boards, each consisting of three members, perform review functions for the Commission in facility licensing proceedings and in others the Commission may specify. Unless the Commission decides to review an Appeal Board decision, that decision becomes the final agency order and is subject only to judicial review in a Federal court of appeals. The board for each proceeding is selected from among the members of the Atomic Safety and Licensing Appeal Panel (ASLAP) by the panel chairman. (See Appendix 2 for further description and membership of the panel.)

Under Commission rules, the Appeal Boards hear appeals from Licensing Board decisions and certain rulings on intervention petitions by members of the public seeking to become parties to the proceeding. They may also review Licensing Board decisions on their own initiative in the absence of any appeal. In limited circumstances, Appeal Boards also consider questions posed by the parties or the Licensing Board, or rulings referred by a Licensing Board, while the proceeding is still in progress. Appeal Boards also occasionally conduct evidentiary hearings, either as part of their appellate review function or on direction from the Commission.

With the gradual completion of facility license proceedings begun years earlier and the licensing of the plants involved, there were fewer active proceedings than usual before the Appeal Boards this past year. The necessity for Appeal Board review of Licensing Board decisions and rulings or other Appeal Board action, however, continued at about the previous level for most of the year. As in the year before, the Appeal Boards issued more than 30 decisions of sufficient significance to be published in the Nuclear Regulatory Commission Issuances, which is the permanent collection of NRC licensing and other decisions.

A substantial amount of Appeal Board time and effort was taken up during the fiscal year in the licensing proceedings for four plants: Limerick (Pa.), Shoreham (N.Y.), Perry (Ohio), and the restart of Three Mile Island Unit 1 (Pa.) These proceedings alone were responsible for over two-thirds of the published decisions. The more significant of these and other Appeal Board decisions are highlighted below.

Limerick

Under Commission regulations, an operating license will not be issued for a nuclear facility unless the NRC finds that there is reasonable assurance that adequate measures to protect the public can and will be taken in the event of a radiological emergency. Before such a finding is made, State and local emergency response plans are reviewed by the Federal Emergency Management Agency (FEMA). The findings of FEMA are then acted on by the NRC.

In *Limerick* (Pa.), several intervenors, including the inmates of a nearby state prison, challenged the adequacy of the emergency response plans for the facility in various respects. The several Licensing Board decisions generally upholding the plans led to a number of appeals. One issue on appeal concerned the adequacy of arrangements for medical services for workers who are both injured and contaminated by radiation while still at the plant site. An issue in a separate appeal involved whether an adequate number of buses and drivers would be available to evacuate students from two school districts located near the plant. With respect to both matters, the Appeal Board found the plan inadequate and returned the cases to the Licensing Board for further development.

One of the issues in still another appeal involved the Licensing Board's rejection of several aspects of the prisoners' contention that the emergency response plan lacked reasonable assurance that the plan would protect them and the prison staff in the event of an emergency at Limerick. The Appeal Board agreed with the Licensing Board's rejection of the claim in all but one respect. According to the Appeal Board, the lower board's refusal to hear the inmates' challenge of the adequacy of the call up system for mobilizing the entire prison work force in a radiological emergency was incorrect. The Appeal Board thus returned this issue to the Licensing Board for hearing.

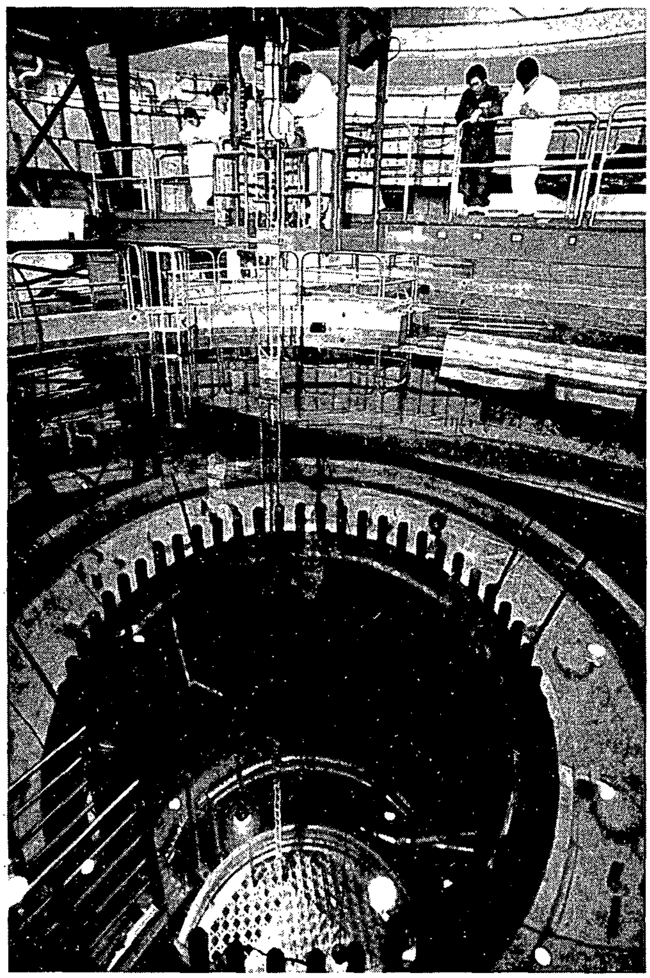
Shoreham

The adequacy of the emergency response plan for the Shoreham (N.Y.) nuclear power plant was also a major point of dispute in the operating license proceeding for the facility on Long Island. The applicant had prepared its plan without the cooperation of the State of New York and the county in which the facility is located. This situation came about because both governmental entities opposed issuance of the license on the ground, among others, that no response plan can adequately protect the health, welfare, and safety of the local residents. A critical issue before the Appeal Board was whether Federal law overrides State law precluding the applicant, using non-government personnel, from carrying out material features of the plan. After extensive review of the applicable law, the Appeal Board agreed with the Licensing Board that, contrary to the applicant's claim, Federal law did not preempt State law in the circumstances. One of the remaining issues involved the applicant's further argument that, in any event, the State and county would respond in good faith in the event of an actual emergency, and thus it would not be necessary to rely on non-governmental personnel to execute the provisions of the plan in question. The Appeal Board upheld the Licensing Board's determination that the prospect of such an uncoordinated and unrehearsed response was insufficient to provide the requisite assurance of public protection. Subsequently, the Commission itself determined that the State and local governments would exert their best effort in an emergency, and sent the matter back for further evidentiary hearing and evaluation on that basis.

In the same proceeding, the Appeal Board was confronted with the question of whether, in an emergency, the Nassau Coliseum, located some 43 miles from the plant, could adequately serve as a reception center for the processing and decontamination of evacuees from the emergency planning zone surrounding the plant. The Appeal Board determined that the matter needed further exploration by the Licensing Board.

Perry

In *Perry* (Ohio), a stay of the issuance of an operating license for the plant was sought on the ground that the intervenor, who lived near the plant, had a family history of cancer and was, therefore, especially at risk. The Appeal Board denied the stay. According to the board, the asserted reason was insufficient to satisfy the most important of the criteria for granting a stay—that the injury claimed was "both certain and great." In another decision involving the Perry plant, the Appeal Board considered whether the plant's system for disposing of any hydrogen that may be generated in the event of an accident is adequate to prevent a severe explosion within the containment building, and whether the plant's diesel generators could be relied on to generate emergency onsite power when needed. As to both, the Appeal Board agreed with the Licensing Board's determination in favor of the applicant.



An NRC Atomic Safety and Licensing Appeal Board continued, in 1986, to consider objections of State and local government to issuance of an operating license for the Shoreham nuclear power plant on Long Island, N.Y. Initial fuel loading of the reactor took place at the start of 1985.

TMI-Restart Proceeding

Vestiges of the *TMI-Restart* (Pa.) proceeding carried over from the previous years into the past year. As a result, the Appeal Board had occasion to review the Licensing Board's final two Partial Initial Decisions in the part of the proceeding that dealt with training and management integrity. In both cases, the Appeal Board upheld the Licensing Board's decision in favor of the licensee.

Other Proceedings

Other proceedings requiring Appeal Board action included *Braidwood* (Ill.), *Catawba* (N.C.), *Midland* (Mich.), *Pilgrim* (Mass.), *Seabrook* (N.H.), *Shearon Harris* (N.C.) and *Turkey Point* (Fla.). These proceedings raised a variety of procedural and substantive issues involving matters of possible safety significance as well as the environment.

COMMISSION DECISIONS

Some of the Commission's more significant decisions during fiscal year 1986 are discussed below. The Commission's actions on export licensing cases are discussed in Chapter 10.

Motion to Reopen Denied

In *Louisiana Power & Light Co.* (Waterford Steam Electric Station, Unit 3), CLI-86-1, 23 NRC 1 (January 30, 1986), the Commission denied intervenors' motion to reopen a hearing, ruling that the fact that an NRC investigation is going on does not in itself mean a violation has occurred.

In ALAB-812, 22 NRC 5 (1985), the Appeal Board denied a motion to reopen except as it pertained to pending investigations by the NRC's Office of Investigations (OI). With regard to the OI investigations, the board had sought more information from OI, but had been unable to obtain sufficient information to rule out all possible grounds for the intervenors' charges. Concluding that only the Commission could obtain full access to information developed by OI, the Appeal Board referred the matter to the Commission for resolution.

The Commission held that the motion to reopen did not meet the Commission's standards for reopening. The Commission found that the material proffered by the intervenors in support of their motion was not the type of relevant, material and reliable new information required to reopen a record. In this regard, the Commission noted that the bare pendency of an OI investigation does not indicate that there has been a violation of NRC requirements involving wrongdoing—all it indicates is that there is an allegation being investigated.

The Commission also concluded that the Appeal Board lacked authority to seek additional information from OI. The Commission held that the Policy Statement on Investigations, Inspections, and Adjudicatory Proceedings, 49 FR 36, 032 (1984), which in certain circumstances allows *ex parte in camera* examination by adjudicatory boards of investigatory information, applies only to issues already accepted by the board for litigation. As to the board's authority to pursue uncontested issues *sua sponte,* it applies where specific facts are brought to its attention indicating that there is a serious safety, environmental, or common defense and security matter. In the present case, the mere allegation of pending OI investigations did not of itself raise a serious safety question.

Construction Permit Expires—Two Major Issues

The Commission issued two major decisions involving the Comanche Peak (Tex.) facility. In *Texas Utilities Electric Co.* (Comanche Peak Steam Electric Station, Unit 1), CLI-86-4, 23 NRC 113 (March 13, 1986), In CLI-86-4, the Commission addressed the question of temporary renewal of an expired construction permit. Comanche Peak Unit 1 had a construction permit with an expiration date of August 1, 1985. Construction continued beyond that date without the plant owner's filing an extension request. Section 185 of the Atomic Energy Act provides that, unless the construction of the facility is completed by the expiration date, the construction permit expires. Some six months after the expiration date, the plant's owner applied for an extension of the construction permit. Two days later, an intervenor in the operating license proceeding for the plant filed a pleading with the Commission seeking: (1) the imposition of a civil penalty against the applicant for engaging in construction activities at Comanche Peak between the construction permit expiration date and extension filing date; (2) a definitive order directing the applicant to file an application for a new construction permit and to cease all construction activities at Comanche Peak; (3) a determination that "significant hazards considerations" existed in any extension of the construction permit; and (4) a hearing before a Licensing Board on the request for an extension. While this matter was pending before the Commission, the NRC staff granted the extension, having decided that there were no significant hazards considerations involved. In response, the intervenor moved that the Commission stay the effectiveness of the construction permit extension.

In CLI-86-4, the Commission denied, *inter alia,* both the request for a halt to construction and the request for the institution of a new construction permit proceeding. In taking this action, the Commission held that, under the Atomic Energy Act, the expiration of the construction permit did not automatically effect its forfeiture and bar the Commission from considering an application for an extension. The Commission, in making this determination, noted that a similar provision of the Communications Act of 1934 (governing the issuance of radio station construction permits by the Federal Communications Commission) had been so interpreted by the Court of Appeals for the D.C. Circuit.

The Commission also denied the intervenor's request for a finding that extension of the construction permit involves significant hazards considerations. The Commission pointed out that the term "no significant hazards consideration" is directed to consideration of radioactive hazards. Because the grant of the extension did not result in any substantive change in the design and construction methods from those permitted under the original permit, the extension did not involve any significant hazards consideration.

Finally, the Commission denied the request because it found that the intervenor did not make a convincing showing on the four traditional stay criteria and referred the intervenor's request for a hearing to the Licensing Board Panel.

In CLI-86-15, 24 NRC ____ (Sept. 19, 1986), also involving Comanche Peak Unit 1, the Commission reviewed a question sent forward by the Appeal Board in the hearing authorized in CLI-86-4. The intervenors in the licensing proceeding attempted to intervene in the construction permit extension hearing. The Licensing Board admitted the intervenors with a consolidated contention. The permittee and the staff filed an interlocutory appeal arguing that the contention was barred under previous Commission decisions. The Appeal Board certified the question to the Commission.

The Commission reviewed the existing case law and decided that a permittee may demonstrate "good cause" for a construction permit extension in one of two ways: the permittee may demonstrate that there was good cause for the past delay in plant completion, or the permittee may show that its current and future actions constitute "good cause" for an allowance of more time for plant completion. Therefore, when the permittee asserts the need for more time to correct safety deficiencies, the Licensing Board should not look only to past conduct to determine whether there is "good cause" for the extension. The Commission then returned the case to the Appeal Board to determine whether the intervenor's contentions met the applicable criteria.

Appeal Board Exceeds Authority

In *Cleveland Electric Illuminating Co.* (Perry Nuclear Power Plant, Units 1 and 2), CLI-86-7, 23 NRC 233 (April 18, 1986), the Commission ruled that an Appeal Board must decide a motion to reopen strictly on the basis of the information provided in the filing.

On January 31, 1986, an earthquake occurred in Ohio near the site of the Perry nuclear plant. Three days later, an intervenor in the proceeding filed with the Appeal Board a motion to reopen the record for the purpose of admitting a new contention challenging the adequacy of the plant's seismic design. Finding itself unable to determine from the papers filed whether the issue raised by the motion had true safety significance, the Appeal Board decided to hold an exploratory hearing to aid it in making that determination and issued orders (unpublished) to that effect.

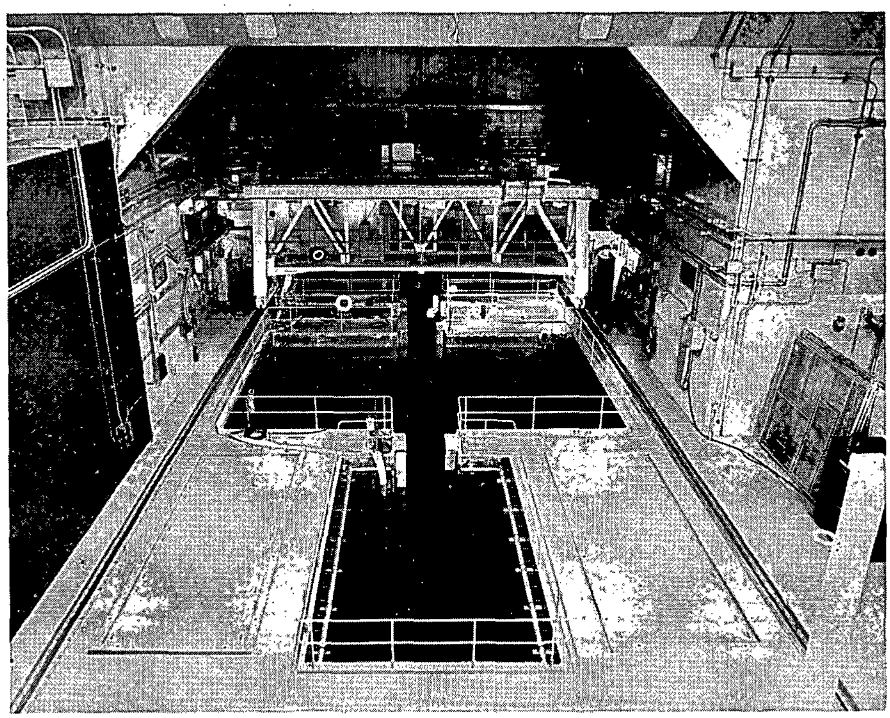
In an exercise of its "inherent supervisory authority over the conduct of NRC adjudications," the Commission took review of the orders on its own motion. Citing *Louisiana Power & Light Co.* (Waterford Steam Electric Station, Unit 3), CLI-86-1, 23 NRC 1 (1986), the Commission emphasized that a board is to decide a motion to reopen on the information before it, and that it has no authority to engage in discovery to supplement that information. As the Commission saw it, when the Appeal Board here had decided it needed additional information to determine whether the new issue raised by the intervenor had true safety significance, it was finding that the intervenor had failed to meet the reopening standard. (Among other criteria, a motion to reopen must raise a significant safety or environmental concern.) Accordingly, the Commission held that the Appeal Board had no authority to act as it did, vacated the board's orders, and denied the motion to reopen.

Shoreham Station—Four Major Cases

In 1986, the Commission issued four major decisions concerning the Shoreham (N.Y.) nuclear power plant, all concerning the emergency plan for the facility. Litigation concerning this plan was still in progress at the close of the report period.

In *Long Island Lighting Company* (Shoreham Nuclear Power Station, Unit 1), CLI-86-14, 24 NRC ___ (January 30, 1986), the Commission denied intervenors' motion to cancel a test of the plan. Intervenors had argued that the test would violate State law, that it would demonstrate nothing of value, and that the plan itself was illegal. In denying the intervenor request, the Commission recognized that while the utility (Long Island Lighting Co., or LILCO) could not lawfully implement all aspects of its emergency plan, conduct of the exercise was necessary because it was expected: (1) to provide information as to whether lack of cooperation in emergency planning by the state and local governments would result in "significant" defects under NRC's emergency planning standards; and (2) to test the utility's ability to accommodate *ad hoc* governmental participation in the event of an actual emergency.

Thus the Commission concluded that even though it might not be possible to test all aspects of an emergency plan, the exercise would not necessarily be useless. The exercise could assist in determining whether any defects that exist as a result of "... limitations of [the utility's] plan when executed under...state and county restrictions..." would be significant under 10 C.F.R 50.47(c)(1).



Following the Ohio earthquake of January 31, 1986, an intervenor filed a motion with the NRC's Appeal Board to reopen the record on the Perry nuclear power plant, challenging the plant's seismic protections. In response to the board's decision to hold an exploratory hearing on the matter, the Commission determined that the board had exceeded its authority and thus vacated the order. The photo shows the interior of the Perry plant's fuel-handling building, located between reactor units. The shipping cast preparation pool is in foreground, and a spent fuel storage pool directly behind it.

In CLI-86-11, 23 NRC 577 (June 6, 1986), the Commission addressed several issues concerning the Shoreham emergency plan. First, the Commission continued the deferral of a remand mandated by the Appeal Board in ALAB-832. Second, the Commission ordered the initiation of a hearing on the results of a February 13, 1986 emergency planning exercise. Third, the Commission established standards for the admission of contentions in the hearing on the exercise, *i.e.*, the Licensing Board was to admit "only those. . . contentions which satisf[ied] the specificity and other requirements of 10 C.F.R. 2.714 by (1) pleading that the exercise demonstrated fundamental flaws in [the applicant's] plan, and (2) by providing bases for the contentions which, if shown to be true, would demonstrate a fundamental flaw in the plan." Last, the Commission ruled that its Rules of Practice did not prohibit the grant of summary disposition on the contentions prior to discovery. Thus, the board might grant such a motion, or, if essential facts were not available for response to the motion, the board could deny the motion or order discovery, if that were likely to produce evidence supporting the existence of a genuine issue of material fact.

In the third decision, CLI-86-13, 24 NRC ____ (July 24, 1986), the Commission reviewed an Appeal Board decision concerning LILCO's legal authority to implement its plan (ALAB-818, 22 NRC 651 (1985)), and two arguments posed by LILCO—based on "realism" and "immateriality"—offered to rebut the inference that a potential lack of such legal authority should result in disapproval of its license application.

In ALAB-818, the Appeal Board had affirmed the Licensing Board's findings that, under State law, the applicant lacked the legal authority to implement material features of its proposed emergency plan, that there was no Federal preemption of those State laws, and therefore that the plan could not be carried out in conformity with Commission regulations. The Appeal Board also rejected the applicant's "realism" and "immateriality" arguments.

In CLI-86-13, the Commission deferred addressing LILCO's argument that the State laws were preempted, but remanded for further evidentiary hearings on issues raised by the "realism" and "immateriality" arguments. The Commission instructed the Licensing Board to consider the realism and immateriality arguments in light of 10 C.F.R. §50.47(c)(1). That regulation authorizes licensing, even where an emergency plan may not comply with the NRC's detailed planning standards, if (1) the defects in the plan are not significant (2) adequate interim compensating actions have been or will be taken promptly, or (3) there are other compelling reasons to permit plant operation.

The Commission advised in CLI-86-13 that the adequacy of a utility plan under §50.47(c)(1) should not be judged by preset standards, but by what is reasonable and feasible for a particular plant. For Shoreham, the Commission assumed that if there were a serious accident, State and County officials would be obliged to assist as a matter of law and as a matter of discharging their public trust, and thus that there would be a "best effort" response. The Commission was unwilling to assume, however, that such response would necessarily be adequate. To obtain more information on this score, the Commission remanded the "realism" and "immateriality" arguments to the Licensing Board for further consideration.

With regard to the "immateriality" argument, the Commission acknowledged that some of the emergency planning measures in question (such as controlling traffic), while not explicitly mentioned in NRC regulations, might nevertheless be required to provide reasonable assurance of adequate protective measures.

In the fourth decision, CLI-86-16, 24 NRC ____ (Sept. 26, 1986), the Commission denied intervenors' motion demanding termination of the adjudication concerning the results of licensee's February 13, 1986 emergency planning exercise. Intervenors argued that two events mandated such termination—Nassau County's revocation of permission to use the Nassau Coliseum as a relocation center, and the State's enactment of legislation for the takeover of LILCO. While recognizing that the status of a relocation center in licensee's plan was now not entirely clear (because of LILCO's loss of the facility on which it had intended to rely), the Commission nevertheless declined to interrupt the hearing, for the reason that the hearing could lead to resolution of issues involving functions not dependent on a specific relocation center. Also, the Commission determined that because the potential takeover of LILCO was subject to many contingencies, the State takeover legislation did not offer a sound basis on which to terminate the hearing.

JUDICIAL REVIEW

The more significant litigation involving the Commission during fiscal year 1986 is summarized below.

Pending Cases

Commonwealth Edison Co. v. NRC, No. 85C09255 (N. D. Ill.) and No. 85-2928 (7th Cir.).

On November 1, 1985, the Commonwealth Edison Co. sued in United States District Court for the Northern District of Illinois for a declaratory judgment that the NRC's application of its current license fee ceilings to license review work done before the effective date of the current ceilings was illegal under the Independent Offices Appropriations Act (IOAA) and contrary to the due process clause of Amendment V to the United States Constitution. Uncertain of what court has jurisdiction over such an issue, Commonwealth Edison filed a similar suit in the Seventh Circuit.

The NRC's current fee ceilings became effective in 1984. By then the costs which the Commission had incurred from the license reviews for Commonwealth Edison's Byron (Ill.) and Braidwood (Ill.) plants had exceeded the previous ceilings which had been imposed by the 1978 regulations. After raising the ceilings in 1984, the NRC then billed Commonwealth Edison for work performed prior to the effective date of the new regulations at a rate which could not have been applied under the old regulations because it exceeded the previous ceilings.

The utility claims that this is an impermissible retroactive application of the current ceilings to review work done after the 1978 ceilings were reached but before the current ceilings became effective. The utility also claims that the NRC has applied the wrong law in calculating interest and penalties on the unpaid bill.

The NRC claims that Commonwealth Edison owes it slightly over $3,000,000, and that, moreover, additional penalties and interest accrue at the rate of over $1,200 per day. More than these assessments are at stake in the case, however, since there are other utilities in Commonwealth Edison's position.

Commonwealth Edison and the NRC agreed to have the U.S. Attorney in Chicago file a joint motion in the District Court to hold the District Court proceeding in abeyance until the Circuit Court has ruled on the petition before it. The parties have filed their briefs, and the Circuit Court heard oral argument in the case on June 4, 1986.

Cuomo, et al. v. NRC (D.C. Cir. No. 85-1042).
This action is a challenge by the State of New York and Suffolk County to the issuance by the NRC of a low-power license to the Shoreham facility on Long Island. Because the low-power authorization was issued in stages, the petitioners three times sought and received permission to amend their petition for review. Petitioners' request for a stay of low-power operation was denied by the D.C. Circuit in an unusually detailed opinion issued in July 1985 (772 F.2d 972 (D.C. Cir. 1985)). On the merits, petitioners claim that the NRC violated the National Environmental Policy Act (NEPA) when it refused to recalculate the cost-benefit balance for the plant, in order to take account of the possibility that petitioners' refusal to participate in emergency planning at Shoreham would preclude issuance of a full-power license.

Florida Power & Light, et al. v. NRC , No. 86-1512.
Pursuant to the consolidated Omnibus Budget Reconciliation Act of 1985, the Commission promulgated a regulation in September 1986 imposing substantially higher user fees on power reactor licensees. Several utilities have filed a petition for review challenging the legality of the Commission's rule.

Sierra Club, et al. v. NRC, et al. (9th Cir. No. 85-7003).
In this case, the Sierra Club filed a petition for review, in January 1985, challenging the Commission's November 21, 1984 decision to rescind in part, without a hearing, an earlier order that suspended operation of San Onofre Unit 1 (Cal.) pending the completion of certain seismic upgrades. The Sierra Club contended that the Commission's November 1984 "Contingent Rescission of Suspension" was a license amendment issued without the prior hearing required by Section 189(a) of the Atomic Energy Act. On February 25, 1985, petitioners sought an emergency stay of the Commission's February 19 order denying a hearing on and stay of the November 21, 1984 order, and an injunction of the operation of San Onofre Unit 1 pending appeal. In March 1985, the Ninth Circuit denied the emergency stay petition.

At oral argument, October 11, 1985, the panel raised a number of difficult and important jurisdictional questions, which the Commission addressed in a supplemental brief filed after the oral argument. The questions were:

(1) Whether the court has jurisdiction over a petition filed before entry of the agency order being challenged (the February 19 order);

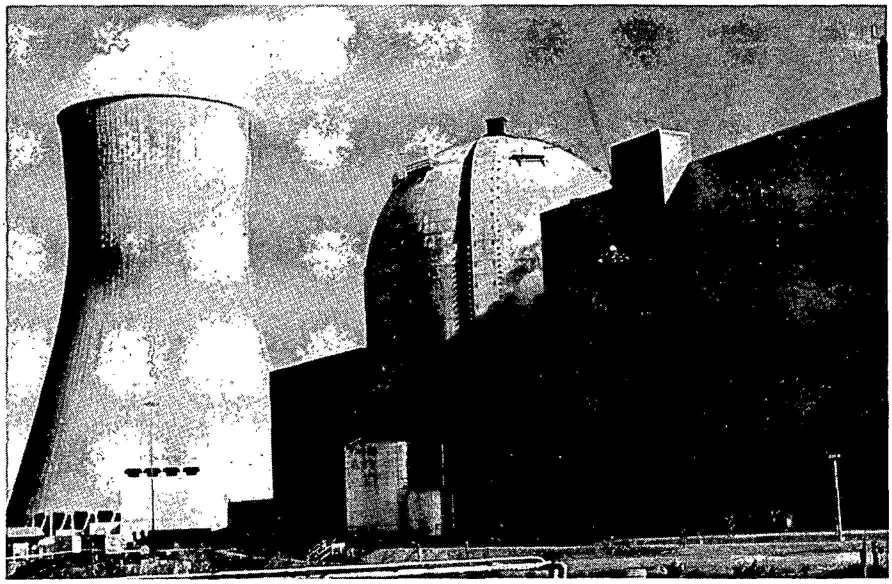
(2) Whether the court can review the legality of an agency order entered in a "proceeding" to which petitioners were not a party (the November 21 order); and

(3) Whether the court has jurisdiction to consider an order which petitioners have asked the agency to reconsider (the November 21 order).

As of mid-1986, the court had not yet issued a decision, but the licensee had completed the seismic upgrades that the NRC had earlier ordered. On July 29, 1986, the NRC filed a motion to dismiss the case, arguing that this subsequent event has rendered the claims of the Sierra Club moot.

Union of Concerned Scientists, et al. v. NRC (D.C. Cir. Nos. 85-1757 and 85-1219).
On November 18, 1985, the Union of Concerned Scientists (UCS), and others filed suit against the Commission, seeking to have the court declare the Commission's "backfitting rule," 10 C.F.R. §50.109, null and void, and direct the Commission to issue a rule that would conform to the requirements of the Atomic Energy Act. The petitioners principal claim is that the Act does not permit the Commission to use cost-benefit analysis in making backfit decisions. The petitioners also assert that the backfit rule prohibits considerations of averted property damage as a benefit and relies on probabilistic risk assessment to estimate risk reduction, and that the backfit rule is therefore arbitrary and capricious. The Nuclear Utility Backfitting and Reform Group (NUBARG) has intervened in the case on the side of the Commission. All the parties have filed their initial briefs in the case, and the Atomic Industrial Forum has filed a brief as friend of the court. Briefing was completed in late July.

On March 19, 1986, UCS filed a motion for leave to amend its petition to seek review of the staff's Manual Chapter 0514, which deals with plant-specific backfitting. The Manual provides a licensee affected by a proposed



In a case challenging a Commission rule regarding financial qualifications and its issuance of an operating license to owners of the Callaway (Mo.) nuclear power plant, shown here, the D.C. Circuit Court upheld both the rule and the issuance of the license.

backfit an opportunity to appeal the staff's backfit decision, but the Manual provides no comparable opportunity for participation by interested members of the general public. The UCS claims that this omission violates the hearing requirements of the Atomic Energy Act. The NRC and NUBARG opposed UCS's March 19 motion, arguing, in part, that the Manual is not judicially reviewable. The UCS raised the same issue in a separate case, No. 86-1219, which UCS initiated in April 1986 to preserve for itself an opportunity to seek review of the Manual in the event the court denied its March 19, 1986 motion, after the statutory deadline applicable to the filing of a petition to review the Manual had passed. NUBARG has intervened in the second case also.

On June 20, 1986, the court dismissed UCS's March 19 motion to amend as moot in light of UCS's later petition raising the same issues. In its ruling, the court also consolidated No. 85-1757 and No. 86-1219. Briefing of the second case was completed in late August. Oral argument for the consolidated cases had not yet been set at the close of the report period.

Significant Judicial Decisions

Coalition for the Environment, St. Louis Region v. NRC, 795 F.2d 168 (D.C. Cir. 1986).

This case challenged the Commission's rule eliminating case-by-case review of financial qualifications for power reactor operating license applicants who are electric utilities. The rule, promulgated on remand of *New England Coalition on Nuclear Pollution v. NRC,* 727 F.2d 1127 (D.C. Cir. 1984), was premised on a Commission finding that the rate regulatory process provides reasonable assurance of sufficient funds for safe operation. The case also challenged the issuance of an operating license to Union Electric Company for the Callaway (Mo.) facility without a separate finding of financial qualification. The court upheld both the new rule and the issuance of the Callaway license, ruling that the rule was adequately supported and not arbitrary or capricious. The court agreed with the First Circuit that the Atomic Energy Act gives the NRC complete discretion to decide what financial qualifications are appropriate (*New England Coalition on Nuclear Pollution v. NRC,* 582 F.2d 87, 93 (1st Cir. 1978)) and that case-by-case review is not required.

Long Island Lighting Company v. County of Suffolk, 628 F. Supp. 654 (E.D.N.Y. 1986).

On behalf of NRC, FEMA and DOE, the Department of Justice brought this case, consolidated with a similar action brought by Long Island Lighting Co., to enjoin Suffolk County from enforcing Local Law 2-86. That law required any person who participated in an emergency plan exercise to submit those plans to the county and made such an exercise subject to the county's disapproval. In addition, it criminalized activity undertaken pursuant to such an exercise. Officials of NRC, FEMA and DOE were covered by the plain language of the law, although later the county stated that it did not intend to apply criminal sanctions to Federal officials.

The local law was enacted within weeks of the emergency plan exercise scheduled at the Shoreham nuclear facility for February 13, 1986, in an obvious attempt to threaten the execution of that exercise. The Department of Justice moved for a preliminary injunction based on Federal preemption. The NRC argued successfully that radiological safety decisions were committed solely to the NRC and that the county lacked the authority to interfere with the Congressionally mandated fact-finding processes necessary to NRC licensing determinations. The preliminary injunction was granted.

Following the exercise, Suffolk County repealed the local law at issue, apparently to avoid a final decision that would create precedent unfavorable to the county. Thus, on June 17, 1986, the judge dismissed the case without prejudice.

Lorion v. NRC, 785 F.2d 1038 (D.C. Cir. 1986).

Ms. Lorion filed a petition on February 8, 1982, to review the NRC's decision denying her request that Turkey Point Unit 4 (Fla.) be shut down for a steam generator inspection. The Commission treated her letter requesting such action as a petition under 10 CFR 2.206 and denied the request. The D.C. Circuit in a sua sponte decision held that the courts of appeals lack subject matter jurisdiction to review denials by the Nuclear Regulatory Commission of 10 CFR 2.206 requests for enforcement action against NRC licensees (712 F.2d 1472). The court stated that jurisdiction to review such denials lies initially in the district court. The Supreme Court reversed the D.C. Circuit and held that courts of appeals do have subject-matter jurisdiction over denials of 10 CFR 2.206 requests.

On remand, the court affirmed the NRC's decision on the merits (785 F.2d 1038, D.C. Cir. 1986). The court left open the question of whether the NRC's decision was judically reviewable.

Oystershell Alliance, et al. v. NRC, et al., 800 F.2d 1201 (D.C. Cir. 1986)

The central issue in this case was whether the Commission erred in issuing its March 15, 1985 effectiveness order allowing the Waterford (La.) nuclear power plant operating license to be issued even though two motions to reopen the Waterford adjudicatory record were still pending before the Appeal Board. In April, the D.C. Circuit denied petitioners' request for an emergency stay. Subsequent to this denial, both motions to reopen were denied. On September 9, 1986, the D.C. Circuit ruled in favor of the Commission. Specifically, the court approved:

(1) The Commission's immediate effectiveness regulations;

(2) The agency's stringent standards for granting a motion to reopen; and

(3) The start-up of the Waterford Unit 3 facility while two motions to reopen were pending before the Appeal Board.

San Luis Obispo Mothers for Peace v. NRC, 789 F.2d 26 (D.C. Cir. 1986)

The D.C. Circuit affirmed the Commission's issuance of low and full-power licenses to Diablo Canyon (Cal.) on December 31, 1984 (751 F.2d 1287 (D.C. Cir. 1984)). That decision rejected a number of challenges to the Commission's licensing including (1) the Commission's refusal to require a supplemental environmental impact statement discussing the worst-case, core-meltdown, accident; (2) its refusal to reopen the record to litigate various QA/QC (quality assurance/quality control) contentions; (3) its refusal to require actual operating experience for individual operator licenses; and (4) its refusal to litigate the possible effects of earthquakes on Diablo Canyon's emergency planning. The panel also held that extending the period of the low-power license was a license amendment; however, because the full-power license had already been issued, the panel found that the error did not warrant redress.

Petitioners requested a rehearing *en banc.* The court granted that petition in part and vacated the part of the panel decision dealing with earthquakes and emergency planning. It also requested briefing by the parties on whether it should consider improperly released transcripts of closed Commission meetings at which the earthquakes/emergency planning issue was deliberated.

On April 25, 1986, by a 5-4 vote, the D.C. Circuit, *en banc,* rejected all challenges to the Commission's earthquakes/emergency planning decision in *Diablo Canyon* (789 F.2d 26 (D.C. Cir. 1986)). On the central question, the majority found the Commission's decision to be "supported by the record, not merely adequately but, we think, conclusively." On the transcripts issue, the majority refused to examine the Commission's deliberations because petitioners made no independent showing of "bad faith or improper behavior." The Supreme Court denied petitioners' request for certiorari on October 20, 1986.

San Luis Obispo Mothers for Peace v. NRC, 799 F.2d 1268 (9th Cir. 1986).

Petitioners filed a petition and motion for a stay seeking to halt the reracking of Diablo Canyon's (Cal.) spent fuel pools. The reracking involved replacing bolted, low density racks with free-standing high density racks. It was authorized by a license amendment based on a "no significant hazards consideration" judgment by the NRC staff. The issue was whether, in view of the uniquely high seismic requirements for Diablo Canyon, the reracking of its spent fuel pools could properly be characterized as involving no significant hazards considerations under 42 U.S.C. 2239a(2)(A) (the "Sholly Amendments"). If so, the license amendment would be followed, rather than preceded, by a hearing. The court stayed, pending judicial review, all reracking for Unit 2 and the placement of any spent fuel in the Unit 1 pool.

On September 11, 1986, the Ninth Circuit, by a 2-1 vote, ruled against the NRC in the case. The court found that the agency improperly characterized the reracking of Diablo Canyon's spent fuel pools as a "no significant hazards consideration" amendment. The court reasoned that the NRC failed to follow its own regulations by not recognizing that the switch from bolted racks (which could not slide in an earthquake) to free-standing racks (which could) created the possibility of a new and different accident, thus meeting one of the criteria for being a significant hazards consideration amendment under 10 C.F.R. §50.92(c)(2). The dissent, which was issued on September 30, 1986, disagreed with the majority and accused them of usurping the agency's technical expertise.

Management and Communication

CHAPTER 13

NRC Offices Consolidated

Nearly a dozen years from the inception of the U.S. Nuclear Regulatory Commission in January 1975, a single site for the agency's manifold offices was selected and secured. In November 1986, the U.S. Government purchased an 18-storey office building at 11555 Rockville Pike in Rockville, Md., and agreed to lease a second building (with an option to purchase) to be constructed at the same site within 30 months. Acquisition of the two buildings will permit full consolidation of the NRC's Headquarters staff of over 2,500 personnel by 1990. Offices of the Chairman and other Commissioners, the Executive Director for Operations and others will begin operations at the first building—the One White Flint North Building—during fiscal year 1988. (See discussion in Chapter 1 regarding NRC relocation and reorganization.)

STRENGTH AND STRUCTURE

Changes Within the Commission and Senior Staff

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

- In July 1986, former Commissioner Lando W. Zech, Jr., was appointed Chairman of the U.S. Nuclear Regulatory Commission, succeeding Nunzio J. Palladino. Chairman Zech was first appointed to the Commission in July 1984.
- In August 1986, Commissioner Kenneth M. Carr was appointed to the Commission, filling the vacancy created when former Chairman Palladino completed his five-year term.
- In July 1986, William C. Parler was appointed General Counsel, succeeding Herzel H.E. Plaine.
- In July 1986, Guy H. Cunningham, former NRC Executive Legal Director, was appointed Deputy General Counsel.
- In April 1986, Victor Stello, Jr., was appointed Executive Director for Operations, succeeding William J. Dircks.
- In May 1986, James H. Sniezek was appointed Deputy Executive Director for Regional Operations and Generic Requirements, succeeding Victor Stello, Jr.
- In July 1986, William G. McDonald was appointed Deputy Executive Director for Operations (Information Resources Management).
- In September 1986, Eric S. Beckjord was appointed Director, Office of Nuclear Regulatory Research, succeeding Robert B. Minogue.

Personnel Management

In fiscal year 1986, the NRC expended 3,445 staff years in carrying out its mission. This number includes work performed by part-time and temporary workers and consultants, as well as full-time permanent staff. The total expenditure of staff years was 1.3 percent below the budgeted target of 3,491 staff years.

Recruitment

In fiscal year 1986, the NRC hired 259 and lost 307 permanent full-time employees through an attrition rate of 9.6 percent per year. The agency's recruitment program included visits to 30 college campuses (including campus "job fairs") and participation in approximately 12 other job fairs during the year. A total of 26 entry-level scientists and engineers were hired.

Training and Development

The NRC provides over 60 different technical courses—in reactor technology, end-user computer applications and probabilistic risk assessment—for its technical and administrative personnel. Twenty-one courses are also available in-house to improve executive, management and administrative skills. NRC employees also participate in a wide range of private sector, college and university, and government-wide educational and development programs directed at improving performance and maintaining up-to-date technical proficiency.

Also in 1986, the NRC continued its emphasis on upward mobility programs and the use of Individual Development Plans to help all employees clarify their career goals and improve their job skills and performance. A Certified

Consolidation of NRC offices at a single location became a realizable prospect in 1986 with the acquisition for use by the NRC of this 18-storey building in Rockville, Md. A second building on the same site was scheduled for construction within 30 months. By the end of 1987, personnel moves to the new location had begun.

Professional Secretary Program, Administrative Skills Enhancement Program and a Computer Science Development Program were all introduced as vehicles by which secretarial/clerical/administrative personnel might expand their sphere of training and advancement opportunities.

ORGANIZATIONAL CHANGES

The Commission unified the former Office of the Executive Legal Director and the Office of the General Counsel, with the combined office continuing under the latter title. The office directs matters of law and legal policy, providing opinions, advice, and assistance to the Commission and staff on all agency activities.

The Office of Nuclear Regulatory Research was reorganized during the report period to provide a more appropriate structure for dealing with and resolving current safety issues. The principal feature of the change was the division of staff into three divisions rather than four. The Division of Regulatory Applications oversees research applications to the regulatory process, provides a base for rulemaking skills, and pursues the application of technology in the regulatory area. The Division of Reactor System Safety concentrates the system disciplines of risk analysis, thermal hydraulics, and severe accident analysis. And the Division of Engineering Safety is responsible for component and structural research and waste management research.

The Office of Nuclear Reactor Regulation established a TVA Project Staff to deal with continuing agency concerns about certain nuclear operations and related activities of the Tennessee Valley Authority. The new staff reports to the Director of the Office in his capacity as Chairman of the NRC's TVA Senior Management Team; the TVA Project Staff replaces the former TVA Oversight Group.

The Commission has abolished the Office of Policy Evaluation.

EMPLOYEE-MANAGEMENT RELATIONS

Incentive Awards

NRC managers recognized high quality work performed by staff members during 1986 with 325 special achievement awards, 340 high quality performance increases, 26 certificates of appreciation, 70 SES bonuses, 5 distinguished service awards, 41 meritorious service awards, and 3 equal employment opportunity awards.

A number of NRC managers were honored by President Reagan, who selected them for Presidential rank awards. Guy H. Cunningham, former Executive Legal Director, and Thomas E. Murley, Regional Administrator, Region I, received the Distinguished Executive Rank Award. Meritorious Executive Rank Awards were received by Darrell G. Eisenhut, former Deputy Director, NRR, William J. Olmstead, Assistant General Counsel for Rulemaking and Fuel Cycle, Associate General Counsel—Licensing & Regulation, OGC, Harold R. Denton, Director, NRR, Clemens J. Heltemes, Jr., Director, AEOD, and Themis P. Speis, Director, Division of Safety Review and Oversight, NRR.

Three NRC Equal Employment Opportunity Awards were conferred in the report year, to Lana C. Cobb, Computer Systems Analyst, ADM, Edward L. Halman, Director of the Division of Contracts, ADM, and James G. Keppler, Regional Adminstrator for Region III (Chicago).

The Distinguished Service Award is the highest honor Award granted by the U.S. Nuclear Regulatory Commission. For fiscal year 1986, the Distinguished Service Award was granted to Robert M. Bernero, Director, Division of BWR Licensing, NRR, Frank J. Congel, Chief, Reliability and Risk Assessment Branch, Division of Safety Review and Oversight, NRR, Charles J. Fitti, Director, Program Support and Analysis Staff, ASLBP, John C. Hoyle, Assistant Secretary, SECY, Chester P. Siess, Member, ACRS.

The Meritorious Service Award is the Commission's second highest honor Award. Recipients during fiscal year 1986 were: Althemese R. Bailey, Building Management Specialist, Building and Operations Branch, Division of Facilities and Operations Support, ADM, Andrew L. Bates, Chief, Operations Branch, SECY, Eugene F. Bates, Emergency Response Coordinator, Emergency Preparedness and Safeguards Programs Section, Division of Radiation Safety and Safeguards, Region IV, Annie M. Bennette, Secretary to the Assistant General Counsel for Hearings, Associate General Counsel—Hearings and Enforcement, OGC, Kevin L. Bohrer, Secretary to the Director, NRR, Robert S. Brown, Jr., former Director, Planning & Program Analysis Staff, NMSS, Roy J. Caniano, Radiation Specialist (Health Physicist), Nuclear Materials Safety Section No. 2, Division of Radiation Safety and Safeguards, Region III, Robert A. Capra, Senior Program Manager, Regional Operations and Generic Requirements Staff, EDO, Edward S. Christenbury, former Assistant General Counsel for Hearings, Associate General Counsel—Hearings and Enforcement, OGC, Douglas M. Collins, Chief, Emergency Preparedness and Radiological Protection Branch, Division of Radiation Safety and Safeguards, Region II, Thomas R. Combs, Chief, Correspondence and Records Branch, SECY, Jesse L. Crews, Senior Reactor Engineer, Office of the Regional Administrator, Region V, Dennis M. Crutchfield, Assistant Director, Division of PWR Licensing-B, NRR, Mark A. Cunningham, Section Leader, Accident Analysis Section, Risk Analysis Branch, Division of Risk Analysis and Operations, RES, Peggy A. Dahlberg, Personnel Management Specialist, Personnel Staff, Division of Resource Management and Administration, Region III, Diane B. Dandois, Chief, Financial Operations Branch, Division of Accounting and Finance, RM, Kitty S. Dragonette, Senior Health Physicist, Low-Level Waste and Uranium, Recovery Projects Branch, Division of Waste Management, NMSS, Abraham L. Eiss, Technical Assistant to the Director, Division of Engineering Technology, RES, Michael Fox, Chief, Staffing and Position Evaluation Branch, Division of Organization and Personnel, ADM, Howard S. Johnson, Office Services Assistant, Property Management Section, Property and Supply Branch, Division of Facilities and Operations Support, ADM, Edward L. Jordan, Director, Division of Emergency Preparedness and Engineering Response, I&E, Harry W. Kerch, Lead Reactor Engineer, Materials and Processes Section, Engineering Branch, Division of Reactor Safety, Region I, Patricia M. Kreutzer, Licensing Assistant, PWR Project Directorate No. 8, Division of PWR Licensing-B, NRR, Dennis J. Kubicki, Fire Protection Engineer, Section B, Plant, Electrical, Instrumentation, and Control Systems Branch, Division of PWR Licensing-B, NRR, Robert G. LaGrange, Section Leader, Section B, Engineering Branch, Division of BWR Licensing, NRR, Richard P. Levi, former Senior Attorney, Assistant General Counsel—Adjudication and Opinion, Associate General Counsel—Licensing and Regulation, OGC, Robert D. Martin, Regional Administrator, Region IV, Thomas T. Martin, Director, Division of Radiation Safety and Safeguards, Region I, Austin L. Newsom, Assistant Executive Director for Operations, ACRS, Meredith E. Rice, Senior Mail Clerk, Central Mail Section, Mail and Messenger Branch, Division of Facilities and Operations Support, ADM, Irwin B. Rothschild, III, Senior Attorney, Assistant General Counsel—Legislation and Government Relations, Associate General Counsel, Licensing and Regulations, OGC, Patrick Rowe, Director, Program Support Staff, ADM, Jerome D. Saltzman, former Assistant Director for State and Licensee Relations, SP, James R. Shea, Director, IP, Steven K. Showe, Chief, Pressurized Water Reactor Branch, Technical Training Center, I&E, Edward Y. Shum, Senior Environmental Engineer, Uranium Fuel Licensing Branch, Division of Fuel Cycle and Material Safety, NMSS, Hazel I. Smith, Senior Management Analyst, Planning and Program Analysis Staff, NRR, Raymond J. Smith, Jr., Reactor System Specialist, Operations Section, Events Analysis Branch, Division of Emergency Preparedness and Engineering Response, I&E, Jill A. Solan, Secretary to the Director, ADM, Bruce A. Wilson, Chief, Reactor Projects Section,

New annual awards instituted in 1986 for NRC resident inspectors include five Regional Resident Inspectors of the Year and, from among them, one NRC Resident Inspector of the Year. The first recipient of the latter award was Robert L. Nelson, Senior Resident Inspector at the Kewaunee (Wis.) nuclear power plant.

The year's annual Management Excellence Award went to two NRC Managers: John E. Glenn, Chief, Nuclear Materials Safety, Region I (above) and Gary G. Zech, Director, TVA Projects, Region II (below).

2C, Operations Branch, Division of Reactor Projects, Region II, and John E. Zerbe, Director, Regional Operations and Generic Requirements Staff, EDO.

The Regional Resident Inspector of the Year Award is a new Award granted annually to one Resident Inspector in each Region for meritorious and outstanding achievement. The Commission also grants an NRC Resident Inspector of the Year Award to one of the five recipients of the Regional Resident Inspector of the Year Award. For fiscal year 1986, the NRC Resident Inspector of the Year Award went to Robert L. Nelson, Senior Resident Inspector at the Kawaunee (Wis.) nuclear power plant, Region III. Regional Resident Inspector of the Year Awards were given to Thomas J. Kenny, Senior Resident Inspector at the Salem (N.J.) nuclear power plant, Region I; Pierce Skinner, Senior Resident Inspector at the Catawba (S.C.) nuclear power plant, Region II; Dennis L. Dubois, Senior Resident Inspector at the Cooper (Neb.) nuclear power plant, Region IV; and Glen P. Perez, Senior Resident Inspector at the Rancho Seco (Cal.) nuclear power plant, Region V.

The Management Excellence Award is an honorary award that recognizes managers and supervisors who have demonstrated the most outstanding managerial accomplishments. Two NRC managers received the Management Excellence Award during fiscal year 1986: John E. Glenn, Chief, Nuclear Materials Safety, Section B, Nuclear Materials Safety and Safeguards Branch, Division of Radiation Safety and Safeguards, Region I, and Gary G. Zech, Director, TVA Projects, Region II.

Donald R. Chapell, former Deputy Director, Division of Fuel Cycle and Material Safety, Office of NMSS, received the NRC Commendation Award from the Executive Director for Operations.

Labor Relations

NRC Management and the National Treasury Employees Union have reached agreement on 48 articles of their new collective bargaining agreement, with five articles yet to be negotiated, as of the close of the report period. The still unresolved articles are: "Performance Appraisal System," "Hours of Work" (including compressed work schedule), "Reduction-in-Force," "Effective Date and Duration," and "Salary" (which management has declared non-negotiable). At fiscal year's end, "Effective Date" and "Duration" were under active negotiation.

Some 50 grievances, 25 mid-contract negotiations, 87 performance/conduct actions, and 9 unfair labor practice charges were handled during fiscal year 1986.

INSPECTION AND AUDIT

The activities of the NRC's Office of Inspector and Auditor (OIA) are intended to assure effectiveness, efficiency and integrity in all NRC operations. In fiscal year 1986, OIA issued 21 audit reports containing 79 recommendations, and 22 follow-up audit reports aimed at improving the operations of various NRC programs and activities. OIA also issued 52 investigative reports in response to allegations regarding the integrity of NRC operations and employees. In addition, 22 matters were referred to the Department of Justice for review and possible criminal prosecution. Some of the reports issued during the report period are summarized below:

Allegations Management for Diablo Canyon

This March 1986 report addresses the processing of allegations by NRC related to activities at the site of the Diablo Canyon (Cal.) nuclear power station. The review was based on concerns raised to OIA by various allegers about the handling of allegations they had previously made to NRC. OIA concluded that, although NRC did a creditable job of managing the vast number of allegations regarding Diablo Canyon, several problems did exist. These problems centered around the method and degree to which several allegations were addressed by personnel at the NRC's Region V office (San Francisco) and the breach of confidentiality of several allegers. The OIA report contained 15 recommendations to improve NRC's overall management and processing of allegations.

NRC Relationships with the Department of Energy

This review was undertaken as a result of a request from Congressman Edward J. Markey to NRC Chairman Nunzio J. Palladino in an April 18, 1985 letter. Congressman Markey's letter contained information regarding the relationship between NRC, the Department of Energy (DOE), and an NRC contractor who also works for DOE. The Congressman's letter also enclosed an article from the publication *Gambit* which, he said, "raised a number of questions about NRC's relationship to DOE." OIA's January 1986 report concluded that there was a need to develop guidelines for NRC employees in their dealings with DOE and that the NRC should reexamine the conflict of interest regulations as they apply to contractors also working for DOE. OIA's review did not substantiate the specific allegations in the *Gambit* article referenced in Congressman Markey's letter.

NRC Relationship with DOE Implementing the NWPA

In a July 1986 audit report, OIA concluded that substantial efforts have been made by NRC and its employees to protect NRC's independence in its dealings with DOE on Nuclear Waste Policy Act (NWPA) issues (see Chapter 7). The report also noted that NRC has managed to effectively carry out its responsibilities in a timely manner to avoid adversely affecting schedules established by the NWPA and DOE. Two areas were identified in which OIA believed corrective action was necessary: the turnover of technical personnel within the Division of Waste Management, and the need for formal controls to ensure that NRC employees will not be directly involved in reviewing work performed while with a previous employer. OIA made recommendations to effect corrective actions in those areas.

NRC's Automatic Data Processing

In an April 1986 audit report, OIA concluded that there was a need to change the NRC's organizational structure for automatic data processing (ADP) to ensure that controls are in place to utilize resources effectively. OIA's report contained 11 recommendations related to (1) changes in the organizational structure of NRC's ADP-related activities, (2) the effectiveness of the ADP Steering Group in establishing the agency's ADP policies, and (3) the management of ADP systems development.

Technical Specifications for Near-Term Operating Licenses

OIA's May 1986 report to the Commission assessed the effectiveness of improvements to NRC's Technical Specification review process, as a result of the agency's experience with the Technical Specifications for the Grand Gulf Unit 1 (Miss.) nuclear power plant. The assessment concluded that, while the Office of Nuclear Reactor Regulation has made significant improvements to the Technical Specifications review process, the schedule established to provide adequate review time was not being followed.

Other areas were identified where improvements had been made but were less effective than they should have been and did not provide for adequate accountability. OIA's report contained recommendations to effect corrective actions in those areas and to enhance the overall effectiveness of Technical Specifications reviews for near-term operating licenses.

NRC Management of Classified Data

In a May 1986 audit report, OIA concluded that NRC was for the most part both efficient and effective in carrying out its responsibilities for implementing and managing NRC's information, and classification and control programs. OIA found, however, that improvements were needed in (1) the process for assigning original classification authority, and (2) the ability of security advisors to provide classification assistance. OIA's report contained six recommendations to enhance NRC's classified information program.

Effect of the CRGR on NRC Staff's Dealing with Safety Issues

Responding to a request from Commissioner James K. Asselstine, OIA conducted a review to determine whether the Committee to Review Generic Requirements (CRGR) has a "chilling effect" on the staff's identification of new safety issues or on the resolution of existing generic safety issues. OIA's review relied primarily on 467 responses to a survey questionnaire sent to 994 technical staff at the branch chief level and below in the Offices of Nuclear Reactor Regulation, Inspection and Enforcement, Nuclear Material Safety and Safeguards, and Nuclear Regulatory Research. OIA's May 1986 report did not reach any definitive conclusions regarding the effect of the CRGR on the staff. The survey showed, however, that nearly all the staff members who responded to the questionnaire believe the CRGR has a positive place in NRC. The survey also indicated that about two-fifths of the staff members who responded (184 respondents) believe that the CRGR is having a deterrent effect on the identification of new generic safety issues or resolution of existing issues—but that number represents only about 19 percent of the total staff to whom questionnaires were sent. The report contained three recommendations to improve the staff's perceptions regarding the CRGR, and to further ensure management's support of the staff's efforts to identify new safety issues and/or find solutions to known generic safety issues.

Review of NRC's Training Function

In a July 1986 audit report, OIA concluded that changes made in NRC's training function as a result of two management studies have resulted in much improvement in NRC's overall training function. OIA found there was a need to evaluate the agency's training programs on a more consistent basis and to have a more systematic approach toward the identification of the training needs of NRC employees. The report contained four recommendations to improve NRC's training program.

CONTRACTING

Contracts with commercial firms for technical assistance, research work, and general purchases totaled approximately $60,100,000 in fiscal year 1985. Contracts under the Small Business Innovation Research Program totaled $1,200,000 and grants with educational and nonprofit institutions totaled $1,900,000.

NRC LICENSE FEES

The NRC is authorized under Title V of the Independent Offices Appropriation Act of 1952 to collect fees for processing applications, permits, licenses and approvals and for routine and non-routine health and safety inspections. Fees billed in fiscal year 1986 totaled $46.4 million. All license and inspection fees are sent to the Department of Treasury for deposit as miscellaneous receipts. Table 1 shows a breakdown of these collections.

The total billings since fees were first imposed (October 1968 through September) 1986 is $309.9 million. Of this amount, $6.5 million has been refunded to licensees, pursuant to a 1974 Supreme Court decision negating annual license fees. The refund program was completed on November 11, 1984, the date the statute of limitations expired.

New Fee Schedule

The Commission, in response to the Consolidated Omnibus Budget Reconciliation Act (Public Law 99-272), adopted a new schedule of annual fees (10 CFR 171), which became effective October 20, 1986. The public law requires the Commission to assess and collect annual charges not to exceed 33 percent of its estimated fiscal year 1987 budgeted costs. The new schedule permits the Commission to collect

Table 1. License Fee Billings—FY 1986

Fees	*Material*	*Facilities*	*Total*
Applications	$ 584,466	—	$ 584,466
Operator Exams	—	$ 3,197,507	3,197,507
Operating Licenses	—	16,094,753	16,094,753
Amendments	543,623	3,500,407	4,044,030
Renewals	340,583	—	340,583
Inspection Fees	1,212,230	19,805,492	21,017,722
Special Projects	44,520	1,043,285	1,087,805
TOTALS	$2,725,422	$43,641,444	$46,366,866

annual charges from utilities licensed to operate nuclear power plants. The annual fee for fiscal year 1987 is $950,000 for every power reactor licensed for operation as of October 1, 1986. The annual fee will be charged on a *pro rata* basis for any power reactor licensed for operation during the year. The annual fee is to be paid in equal installments of 25 percent.

The annual fee was calculated by taking 33 percent ($133 million) of the NRC's estimated budget for fiscal year 1987 ($405 million), subtracting the estimated amount of the fees to be collected during the fiscal year under 10 CFR 170 ($37 million) and dividing the remainder ($96 million) by the number of power reactors currently licensed for operation. The annual fees reflect the generic costs of providing regulatory services to power reactor licensees in the following five areas: nuclear regulatory research, safeguards, reactor regulation, inspection and enforcement and analysis and evaluation of operational data. The total cost of providing these services during fiscal year 1987 is estimated to be about $124 million.

The new rule does not alter the fees assessed in accordance with 10 CFR 170 for regulatory services related to all facilities and materials licensees, including utilities licensed to operate nuclear power plants.

Litigation Concerning Fees

As noted above, the Commission published a Final Notice of Rulemaking in the Federal Register on September 18, 1986, related to annual fees for licensed power reactors (10 CFR 171), which became effective October 20, 1986. On behalf of numerous utilities, three law firms have petitioned the U.S. Court of Appeals for the District of Columbia Circuit for review of the final rule.

A second law suit has been filed with the U.S. Court of Appeals, Seventh Circuit, related to the Commission's revised 10 CFR 170 schedule of fees, which became effective June 20, 1984. The revised schedule was designed to recover more completely the NRC licensing and inspection costs incurred for providing services to identifiable recipients. Under the revised schedule, the Commission raised some of the maximum fee ceilings and eliminated the fee ceilings in other areas. The billing procedure for those applications subject to full cost reviews was also revised. Under the revised rule, the applicant pays for the cost of the application review every six months as the work progresses. On November 4, 1985, Commonwealth Edison Company petitioned the court to review the NRC's application of the 1984 amendment to 10 CFR 170 as it relates to the billing of licensing review activities for that company's Byron and Braidwood (Ill.) nuclear power stations. Oral arguments were heard by the court on June 4, 1986.

PUBLIC COMMUNICATION

Public Information

Public Affairs. The NRC Office of Public Affairs informed the public about the activities and programs of the Nuclear Regulatory Commission in daily contacts with the news media, at NRC meetings or hearings, and by telephone response. The Office issued public press releases on Commission programs, rulemakings, public hearings, proposed fines against licensees, and other agency activities to the news media, the scientific community, universities and the general public.

Media Seminars. Four of the Regional Offices of the NRC conducted a series of one-day education workshops, for the sixth consecutive year, for reporters and news editors from national wire services, broadcast networks, news magazines and daily newspapers. The workshops, dealing with the fundamentals of nuclear power and the risks of exposure to radiation, were held in Chattanooga, Tenn., on January 23; in Detroit, Mich., on April 10; in San Clemente, Cal., on August 25; and in Philadelphia, Pa., on November 20, 1986.

Headquarters Public Document Room

Persons interested in detailed information about commercial nuclear facilities have found the NRC's principal Public Document Room (PDR) a source of very extensive and useful material. Located at 1717 H Street, N.W., in Washington, D.C., the PDR is a specialized documentation center that houses significant documents on nuclear regulation which have been made available to the public. Users of the center can have documents reproduced for a nominal fee.

Researches in the PDR can examine copies of a wide variety of materials: NRC reports; transcripts and summaries of meetings; licenses and their amendments; existing and proposed regulations; and correspondence on technical, legal and administrative matters. Most of these documents are related specifically to nuclear power plants (their design, construction, operation and inspection) and to nuclear materials, including radioactive wastes (their use, transport and disposal). The PDR features extensive accession listings and an on-line computer data base.

The PDR contains about 1.4 million documents; the collection is enlarged by an average of 274 new items every day. During an average month, the PDR serves about 1,200 users. The staff retrieves an average of 5,400 files per month containing multiple documents or microfiche for researches on-site and provides about 2,100 documents in response to letters and telephone requests. The public purchased 3.1 million pages of documents and about 6,000 microfiche cards in fiscal year 1986.

Media seminars and local press conferences were conducted by NRC Regional Offices and Headquarters public affairs personnel throughout 1986. Shown here are NRC Director of Reactor Regulation Harold R. Denton (left) and Region II Administrator J. Nelson Grace at a press conference following a visit to the Shearon Harris plant near Raleigh, N.C.

Persons wishing to use or obtain additional information regarding the holdings, file organization, reference, reproduction services and procedures of the PDR may call (202) 634-3274, or write to the U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. A "Public Document Room Users' Guide" and "Public Document Room File Classification System" guide are available upon request. In addition, orientation sessions are provided for individuals or groups interested in using the facility, and training sessions are scheduled regularly for users in how to search the PDR automated bibliographic retrieval system (an on-line card catalogue).

Local Public Document Rooms

Through the local public document room (LPDR) program, the NRC makes documents available to the public near the sites of proposed and operating nuclear power plants and certain fuel cycle and low- and high-level nuclear waste repositories. The collections contain information about the licensing, construction, operation, inspection and regulation of nearby nuclear facilities. LPDR materials include documents dealing with health and safety, safeguards, environmental and antitrust considerations. The collections are usually located in university or public libraries that have copying facilities and are open to the public during the evening and on weekends. There are currently about 100 LPDRs with documentation related to nuclear power plants in operation or under construction. (See Appendix 3 for a complete listing of LPDR locations and addresses.)

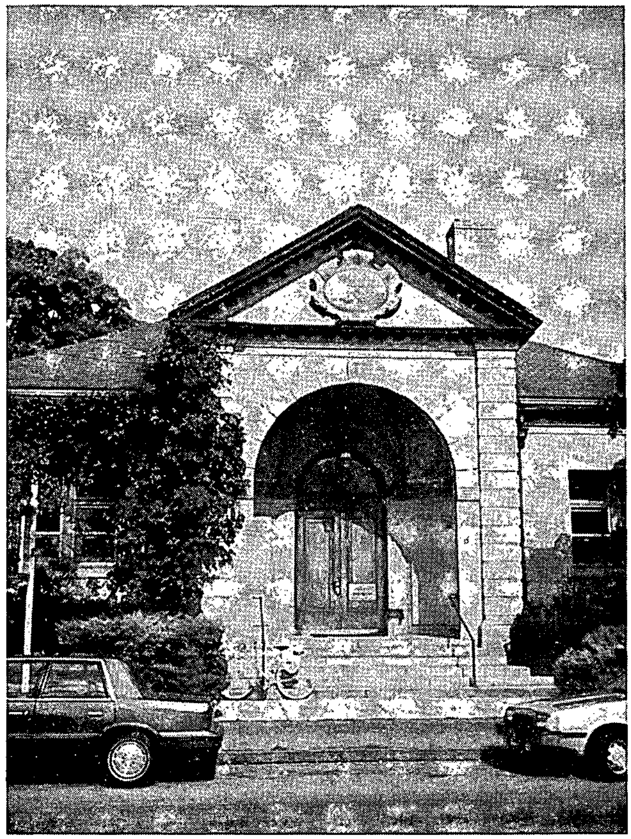
As part of its program to inform the public about the availability of these kinds of documents at the local level, the NRC publishes an LPDR newsletter and conducts evening workshops at LPDR libraries. The workshops provide instruction to the public in procedures for identifying, locating and securing information. A toll-free telephone number (1-800-638-8081) is available to library staff personnel and individuals who seek guidance with respect to collection content, search strategies, the use of reference tools and indices, and locating and retrieving information at LPDR facilities.

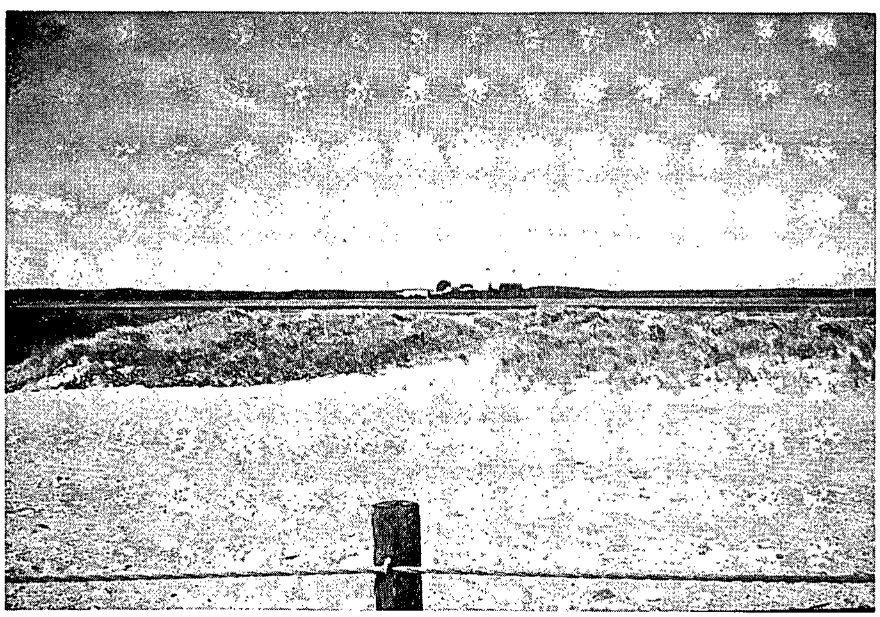
COMMISSION HISTORY PROGRAM

The Commission History Program studies the origins and evaluation of regulatory policies and programs. The History Staff continued to conduct research on a sequel to its book, Controlling the Atom: the Beginning of Nuclear Regulation, 1946-1962, published in 1984 by the University of California Press. Drawing on a broad array of official records and private manuscripts, the new volume will cover the period from 1963 into the early 1970s, a period that witnessed vital changes in and heated controversies over the commercial development of nuclear power. Like the first volume, it is intended to serve as a reference for general readers as well as for the agency staff.

OFFICE OF INVESTIGATIONS

The Office of Investigations (OI) continues to perform investigations of allegations of wrongdoing by individuals





Among the busiest of NRC's local public document rooms during 1986 was the one at Exeter, N.H., where a continuing controversy has attended the construction and licensing of the Seabrook nuclear power plant. NRC documents are housed in the Exter Public Library (above right) and cared for by, among others, Mrs. Lee Perkins, Library Assistant (above). A long-range photo of the facility (at right) shows remoteness of the plant site.

or organizations other than NRC employees or NRC contractors (including licensees, applicants and vendors, or their contractors), as described in the *1985 NRC Annual Report*, pp. 193-195.

In fiscal year 1986, OI opened 95 new cases and closed 141 cases. Eleven of the closed cases were referred to the Department of Justice for prosecutive consideration. Five cases are presently pending before Grand Juries.

Indictments

On September 20, 1986, a Federal Grand Jury in Grand Rapids, Mich., acting on information developed over a two-year investigation by the OI Field Office in Region III (Chicago), returned a nine-count indictment charging both the licensee, the American Electric Power Company, and an individual engineer with violation of the Federal law associated with NRC fire protection requirements. The charges allege willful material false statements to the NRC and operation of the D.C. Cook (Mich.) facility in violation of Federal regulations.

On September 26, 1986, a Federal Grand Jury returned a 42-count indictment charging YOH Security, two of its former managers, and the operator of a training school with conspiracy and false statements. YOH is the primary site-security contractor employed by Philadelphia Electric Company (PECO).

On June 26, 1986, a Federal Grand Jury returned a nine-count indictment charging International Neutronics, Inc.(INI), and two of its managers with false statements, mail fraud and conspiracy related to a spill of contaminated water at INI's Dover, N.J., facility and INI's alleged efforts to conceal the spill from the NRC.

Convictions

In December 1985, the president of Topping Associates/American Filter Company was sentenced to one year probation, $34,000 restitution, and a $1,000 fine after his conviction on one count of mail fraud (18 U.S.C. 1341). This action came out of of an OI investigation into allegations that Topping had substituted commercial grade filters for nuclear grade filters for use by the Calvert Cliffs (Md.) nuclear power plant.

On February 19, 1986, the manager of Pittsburgh Testing Laboratory (PTL) was fined $2,500, and PTL was fined $15,000 for violations of 42 U.S.C. 2273 and 18 U.S.C. 1001. The case resulted from an OI investigation into allegations that the manager of PTL had misled NRC inspectors with respect to the practices of an unqualified radiographer, and had destroyed or concealed official records. The manager and PTL pled guilty to the charges.

On March 26, 1986, acting on information developed during an OI investigation, FBI agents arrested a materials licensee and charged him with the illegal possession of americium-241 and with making material false statements to the NRC.

During a series of searches related to the arrest, approximately 21 curies of americium-241 (a man-made, radioactive element, atomic number 95, which is potentially very hazardous in powder form) was seized. The materials licensee pled guilty to the charges and is currently on court-supervised probation.

OFFICE OF SMALL AND DISADVANTAGED BUSINESS UTILIZATION/CIVIL RIGHTS

Small and Disadvantaged Business Utilization Program

The Small and Disadvantaged Business Utilization Program annually establishes procurement preference goals in response to provisions of Public Law 95-507, amending the Small Business Investment Act of 1957. During fiscal year 1986:

- It was estimated that $53 million in total prime contracts would be awarded in fiscal year 1985, and that the total amount of all prime contracts with individual dollar values over $10,000 would be $49 million. The actual total prime contracts and actual dollar awards over $10,000 were $46,060,490 and $42,703,835, respectively.
- It was estimated that small business prime awards with dollar values over $10,000 would be $19,350,000, or 39.5 percent. The actual achievement for small business prime awards with dollar values over $10,000 was $18,454,400, or 43.21 percent of the dollars reflected in the item above.
- The NRC estimated that awards to 8(a) firms would be $6,400,000 or 12.08 percent in fiscal year 1986. Awards to 8(a) firms were actually 7,501,640 or 16.29 percent of the total dollar amount of all prime contracts regardless of dollar value.
- The goal for prime contract awards having a value of $10,000 or more to small disadvantaged business firms other than 8(a) was $1,000,000 or 2 percent. The actual achievement was $208,813 or 0.49 percent of the dollars reported in the first item above, using awards over $10,000 as the base.
- The estimate for prime contract awards to small business concerns owned and controlled by women was $1,060,000 or 2 percent. Awards to such firms were $1,021,854 or 2.22 percent of the total dollar amount of all prime contracts regardless of dollar value.

- The goal for subcontract awards to small business was $1,300,000 or 65 percent of total subcontracts awarded. Subcontracting achievement to small businesses was $1,218,482 or 52.67 percent of total subcontracts awarded. The NRC's total subcontract dollar awards goal in fiscal year 1986 was $2,000,000.
- The goal for subcontract awards to small disadvantaged businesses was $47,000 or 2.35 percent. Subcontracting awards to small disadvantaged businesses was $31,170 or 1.35 percent of total subcontract dollars awarded.

During the year, 90 interviews were conducted with firms wanting to do business with the NRC, and 45 follow-up meetings were arranged with NRC technical personnel. The OSDBU/CR staff also participated in five major small business conferences, or "job fairs." Most noteworthy among these were the annual Minority Enterprise Development Week (MED Week) in October 1986 and the MEGA Market-place observance for women business owners in April 1986.

In response to Executive Order 12320, the OSDBU/CR staff implemented several activities designed to increase agency support to Historical Black Colleges and Universities (HBCUs). These activities—which included an NRC-sponsored Symposium on Technology Transfer and the United States Nuclear Power Industry, the initiation of a program that will focus on a series of Conferences to be held on Nuclear Waste Management, and an NRC Training Workshop for HBCUs—are all part of a continuing commitment by NRC to support HBCUs. The agency committed a total of $242,500 for the HBCUs Program for fiscal year 1986.

Carol J. Orkin of the Office of Personnel Management spoke to NRC employees in August 1986 on the subject: "Women Taking Charge of Their Lives." Her presentation was part of the continuing effort under the Commission's Federal Women's Program (FWP) to keep employees informed on women's issues and career-enhancing opportunities. Ten such programs were presented by the FWP during the year.

Federal Women's Program

Several initiatives and programs were established and continued this year which will have long range and positive impact on the goals of the Federal Women's Program (FWP). A new reporting requirement was placed on Regional Administrators and Office Directors to provide a quarterly account of their initiatives to improve the status of women. This effort has been most successful in helping the FWP Manager to focus attention on areas needing improvement and to provide assistance where needed. Recruitment trips were made to local universities, in conjunction with the Division of Organization & Personnel (O&P), to attract qualified technical women to NRC, an effort which resulted in several applications for employment by minority females. On-site visitations to Regional Offices also proved useful in establishing program direction for both the Program Manager and Regional Administrators.

Efforts to keep employees informed about women's issues and career-enhancing opportunities continued to be successful. In addition to the publication of several articles in the agency's newsletter "News, Reviews, and Comments," the Federal Women's Program Manager teamed up with the Federal Women's Program Advisory Committee in the presentation of 10 training and awareness programs. One of these activities was a presentation by Channel 9 News (CBS television in Washington, D.C.) Co-Anchor Maureen Bunyan during National Women's History Week. (See *1085 NRC Annual Report,* p. 196.) Another highlight was an address and discussion by Carol J. Okin, an official of the Office of Personnel Management, presented on August 26, 1986. As many as 500 employees attended some of these functions. Similarly successful programs were presented to commemorate National Secretaries Week and Women Equality Day.

Several NRC employees, including members of the Federal Women's Program Advisory Committee, attended the Federally Employed Women National training conference during the report period. The Federal Women's Program Manager and Regional Coordinators also attended the (FEW) Conference and participated in a two-day pre-training conference workshop to address NRC's women's issues and concerns. Recommendations from the pre-training conference were presented to management for action. The recommendations were well received by management and appropriate follow-up actions were taken.

The Federal Women's Program Manager played an active role in securing "upward mobility" positions and career-enhancing opportunities for women by working with managers, supervisors, the Division of Organization and Personnel, and the Employee Development and Training Staff. These and other initiatives on the part of the entire management team have brought about a number of positive developments. For instance, several offices have instituted developmental assignments, reassignments, job restructuring, and developmental details. In addition, promotions for women have increased over the past two years causing a steady increase in the number of women on the way to the top. Four out of five participants in the OPM Executive Potential Program are women and three women are participating in the Senior Executive Service Candidate Development Program. Thirteen women are taking part in the current formal upward mobility program and several others are reportedly receiving similar benefits through informal upward mobility efforts. Women are well represented in the Cooperative Education Program, with 11 women currently participating. In many NRC offices, up to 75 percent of agency-sponsored training is of women employees.

The availability of part-time employment for a number of women has raised the number of participants in that category from 144 to 153.

Several women have achieved permanent placement from secretarial to para-professional positions. An increased number of other women have received merit promotions on their career-ladders and selection for supervisory positions.

The number of women employed by NRC increased 7 percent during the fiscal year, while the number of total NRC employees increased 6 percent. During the same period, the number of women at the GG-11 level and above increased 2 percent, while the agency's total permanent full-time workforce declined 2 percent.

Another important feature of the Federal Women's Program at the NRC is the activity of its FWP Advisory Committee. In November 1986, the FWPAC conducted an Open Forum for NRC employees, answering questions and engaging in discussions of topics of particular interest to women in the agency. Shown here are members of the committee (left to right): Cheryl Sakenas, Ira Radden, Patricia Tana, Victor Zeoli, Melinda Malloy, Committee Chairman Cherie Siegel, Betty Cianci, FWP Manager Era Marshall, and Paula Frohman.

One of the most noteworthy events for the Federal Women's Program during the fiscal year was the expression of commitment to the goals of the Federal Women's Program on the part of the NRC Executive Director for Operations, who submitted a plan to the Commission aimed at increasing the representation of women and minorities at the level of GG-13 and above. This plan, among other things, provides for rotating assignments in order to promote greater utilization and training—as well as increased hiring goals—for women and minorities in offices where participation is currently low-to-non-existent.

Civil Rights Program

The second Consolidated Equal Employment Opportunity (EEO) Program Plan was implemented in order to continue to promote affirmative action in NRC employment practices.

On July 23, 1986, the Commission was briefed on the status of NRC's EEO/Affirmative Action Plan goals, programs, and accomplishments. During fiscal year 1986, there was a continuing and significant growth of minority employees both in overall employment and at higher grades in the agency. While the agency's permanent full-time workforce declined by 2 percent, minority employees increased by 1 percent. Minorities at grade levels of GG-11 and above also increased at a rate of 1 percent, as against NRC's 2 percent loss overall in those grades. In grades GG-12 and -13, minority employee representation rose 10 and 12 percent, respectively, against agency growth of 4 and 6 percent, respectively.

The NRC conducted an active recruitment program during the report period, placing strong emphasis on colleges with a high quality engineering program and a good minority and/or female representation. NRC representatives made 33 visits to college campuses, seven of them predominately minority schools. These recruitment visits included participation in 12 campus "job fairs" for minorities or women.

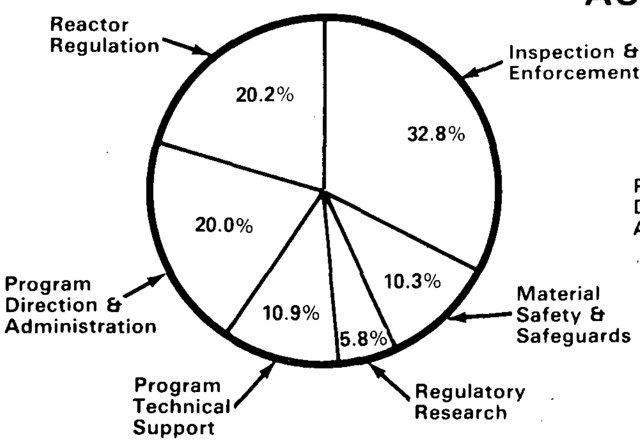
The agency has established a revised Upward Mobility Plan (UMP) to provide developmental opportunities to lower-level employees. The program focuses on the selection of those employees who show the potential to function effectively in professional or para-professional positions with greater growth potential. During fiscal year 1986, 13 UMP positions were filled.

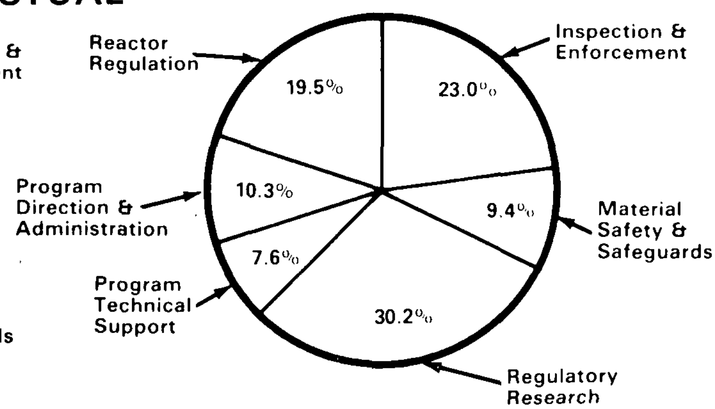
An analysis of the EEO accomplishment report—submitted annually by Office Directors and Regional Administrators to the Director, Office of Small and Disadvantaged Business Utilization/Civil Rights—was provided to the Chairman and the Executive Director for Operations to apprise them of the performance of managers in achieving their assigned goals. The Director, Office of Small and Disadvantaged Business Utilization/Civil Rights, continues to function as a non-voting, ex-officio member of the SES Performance Review Board.

Of the 62 discrimination complaints filed with the agency since 1975, 13 were pending as of September 30, 1986, including six new complaints filed in fiscal year 1986.

As part of the continuing effort to provide EEO training for NRC personnel, a training course was developed and conducted by OSDBU/CR staff at the request of the Office of Nuclear Material Safety and Safeguards, Waste Management Division. This course was designed to assure understanding of Federal EEO policy and Affirmative Action programs.

NRC RESOURCES
FY 1986
ACTUAL

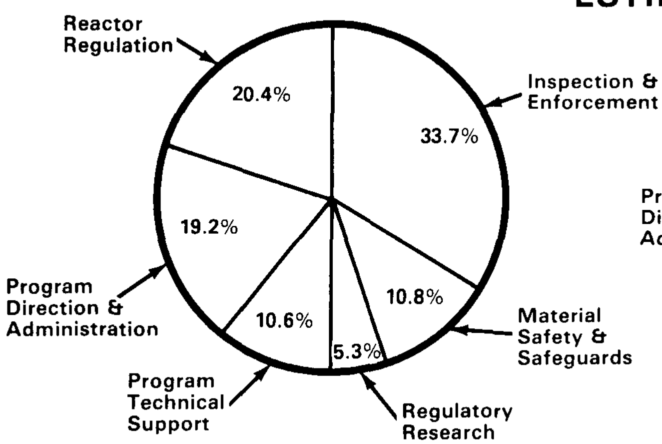


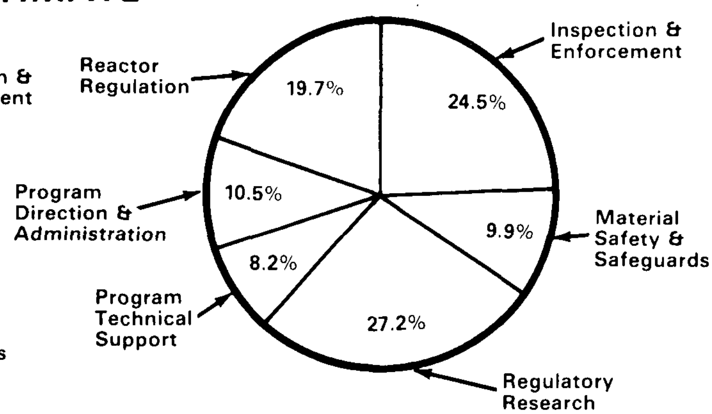


PERSONNEL 3445
(Full Time Equivalent)

FUNDS $410 MILLION

NRC RESOURCES
FY 1987
ESTIMATE





PERSONNEL 3369
(Full Time Equivalent)

FUNDS $410 MILLION

FY 1985/1986 Financial Statements

Balance Sheet (in thousands)

Assets	*September 30, 1986*	*September 30, 1985*
Cash:		
Appropriated Funds In U.S. Treasury	$ 129,055	$ 149,975
Other—Notes 1 & 3	4,540	5,837
Imprest Fund Balance	238	-0-
	133,833	155,812
Accounts Receivable:		
Federal Agencies	184	406
Miscellaneous Receipts—Note 2	14,415	15,631
Other	236	110
Less—Allowance For Uncollectibles	306	306
	14,529	15,841
Plant:		
Completed Plant and Equipment	30,675	27,534
Less—Accumulated Depreciation	13,881	10,583
	16,794	16,951
Advances and Prepayments:		
Federal Agencies	-0-	-0-
Other	4,374	3,370
	4,374	3,370
Total Assets	169,530	$ 191,974

Liabilities and NRC Equity	*September 30, 1986*	*September 30, 1985*
Liabilities:		
Funds Held for Others—Notes 1 & 3	$ 4,540	$ 5,837
Accounts Payable and Accrued Expenses:		
Federal Agencies	18,301	24,058
Other	26,833	26,729
Accrued Annual Leave of NRC Employees	13,222	13,095
Deferred Revenue—Note 3	-0-	-0-
Total Liabilities	$ 62,896	$ 69,719
NRC Equity: Balance at October 1	122,255	117,405
Additions:		
Funds Appropriated—Net	400,026	448,200
Non-Reimbursable Transfers from Other Gov't Agencies	-0-	-0-
	522,281	565,605
Deductions:		
Net Cost of Operations	363,788	361,690
Funds Returned to U.S. Treasury—Note 2	51,859	81,660
	415,647	443,350
Total NRC Equity	106,634	122,255
Total Liabilities and NRC Equity	$ 169,530	$ 191,974

Note 1. As of September 30, 1986, includes $3,431,169.84 of funds received under cooperative research agreements involving NRC, DOE, Euratom, France, Federal Republic of Germany, Japan, Austria, the Netherlands, Belgium, and the United Kingdom.
Also included is $250,386.00 of funds received from deferred revenue billings. These funds will be refunded and/or recorded as earned revenue after the cost of processing the applications has been finalized and, accordingly, are not available for NRC use. See Note 3.

Note 2. These funds are not available for NRC use.

Note 3. On March 24, 1978, 10 CFR 1 was revised. Contained therein by category of license are maximum fee amounts to be paid by applicants at the time a facility or material license is issued. Also, after the review of the license application is complete, the expenditures for professional manpower and appropriate support services are to be determined and the resultant fee assessed. In no event will the fee exceed the maximum fee for that license category, which generally has been paid. This could involve the refunding of a significant portion of the initial amount paid. Therefore, the revenue is recorded in a Deferred Revenue account at the time of billing and is removed from this account and recorded in Funds Held for Others when the bill is paid. The balance in the Deferred Revenue account consists of deferred revenue on billings issued but not collected. See Note 1.

Note 4. Represents current year cost of plant and equipment acquisitions for use at DOE facilities.

FY 1985/1986 NRC Statement of Operations
(in thousands)

	Fiscal Year, 1986 (October 1, 1985, thru September 30, 1986)	*Fiscal Year, 1985 (October 1, 1984, thru September 30, 1985)*
Personnel Compensation	$ 153,438	$ 152,943
Personnel Benefits	18,301	19,150
Program Support	180,808	219,700
Administrative Support	48,456	49,558
Travel of Persons	9,593	8,887
Equipment (Technical)—Note 4	210	295
Construction—Note 4	-0-	-0-
Taxes and Indemnities	30	749
Refunds to Licensees	-0-	-0-
Representational Funds	3	3
Reimbursable Work	92	76
Increase in Annual Leave Accrual	128	810
Depreciation Expense	3,297	2,679
Equipment Write-Offs and Adjustments	167	278
Allowance For Uncollectibles	-0-	306
Total Cost of Operations	$ 414,523	$ 455,434
Less Revenues:		
Reimbursable Work for Other Federal Agencies	92	76
Fees (Deposited in U.S. Treasury as Miscellaneous Receipts)—Note 2:		
Material Licenses	3,151	3,062
Facility Licenses	43,636	87,890
Other	3,855	2,716
Total Revenue	50,734	93,744
Net Cost of Operations Before Prior Year Adjustments	363,789	361,690
Prior Year Adjustment	-0-	-0-
Net Cost of Operations	$ 363,789	$ 361,690

U.S. Government Investment in the Nuclear Regulatory Commission
(in thousands)
(From January 19, 1975 through September 30, 1986)

Appropriation Expenditures:	
Fiscal Year 1975 (January 19, 1975 through June 30, 1975)	$ 52,792
Fiscal Year 1976 (July 1, 1975 through September 30, 1976)	226,248
Fiscal Year 1977 (October 1, 1976 through September 30, 1977)	230,559
Fiscal Year 1978 (October 1, 1977 through September 30, 1978)	270,877
Fiscal Year 1979 (October 1, 1978 through September 30, 1979)	309,493
Fiscal Year 1980 (October 1, 1979 through September 30, 1980)	377,889
Fiscal Year 1981 (October 1, 1980 through September 30, 1981)	416,867
Fiscal Year 1982 (October 1, 1981 through September 30, 1982)	441,902
Fiscal Year 1983 (October 1, 1982 through September 30, 1983)	514,613
Fiscal Year 1984 (October 1, 1983 through September 30, 1984)	462,084
Fiscal Year 1985 (October 1, 1984 through September 30, 1985)	467,902
Fiscal Year 1986 (October 1, 1985 through September 30, 1986)	420,946
	$4,192,172
Unexpended Balance of Appropriated Funds in U.S. Treasury September 30, 1986	129,055
Transfer of Refunds Receivable from Atomic Energy Commission, January 19, 1975	429
Funds Appropriated—Net	4,321,656
Less:	
Funds Returned to U.S. Treasury—Note 2	274,230
Assets and Liabilities Transferred from Other Federal Agencies Without Reimbursement	1,673
Net Cost of Operations from January 19, 1975 through September 30, 1986	3,939,119
Total Deductions	4,215,022
NRC Equity at September 30, 1986 as Shown on Balance Sheet	$ 106,634

Appendix 1

NRC ORGANIZATION

(As of December 31, 1986)

COMMISSIONERS

Lando W. Zech, Jr., Chairman
Thomas M. Roberts
James K. Asselstine
Frederick M. Bernthal
Kenneth M. Carr

The Commission Staff

General Counsel, William C. Parler
Office of Public Affairs, Joseph J. Fouchard, Director
Office of Congressional Affairs, Carlton C. Kammerer, Director
Office of Inspector and Auditor, Sharon R. Connelly, Director
Secretary of the Commission, Samuel J. Chilk
Office of Investigations, Ben B. Hayes, Director

Other Offices

Advisory Committee on Reactor Safeguards, William Kerr, Chairman
Atomic Safety & Licensing Board Panel, B. Paul Cotter, Jr., Chairman
Atomic Safety & Licensing Appeal Panel, Alan S. Rosenthal, Chairman

EXECUTIVE DIRECTOR FOR OPERATIONS

Executive Director for Operations, Victor Stello, Jr.
Deputy Executive Director for Operations, Jack W. Roe
Deputy Executive Director for Regional Operations
and Generic Requirements, James H. Sniezek
Deputy Executive Director for Operations (Information Resources Management),
William G. McDonald
Assistant for Operations, Thomas A. Rehm

Program Offices

Office of Nuclear Reactor Regulation, Harold R. Denton, Director
Office of Nuclear Material Safety and Safeguards, John G. Davis, Director
Office of Nuclear Regulatory Research, Eric S. Beckjord, Director
Office of Inspection and Enforcement, James M. Taylor, Director

Staff Offices

Office of Administration, Patricia G. Norry, Director
Office of Resource Management/Controller, Ronald M. Scroggins
Office of International Programs, James R. Shea, Director
Office of State Programs, G. Wayne Kerr, Director
Office for Analysis and Evaluation of Operational
Data, Clemens J. Heltemes, Jr., Director
Office of Small and Disadvantaged Business Utilization/
Civil Rights, William B. Kerr, Director

Regional Offices

Region I—Philadelphia, Pa., Thomas E. Murley, Regional Administrator
Region II—Atlanta, Ga., J. Nelson Grace, Regional Administrator
Region III—Chicago, Ill., James G. Keppler, Regional Administrator
Region IV—Dallas, Tex., Robert D. Martin, Regional Administrator
Region V—San Francisco, Cal., John B. Martin, Regional Administrator

The NRC is responsible for licensing and regulating nuclear facilities and materials and for conducting research in support of the licensing and regulatory process, as mandated by the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and the Nuclear Nonproliferation Act of 1978; and in accordance with the National Environmental Policy Act of 1969, as amended, and other applicable statutes. These responsibilities include protecting public health and safety, protecting the environment, protecting and safeguarding materials and plants in the interest of national security, and assuring conformity with antitrust laws. Agency functions are performed through: standards-setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits and licenses; inspection, investigation and enforcement; evaluation of operating experience; and regulatory research. The Commission itself is composed of five members, appointed by the President and confirmed by the Senate, one of whom is designated by the President as Chairman. The Chairman is the principal executive officer and the official spokesman of the Commission.

The Executive Director for Operations directs and coordinates the Commission's operational and administrative activities among the program and support staff offices described below and also coordinates the development of policy options for Commission consideration. The EDO reports directly to the Chairman.

The Office of Nuclear Reactor Regulation licenses nuclear power, test and research reactors under a two-phase process. A construction permit is granted before facility construction can begin and an operating license is issued before fuel can be loaded. NRR reviews license applications to assure that each proposed facility can be built and operated without undue risk to the health and safety of the public and with minimal impact on the environment. NRR monitors operating reactor facilities during their lifetime through decommissioning.

The Office of Nuclear Material Safety and Safeguards is responsible for the licensing and regulation of facilities and materials associated with the processing, transport and handling of nuclear materials, and with the disposal of nuclear waste; the office also regulates uranium recovery facilities. NMSS reviews and assesses safeguards against potential threats, thefts and sabotage for licensed facilities, including reactors, working closely with other NRC offices in coordinating safety and safeguards programs and in recommending research, standards and policy options necessary for their successful operation.

The Office of Nuclear Regulatory Research plans and conducts a comprehensive research and standards program that is deemed necessary for the performance of the Commission's licensing and regulatory functions and that is responsive to current and future NRC needs. The program covers such areas as facility operation, engineering technology, accident evaluation, probabilistic risk analysis, siting, health, and waste management.

The Office of Inspection and Enforcement develops and oversees programs of inspection of nuclear facilities and materials licensees in order to determine whether those facilities are constructed and operations are conducted in compliance with license provisions and Commission regulations; in order to identify conditions that may adversely affect the protection of the public health and safety, of nuclear materials and facilities, or of the environment; and in order to provide a basis for recommending issuance or denial of licenses. IE develops and oversees a program of investigation of accidents, incidents and allegations of improper actions that involve nuclear material and facilities; enforces NRC regulations and license provisions; and manages and directs all NRC actions related to emergency preparedness, including evaluations of State and local emergency plans performed by the Federal Emergency Management Agency (FEMA). It also performs audits of its programs as carried out by NRC Regional Offices.

The Regional Offices are under the supervision and direction of the Executive Director for Operations and carry out NRC regulatory programs originating in various Headquarters Offices.

THE COMMISSION STAFF

The Office of the Secretary provides general management services to support the Commission and to implement Commission decisions, advises and assists the Commission and staff on the planning, scheduling and conduct of Commission business; prepares for and records Commission meetings; manages the Commission staff paper system and monitors the status of all items requiring action; integrates automated data processing and office automation initiatives into the Commission's administrative system, maintains a forecast of matters for future Commission consideration; processes and controls Commission correspondence; maintains the Commission's official records; maintains the official Commission adjudicatory and rulemaking dockets and serves Commission issuances in all adjudicatory matters and public proceedings; administers the NRC Historical Program; and directs and administers the NRC Public Document Room.

The Office of the General Counsel directs matters of law and legal policy, providing opinions, advice, and assistance to the Commission and staff with respect to all activities of the Agency."

The Office of Investigations conducts, supervises and assures quality control of investigations of licensees, applicants, contractors or vendors, including the investigation of all allegations of wrongdoing by other than NRC employees and contractors. The Office develops policy, procedures and standards for these activities.

The Office of Inspector and Auditor investigates to ascertain the integrity of all NRC operations; investigates allegations of NRC employee misconduct, equal employment and civil rights complaints, and claims for personal property loss or damage; conducts the NRC's internal audit activities; and hears individual employee concerns regarding Commission activities, under the Agency's "open door" policy. The Office develops policies governing the Commission's financial and management audit program and is the agency contact with the General Accounting Office on this function. The Office refers criminal matters to the Department of Justice and maintains liaison with law enforcement agencies.

The Office of Public Affairs plans and administers NRC's program to inform the public of Commission policies, programs and activities and keeps NRC management informed of public affairs activities of interest to the Commission. OPA reports directly to the Chairman.

The Office of Congressional Affairs provides advice and assistance to the Commission and senior staff on congressional matters, coordinates NRC's congressional relations activities, and maintains liaison for the Commission with congressional committees and members of Congress. OCA reports directly to the Chairman.

SUPPORT STAFF

The Office of Administration directs the Agency's programs for organization and personnel management; security and classification; technical information and document control; facilities and materials license fees; contracting and procurement; rules, proceedings and document services, including administration of Freedom of Information Act and Privacy Act requests; management development and training; telecommunications; transportation services; management of space; and other administrative services.

The Office of Resource Management develops and maintains NRC's financial and manpower management programs, including policies, procedures and standards of accounting, budgeting, cost analysis, resource planning and analysis, and automated data processing systems development and support. The Office provides management information for other offices and issues special reports from the NRC to Congress, other government agencies and the public. The Office assists NRC offices in statistical matters and in the budget process, keeping the EDO and Commission informed on programs and issues of significance. RM also maintains liaison with the Office of Management and Budget, the Congress, other agencies of government, and the private sector.

The Office of International Programs plans and implements programs of international nuclear safety cooperation, creating and maintaining relationships with foreign regulatory agencies and international organizations; coordinates NRC export-import and international safeguards policies; issues export and import licenses; and coordinates responses by NRC to other agencies related to export-import actions and issues.

The Office of State Programs directs programs related to regulatory relationships with State governments and organizations and interstate bodies, manages the NRC State Agreements program, administers the indemnification program and performs financial qualifications reviews of applicants and licensees. The Office also verifies that applicants are not in violation of the antitrust laws.

The Office for Analysis and Evaluation of Operational Data provides agency coordination for the collection, storage, and retrieval of operational data associated with licensed activities, analyzes and evaluates such operational experience and feeds back the lessons of that experience to NRC licensing, standards and inspections activities. The Office oversees action taken in response to the feedback and assesses the overall effectiveness of the agency-wide operational safety data program, serving as a focal point for interaction with the ACRS and industry groups involved in operational safety data analysis and evaluation.

The Office of Small and Disadvantaged Business Utilization/Civil Rights develops and implements the NRC's program in accordance with the Small Business Act, as amended, insuring that appropriate consideration is given to labor surplus area firms and women-owned businesses. The Office develops and recommends NRC policy providing for equal employment opportunity and develops, monitors and evaluates the affirmative action program to assure compliance with the policy. The Office also serves as contact with local and national public and private organizations with related interests.

OTHER ORGANIZATIONAL ELEMENTS

The Advisory Committee on Reactor Safeguards is a statutory committee of 15 scientists and engineers advising the Commission on safety aspects of proposed and existing nuclear facilities and on the adequacy of proposed reactor safety standards and performing such other duties as the Commission may request. The Committee conducts a continuing study of reactor safety research and submits an annual report to the Congress. The Committee also administers the ACRS Fellowship Program.

The Atomic Safety and Licensing Board Panel is a panel of lawyers and others with expertise in various technical fields from which three-member Licensing Boards are drawn to conduct public hearings and make such intermediate or final decisions as the Commission may authorize in proceedings to grant, amend, suspend or revoke NRC licenses.

The Atomic Safety and Licensing Appeal Panel is a panel from which three-member Appeal Boards are selected to exercise the authority and perform the review functions which would otherwise be carried out by the Commission in certain licensing proceedings. Licensing Board decisions are reviewable by an Appeal Board, either in response to an appeal or on its own initiative. The Appeal Board's decision is also subject to review by the Commission in response to an appeal for discretionary review or on its own initiative.

Appendix 2

NRC Committees and Boards

Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS) is a statutory committee established to advise the Commission on the safety aspects of proposed and existing nuclear facilities and the adequacy of proposed reactor safety standards, and to perform such other duties as the Commission may request. As of December 31, 1986, the members were:

CHAIRMAN: DR. WILLIAM KERR, Professor of Nuclear Engineering and Director of the Office of Energy Research, University of Michigan, Ann Arbor, Mich.
VICE-CHAIRMAN: DR. FORREST J. REMICK, Acting Vice President for Research and Graduate Studies and Professor of Nuclear Engineering, Pennsylvania State University, University Park, Pa.
DR. MAX W. CARBON, Professor and Chairman of Nuclear Engineering Department, University of Wisconsin, Madison, Wis.
MR. JESSE C. EBERSOLE, retired Head Nuclear Engineer, Division of Engineering Design, Tennessee Valley Authority, Knoxville, Tenn.
DR. HAROLD W. LEWIS, Professor of Physics, Department of Physics, University of California, Santa Barbara, Cal.
DR. CARSON MARK, Retired Division Leader, Los Alamos Scientific Laboratory, Los Alamos, N.M.
MR. CARLYLE MICHELSON, retired Principal Nuclear Engineer, Tennessee Valley Authority and Retired Director, Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, Washington, D.C.
DR. DADE W. MOELLER, Professor of Engineering in Environmental Health and Director, Office of Continuing Education, School of Public Health, Harvard University, Boston, Mass.
DR. DAVID OKRENT, Chairman, School of Engineering and Applied Science, University of California, Los Angeles, Cal.
MR. GLENN A. REED, Retired Plant Manager, Pt. Beach Nuclear Power Plant, Wisconsin Electric Power Company, Two Rivers, Wis.
DR. PAUL G. SHEWMON, Professor and Chairman of Metallurgical Engineering Department, Ohio State University, Columbus, Ohio
DR. CHESTER P. SIESS, Professor Emeritus of Civil Engineering, University of Illinois, Urbana, Ill.
MR. DAVID A. WARD, Chairman, Research Manager, Reactor Safety Research, E. I. du Pont de Nemours & Company, Savannah River Laboratory, Aiken, S.C.
MR. CHARLES J. WYLIE, Retired Chief Engineer, Electrical Division, Duke Power Company, Charlotte, N.C.

Atomic Safety and Licensing Board Panel

PANEL MEMBERS:

CHIEF ADMINISTRATIVE JUDGE B. PAUL COTTER, JR., ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Executive) ROBERT M. LAZO, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DEPUTY CHIEF ADMINISTRATIVE JUDGE—(Technical) FREDERICK J. SHON, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE GEORGE C. ANDERSON, Marine Biologist, University of Washington, Seattle, Wash.
JUDGE CHARLES BECHHOEFER, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE PETER B. BLOCH, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE GLENN O. BRIGHT, ASLBP Engineer, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE A. DIXON CALLIHAN, Retired Physicist, Union Carbide Corporation, Oak Ridge, Tenn.
JUDGE JAMES H. CARPENTER, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE HUGH K. CLARK, Retired Attorney, E.I. duPont deNemours & Company, Kennedyville, Md.
JUDGE RICHARD F. COLE, ASLBP Environmental Scientist U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MICHAEL A. DUGGAN, Economist, University of Texas, Austin, Tex.
JUDGE GEORGE A. FERGUSON, Physicist, Howard University, Washington, D.C.
JUDGE HARRY FOREMAN, Medical Doctor (retired), University of Minnesota, Minneapolis, Minn.
JUDGE RICHARD F. FOSTER, Environmental Scientist, Sunriver, Ore.
JUDGE JOHN H. FRYE, III, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JAMES P. GLEASON, Attorney, Silver Spring, Md.
JUDGE HERBERT GROSSMAN, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE CADET H. HAND, JR., Marine Biologist, University of California, Bodega Bay, Cal.
JUDGE JERRY HARBOUR, ASLBP Environmental Scientist U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE DAVID L. HETRICK, Nuclear Engineer, University of Arizona, Tucson, Ariz.

JUDGE ERNEST E. HILL, Nuclear Engineer, Lawrence Livermore Laboratory, Livermore, Cal.
JUDGE FRANK F. HOOPER, Marine Biologist, University of Michigan, Ann Arbor, Mich.
JUDGE HELEN F. HOYT, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE ELIZABETH B. JOHNSON, Nuclear Engineer, Oak Ridge National Laboratory, Oak Ridge, Tenn.
JUDGE WALTER H. JORDAN, Retired Physicist, Oak Ridge Laboratories, Oak Ridge, Tenn.
JUDGE JAMES L. KELLEY, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JERRY R. KLINE, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE JAMES C. LAMB, III, Sanitary Engineer, University of North Carolina, Chapel Hill, N.C.
JUDGE GUSTAVE A. LINENBERGER, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE LINDA W. LITTLE, Environmental Biologist, L.W. Little Associates, Raleigh, N.C.
JUDGE EMMETH A. LUEBKE, ASLBP Physicist, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MORTON B. MARGULIES, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE KENNETH A McCOLLOM, Electrical Engineer (retired), Oklahoma State University, Stillwater, Okla.
JUDGE GARY L. MILHOLLIN, Attorney, Catholic University of America, Washington, D.C.
JUDGE MARSHALL E. MILLER, Attorney (retired), Summerland, Fla.
JUDGE PETER A. MORRIS, ASLBP, Physicist, US Nuclear Regulatory Commission, Bethesda, Md.
JUDGE OSCAR H. PARIS, ASLBP Environmental Scientist, U.S. Nuclear Regulatory Commission Bethesda, Md.
JUDGE DAVID R. SCHINK, Oceanographer, Texas A&M University College Station, Tex.
JUDGE IVAN W. SMITH, ASLBP Administrative Law Judge, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JUDGE MARTIN J. STEINDLER, Chemist, Argonne National Laboratory, Argonne, Ill.
JUDGE SEYMOUR WENNER, Retired Administrative Law Judge, Postal Rate Commission, Chevy Chase, Md.
JUDGE SHELDON J. WOLFE, ASLBP Attorney, U.S. Nuclear Regulatory Commission, Bethesda, Md.

PROFESSIONAL STAFF

CHARLES J. FITTI, Director, Program Support and Analysis Staff, U.S. Nuclear Regulatory Commission, Bethesda, Md.
ELVA W. LEINS, Director, Program Support and Analysis STaff, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DAVID L. PRESTEMON, Legal Counsel to the Panel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
JACK G. WHETSTINE, Hearing Support Supervisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.

Atomic Safety and Licensing Appeal Panel

An Atomic Safety and Licensing Appeal Board, established September 18, 1969, was delegated the authority to perform the review function that would otherwise be performed by the Atomic Energy Commission in proceedings on applications for licenses or authorizations in which the Commission had a direct financial interest, and in such other licensing proceedings as the Commission might specify.

In view of the increase in the number of proceedings subject to administrative appellate review, the Atomic Safety and Licensing Appeal Panel was established on October 25, 1972, from whose membership three-member panel Appeal Boards could be designated for each proceeding in which the Commission had delegated its authority to an Appeal Board. At the same time, the Commission modified its rules to delegate authority to Appeal Boards in all proceedings involving the licensing of production and utilization facilities (for example, power reactors).

Pursuant to subsection 201(g)(1) of the Energy Reorganization Act of 1974, the functions performed by Appeal Boards were specifically transferred to the Nuclear Regulatory Commission. The Commission appoints members to the Appeal Panel, and the Chairman of the panel designates a three-member Appeal Board for each proceeding. The Commission retains review authority over decisions and actions of Appeal Boards. The Appeal Panel, on October 1, 1986, was composed of the following persons:

FULL-TIME MEMBERS:

ALAN S. ROSENTHAL, Appeal Panel Chairman, U.S. Nuclear Regulatory Commission, Bethesda, Md.
GARY J. EDLES, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DR. REGINALD L. GOTCHY, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.*
CHRISTINE N. KOHL, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS S. MOORE, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.
HOWARD A. WILBER, Appeal Panel Member, U.S. Nuclear Regulatory Commission, Bethesda, Md.

PART-TIME MEMBER:

DR. W. REED JOHNSON, Professor of Nuclear Engineering, University of Virginia, Charlottesville, Va.

PROFESSIONAL STAFF:

JOHN CHO, Counsel, U.S. Nuclear Regulatory Commission, Bethesda, Md.
THOMAS G. SCARBROUGH, Technical Advisor, U.S. Nuclear Regulatory Commission, Bethesda, Md.
DAVID P. GESSER, Law Clerk, U.S. Nuclear Regulatory Commission, Bethesda, Md.
*Resigned as of October 2, 1986.

Advisory Committee on Medical Uses of Isotopes

The Advisory Committee on Medical Uses of Isotopes (ACMUI) was established in July 1958. The ACMUI, composed of qualified physicians and scientists, considers medical questions referred to it by the NRC staff and renders expert opinions regarding the medical uses of radioisotopes. The ACMUI also advises the NRC staff, as required, on matters of policy. Members are employed under yearly personal services contracts. As of September 30, 1986, the members were:

RICHARD E. CUNNINGHAM, Chairman, ACMUI, Director, Division of Fuel Cycle and Material Safety, U.S. Nuclear Regulatory Commission, Silver Spring, Md.
DR. VINCENT P. COLLINS, Medical Director, Houston Institute for Cancer Research, Diagnosis and Treatment, Houston, Tex.
DR. SALLY J. DE NARDO, Director, Nuclear Hematology-Oncology, Department of Nuclear Medicine, University of California Davis Medical Center, Sacramento, Cal.
DR. JACK K. GOODRICH, Radiology Associates of Erie, Erie, Pa.
DR. MELVIN L. GRIEM, Professor and Director, Chicago Tumor Institute, University of Chicago, Chicago, Ill.
DR. NILO E. HERRERA, Director, Department of Laboratory Medicine, Danbury Hospital, Danbury, Conn.
DR. B. LEONARD HOLMAN, Chief, Clinical Nuclear Medicine, Department of Radiology, Brigham and Women's Hospital Boston, Mass.
DR. GERALD M. POHOST, Director, Division of Cardiovascular Disease, University of Alabama at Birmingham, Birmingham, Ala.
DR. EDWARD W. WEBSTER, Director, Department of Radiation Physics, Massachusetts General Hospital, Boston, Mass.
DR. DAVID H. WOODBURY, Director, Nuclear Medicine Section, Wayne County General Hospital, Westland, Mich.

Advisory Panel for the Decontamination of Three Mile Island Unit 2

The Advisory Committee for the Decontamination of Three Mile Island, Unit 2, was established in October 1980. Its purpose is to obtain input and views from the residents of the Three Mile Island area and afford Pennsylvania government officials an opportunity to participate in the Commission's decision-making process regarding cleanup plans for Three Mile Island Unit 2. The Panel consists of the following members representing agencies of the Commonwealth of Pennsylvania, local government authorities in the vicinity of the Three Mile Island facility, the scientific community and persons having their principal place of residence in the vicinity of the facility.

ARTHUR E. MORRIS, Panel Chairman, Mayor of Lancaster, Pa.
JOSEPH J. DINUNNO, Private Consultant, Annapolis, Md.
THOMAS GERUSKY, Director of the Pennsylvania Bureau of Radiation Protection, Department of Environmental Resources, Harrisburg, Pa.
JOHN LUETZELSCHWAB, Professor of Physics, Dickinson College, Carlisle, Pa.
ELIZABETH MARSHALL, resident of York, Pa.
KENNETH L. MILLER, Director of the Division of Health Physics and Associate Professor of Radiology, Milton S. Hershey Medical Center, Hershey, Pa.
FREDERICK S. RICE, Chairman, Dauphin County Board of Commissioners, Harrisburg, Pa.
GORDON ROBINSON, Associate Professor of Nuclear Engineering, Pennsylvania State University, Univeristy Park, Pa.
JOEL ROTH, resident of Elizabethville, Pa.
THOMAS SMITHGALL, resident of Lancaster, Pa.
ANN TRUNK, resident of Middletown, Pa.
NEIL WALD, Professor of Radiation Health, Department of Radiology, University of Pittsburgh, Pa.

Appendix 3

Local Public Document Rooms

Copies of most documents originating in the NRC or submitted to it for review are placed in the Commission's Public Document Room (PDR) at 1717 H Street, N.W., Washington, D.C., for public inspection. Other PDRs on NRC premises include the rooms at the Willste Building, 7915 Eastern Avenue, Silver Spring, Md., and in the five Regional Offices (the latter for documents related to nuclear material licenses, i.e., most byproduct and source material licenses). In addition, documents related to licensing proceedings or licensed operation of specific facilities are made available in local PDRs established in the vicinity of each proposed or existing nuclear facility. The locations of the local PDRs, the names of the persons to contact, and the names of the facilities for which documents are retained are listed below. (N.B. Updated listings of local PDRs may be obtained by writing to the Local Public Document Room Branch, Division of Rules and Records, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.)

ALABAMA

- Mrs. Maude S. Miller, Head Librarian
  Athens Public Library
  South Street
  Athens, Ala. 35611
  Browns Ferry Nuclear Power Station
  Browns Ferry Low-Level Waste Storage

- Ms. Bettye Forbus, Director
  Houston-Love Memorial Library
  212 W. Burdeshaw Street
  P.O. Box 1369
  Dothan, Ala. 36302
  Jospeh M. Farley Nuclear Plant

- Mrs. Peggy McCutchen, Director
  Scottsboro Public Library
  1002 South Broad Street
  Scottsboro, Ala. 35768
  Bellefonte Nuclear Plant

ARIZONA

- Mrs. Shera Farnham, Head
  Business and Science Division
  Phoenix Public Library
  12 East McDowell Road
  Phoenix, Ariz. 85004
  Palo Verde Nuclear Station

ARKANSAS

- Mrs. Delores Pollard, Serials Librarian
  Tomlinson Library
  Arkansas Tech. University
  Russellville, Ark. 72801
  Arkansas Nuclear One

CALIFORNIA

- Ms. Margaret J. Nystrom, Director
  Eureka-Humboldt County Library
  636 F Street
  Eureka, Cal. 95501
  Humboldt Bay Power Plant

- Mrs. Anne Connor, Senior Librarian
  West Los Angeles Regional Library
  11360 Santa Monica Boulevard
  Los Angeles, Cal. 90025
  UCLA Training Reactor

- Ms. Marilee Cogswell, Documents Librarian
  Sacramento Public Library
  828 I Street
  Sacramento, Cal. 95814
  Rancho Seco Nuclear Generating

- Ms. Judy Horn, Department Head
  University of California
  General Library
  P.O. Box 19557
  Irvine, Cal. 92713
  San Onofre Nuclear Station

- Mr. Chi Su Kim, Head
  Government Documents and Maps Dept.
  Robert E. Kennedy Library
  California Polytechnic State University
  San Luis Obispo, Cal. 93407
  Diablo Canyon Nuclear Power Plant

COLORADO

- Miss Shirley Soenksen, Acting Director
  Greeley Public Library
  City Complex Building
  919 7th Street
  Greeley, Colo. 80631
  Fort St. Vrain Nuclear Generating Station

CONNECTICUT

- Ms. Marcella Kenney, Reference Librarian
  Russell Library
  123 Broad Street
  Middletown, Conn. 06457
  Haddam Neck Plant

- Mr. Vincent Juliano, Director
  Waterford Public Library
  49 Rope Ferry Road
  Waterford, Conn. 06385
  Millstone Nuclear Power Station

FLORIDA

- Ms. Vanda Carnes, Librarian
  Crystal River Public Library
  668 N.W. First Avenue
  Crystal River, Fla. 32629
  Crystal River Nuclear Plant

- Ms. Jimmie Anne DeRoss, Librarian
  Charles S. Miley Learning Resources Ctr.
  Indian River Community College
  3209 Virginia Avenue
  Ft. Pierce, Fla. 33450
  St. Lucie Plant

- Ms. Karlinne Wulf, Librarian
  Miami-Dade Public Library
  Homestead Branch
  700 North Homestead Blvd.
  Homestead, Fla. 33030
  Turkey Point Plant

- Ms. Esther B. Gonzalez, Librarian
  Urban and Regional Documents Collection Library
  Florida International University
  Tamiami Campus
  Miami, Fla. 33199
  Turkey Point Plant

GEORGIA

- Mrs. Wynell Bush, Librarian
  Appling County Public Library
  301 City Hall Drive
  Baxley, Ga. 31513
  Edwin I. Hatch Nuclear Plant

- Mrs. Velna R. Glisson
County Librarian
412 4th Street
Waynesboro, Ga. 30830
Alvin W. Vogtle Nuclear Plant

ILLINOIS

- Mrs. Penny A. O'Rourke, Librarian
Byron Public Library District
109 N. Franklin Street
Byron, Ill. 61010
Byron Station

- Ms. Cheryle Rae Nyberg
Assistant Law Librarian
University of Illinois Law Library
504 East Pennsylvania Avenue
Champaign, Ill. 61820
Clinton Power Station

- Mrs. Malinda Evans, Librarian
Vespasian Warner Public Library
120 West Johnson Street
Clinton, Ill. 61727
Clinton Power Station

- Mr. Earl R. Shumaker, Head
Government Publications Department
Founder's Memorial Library
Northern Illinois University
DeKalb, Ill. 60115
Byron Station

- Mrs. Marie Hoscheid, Head
Reference Department
Moline Public Library
504 17th Street
Moline, Ill. 61265
Quad Cities Station
Sheffield Low-level Waste Burial Site

- Mrs. Pam Wilson
Head Librarian
Morris Public Library
604 Liberty Street
Morris, Ill. 60450
Dresden Nuclear Power Station
Morris Spent Fuel Storage Facility

- Ms. Evelyn Moyle, Documents Librarian
Jacobs Memorial Library
Illinois Valley Community College
Rural Route 1
Oglesby, Ill. 61348
LaSalle County Station

- Mr. Dave Sweet, Librarian
Business, Science and Technology Dept.
Rockford Public Library
215 North Wyman Street
Rockford, Ill. 61101
Byron Station

- Mrs. Sharon Ruda, Head Librarian
Government Documents Collection
Wilmington Public Library
201 South Kankakee Street
Wilmington, Ill. 60481
Braidwood Station

- Mrs. Joan Wilts, Head
Adult Services
Waukegan Public Library
128 N. County Street
Waukegan, Ill. 60085
Zion Nuclear Power Station

- Mrs. Charlene Sanders, Director
West Chicago Public Library
332. E. Washington Street
West Chicago, Ill. 60185
Keer-McGee West Chicago

INDIANA

- Mrs. Charlene M. Peters, Head
Adult Services Librarian
Madison-Jefferson County Public Library
420 West Main Street
Madison, Ind. 47250
Marble Hill Nuclear Generating Station

IOWA

- Mr. Roger Rayborn, Reference Librarian
Cedar Rapids Public Library
500 1st Street, S.E.
Cedar Rapids, Ia. 52401
Duane ARnold Energy Center

KANSAS

- Ms. Vickie Bozarth, Documents Librarian
Government Documents Division
William Allan White Library
Emporia State University
1200 Commercial Street
Emporia, Kans. 66801
Wolf Creek Generating Station

- Ms. Jan Brown, Documents Librarian
NRC-LPDR Documents Collection
Washburn University School of Law
Topeka, Kans. 66621
Wolf Creek Generating Station

LOUISIANA

- Mrs. Smittie Bolner, Head
Government Documents Department
Troy H. Middleton Library
Louisiana State University
Baton Rouge, La. 70803
River Bend Station

- Mr. Kenneth E. Owen, Head
Louisiana Collection
Earl K. Long Library
University of New Orleans
Lakefront Drive
New Orleans, La. 70148
Waterford Generating Station

MAINE

- Mrs. Barbara Shelton, Librarian
Wiscasset Public Library
High Street
P.O. Box 367
Wiscasset, Me. 04578
Maine Yankee Atomic Power Plant

MARYLAND

- Ms. Patricia Hofmann, Reference Librarian
Calvert County Public Library
Fourth Street
P.O. Box 300
Prince Frederick, Md. 20678
Calvert Cliffs Nuclear Power Plant

MASSACHUSETTS

- Mrs. Margaret E. Howland, Director
Library/Learning Resource Center
Greenfield Community College
One College Drive
Greenfield, Mass. 01301
Yankee Rowe Nuclear Power Station

- Ms. Grace E. Karbott, Library Technician
Plymouth Public Library
11 North Street
Plymouth, Mass. 02360
Pilgrim Nuclear Power Station

MICHIGAN

- Dr. Carol Juth, Reference Librarian
Van Zoeren Library
Hope College
Holland, Mich. 49423
Palisades Nuclear Plant

- Ms. Sandra L. Cook, Reference Librarian
Grace A. Dow Memorial Library
1710 W. St. Andrews Road
Midland, Mich. 48640
Midland Plant

- Mr. Gig Stewart, Director
North Central Michigan College
1515 Howard Street
Petoskey, Mich. 49770
Big Rock Point Nuclear Plant

- Mrs. Marie D. Chulski, Head
Reference and Information
Reference/Government Documents Coordinator
Ellis Reference and Information Center
Monroe County Library System
3700 S. Custer Road
Monroe, Mich. 48161
Enrico Fermi Atomic Power Plant

- Ms. Bea Rodgers, Library Assistant
Maude Preston Palenske Memorial Library
500 Market Street
St. Joseph, Mich. 49085
Donald C. Cook Nuclear Power Plant

MINNESOTA

- Mr. Thomas Smisek, Librarian
Technology and Science Department
Minneapolis Public Library
300 Nicollet Mall
Minneapolis, Minn. 55401
Monticello Nuclear Generating Plant
Prairie Island Nuclear Station

MISSISSIPPI

- Mrs. Gayle Keefe
Library Technical Assistant
George M. McLendon Library
Hinds Junior College
Main Street
Raymond, Miss. 39154
Grand Gulf Nuclear Station

MISSOURI

- Mrs. Evelyn Hillard
Public Services Librarian
Callaway County Public Library
710 Court Street
Fulton, Mo. 65251
Callaway Plant

- Ms. Jerry L. Ewing
Documents Librarian
John M. Olin Library
Washington University
Skinker and Lindell Boulevards
St. Louis, Mo. 63130
Callaway Plant

NEBRASKA

- Mrs. Loy Mowery, Director
Auburn Public Library
1118 15th Street
P.O. Box 324
Auburn, Neb. 68305
Cooper Nuclear Station

- Mrs. Theresa Jehlik, Librarian
Business, Science and Technology Dept.
W. Dale Clark Library
215 S. 15th Street
Omaha, Neb. 68102
Fort Calhoun Station

NEW HAMPSHIRE

- Ms. Pamela Gjettum, Director
Exeter Public Library
Front Street
Exeter, N.H. 03833
Seabrook Nuclear Station

NEW JERSEY

- Ms. Joanne L. Owens, Librarian
Pennsville Public Library
190 S. Broadway
Pennsville, N.J. 08070
Hope Creek Nuclear Station

- Ms. Elizabeth C. Fogg, Director
Salem Free Public Library
112 West Broadway
Salem, N.J. 08079
Salem Nuclear Generating Station

- Ms. Lois J. Brown, Reference Librarian
Reference Department
Ocean County Library
101 Washington Street
Toms River, N.J. 08753
Oyster Creek Nuclear Power Plant

NEW YORK

- Ms. Blanche Judd, Coordinator
Reference and Documents Department
Penfield Library
State University of New York
Oswego, N.Y. 13126
James A. Fitzpatrick Nuclear Power Plant
Nine Mile Point Nuclear Station

- Ms. Carolyn Johnson, Head
Business and Social Science Division
Rochester Public Library
115 South Avenue
Rochester, N.Y. 14610
Robert Emmet Ginna Nuclear Plant

- Mr. Donald Cloudsley, Director
Buffalo and Erie County Public Library
Lafayette Square
Buffalo, N.Y. 14203
West Valley Demonstration Project

- Ms. Francine Dimmick, Reference Clerk
Shoreham-Wading River Public Library
Route 25 A
Shoreham, N.Y. 11786
Shoreham Nuclear Power Station

- Mr. Oliver F. Swift
Municipal Reference Librarian
White Plains Public Library
100 Martine Avenue
White Plains, N.Y. 10601
Indian Point Station

NORTH CAROLINA

- Ms. Dawn Hubbs, Documents Librarian
J. Murrey Atkins Library
University of North Carolina at Charlotte—UNCC Station
Charlotte, N.C. 28223
William B. McGuire Nuclear Station

- Mrs. Joe Ann Stephens
Reference Librarian
Richard B. Harrison Library
1313 New Bern Avenue
Raleigh, N.C. 27610
Shearon Harris Nuclear Power Plant

- Mrs. Arlene Hanerfeld
Assistant Reference/Documents Librarian
William Madison Randall Library
University of North Carolina at Wilmington
601 S. College Road
Wilmington, N.C. 28403-3297
Brunswick Steam Electric Plant

OHIO

- Ms. Ann Freed, Reference Librarian
Perry Public Library
3753 Main Street
Perry, Ohio 44081
Perry Nuclear Power Plant

- Mrs. Julia Baldwin, Documents Librarian
  Government Documents Collection
  William Carlson Library
  University of Toledo
  2801 West Bancroft Avenue
  Toledo, Ohio 43606
  Davis-Besse Nuclear Power Station

OKLAHOMA

- Mrs. O. J. Grosclaude, Branch Director
  Sallisaw City Library
  101 E. Cherokee St.
  Sallisaw, Okla. 74995
  Kerr-McGee Sequoyah

OREGON

- Ms. Arlys Fones, Head
  Social Science and Science Dept.
  Government Documents Collection
  Library Association of Portland
  801 S.W. 10th Avenue
  Portland, Ore. 97205
  Trojan Nuclear Plant

PENNSYLVANIA

- Ms. Mary E. Colombo, Director
  B.F. Jones Memorial Library
  663 Franklin Avenue
  Aliquippa, Pa. 15001
  Beaver Valley Power Station

- Mr. John E. Geschwindt, Head
  Government Publications Section
  State Library of Pennsylvania
  Walnut Street and Commonwealth Avenue
  Box 1601
  Harrisburg, Pa. 17105
  Three Mile Island Nuclear Station
  Peach Bottom Atomic Power Station

- Ms. Yu-Wan Cheng
  Apollo Memorial Library
  219 N. Pennsylvania
  Apollo, Pa. 15613
  Babcock & Wilcox Parks Township and B&W Apollo

- Mr. William A. Felker
  Reference and Information Manager
  Government Publication Department
  Free Library of Philadelphia
  19th and Vine Streets
  Philadelphia, Pa. 19103
  Limerick Generating Station

- Mrs. Julia Albright
  Interlibrary Loan Librarian
  Pottstown Public Library
  500 High Street
  Pottstown, Pa. 19464
  Limerick Generating Station

- Mr. Ernest Fuller
  NRC Materials Aide
  Saxton Community Library
  911 Church Street
  Saxton, Pa. 16678
  Saxton Nuclear Experimental Facility

- Ms. Diane H. Smith, Head
  Government Documents
  Pattee Library
  Room C 207
  Pennsylvania State University
  University Park, Pa. 16802
  Beaver Valley Power Station
  Susquehanna Steam Electric Station

- Mr. Ray Van de Castle
  Reference Librarian
  Reference Department
  Osterhout Free Library
  71 South Franklin Street
  Wilkes-Barre, Pa. 18701
  Susquehanna Steam Electric Station
  Susquehanna Low-Level Waste Storage

RHODE ISLAND

- Ms. Ann Crawford, Director
  Cross Mill Public Library
  Old Post Road
  Charlestown, R.I. 02813
  Wood River Junction

SOUTH CAROLINA

- Mrs. Sandra Hummel, Director
  Barnwell County Public Library
  Hagood Avenue
  Barnwell, S.C. 29812
  Barnwell Reprocessing Plant
  Barnwell Low-Level Waste Burial Site

- Ms. Maureen Harris, Head
  Public Documents Collection
  Robert M. Cooper Library
  Clemson University
  Clemson, S.C. 29631
  Oconee Nuclear Plant

- Ms. Mary Toll, Reference Librarian
  Technical Services Department
  South Carolina State Library
  1500 Senate Street
  Columbia, S.C. 29201
  Catawba Nuclear Station

- Ms. Virginia Warr, Librarian
  Nuclear Information Depository
  Hartsville Memorial Library
  220 N. Fifth Street
  Hartsville, S.C. 29550
  H.B. Robinson Plant

- Mrs. Mary Mallaney
  Assistant Reference Librarian
  York County Library
  138 East Black Street
  P.O. Box 10032
  Rock Hill, S.C. 29730
  Catawba Nuclear Station

- Ms. Joyce McCall, Librarian
  Oconee County Library
  501 W. South Broad Street
  Walhalla, S.C. 29691
  Oconee Nuclear Plant

- Ms. Sarah D. McMaster, Director
  Fairfield County Library
  Garden and Washington Streets
  Winnsboro, S.C. 29180
  Virgil C. Summer Nuclear Station

TENNESSEE

- Ms. Patricia Maroney, Head
  Business, Science and Technology Dept.
  Chattanooga-Hamilton County Library
  1001 Broad Street
  Chattanooga, Tenn. 37402
  Sequoyah Nuclear Plant
  Watts Bar Nuclear Plant
  TVA Sequoyah Low-Level Waste Storage

TEXAS

- Miss Willie Paredez
  Documents Assistant
  Library—Documents
  University of Texas at Arlington
  701 South Cooper
  P.O. Box 19497
  Arlington, Tex. 76019
  Comanche Peak Steam Electric Station

- Mrs. Audray Bateman, Curator
  Austin-Travis County Collection
  Austin History Center
  Austin Public Library
  810 Guadalupe Street
  P.O. Box 2287
  Austin, Tex. 78701
  South Texas Project

- Mrs. Freda Whitworth
  Librarian
  Glen Rose-Somervell Library
  Barnard and Highway 144
  P.O. Box 417
  Glen Rose, Tex. 76043
  Comanche Peak Steam Electric Station

WASHINGTON

- Mr. John R. Deosdade
  Documents Librarian
  Business and Science Dept.
  San Antonio Public Library
  203 S. St. Mary's Street
  San Antonio, Tex. 78205
  South Texas Project

- Ms. Patsy G. Norton, Director
  Wharton County Junior College
  J.M. Hodges Learning Center
  911 Boling Highway
  Wharton, Tex. 77488
  South Texas Project

VERMONT

- Mrs. Junia A. Bryant
  Assistant Librarian
  Brooks Memorial Library
  224 Main Street
  Brattleboro, Vt. 05301
  Vermont Yankee Nuclear Power Station

VIRGINIA

- Mr. Gregory A. Johnson
  Senior Public Services Assistant
  Manuscripts Dept.
  Alderman Library
  University of Virginia
  Charlottesville, Va. 22901
  North Anna Power Station

- Mr. Alan Zoellner
  Documents Librarian
  Documents Dept.
  Swem Library
  College of William and Mary
  Williamsburg, Va. 23185
  Surry Power Station
  Surry Independent Spent Fuel Storage

WASHINGTON

- Mrs. Lois McCleary
  Library Assistant
  W.H. Abel Memorial Library
  125 Main Street
  South Montesano, Wash. 98563
  WPPSS Nuclear Projects 3 & 5

- Ms. Janet Fullerton
  Reference Librarian
  Richland Public Library
  Swift and Northgate Streets
  Richland, Wash. 99352
  WPPSS Nuclear Projects 1, 2, & 4
  Basalt Waste Isolation Project,
  Richland Low-level Waste Burial Site

WISCONSIN

- Mrs. Kathy Pletcher, Head
  Government Documents Section
  Library Learning Center
  University of Wisconsin
  2420 Nicolet Drive
  Green Bay, Wis. 54301
  Kewaunee Nuclear Power Plant

- Ms. Elizabeth J. Helfrich
  Reference Librarian
  LaCrosse Public Library
  800 Main Street
  LaCrosse, Wis. 54601
  LaCrosse Nuclear Power Plant

- Ms. Joan Schmid, Head
  Adult Services
  Joseph Mann Library
  1516 16th Street
  Two Rivers, Wis. 54241
  Point Beach Nuclear Plant

Appendix 4

Regulations and Amendments—Fiscal Year 1986

The regulations of the Nuclear Regulatory Commission are contained in Title 10, Chapter 1, of the Code of Federal Regulations. Effective and proposed regulations concerning licensed activities, and certain policy statements related to them, which were published in the Federal Register during fiscal year 1986, are described briefly below.

REGULATIONS AND AMENDMENTS PUT INTO EFFECT

Minor Clarifying Amendments—Part 9

On October 9, 1985 (50 FR 41127), the NRC published an amendment to its regulations pertaining to the availability of records under the Freedom of Information Act to clarify and conform them to existing case law and to reflect long-standing agency practice. The amendment, effective immediately, also conformed the reproduction costs for Privacy Act records to costs currently charged at the NRC's Public Document Room and other NRC offices for publicly available documents.

Regional Licensing Program; Fort St. Vrain Nuclear Generating Station—Part 50

On October 9, 1985 (50 FR 41128), the NRC published an amendment to its regulations concerning the domestic licensing of utilization facilities to reflect the NRC's regional licensing program. The amendment, effective October 4, 1985, states that authority and responsibility for implementing NRC's nuclear reactor licensing program pertaining to the Fort St. Vrain Nuclear Generating Station will be carried out by the Director of Nuclear Reactor Regulation.

Hybrid Hearing Procedures for Expansion of Spent Nuclear Fuel Storage Capacity at Civilian Nuclear Power Reactors—Parts 2 and 72

On October 15, 1985 (50 FR 41662), the NRC amended its regulations, effective November 14, 1985, to implement Section 134 of the Nuclear Waste Policy Act of 1982. The Act permits the use of a modified hearing process in certain contested proceedings on an application for a license or license amendment to expand the spent nuclear fuel storage capacity at the site of a civilian nuclear power reactor. The amendment changes existing agency practice by employing less formal procedures in the early stage of the hearing process and by designating only genuine and substantial issues for resolution in an adjudicatory hearing.

Uranium Mill Tailing Regulations: Conforming NRC Requirements to EPA Standards—Parts 40 and 150

On October 16, 1985 (50 FR 41852), the NRC published an amendment to its regulations governing the disposal of uranium mill tailings. The regulation, effective November 15, 1985, conforms existing NRC regulations to the regulations published by the Environmental Protection Agency for the protection of the environment from these wastes.

Statement of Organization and General Information—Part 1

On October 18, 1985 (50 FR 42145), the NRC published an amendment to its regulations, effective immediately, that granted additional rulemaking authority to its Executive Director for Operations.

General Statement of Policy and Procedure for Enforcement Actions—Part 2

On November 20, 1985 (50 FR 47716), the NRC published an amendment to the regulations that describes the agency's enforcement policy. The amendment, effective February 18, 1986, describes how the enforcement policy applies to vendors of products or services that are supplied to the nuclear industry for ultimate use in facilities or activities that are licensed by the NRC.

Miscellaneous Amendments—Part 9

On December 10, 1985 (50 FR 50283), the NRC published an amendment to its regulations pertaining to public records. The amendment, effective immediately, indicates that Executive Order 12356 governs classified documents and removes references to a defunct NRC Committee.

Specific Exemptions; Clarification of Standard—Part 50

On December 12, 1985 (50 FR 50764), the NRC published an amendment to its regulations for the licensing of production and utilization facilities. The amendment, effective January 13, 1986, clarifies the standards applied by the NRC when it considers whether to grant an exemption from the requirements set out in 10 CFR Part 50.

Limiting the Use of Highly Enriched Uranium in Domestically Licensed Research and Test Reactors—Part 50

On February 25, 1986 (51 FR 6514), the NRC published an amendment to its regulations that limits the use of highly enriched uranium (HEU) fuel in domestically licensed research and test reactors (non-power reactors). The amendment, effective March 27, 1986, generally requires that newly licensed non-power reactors use low enriched uranium (LEU) fuel and, contingent on Federal funding for conversion costs, that licensees of existing non-power reactors replace HEU fuel with LEU fuel. The amendment is intended to reduce the risk of theft or diversion of HEU fuel used in non-power reactors.

Final Procedures and Standards on No Significant Hazards Considerations—Parts 2 and 50

On March 6, 1986 (51 FR 7744), the NRC published an amendment to its regulations providing procedures under which the agency normally would give notice of opportunity for a hearing on applications it receives to amend operating licenses for nuclear power reactors and testing facilities and prior notice and reasonable opportunity for comment on proposed determinations about whether these amendments involve no significant hazards considerations. The amendments, effective May 5, 1986, also specify criteria for dispensing with prior notice and opportunity for comment if emergency conditions exist, for shortening the comment period if exigent circumstances exist, and for furnishing procedures for consultation with the State in which the facility requesting the amendment is located.

Material Balance Reports of Source Material and Special Nuclear Material - Parts 40, 51, 74, and 150

On March 21, 1986 (51 FR 9763), the NRC published an amendment to its regulations concerning the submission of material balance inventory reports. The final rule, effective April 21, 1986, eliminates the requirement to submit a statement of material balance for U.S. origin source material for all licenses except those reporting under the US/IAEA Safeguards Agreement. The final rule also eliminates the requirement to report the composition of ending inventory for all except nuclear reactor licensees and licensees reporting under the Agreement.

Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures—Part 50

On April 11, 1986 (51 FR 12502), the NRC published an amendment to its regulations, effective May 12, 1986, modifying General Design Criterion 4 of Appendix A to Part 50. The amendment allows the use of leak-before-break technology for excluding from the design basis the dynamic effects of postulated ruptures in primary coolant loop piping in pressurized water reactors.

Licensing Requirements for the Export of Nuclear Equipment and Material - Part 110

On April 14, 1986 (51 FR 12598), the NRC published an amendment to its regulations, effective immediately, pertaining to the export of nuclear equipment and material. The amendment requires certain holders of export licenses to notify the Commission, in writing, at least 40 days prior to exporting Canadian-origin nuclear material or equipment. The amendment also expands the general license for byproduct material to cover the export of americium-241 contained in industrial process control equipment and updates the list of countries in the provisions setting out restricted destinations.

Criteria for Reopening Records in a Formal Licensing Proceeding—Part 2

On May 30, 1986 (51 FR 19535), the NRC published an amendment to its regulations, effective June 30, 1986, that codified and refined NRC case law criteria for reopening a closed evidentiary record in a formal licensing proceeding. The amendment will ensure that no uncertainty exists concerning requirements for a motion to reopen a record and will facilitate proper and timely consideration of motions to reopen by an adjudicatory board.

Industrial Radiography Radiation Surveys and Licensee's Performance Inspection Programs—Part 34

On June 16, 1986 (51 FR 21736), the NRC published an amendment to its regulations concerning industrial radiography. The amendment, effective July 16, 1986, requires industrial radiography licensees to perform an additional survey of a radiography device at the time the device is put into storage and to record the results of the time-of-storage survey in place of the previous requirement to record the last-use survey. The amendment also requires each applicant for a radiography license to describe the program it will use to evaluate the safety performance of each radiographer and radiographer's assistant.

Enforcement of Nondiscrimination on the Basis of Handicap in Federally Conducted Programs—Part 4

On June 23, 1986 (51 FR 22880), the NRC published an amendment to its regulations requiring that the NRC ensure nondiscrimination against qualified handicapped persons in programs or activities it conducts. The amendment, effective August 22, 1986, sets forth standards for what constitutes discrimination on the basis of handicap, provides a definition of handicapped person and qualified handicapped person, and establishes a complaint mechanism for resolving allegations of discrimination.

Disposal of High-Level Radioactive Wastes in Geologic Repositories: Amendments to Licensing Procedures—Parts 2 and 60

On July 30, 1986 (51 FR 27158), the NRC published an amendment to its regulations applicable to the disposal of high-level radioactive wastes in geologic repositories. The amendments, effective August 29, 1986, deal with the procedural aspects of site characterization and the participation of States and Indian Tribes.

Miscellaneous Amendments Concerning Physical Protection of Nuclear Power Plants - Parts 50 and 73

On August 4, 1986 (51 FR 27817), the NRC published an amendment to its regulations, effective September 3, 1986, that provides a more safety conscious safeguards system for the physical protection of nuclear power plants. The revised requirements include refined policy on vital area access controls, authority to suspend safeguards measures during safety emergencies, protection of certain security equipment, and key and lock controls.

Searches of Individuals at Power Reactor Facilities—Part 73

On August 4, 1986 (51 FR 27822), the NRC amended its requirements for entry searches at power reactor facilities, effective September 3, 1986. The amendment requires equipment searches of all individuals, except on-duty law enforcement officers, seeking access to protected areas and pat-down searches when detection equipment fails or cause for suspicion exists.

Licensing Requirements for the Export of Tritium—Part 110

On August 4, 1986 (51 FR 27825), the NRC published an amendment to its regulations concerning the export of tritium under a general license. The amendment, effective immediately, imposes more restrictive limits on the amount of dispersed tritium which can be contained in luminescent light sources and other items exported under the general license.

Radioactive Waste Below Regulatory Concern; Policy Statement—Part 2

On August 29, 1986 (51 FR 30839), the NRC published a policy statement, codified as Appendix B to Part 2, concerning the expeditious handling of petitions for rulemaking requesting that a specific waste stream be exempted from disposal in a low-level waste disposal facility. The policy statement, effective October 27, 1986, and the accompanying staff implementation plan describe the information petitioners should file and the decision criteria and administrative procedures the Commission will use to act on the petition in an expeditious manner.

Annual Fee for Power Reactor Operating Licenses and Conforming Amendment - Parts 51 and 171

On September 18, 1986 (51 FR 33224), the NRC published an addition to its regulations, effective October 20, 1986, that imposes an annual fee on power reactors with operating licenses. The annual fee will recover allowable NRC budgeted costs for providing regulatory services to power reactors with operating licenses and will not alter the existing fee schedule under 10 CFR Part 170.

REGULATIONS AND AMENDMENTS PROPOSED

Standards for Production Against Radiation—Parts 10, 20, 30, 31, 32, 34, 40, 50, 61, and 70

On December 20, 1985 (50 FR 51992), the NRC published a notice of proposed rulemaking that would revise its regulations that set out requirements for the protection of individuals exposed to ionizing radiation from routine activities licensed by the NRC. The proposed rule is intended to improve NRC radiation protection standards by reflecting developments in the principles that underlie radiation protection and advances in related sciences since the current regulations were promulgated nearly thirty years ago. Because of numerous typesetting errors, this document was republished in its entirety on January 9, 1986 (50 FR 1092).

Station Blackout—Part 50

On March 21, 1986 (51 FR 9829), the NRC published a notice of proposed rulemaking that would require that light-water-cooled nuclear power reactors be capable of withstanding a total loss of alternating current (AC) electric power for a specified period and maintaining reactor core cooling during that period.

Revision to Ex Parte and Separation of Function Rules Applicable to Formal Adjudicatory Proceedings—Parts 0 and 2

On March 26, 1986 (51 FR 10393), the NRC published a notice of proposed rulemaking that would amend its regulations dealing with ex parte communications and separation of adjudicatory and nonadjudicatory functions in formal agency proceedings. The proposed rule would update agency rules of practice and incorporate requirements imposed by the Government in the Sunshine Act.

Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste—Parts 2, 19, 20, 21, 51, 70, 72, 73, 75, and 150

On May 27, 1986 (51 FR 19106), the NRC published a notice of proposed rulemaking that would subject monitored retrievable storage facilities for spent nuclear fuel and high-level radioactive waste to NRC licensing requirements if Congress approves construction of such a facility. The proposed action is necessary for the NRC to have the regulations required by the Nuclear Waste Policy Act of 1982 in place in a timely manner.

Disposal of High-Level Radioactive Wastes in Geologic Repositories; Conforming Amendments—Part 60

On June 19, 1986 (51 FR 22288), the NRC published a notice of proposed rulemaking that would amend its regulations for the disposal of high-level radioactive wastes in geologic repositories. The proposed rule would conform existing NRC regulations to the environmental standards for management and disposal of high-level radioactive wastes issued by the Environmental Protection Agency by incorporating the substantive requirements of the environmental standards into NRC's regulations.

Bankruptcy Filing; Notification Requirements—Parts 30, 40, 50, 61, 70, and 72

On June 20, 1986 (51 FR 22531), the NRC published a notice of proposed rulemaking that would require a licensee to inform the NRC in the event that the licensee is involved in a bankruptcy proceeding. The proposed rule is needed because a licensee's severe financial condition could affect its ability to handle licensed radioactive material and because the NRC must be aware of the situation so that appropriate measures to protect public health and safety can be taken.

Annual Fee for Nuclear Power Reactors Operating Licenses or Applications and Major Materials Licenses and Conforming Amendment—Parts 51 and 171

On July 1, 1986 (51 FR 24078), the NRC published a notice of proposed rulemaking that would prescribe an annual fee for all persons who have applied for or hold a license to operate a nuclear power reactor and for major materials licensees. The annual fee would recover allowable NRC budgeted costs for reactor-related and certain materials-related regulatory services.

Rules of Practice for Domestic Licensing Proceedings—Procedural Changes in the Hearing Process—Part 2

On July 3, 1986 (51 FR 24365), the NRC published a notice of proposed rulemaking that would make procedural changes in the hearing process that are intended to improve nuclear power plant licensing. The proposed rule would address the admission of contentions, discovery against the NRC staff, use of cross-examination plans, timing of motions for summary disposition, and limitations on intervenors' filings of proposed findings of fact, conclusions of law, and appellate briefs.

Uranium Mill Tailings Regulations: Ground-Water Protection and Other Issues - Part 40

On July 8, 1986 (51 FR 24697), the NRC published a notice of proposed rulemaking that would amend its regulations governing the disposal of uranium mill tailings. The proposed amendments are intended to incorporate the ground-water protection regulations published by the Environmental Protection Agency into existing NRC regulations. This action is taken in order to comply with the statutory requirement that NRC conform its regulations to EPA standards.

Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures—Part 50

On July 23, 1986 (51 FR 26393), the NRC published a notice of proposed rulemaking that would expand the scope of a previous amendment concerning the protection of structures, systems, and components important to safety against dynamic effects from postulated pipe ruptures. The proposed rule, which would cover all high energy piping in all nuclear power plants, is based on the results of recent research and insights from probabilistic risks analyses.

ADVANCE NOTICES OF PROPOSED RULEMAKING

Degree Requirements for Senior Operators at Nuclear Power Plants—Parts 50 and 55

On May 30, 1986 (51 FR 19561), the NRC published an advance notice of proposed rulemaking to request public comment on a contemplated amendment to its regulations that would require an applicant for a license as a Senior Operator of a nuclear power plant to hold a baccalaureate degree in engineering or the physical sciences from an accredited institution.

Seismic and Geologic Siting Criteria for Nuclear Power Plants; Reassessment of Current Criteria—Part 100

On August 25, 1986 (51 FR 30224), the NRC published a notice to request public comment and announce a public meeting on the initiation of its reassessment of the current seismic and geologic siting criteria for nuclear power plants.

Appendix 5

Regulatory Guides—Fiscal Year 1986

NRC regulatory guides describe methods for implementing specific parts of the Commission's regulations and, in some cases, describe techniques used by the staff in evaluating specific problems or postulated accidents. Guides also may advise applicants regarding information the NRC staff needs in reviewing applications for permits and licenses.

Comments on the guides are encouraged, and the guides are revised whenever appropriate to reflect new information or experience. NRC issues the guides for public comment in draft form before they have received complete staff review and an official staff position has been established.

Once issued, regulatory guides may be withdrawn when superseded by Commission regulations, when equivalent recommendations have been incorporated in applicable approved codes and standards, or when changes make them obsolete.

When guides are issued, revised, or withdrawn, notices are placed in the *Federal Register*.

To reduce the burden on the taxpayer, the NRC has made arrangements for the sale of active regulatory guides by both the U.S. Government Printing Office (on an individual guide basis) and the National Technical Information Service (on a standing order basis). Draft guides issued for public comment receive free distribution. NRC licensees receive, at no cost, pertinent draft and active regulatory guides as they are issued.

The following guides were issued, revised, or withdrawn during the period October 1, 1985, to September 30, 1986.

Division 1—Power Reactor Guides

1.82 Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident (Revision 1)

1.84 Design and Fabrication Code Case Acceptability—ASME Section III, Division 1 (Revision 24)

1.85 Materials Code Case Acceptability—ASME Section III, Division 1 (Revision 24)

1.105 Instrument Setpoints for Safety-Related Systems (Revision 2)

1.147 Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1 (Revision 5)

1.152 Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants

1.153 Criteria for Power, Instrumentation, and Control Portions of Safety Systems

Division 2—Research and Test Reactor Guides

NONE

Division 3—Fuels and Materials Facilities Guides

3.4 Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities (Revision 2)

3.41 WITHDRAWN. Validation of Calculational Methods for Nuclear Criticality Safety

3.56 General Guidance for Designing, Testing, Operating, and Maintaining Emission Control Devices at Uranium Mills

Division 4—Environmental and Siting Guides

4.16 Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants (Revision 1)

4.16 Monitoring and Reporting Radioactivity in Releases of Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Processing and Fabrication Plants and Uranium Hexafluoride Production Plants (Revision 1 Errata)

Division 5—Materials and Plant Protection Guides

5.65 Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls

Division 6—Product Guides

NONE

Division 7—Transportation Guides

7.10 Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material (Revision 1)

Division 8—Occupational Health Guides

NONE

Division 9—Antitrust and Financial Review Guides

NONE

Division 10—General Guides

NONE

Draft Guides

Division 1

EE 405-4 Proposed Revision 3 to Regulatory Guide 1.63, Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants

ES 926-4 Second Proposed Revision 1 to Regulatory Guide 1.23, Meteorological Measurement Program for Nuclear Power Plants

ME 305-4 Proposed Revision 2 to Regulatory Guide 1.99, Radiation Damage to Reactor Vessel Materials

SI 501-4 Station Blackout

SI 502-4 Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors

Division 3

CE 301-4 Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks

CE 304-4 Standard Format and Content of Decommissioning Plans for 10 CFR Parts 30, 40, and 70 Licensees

CE 306-4 Standard Format and Content for a Topical Safety Analysis Report for a Dry Spent Fuel Storage Cask

CE 402-4 Proposed Revision 1 to Regulatory Guide 3.50, Guidance on Preparing a License Application To Store Spent Fuel and High-Level Radioactive Waste

CE 410-4 Design of an Independent Spent Fuel Storage Installation (Dry Storage)

CE 501-4 Administrative Practices for Nuclear Criticality Safety at Fuels and Materials Facilities

WM 407-4 Methods for Estimating Radioactive and Toxic Airborne Source Terms for Uranium Milling Operations

Division 5

SG 901-4 Proposed Revision 1 to Regulatory Guide 5.62, Reporting of Physical Security Events

Division 7

FC 416-4 Proposed Revision 2 to Regulatory Guide 7.9, Standard Format and Content of Part 71 Applications for Approval of Packaging for Radioactive Material

Division 7 (Continued)

MS 501-4 Fracture Toughness Criteria for Ferritic Steel Shipping Containers with a Wall Thickness Greater than Four Inches (0.1 m)

Division 10

FC 414-4 Guide for the Preparation of Applications for Licenses for Medical Teletherapy Programs

FC 601-4 Guide for the Preparation of Applications for Radiation Safety Evaluation and Registration of Devices Containing Byproduct Material

Appendix 6

Nuclear Electric Generating Units in Operation Or Under Construction

(As of December 31, 1986)

The following listing includes nuclear power reactor electrical generating units which were in operation or under construction or under construction in the United States as of December 31, 1985, representing a total capacity of approximately 115,000 MWe. Reactor types are indicated as follows: BWR—boiling water reactor, PWR—pressurized water reactor, HTGR—high temperature gas-cooled reactor. Plant status is indicated as follows: OL—has operating license, CP—has construction permit. The dates for operation are either actual or as scheduled by the utilities as of December 31, 1986.

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
ALABAMA						
Decatur	Browns Ferry Nuclear Power Plant Unit 1	1,065	BWR	OL 1973	Tennessee Valley Authority	1974
Decatur	Browns Ferry Nuclear Power Plant Unit 2	1,065	BWR	OL 1974	Tennessee Valley Authority	1975
Decatur	Browns Ferry Nuclear Power Plant Unit 3	1,065	BWR	OL 1976	Tennessee Valley Authority	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 1	804	BWR	OL 1977	Alabama Power Co.	1977
Dothan	Joseph M. Farley Nuclear Plant Unit 2	814	PWR	OL 1981	Alabama Power Co.	1981
Scottsboro	Bellefonte Nuclear Plant	1,235	PWR	CP 1974	Tennessee Valley Authority	1993
Scottsboro	Bellefonte Nuclear Plant Unit 2	1,235	PWR	CP 1974	Tennessee Valley Authority	1995
ARIZONA						
Wintersburg	Palo Verde Nuclear Generating Station Unit 1	1,304	PWR	OL 1984	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 2	1,304	PWR	OL 1985	Arizona Public Service Co.	1986
Wintersburg	Palo Verde Nuclear Generating Station Unit 3	1,304	PWR	CP 1976	Arizona Public Service Co.	1987
ARKANSAS						
Russelville	Arkansas Nuclear One Unit 1	836	PWR	OL 1974	Arkansas Power & Light Co.	1974
Russelville	Arkansas Nuclear One Unit 2	858	PWR	OL 1978	Arkansas Power & Light Co.	1980

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
CALIFORNIA						
San Clemente	San Onofre Nuclear Generating Station Unit 1	436	PWR	OL 1967	So. Calif. Ed. & San Diego Gas & Electric Co.	1968
San Clemente	San Onofre Nuclear Generating Station Unit 2	1,100	PWR	OL 1982	So. Calif. Ed. & San Diego Gas & Electric Co.	1983
San Clemente	San Onofre Nuclear Generating Station Unit 3	1,100	PWR	OL 1983	So. Calif. Ed. & San Diego Gas & Electric Co.	1984
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 1	1,084	PWR	OL 1984	Pacific Gas & Electric Co.	1985
Diablo Canyon	Diablo Canyon Nuclear Power Plant Unit 2	1,106	PWR	OL 1985	Pacific Gas & Electric Co.	1986
Clay Station	Rancho Seco Nuclear Generating Station Unit 1	873	PWR	OL 1974	Sacramento Municipal Utility District	1975
COLORADO						
Platteville	Fort St. Vrain Nuclear Generating Station	330	HTGR	OL 1973	Public Service Co. of Colorado	1979
CONNECTICUT						
Haddam Neck	Haddam Neck Generating Station	555	PWR	OL 1967	Conn. Yankee Atomic Power Co.	1968
Waterford	Millstone Nuclear Power Station Unit 1	654	BWR	OL 1970	Northeast Nuclear Energy Co.	1971
Waterford	Millstone Nuclear Power Station Unit 2	864	PWR	OL 1975	Northeast Nuclear Energy Co.	1975
Waterford	Millstone Nuclear Power Station Unit 3	1,156	PWR	OL 1985	Northeast Nuclear Energy Co.	1986
FLORIDA						
Florida City	Turkey Point Station Unit 3	646	PWR	OL 1972	Florida Power & Light Co.	1972
Florida City	Turkey Point Station Unit 4	646	PWR	OL 1973	Florida Power & Light Co.	1973
Red Level	Crystal River Plant Unit 3	806	PWR	OL 1977	Florida Power Corp.	1977
Ft. Pierce	St. Lucie Plant Unit 1	817	PWR	OL 1976	Florida Power & Light Co.	1976
Ft. Pierce	St. Lucie Plant Unit 2	842	PWR	OL 1983	Florida Power & Light Co.	1983
GEORGIA						
Baxley	Edwin I. Hatch Plant Unit 1	757	BWR	OL 1974	Georgia Power Co.	1975
Baxley	Edwin I. Hatch Plant Unit 2	771	BWR	OL 1978	Georgia Power Co.	1979

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
GEORGIA—(continued)						
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 1	1,100	PWR	CP 1974	Georgia Power Co.	1987
Waynesboro	Alvin W. Vogtle, Jr. Plant Unit 2	1,100	PWR	CP 1974	Georgia Power Co.	1988
ILLINOIS						
Morris	Dresden Nuclear Power Station Unit 2	772	BWR	OL 1969	Commonwealth Edison Co.	1970
Morris	Dresden Nuclear Power Station Unit 3	773	BWR	OL 1971	Commonwealth Edison Co.	1971
Zion	Zion Nuclear Plant Unit 1	1,040	PWR	OL 1973	Commonwealth Edison Co.	1973
Zion	Zion Nuclear Plant Unit 2	1,040	PWR	OL 1973	Commonwealth Edison Co.	1974
Cordova	Quad-Cities Station Unit 1	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Cordova	Quad-Cities Station Unit 2	769	BWR	OL 1972	Comm. Ed. Co.-Iowa-Ill Gas & Elec. Co.	1973
Seneca	LaSalle County Nuclear Station Unit 1	1,078	BWR	OL 1982	Commonwealth Edison Co.	1984
Seneca	LaSalle County Nuclear Station Unit 2	1,078	BWR	OL 1983	Commonwealth Edison Co.	1984
Bryon	Byron Station Unit 1	1,120	PWR	OL 1984	Commonwealth Edison Co.	1985
Byron	Byron Station Unit 2	1,120	PWR	OL 1986	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 1	1,120	PWR	OL 1986	Commonwealth Edison Co.	1987
Braidwood	Braidwood Unit 2	1,120	PWR	CP 1975	Commonwealth Edison Co.	1988
Clinton	Clinton Nuclear Power Plant Unit 1	950	BWR	OL 1986	Illinois Power Co.	1986
IOWA						
Pala	Duane Arnold Energy Center Unit 1	515	BWR	OL 1974	Iowa Elec. Power & Light Co.	1975
KANSAS						
Burlington	Wolf Creek	1,150	PWR	OL 1985	Kansas Gas & Elec. Co.	1985

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
LOUISIANA						
Taft	Waterford Steam Electric Station	1,151	PWR	OL 1984	Louisiana Power & Light Co.	1985
St. Francisville	River Bend Station Unit 1	934	BWR	OL 1985	Gulf States Utilities Co.	1986
MAINE						
Wiscasset	Maine Yankee Atomic Power	810	PWR	OL 1972	Maine Yankee Atomic Power Co.	1972
MARYLAND						
Lusby	Calvert Cliffs Nuclear Power Plant Unit 1	825	PWR	OL 1974	Baltimore Gas & Elec. Co.	1975
Lusby	Calvert Cliffs Nuclear Power Plant Unit 2	825	PWR	OL 1976	Baltimore Gas & Elec. Co.	1977
MASSACHUSETTS						
Rowe	Yankee Nuclear Power Station	175	PWR	OL 1960	Yankee Atomic Elec. Co.	1961
Plymouth	Pilgrim Station Unit 1	670	BWR	OL 1972	Boston Edison Co.	1972
MICHIGAN						
Big Rock Point	Big Rock Point Nuclear Plant	64	BWR	OL 1962	Consumers Power Co.	1963
South Haven	Palisades Nuclear Power Station	635	PWR	OL 1971	Consumers Power Co.	1971
Laguna Beach	Enrico Fermi Atomic Power Plant Unit 2	1,093	BWR	OL 1985	Detroit Edison Co.	1986
Bridgman	Donald C. Cook Plant Unit 1	1,044	PWR	OL 1974	Indiana & Michigan Elec. Co.	1975
Bridgman	Donald C. Cook Plant Unit 2	1,082	PWR	OL 1977	Indiana & Michigan Elec. Co.	1978
MINNESOTA						
Monticello	Monticello Nuclear Generating Plant	525	BWR	OL 1970	Northern States Power Co.	1971
Red Wing	Prairie Island Nuclear Generating Plant Unit 1	503	PWR	OL 1973	Northern States Power Co.	1973
Red Wing	Prairie Island Nuclear Generating Plant Unit 2	500	PWR	OL 1974	Northern States Power Co.	1974
MISSISSIPPI						
Port Gibson	Grand Gulf Nuclear Station Unit 1	1,250	BWR	OL 1982	Mississippi Power & Light Co.	1985
Port Gibson	Grand Gulf Nuclear Station Unit 2	1,250	BWR	CP 1974	Mississippi Power & Light Co.	Indef.
MISSOURI						
Fulton	Callaway Plant Unit 1	1,188	PWR	OL 1984	Union Electric Co.	1985

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
NEBRASKA						
Fort Calhoun	Fort Calhoun Station Unit 1	478	PWR	OL 1973	Omaha Public Power District	1973
Brownville	Cooper Nuclear Station	764	BWR	OL 1974	Nebraska Public Power District	1974
NEW HAMPSHIRE						
Seabrook	Seabrook Nuclear Station Unit 1	1,198	PWR	OL 1986	Public Service of N.H.	Indef.
Seabrook	Seabrook Nuclear Station Unit 2	1,198	PWR	CP 1976	Public Service of N.H.	Indef.
NEW JERSEY						
Toms River	Oyster Creek Nuclear Power Plant Unit 1	620	BWR	OL 1969	GPU Nuclear Corp.	1969
Salem	Salem Nuclear Generating Station Unit 1	1,079	PWR	OL 1976	Public Service Elec. & Gas Co.	1977
Salem	Salem Nuclear Generating Station Unit 2	1,106	PWR	OL 1980	Public Service Elec. & Gas Co.	1981
Salem	Hope Creek Generating Station Unit 1	1,067	BWR	OL 1986	Public Service Elec. & Gas Co.	1986
NEW YORK						
Indian Point	Indian Point Station Unit 2	864	PWR	OL 1973	Consolidated Edison Co.	1974
Indian Point	Indian Point Station Unit 3	891	PWR	OL 1975	Power Authority of the State of New York	1976
Scriba	Nine Mile Point Nuclear Unit 1	610	BWR	OL 1969	Niagara Mohawk Power Co.	1969
Scriba	Nine Mile Point Nuclear Unit 2	1,080	BWR	OL 1986	Niagara Mohawk Power Co.	1986
Ontario	R. E. Ginna Nuclear Power Plant Unit 1	470	PWR	OL 1969	Rochester Gas & Elec. Co.	1970
Brookhaven	Shoreham Nuclear Power Station	820	BWR	OL 1984	Long Island Lighting Co.	Indef.
Scriba	James A. FitzPatrick Nuclear Power Plant	810	BWR	OL 1974	Power Authority of the State of New York	1975
NORTH CAROLINA						
Southport	Brunswick Steam Electric Plant Unit 2	790	BWR	OL 1974	Carolina Power & Light Co.	1975
Southport	Brunswick Steam Electric Plant Unit 1	790	BWR	OL 1976	Carolina Power & Light Co.	1977
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 1	1,180	PWR	OL 1981	Duke Power Co.	1981
Cowans Ford Dam	Wm. B. McGuire Nuclear Station Unit 2	1,180	PWR	OL 1983	Duke Power Co.	1984
Bonsal	Shearon Harris Plant Unit 1	915	PWR	OL 1986	Carolina Power & Light Co.	1987

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
OHIO						
Oak Harbor	Davis-Besse Nuclear Power Station Unit 1	874	PWR	OL 1977	Toledo Edison-Cleveland Electric Illum. Co.	1977
Perry	Perry Nuclear Power Plant Unit 1	1,205	BWR	OL 1986	Toledo Edison-Cleveland Elec. Illum. Co.	1987
Perry	Perry Nuclear Power Plant Unit 2	1,205	BWR	CP 1977	Toledo Edison-Cleveland Elec. Illum. Co.	Indef.
OREGON						
Prescott	Trojan Nuclear Plant Unit 1	1,080	PWR	OL 1975	Portland General Elec. Co.	1976
PENNSYLVANIA						
Peach Bottom	Peach Bottom Atomic Power Station Unit 2	1,051	BWR	OL 1973	Philadelphia Elec. Co.	1974
Peach Bottom	Peach Bottom Atomic Power Station Unit 3	1,035	BWR	OL 1974	Philadelphia Elec. Co.	1974
Pottstown	Limerick Generating Station Unit 1	1,065	BWR	OL 1984	Philadelphia Elec. Co.	1986
Pottstown	Limerick Generating Station Unit 2	1,065	BWR	CP 1974	Philadelphia Elec. Co.	1990
Shippingport	Beaver Valley Power Station Unit 1	810	PWR	OL 1976	Duquesne Light Co. Ohio Edison Co.	1976
Shippingport	Beaver Valley Power Station Unit 2	852	PWR	CP 1974	Duquesne Light Co. Ohio Edison Co.	1987
Goldsboro	Three Mile Island Nuclear Station, Unit 1	776	PWR	OL 1974	GPU Nuclear Corp.	1974
Berwick	Susquehanna Steam Electric Station Unit 1	1,052	BWR	OL 1982	Pennsylvania Power & Light Co.	1983
Berwick	Susquehanna Steam Electric Station Unit 2	1,052	BWR	OL 1984	Pennsylvania Power & Light Co.	1985
SOUTH CAROLINA						
Hartsville	H. B. Robinson S.E. Plant Unit 2	665	PWR	OL 1970	Carolina Power & Light Co.	1971
Seneca	Oconee Nuclear Station Unit 1	860	PWR	OL 1973	Duke Power Co.	1973
Seneca	Oconee Nuclear Station Unit 2	860	PWR	OL 1973	Duke Power Co.	1974

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
SOUTH CAROLINA—*(continued)*						
Seneca	Oconee Nuclear Station Unit 3	860	PWR	OL 1974	Duke Power Co.	1974
Broad River	Virgil C. Summer Nuclear Station Unit 1	900	PWR	OL 1982	So. Carolina Elec. & Gas Co.	1984
Lake Wylie	Catawba Nuclear Station Unit 1	1,145	PWR	OL 1984	Duke Power Co.	1985
Lake Wylie	Catawba Nuclear Station Unit 2	1,145	PWR	OL 1986	Duke Power Co.	1986
TENNESSEE						
Daisy	Sequoyah Nuclear Power Plant Unit 1	1,128	PWR	OL 1980	Tennessee Valley Authority	1981
Daisy	Sequoyah Nuclear Power Plant Unit 2	1,148	PWR	OL 1981	Tennessee Valley Authority	1982
Spring City	Watts Bar Nuclear Plant Unit 1	1,165	PWR	CP 1973	Tennessee Valley Authority	1986
Spring City	Watts Bar Nuclear Plant Unit 2	1,165	PWR	CP 1973	Tennessee Valley Authority	1987
TEXAS						
Glen Rose	Comanche Peak Steam Electric Station Unit 1	1,150	PWR	CP 1974	Texas Utilities	1987
Glen Rose	Comanche Peak Steam Electric Station Unit 2	1,150	PWR	CP 1974	Texas Utilities	1988
Bay City	South Texas Nuclear Project Unit 1	1,250	PWR	CP 1975	Houston Lighting & Power Co.	1987
Bay City	South Texas Nuclear Project Unit 2	1,250	PWR	P 1975	Houston Lighting & Power Co.	1989
VERMONT						
Vernon	Vermont Yankee Generating Station	504	BWR	OL 1972	Vermont Yankee Nuclear Power Corp.	1972
VIRGINIA						
Gravel Neck	Surry Power Station Unit 1	775	PWR	OL 1972	Va. Electric & Power Co.	1972
Gravel Neck	Surry Power Station Unit 2	775	PWR	OL 1973	Va. Electric & Power Co.	1973
Mineral	North Anna Power Station Unit 1	865	PWR	OL 1976	Va. Electric & Power Co.	1978
Mineral	North Anna Power Station Unit 2	890	PWR	OL 1980	Va. Electric & Power Co.	1980

Site	Plant	Capacity (Net MWe)	Type	Status	Utility	Commercial Operation
WASHINGTON						
Richland	WPPSS No. 1 (Hanford) Supply System	1,266	PWR	CP 1975	Wash. Public Power	Indef.
Richland	WPPSS No. 2 (Hanford) Supply System	1,103	BWR	OL 1983	Wash. Public Power	1984
Satsop	WPPSS No. 3 Supply System	1,242	PWR	CP 1978	Wash. Public Power	Indef.
WISCONSIN						
LaCrosse	LaCrosse (Genoa) Nuclear Generating Station	48	BWR	OL 1967	Dairyland Power Corp.	1969
Two Creeks	Point Beach Nuclear Plant Unit 1	495	PWR	OL 1970	Wisconsin Electric Power Co.	1970
Two Creeks	Point Beach Nuclear Plant Unit 2	495	PWR	OL 1971	Wisconsin Electric Power Co.	1972
Kewaunee	Kewaunee Nuclear Power Plant	515	PWR	OL 1973	Wisconsin Public Svc. Corp.	1974

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