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ABSTRACT

This report, the tenth in a series of annual reports, was prepared in response to congressional inquiries concerning how nuclear regulatory research is used. It summarizes the accomplishments of the Office of Nuclear Regulatory Research during FY 1994.

The goal of the Office of Nuclear Regulatory Research (RES) is to ensure the availability of sound technical bases for timely rulemaking and related decisions in support of NRC regulatory/licensing/inspection activities. RES also has responsibilities related to the resolution of generic safety issues and to the review of licensee submittals regarding individual plant examinations. It is the responsibility of RES to conduct the NRC's rulemaking process, including the issuance of regulatory guides and rules that govern NRC licensed activities.

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- Based on public comments, modifications were made to the draft regulatory guide for evaluating pressure vessels whose beltline materials have Charpy upper-shelf energy (the indication of reactor vessel toughness) that has fallen below the 50 ft-lb regulatory limit prescribed in Appendix G to 10 CFR Part 50. Generic analyses using the draft regulatory guide methodology demonstrated that adequate pressure vessel integrity exists for pressure vessels with Charpy upper-shelf energy values well below the 50 ft-lb value specified in Appendix G. It is expected that the final guide will be published in FY 1995.

- Based on public comments, modifications were made to the draft regulatory guide for calculational and dosimetry methods for determining pressure vessel neutron fluence for power reactors. In support of this draft guide, new neutron cross-section libraries that apply the latest evaluated nuclear data files were published.

- Interim fatigue design curves were revised to more accurately describe fatigue life of primary pressure boundary components exposed to the high-temperature coolant in light-water reactor (LWR) systems. Fatigue tests are in progress to validate and/or update the proposed design curves.

- A Synthetic Aperture Focusing Technique for Ultrasonic Testing (SAFT-UT) system, which provides a method for more reliable detection and sizing of flaws, was fabricated for the NRC's nondestructive examination mobile laboratory. Specialized training was provided for the mobile laboratory personnel. The system was successfully used by the mobile laboratory personnel for the first time in 1994 in performing audit inspections of piping at an operating nuclear power plant.

- A collaborative research agreement was reached between the Japan Atomic Energy Research Institute (JAERI) and the Oak Ridge National Laboratory (ORNL) to exchange information gained from ORNL examination of material from the decommissioned Japan Power Demonstration reactor. The information exchange will result in a better understanding of changes in reactor pressure vessel material properties caused by long-term irradiation.

- An improved method for performing inservice inspections of steam generator tubing was successfully demonstrated at two operating nuclear power plants. The method, which employs multiple "pancake" type coils and multifrequency data analysis for better sensitivity, produced significantly higher signal levels with a greatly enhanced inspection speed over other currently produced improved techniques.

- A draft regulation and draft regulatory guide addressing the engineering and metallurgical aspects of thermal annealing for U.S. plants were developed and published for public comment. The proposed regulation provides the administrative and technical basis for performing thermal annealing of reactor pressure vessels in U.S. commercial nuclear power plants. The draft regulatory guide elaborates the information licensees should develop and provide as part of the thermal annealing plan development, including detailed information concerning the pressure vessel and other components that could be affected by the high-temperature annealing, measurements that are to be made before, during, and after the annealing, and the method for determining the post-anneal material properties for the pressure vessel beltline materials.

- Amendments to 10 CFR 50.61 and to Appendices G and H of 10 CFR Part 50 were proposed and published for public comment.

The proposed amendments would clarify the pressurized thermal shock requirements, the fracture toughness requirements, and the reactor vessel materials surveillance program requirements.

Electrical and Mechanical Components

- Research continued into the aging-related degradation of performance of safety-significant components and systems. The research also addresses methods for mitigating and managing the aging process in these components and systems. Draft reports were issued addressing the chemical and volume control systems for pressurized water reactors, containment cooling systems, reactor core isolation cooling systems, accumulators, air-operated valves, and isolation condenser systems.

- Results from work completed on the risk-based methodology for assessing aging effects in nuclear power plants have been used for identifying safety-related motor-operated valves (MOVs) having the most impact on plant risk. The valves being considered are those covered by Generic Letter 89–10, "Safety-Related Motor-Operated Valve Testing and Surveillance." Dynamic tests and surveillance tests, in accordance with GL 89–10, could then be performed on those MOVs with the largest risk impact. Relative risk importance of single MOVs and the interactions of MOVs with other components, including other MOVs, can be analyzed using this approach. A draft report documenting the results of this work was issued for review in FY 1994. The results have provided the technical basis for evaluating licensees' submittals for ranking their respective MOVs for tests in accordance with GL 89–10.

- Research continued into the factors affecting the performance of motor-operated valves (MOVs), specifically addressing whether corrosion in internal valve parts can significantly affect the torque and thrust required for MOVs to perform their functions when called upon to do so. Additionally, this information will be used in determining if the MOVs comply with GL 89–10. Friction experiments were conducted on samples of corroded materials typical of certain valves. Additional experiments are continuing to determine the effects of corrosion on friction and thrust.

Standard Reactor Designs

Systems Performance of Advanced Reactors

- Modification of the ROSA facility to validate the applicability of NRC codes to the AP–600 design and provide independent confirmation of the predicted performance of AP–600 plant safety systems was completed. Twelve of the 14 Phase 1 tests were successfully carried out. Results thus far suggest that this reactor can be successfully cooled under a variety of postulated accident scenarios. Two potential safety issues were identified during testing (i.e., a large thermal gradient occurred in piping where cold water from the passive heat removal systems enters and the possibility of a water hammer resulting from contact between subcooled water and steam) and are being carefully evaluated for their safety implications.

- Construction of an integral test facility is nearing completion at Purdue University to carry out tests on a broad spectrum of loss-of-coolant accidents and transients postulated for the General Electric simplified boiling water reactor. Testing is scheduled to begin in the summer of 1995 and continue into 1996.

- The program to demonstrate a method for estimating the reliability of passive safety systems in advanced reactor designs such as the Westinghouse AP–600 and the General Electric SBWR was continued. A peer review meeting to discuss the approach and its status was held. The demonstration project will be completed in late 1995.

Engineering Issues for Advanced Reactor Designs

- The Advanced Light-Water Reactor (ALWR) Equipment Qualification Panel has resolved all issues pertaining to the experience-based approach for seismic qualification of Group 1 items. Group 1 equipment has a mature design with little design variability and, in general, has demonstrated characteristics of inherent seismic ruggedness. The ALWR Equipment Qualification Panel met with vendors of batteries and transformers to discuss design variability and failure modes for ALWR designs. In addition, the panel toured a select group of the candidate facilities that had experienced earthquake damage and would be used in the EPRI/ARC data base.

Regulatory Applications of New Source Terms

- A revised accident source term document (NUREG-1465) reflecting severe accident research insights gained over the last 30 years was issued. The new source term provides more realistic estimates of fission product releases into containment in terms of timing, nuclide types, quantities, and chemical form than the current source term which dates from 1962.

- Public comments on a proposed revision to NRC's siting criteria (10 CFR Part 100) are being analyzed, and a final rule is being developed reflecting comments received. This rule will incorporate basic reactor siting criteria and the continued use of accident source terms and dose calculations for siting of certain custom plants.

Part 2--NUCLEAR SAFETY RESEARCH--REACTOR REGULATION SUPPORT

Plant Performance

- Changes in fuel pellets and cladding that appear to reduce the fuel's resistance to damage occur at high burnups. These changes affect fuel behavior computer codes that are used in licensing safety analyses and affect a number of fuel damage criteria that appear in regulations, regulatory guides, and the standard review plan. Research programs are now in place to update the NRC's fuel behavior codes for application at high burnups and to revise, as necessary, the licensing fuel damage criteria.

- Developmental assessment, peer review, and documentation of the revised RELAP5 code have been completed with NRC participation in the International Code Assessment and Maintenance Program continuing. The RELAP5 code is being used for the certification of the AP-600 and SBWR as well as to support licensing activities related to operating plants, e.g., pressurized thermal shock.

Human Reliability

- A method has been developed for assessing the effectiveness of training programs at nuclear power plants and efforts are continuing to establish a technical base for setting minimum staffing levels for both control rooms and operating support staff.

- A handbook detailing the effects on human performance of environmental factors such as light, heat, etc., has been published for use by NRC inspectors.

- The proceedings of the NRC/NIST workshop on digital system reliability have been issued and the first phase of a study by the National Academy of Sciences to define an effective approach to the regulation of computer-based (digital) technology in nuclear safety and control systems was initiated.

- Work continued on the project cooperatively sponsored with EPRI to develop guidelines for the verification, validation, and quality assurance for certification of high-integrity software for use in plant safety systems. Similar studies have been initiated to develop the bases for the environmental qualification of advanced instrumentation and control systems.

- Reliability and risk analysis tools have been developed to permit ready evaluation of the risk impact of changes to plant technical specifications.

- "Advanced Human-System Interface Design Review Guidelines" (NUREG/CR-5908), in support of the standard review plan, was issued setting forth guidance for staff consideration of proposals for control room designs or modifications.

Reactor Accident Analysis

Reactor Risk Analysis

- Results of an analysis of the risks of potential occurrences during low-power and shutdown operations have been completed. These analyses suggest that traditional technical specifications may not always be adequate to accommodate potential accidents occurring during such operations.

- An analysis of the South Texas nuclear project to support its request for modifications of plant technical specifications, based in part on risk, was completed. The analysis determined that the applicant's assessment of core damage frequency was within the range of estimates for other similar facilities, and this information has been used by NRR to support their regulatory decision.

- The cooperative program with the European Communities and the Organization for Economic Cooperation and Development to carry out an intercomparison exercise on six accident consequence codes, including the NRC-developed MACCS code, was successfully completed. The study indicated substantial agreement in the reactor accident consequence predictions made by these codes.

- Probabilistic risk assessment (PRA) data from four more licensed power plants were added to the SAPHIRE (Systems Analysis Programs for Hands-on Integrated Reliability Evaluation) data base to bring the total to 17. SAPHIRE is a set of codes used in performing PRAs and allows the staff to create, quantify, and evaluate accident risks.

Containment Performance

- The culmination of extensive experimental and analytical research on the issue of direct containment heating (DCH), principally for the Zion reactor, determined that the pressure loads produced by DCH are significantly lower than earlier estimates and present a negligible threat to reactor containment integrity. Peer-reviewed reports (NUREG/CR-6075 and Supplement 1 to NUREG/CR-6075) documenting these findings have been issued.

- In cooperation with the Japanese Ministry of International Trade and Industry, construction has been completed on the High-Temperature Hydrogen Combustion facility. This facility will study the modes of high temperature H_2 combustion (deflagration/detonation) to assess the possible threat of hydrogen detonations on containment integrity. Lower temperature testing to characterize H_2 combustion under conditions simulating accident environments (steam) suggest that mixtures initially nonflammable (because of steam concentrations) became flammable as steam was condensed and that glow plugs were successful in igniting the gas with no detonation or accelerated flame propagation observed.

- Under an agreement between the Commission of European Communities Joint Research Center (JRC) and the NRC in the field of severe accident research related to molten fuel-coolant interactions (FCIs), four successful tests have been performed to date in the large-scale (150-kg reactor prototypic melts) FARO test facility. These tests investigated non-explosive melt breakup and quenching, structure heating, and the limits of melt-coolant mixing at high pressure. Additional tests have been performed in the small-scale KROTOS facility (one-dimensional shock tube geometry) to investigate steam explosion phenomenology at low pressure.

- The OECD RASPLAV project is investigating melt-vessel interactions and providing data on internal natural convection flow and the local heat flux distribution inside the lower head of the reactor pressure vessel for various melt compositions. This project involves large-scale integral experiments using UO_2 in representative lower head reactor pressure vessel geometries (i.e., slice geometry), analytical studies, and a number of small-scale separate effects experiments. Recently, a small-scale experiment that showed the feasibility of using slice geometry for large-scale experiments was carried out.

Reactor Containment Structural Integrity

- In order to improve the state of practice in inspection of containments to reduce the chances of having significant undetected degradation due to corrosion, work continued in 1994 on the rulemaking to incorporate by reference Subsections IWE and IWL of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code into 10 CFR 50.55a. Subsection IWE provides rules for the inservice inspection of metal containments and the liners of concrete containments. Subsection IWL provides rules for the inservice inspection of the reinforced concrete and the post-tensioning systems of concrete containments. The proposed rule does not address inspection of inaccessible areas; the state of practice for inspection of inaccessible areas will have to be improved before a resolution of this issue is achieved.

Severe Accident Policy Implementation

- Seventy-six IPE (internal-event) submittals have been received to date, with one due in CY 1995. Staff evaluation reports (SERs) have been issued for 27 submittals. It is expected that all IPE submittals will be reviewed and SERs issued by the end of CY 1996.

- Sixteen complete and four partial IPEEE (external-event) submittals have been received of which four are being evaluated.

- Studies began of the IPE results to gain more generic insights. Issues such as the plant-to-plant variability in estimated core damage frequency results and the reasons for this variability are being studied.

Safety Issue Resolution and Regulation Improvements

Earth Sciences

- Archaeological and geological evidence identified during investigations in the New Madrid seismic zone, along with geological evidence from other studies, demonstrate that prehistoric earthquakes (at least two) large enough to cause liquefaction (magnitude 5.5 to 6.0 or larger) have occurred in the New Madrid seismic zone in the late Holocene. These findings contribute toward providing a basis for estimating the seismic hazard in the south central United States.

- The panel of experts assembled for the joint NRC/DOE/EPRI study of methods for probabilistic seismic hazard analysis (PSHA) has completed the development of new guidelines for performing PSHA. The panel has issued a draft report describing the guidelines, which emphasize methods of eliciting expert opinions. A peer review by the National Academy of Sciences/National Research Council is in progress.

Plant Response to Seismic and Other External Events

- The Commission approved the staff's recommendation to issue a second proposed revision of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," for public comment. This revision reflects new information and research results available since the first proposed revision of the regulations was issued and comments received from the public on that proposed revision of the regulations. Draft regulatory guides and standard review plan sections providing methods acceptable to the NRC staff for implementing the proposed regulations were issued for public comment in February 1995.

- On January 17, 1994, a magnitude 6.7 earthquake occurred in the San Fernando Valley near the town of Northridge, California. This is the same general area affected by the magnitude 6.5 San Fernando earthquake of 1971. Representatives from the NRC Offices of Nuclear Regulatory Research (RES) and Nuclear Reactor Regulation, and an RES contractor, Lawrence Livermore National Laboratory, toured the damaged area. In general, well-engineered structures and equipment that may have experienced ground motion far in excess of their design remained functional. However, extensive damage to several modern mid-rise and low-rise steel-moment frame buildings was observed. Components made of brittle materials, such as ceramic insulators and cast iron components, received significant damage consistent with that observed after other earthquakes.

Generic Safety Issue Resolution

- During FY 1994, the NRC identified no new generic issues, established priorities for three issues (see Table 6.1), and resolved five issues (see Table 6.2). Table 6.3 contains the schedules for resolution of all unresolved issues.

Reactor Regulatory Standards

- A final rule, 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses was issued in February 1994. The rule deletes the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test during the 6-year license term as a prerequisite for license renewal.

- An advance notice of proposed rulemaking, 10 CFR Part 52, concerning standard design certification for evolutionary LWRs was issued in November 1993. The contemplated rulemaking would define the form and content of the rules that would certify the design.

- A proposed rule, 10 CFR Parts 19 and 20, regarding the use of "controlled areas," the definition of occupational and public exposure, and training requirements was issued in February 1994.

- A proposed rule, 10 CFR Part 20, on frequency of medical examinations for use of respiratory protection equipment was issued in September 1994. The proposed amendment would remove the requirement for an annual medical examination and allow for alternative timeframes.

- A proposed rule, 10 CFR Part 21, on procurement of commercial grade items by reactor licensees was issued in October 1994. The proposed rule responds to a petition for rulemaking (PRM-21-02) submitted by the Nuclear Management and Resources Council (NUMARC), which is now incorporated into the Nuclear Energy Institute (NEI). The proposed amendment would clarify and add flexibility for procuring items for safety-related service.

- A proposed rule, 10 CFR Parts 50, 55, and 73, on reduction of reporting requirements imposed on NRC licensees was issued in November 1994. The proposed amendments would reduce reporting requirements on power reactors, research and test reactors, and nuclear material licensees.

Part 3—NUCLEAR MATERIALS LICENSING AND REGULATION SUPPORT

Nuclear Materials

- A proposed and a final rule, 10 CFR Part 40, Appendix A, on uranium mill tailings were issued in November 1993 and June 1994, respectively. The final rule conforms NRC regulations to Environmental Protection Agency (EPA) regulations under the Clean Air Act and supports recision of certain EPA Clean Air Act requirements.

- A final rule, 10 CFR Parts 30, 40, 50, 70, and 72, to allow self-guarantee as an additional mechanism for financial assurance for decommissioning was issued in December 1993. This rulemaking is in response to a petition for rulemaking (PRM-30-59) submitted by the General Electric Company and Westinghouse Electric Corporation. The final rule applies to certain financially strong, non-electric utility licensees and allows the use of self-guarantee as financial assurance for decommissioning funding. It does not apply to electric utility licensees.

- An advance notice of proposed rulemaking (ANPR), 10 CFR Part 20, on disposal of radioactive material by release in sanitary sewer systems was issued in February 1994. The ANPR requested comments on the appropriateness of current NRC regulations and solicited comments on possible alternative approaches.

- A proposed rule, 10 CFR Part 34, on the conduct of radiography using sealed sources was issued in February 1994. The proposed rule responds to a petition for rulemaking (PRM–34–04) submitted by the International Union of Operating Engineers, Local No. 2. The proposed rule represents a complete revision to this part of the Commission's regulations, including certification of radiographers and implementation of a two-person rule with radioactive sources.

- A proposed rule, 10 CFR 72.214, adding a standardized HUHOMS cask to the list of approved spent fuel storage casks was issued in June 1994. The rule will increase the number of NRC-certified spent fuel storage casks available under a general license.

- A proposed rule, 10 CFR 35.75, on criteria for release of patients administered radioactive material was issued in June 1994. The proposed rule addresses the requests of three petitions for rulemaking: PRM–20–20 from Dr. Carol S. Marcus and PRM–35–10/10a from the American College of Nuclear Medicine. The proposed amendment would specify a dose limit of 5 mSv (0.5 rem) rather than the limit of 30 mCi currently specified.

- A final rule, 10 CFR Part 73, on physical fitness programs for security personnel at Category I fuel cycle facilities was issued in July 1994. The amendment requires physical fitness training programs as well as annual performance testing for specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.

Uranium Enrichment

- A proposed and a final rule, 10 CFR Parts 19, 20, 21, 26, 51, 70, 71, 73, 74, 76, and 95, for certification of the operations of gaseous diffusion enrichment facilities were issued in February 1993 and September 1994, respectively. The rule covers both the certification process and the standards to be used to judge acceptable performance for certification of the operations of the gaseous diffusion enrichment facilities leased by the U.S. Enrichment Corporation from the Department of Energy.

Low-Level Waste Disposal

- A proposed rule, 10 CFR Parts 30, 40, 70, and 73, on clarification of decommissioning funding requirements was issued in June 1994. The amendments would clarify when decommissioning funding assurance was required and provide that assurance would be available after operations were terminated and decommissioning initiated.

- A final rule, 10 CFR Parts 2, 30, 40, 70, and 72, on timeliness in decommissioning of materials facilities was issued in July 1994. The rule establishes timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities.

- A proposed rule, 10 CFR Parts 20, 30, 40, 50, 51, 70, and 72, on radiological criteria for decommissioning was issued in August 1994. The proposed rule is based on comments received from seven workshops and a draft of the rulemaking published in February 1994.

Office Program

- The NRC supports the Small Business Innovation Research (SBIR) program to stimulate technological innovation by small businesses, strengthen the role of small business in meeting Federal research and development needs, increase the commercial application of NRC-supported research results, and improve the return on investment from Federally funded research for economic and social benefits to the nation.

Participation in this program has continued since the program was established in FY 1982. In FY 1994, the NRC was supporting 17 SBIR projects-in-progress.

PART 1--NUCLEAR SAFETY RESEARCH--REACTOR LICENSING SUPPORT

1. REACTOR AGING AND RENEWAL

This program is conducted to ensure that reactor plant systems and related components perform as designed during normal operation and transient and accident conditions and ensure that their functional integrity and operability can be maintained over the operating life of the plant. The program includes the reactor system pressure boundary. Failure to maintain pressure boundary integrity could compromise the ability to cool the reactor core and could lead to a loss-of-coolant accident accompanied by the release of hazardous fission products.

1.1 Reactor Vessel Safety and Piping Integrity

1.1.1 Statement of Problem

The reactor system pressure boundary of a light-water reactor (LWR) is the principal boundary enclosing the nuclear fuel core and the water coolant used to maintain suitably low temperatures of the fuel cladding and to conduct the heat from the fission reaction and convert the water coolant into steam for electricity generation. The primary system includes the reactor pressure vessel, primary coolant piping, primary pumps, and steam generators for pressurized water reactors (PWRs). For boiling water reactors (BWRs), the primary system includes the steam line piping out to the first isolation valve. This boundary must be kept intact and fully serviceable at all times to ensure that water coolant is always available to cover the fuel core so that the heat generated during power operation or from decay following shutdown can always be safely conducted away, thus precluding a core meltdown accident. The principles of ensuring the structural integrity of the primary system components are embodied in the elements of fracture mechanics used to predict conditions for failure. These elements are (1) knowledge of the material properties (strength, toughness, embrittlement, etc.), especially the changes in those properties that can occur as a consequence of nuclear operations; (2) knowledge of the pressure and other loadings that can be applied to the components either from normal operations or from accidents; and (3) knowledge of the presence and size of cracks or other flaws in the components. The regulations, codes, or guides that pertain to the structural integrity of LWRs were written to ensure that possible combinations of material properties, loads, and flaws will yield adequate margins against failure of primary system components. The goal of the reactor vessel safety and piping integrity element is to ensure that appropriate analytical procedures and inspection methods exist for assessing the safety of components during normal operation and transient and accident conditions and that sufficient critical experiments are conducted to validate those procedures and methods.

Ensuring the structural integrity of the pressure boundary has been at the center of several recent well-publicized regulatory issues—for example, the 1984 decision to require an accelerated schedule of five BWR inspections due to cracking in the coolant pipes; the 1991 review of the Yankee Rowe plant; and the 1992 review of the Trojan plant steam generators. Additionally, incidents of cracks and leaks in piping and steam generator tubes have highlighted the need for materials data, analysis methods, and inspection techniques for these components.

Much of the work is completed and has been put in practice through several regulations, regulatory guides, and parts of the standard review plan, as well as through national codes and standards. The remaining work is providing the basis for both confirming and revising some of the earlier regulatory positions, with the overall aim of providing a stable, fully validated regulatory framework for ensuring the integrity of the primary pressure boundary for the foreseeable future.

1.1.2 Program Strategy

The approach used for this element is to develop analytical procedures for predicting continuing integrity or conditions-for-failure and to ensure that an adequate experimental basis exists to validate those procedures. The most critical facet of pressure vessel integrity is embrittlement of the pressure vessel steel as a result of exposure to neutrons escaping from the fuel core during normal service. Experiments are conducted to develop a base of information on all the factors that will cause embrittlement to increase during service life. Much work is done to establish correlations between small-specimen behavior and thick-section behavior to ensure that the analyses performed to assess structural integrity are valid. Similarly, the ability to predict integrity in piping has required testing of full-sized sections of pipe having a variety of cracks to determine if such cracks could cause failure during either normal service or accident conditions. For both vessels and piping, knowledge of the rate at which cracks grow is very important to ensure that a component will not fail during its forthcoming operational period. Experiments are conducted on a wide variety of pertinent materials under a range of typical and expected exposure conditions to determine the maximum bounding rates of crack growth. Detection and sizing of flaws and cracks in all primary system components are conducted by the industry through periodic inservice inspections at shutdowns. To ensure that the inspections reliably detect and accurately size the flaws, extensive tests are conducted with inspection teams drawn from the industry using typical equipment and techniques on samples whose flaw conditions are known. From the results, it is possible to determine which techniques are effective and the magnitude of the error bands for flaw detection and sizing. Improvements in methods are proposed and qualification procedures developed that can provide better assurance of flaw detection in future inspections and for sizing flaws more accurately. Materials and components removed from actual service are used to measure material properties after years of service, to evaluate the extent of corrosion, and to validate the existence of flaws that have previously been identified and had their size estimated.

1.1.3 Research Accomplishments in FY 1994

1.1.3.1 Pressure Vessel Safety

This area of NRC research focuses on ensuring the structural integrity of the reactor system pressure boundary, i.e., keeping it free from damage and leaktight. Ensuring the structural integrity of the pressure boundary has been at the center of several recent well-publicized regulatory issues—for example, the 1984 decision to require an accelerated schedule of five BWR inspections because of cracking in the coolant pipes; the 1991 review of the Yankee Rowe (Mass.) plant; and the 1992 review of the Trojan (Ore.) plant steam generators. The underlying concern in ensuring the integrity of the pressure boundary is that failure to do so could compromise the operator's ability to cool the reactor core and possibly bring about a loss-of-coolant accident (LOCA) that could be accompanied by a release of hazardous fission products.

Research in this area is a broad-based program initiated in 1967. The original program was focused solely on the properties and fracture behavior of the reactor pressure vessel—the large, thick-walled steel cylinder that houses and supports the reactor core. As the full challenge of ensuring the integrity of this critical component was realized, the scope of the research program was expanded to include irradiation damage, service-induced cracking mechanisms, and methods for periodically inspecting the pressure vessel. Incidents of cracking and leaking in pipes and steam generator tubes have accentuated the need for materials data, analysis methods, and inspection techniques relevant to these components.

The research program on pressure vessel safety has expanded to meet these additional challenges. However, much of the work is completed and has been put in practice through several regulations, regulatory guides, and parts of the standard review plan, as well as through national codes and standards. The remaining work is providing the bases for both confirming and revising some of the earlier regulatory positions, with the overall aim of providing a stable, fully validated regulatory environment for ensuring the integrity of the primary pressure boundary for the foreseeable future. The technical efforts in the research program—fracture evaluation and radiation embrittlement—are central to sound regulatory positions addressing the safe operation of the pressure vessel. For example, efforts to revise the basis for determining the allowable operating pressure and temperature to preclude brittle failure of the pressure vessel drew on research results from the pressure vessel safety program.

1. *Fracture Evaluation.* Addressing fracture analysis methods assumed a particularly large role in the overall program during FY 1994. Fracture analysis work involves an ongoing program to develop and reduce to practice advanced analysis

methods that will improve the ability to predict the allowable pressures and temperatures for the pressure vessel and the ability to evaluate the integrity of the pressure vessels under design basis and hypothetical accident conditions. Basic work is being performed by researchers at Oak Ridge National Laboratory (ORNL), augmented by research being performed at Brown University, the University of Illinois, Texas A&M University, and the U.S. Navy's Naval Surface Warfare Center (NSWC). The researchers are developing state-of-the-art analysis methods and evaluating them against test data developed at ORNL, the National Institute for Standards and Technology (NIST), and the NSWC. The work performed during FY 1994 has been very promising, and the programs have been vigorously pursued to permit evaluation of test geometries and loadings that are more typical of reactor pressure vessels in the ductile-to-brittle transition region of the material's fracture toughness versus operating temperature behavior in the beltline region of the aged reactor vessels. The researchers are also coordinating their work with international efforts through a cooperative project on fracture analysis of large-scale experiments, under the auspices of the Committee on Safety of Nuclear Installations. Collaborative efforts with another European Community program are well under way and are expected to provide results from a large-scale test that will closely simulate a reactor pressure vessel subjected to accident loads. This will provide a more realistic validation of the revised analysis methods.

During FY 1994, the results of several efforts were put to use in performing generic analyses of reactor pressure vessels fabricated from materials with a low resistance to a "ductile tearing" failure mode. In the early 1970s, the NRC recognized that some pressure vessels were fabricated using steel plates and some weld types that did not provide the high resistance to this failure mode exhibited by most of the plates, forgings, and welds used in reactor pressure vessels. The NRC issued Appendix G to 10 CFR Part 50 in 1973 to provide explicit requirements on the Charpy upper-shelf energy—a measure of the ductile tearing resistance of these materials—for both new construction and for operating plants. The American Society of Mechanical Engineers (ASME) published a Code Case N-512 (Section XI, Division 1, February 1993) on this issue, but it did not address complete details of all the potential loading conditions for reactor pressure vessels, nor did it include guidance on determining appropriate material properties for use in the evaluation method.

The RES staff published a draft regulatory guide at the end of September 1993 expanding the ASME Code's guidance to include evaluation methods pertinent to all service loading conditions, guidance on selection of transients for consideration at various service load levels, and specific guidance on estimating material properties. During FY 1994, public comments on the draft guide were received, analyzed, and applied to revising the draft guide for final reviews for publication during FY 1995.

The RES staff worked with researchers at ORNL to initiate development of technical bases in probabilistic fracture mechanics for revising Regulatory Guide 1.154 on plant-specific evaluation of pressurized thermal shock in PWR pressure vessels. Additional research is being coordinated with efforts of the staff in thermal-hydraulics and probabilistic risk assessment to revise Regulatory Guide 1.154, in accordance with the SECY-92-283 document on the lessons learned from the Yankee-Rowe reactor pressure vessel integrity evaluation. It is planned that the development of technical bases will be completed in FY 1995, and draft revisions to the guide will be completed and published in 1996.

2. *Radiation Embrittlement.* Of special concern in ensuring the integrity of the reactor pressure vessel is the embrittlement of the pressure vessel steel caused by neutrons escaping from the reactor core during normal operation. These neutrons impinge on the pressure vessel wall and, through a complex process, reduce the ability of the steel to resist fracture. The embrittlement increases with continued operation. To ensure the continued safe operation of pressure vessels, the research program includes a significant effort to quantify the effects of neutron radiation embrittlement, to understand the mechanisms that control this process, and to find methods to mitigate the embrittlement and restore the original fracture toughness.

During FY 1994, the radiation embrittlement research efforts moved forward on several fronts. Test reactor irradiations were completed by

ORNL, using the University of Michigan test reactor, to evaluate the effects of neutron radiation on weld materials removed from the canceled Midland Unit 1 (Mich.) reactor pressure vessel. The materials are representative of the so-called "limiting" material in several operating nuclear power plants. The materials are also being irradiated in the surveillance programs of an operating power plant as part of an NRC-industry coordinated research effort. When the results from each of these programs are available in the late 1990s, they will provide important information about the embrittlement trends for these materials and equally important information about the differences between test reactor and power reactor irradiation conditions, as well as about the mechanisms controlling embrittlement of these materials. Radiation embrittlement research also includes a study of the effects of thermal annealing and reirradiation on fracture toughness properties. Annealing tests of irradiated test specimens of typical reactor vessel steels have demonstrated the degree of recovery of fracture properties using conditions generally proposed for annealing. Studies on the effects of reirradiation are in progress.

During FY 1994, a collaborative research agreement was reached between the Japan Atomic Energy Research Institute (JAERI) and ORNL that provides for ORNL to examine pieces of the pressure vessel from the decommissioned Japan Power Demonstration reactor. This examination will focus on the changes in microstructure and fracture properties caused by long-term exposure to irradiation and provides an opportunity to examine in depth a reactor vessel from an actual nuclear power reactor. The ORNL studies will complement ongoing research being conducted by JAERI at their Tokai research establishment.

During FY 1994, ORNL published neutron cross-section libraries, BUGLE-93 and VITAMIN-B6, that can be used in evaluating the neutron fluence for power reactors, which is an essential input in estimating the level of radiation embrittlement for reactor pressure vessels. In addition to the cross-section library work, researchers at ORNL have worked with researchers in the Czech Republic, and other East European researchers, in performing calculations to predict the results of carefully controlled "benchmark" experiments conducted by the Czech researchers. This continuing work is generating important data relevant to the NRC's program to validate neutron fluence calculation methods and is providing technology transfer and validation of the methods being used by the different laboratories. This work contributed to the staff's effort to evaluate public comments and to revise a draft regulatory guide on calculational and dosimetry methods for determining pressure vessel neutron fluence.

Work continued in FY 1994 to compile and evaluate embrittlement trends using the power reactor pressure vessel material surveillance data. These data are reported to the NRC in accordance with Appendix H to 10 CFR Part 50 and reflect embrittlement trends for reactor pressure vessels irradiated under typical power reactor conditions. The work by ORNL to compile these data into a comprehensive data base has provided the basis for work by Modeling and Computing Services to develop statistically based models for predicting radiation embrittlement. The ORNL data base has also been used by the regulatory staff in both plant-specific and generic evaluations. The ORNL work is a continuing effort while the Modeling and Computing Services work is expected to be completed in 1995. This work will enable the NRC to evaluate the need for further revision to Regulatory Guide 1.99, which provides the methods for estimating radiation embrittlement and is a fundamental part of the NRC's approach to ensuring pressure vessel safety.

Research to better understand the mechanisms of radiation embrittlement continued in FY 1994, with significant advances being made by ORNL and the University of California at Santa Barbara, in conjunction with researchers in the United Kingdom, in modeling the complex interactions among the impinging neutrons and the atoms in the pressure vessel steel. This work is closely integrated with the experimental work being done in Europe. Understanding the controlling mechanisms is essential to confidently extrapolating empirical models of radiation embrittlement to unique operating circumstances. The progress in the mechanisms research is providing assurance that the empirical models are conservative and is helping to define the limits of extrapolation for those models.

In 1994, a proposed amendment to the NRC regulations for nuclear power plants was issued to clarify several items related to the fracture toughness requirements for reactor pressure vessels. The proposed amendment would clarify the pressurized thermal shock requirements for thermal annealing of a reactor pressure vessel. In addition, a draft regulatory guide was issued to provide the information and criteria needed to evaluate an application to perform a thermal annealing treatment for a reactor vessel when neutron radiation has reduced the fracture toughness of the vessel material. The thermal annealing treatment is expected to restore the fracture properties to acceptable levels.

1.1.3.2 Piping Integrity

During the 1980s, increased concern with intergranular stress corrosion cracking in BWR piping systems and increased needs for research on other aspects of environmentally assisted cracking and pipe fracture behavior led to increased research on piping integrity as part of an overall pressure boundary integrity research program.

In FY 1994, work on large-scale pipe fracture and fracture characteristics of cast stainless steels wound down. However, other concerns in the piping system are still being investigated. The potential for fatigue damage in reactor systems has long been recognized, but data from Japan indicate that the effects of the water coolant on the expected fatigue lives may not be adequately accommodated by the present ASME design rules. The Argonne National Laboratory (ANL) is collecting additional data on coolant effects on fatigue in BWR and PWR chemistries and analyzing these data, and data obtained from other sources, as part of a program to develop better characterizations of fatigue behavior for NRC use in evaluating remaining service life in aging plants. During FY 1994, new interim design curves that incorporate the effects of reactor environments were published and are being updated using newly developed ANL data.

Pipe fracture research continued during FY 1994, with one of the major programs at Battelle Memorial Institute drawing to a close. This research has provided the technical basis for the flaw evaluation methodologies for piping that are contained in the ASME Code Section XI and the technical basis for the NRC's leak-before-break evaluation methodology. The balance of the research is being conducted as part of an internationally funded research project, the Second International Piping Integrity Research Group program, also being conducted by Battelle. That work is examining the effects of simulated seismic loading on the fracture behavior of cracked pipe and piping fittings. It is anticipated that all the NRC's large-scale pipe fracture research will be completed by early FY 1996.

1.1.3.3 Core Internal Components

Irradiation-assisted stress corrosion cracking (IASCC) of core internal components of both BWRs and PWRs has been observed and is becoming a more common problem as reactors age and core materials accumulate higher fluence. Although many of the affected components can be replaced, others are difficult or impractical to replace. The susceptibility of materials to IASCC seems to be strongly dependent on minor variations in material composition and microstructure. Thus, nominally identical materials show large differences in resistance to IASCC. Ongoing research by the NRC and by others is attempting to identify those characteristics that make materials susceptible to IASCC. In particular, the NRC is sponsoring the irradiation of a large group of materials in the Halden reactor in Norway; examination of these materials should help to clarify the role of material composition, fluence, and the operating environment.

1.1.3.4 Inspection Procedures and Technologies

The NRC's approach to ensuring the integrity of the reactor pressure boundary builds on the overall "defense-in-depth" concept. The research program parallels this fundamental approach and includes programs geared to each of the major considerations in providing structural integrity—analysis methods, material properties, and inspection techniques. The research program addressing inspection procedures and technologies provides an independent basis for evaluating the efficacy and reliability of industry inspection programs. The program includes studies of improved methods for selecting components for inspection and strategies for setting the required capability for the inspection method and inspection periods to provide a reliable overall inspection. The program also deals with the inspection technologies and methods necessary to ensure reliable detection and accurate sizing of flaws. Finally, the program includes a focused effort to transfer this technology to practitioners in the NRC regional and headquarters offices.

International Studies. The NRC is an active participant and a leader in the Program for the Inspection of Steel Components, Phase III (PISC III). This international program, organized in 1986, is assessing the effectiveness of nondestructive testing technologies and procedures for the inservice inspection (ISI) of nuclear power plant components. The participants in this program have invested an estimated $40 million in the program, including contributions of materials, inspection services, and manpower. The products from this program will assist regulators and code bodies in establishing technical bases for improving ISI requirements.

The focus of the PISC III program is on the nondestructive testing of realistic LWR primary circuit components containing realistic flaws. During FY 1994, results were reported for a flaw sizing study in a reactor pressure vessel, detection and sizing of flaws in dissimilar metal weldments, and the detection and sizing of flaws in stainless steel piping. This work shows that some inspectors were effective and had a high flaw detection rate with a corresponding low false call rate. However, other inspectors demonstrated an ineffective performance with a low flaw detection rate and high false call rate. For flaw depth sizing, there were a few inspectors and conditions in which performance was acceptable. However, the overall performance was poor with low correlation and large errors between the depth estimates and the true depth size.

Improved Ultrasonic Detection and Sizing of Flaws. An improved method for more reliably detecting flaws and sizing them with greater accuracy in LWR primary circuit components is the Synthetic Aperture Focusing Technique for Ultrasonic Testing (SAFT–UT). The SAFT–UT technology is based on physical principles of ultrasonic wave propagation and uses computers to process the data to produce high-resolution, three-dimensional images of flaws to aid the inspector in locating and sizing them. The SAFT–UT technology has been developed through extensive laboratory testing and validated through blind trials. The technique performed well in the PISC III pressure vessel flaw sizing studies. A SAFT–UT system was fabricated for the NRC's nondestructive examination mobile laboratory and operational training was provided to the NRC personnel who conduct independent field audits of ISI results. This system was successfully used for the first time in 1994 by the NRC staff for performing audit inspections of piping at the Peach Bottom (Pa.) nuclear power plant.

Field Trials for Improved Eddy Current Inspection of Steam Generator Tubing. Researchers from ORNL participated in two inservice inspections of steam generator tubes at the Prairie Island 1 and 2 (Minn.) plants to test new eddy current probes and signal analyses techniques and instrumentation under field conditions. The new probes use multiple "pancake" type coils for better sensitivity. The design incorporates several coils around the circumference of the probe so that rotation is not needed, and the probes can be translated along the tube at high speed. Thus, the new array probes offer the best features of the two currently used probes: high speed and sensitivity. In the field tests, the new probes produced signal levels from flaws that were 5 to 10 times greater than the current-practice rotating pancake coil (RPC) inspection. The ORNL inspection was 75 times faster than the current RPC inspection and nearly as fast as the bobbin coil inspection. The high sensitivity and high inspection speeds that can be achieved with these probes would permit inspection of the entire length of tubes in generators that are experiencing considerable degradation in a fraction of the time currently required for inspecting just short lengths of tubing with the RPC probes. The new array probes are also sensitive to axial cracks, circumferential cracks, and volumetric defects, which is a significant improvement over current bobbin coil probes.

1.1.3.5 United States-Russian Federation/Ukraine Cooperative Agreement

The NRC staff and researchers from ORNL and the University of California at Santa Barbara participated in September 1994 workshops and meetings in Kiev, Ukraine, and Moscow, Russia, as part of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS). Working Group 3 on "Radiation Embrittlement" held a 2-day workshop in Kiev to discuss pressure vessel integrity issues, followed by a 4-day working group meeting in Moscow. A total of 16 papers were presented during the workshop (eight from the United States and eight from the Ukrainian participants), with a total of 24 papers presented during the working group meeting (eight from the United States and 12 from the Russian participants).

NRC staff members and representatives of the Department of Energy and the national laboratories participated in a September 1994 meeting in Moscow of the JCCCNRS Working Group 12 to discuss issues related to nuclear power plant aging and plant life extension. The United States delegation presented 10 papers during the working group meeting. Subsequent Working Group 12 activities have included the exchange of more information on special topics and preparations for the sixth Working Group 12 meeting to be held in the United States in the summer of 1995.

1.2 Aging of Reactor Components

1.2.1 Statement of Problem

Aging affects all reactor structures, systems, and components in various degrees and has the potential to increase risk to public health and safety if its effects are not controlled. In order to ensure continuous safe operation, measures must be taken to monitor key structures, systems, and components and interfaces to detect aging degradation and to mitigate its effects through maintenance, repair, or replacement. For an older plant approaching the end of its design life and for which extended operation beyond its original license period of 40 years is contemplated, aging becomes a critical concern and will clearly be crucial to any assessment of the safety implications of license renewal.

The NRC and the nuclear industry have initiated a significant effort aimed at renewing plant licenses beyond their original term of 40 years. According to an early Department of Energy study, the projected net benefit to the United States economy can be on the order of $230 billion through the year 2030, assuming a 20-year period of extended operation for current plants. The benefit reflects both the lower fuel cost of the nuclear plants and reduced outlays for replacement of generating capacity. The license renewal rule, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," was issued in final form in December 1991. The initial form of draft Regulatory Guide DG–1009, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," was issued for comment in 1991. Since publication of the final license renewal rule, a number of significant policy issues have been identified. As a result, the Commission is in the process of amending the license renewal rule. Proposed revisions were published in September 1994.

1.2.2 Program Strategy

NRC staff effort in aging is being pursued in several areas, including technical and scientific research to identify the effects of aging on the key safety-related components of the plant and to examine methods for mitigating such effects. Specifically, the strategy is to achieve, relative to each component, the following results:

1. Identify and characterize aging and service wear effects that, if unmitigated, could cause degradation of structures, systems, and components and thereby impair plant safety.

2. Develop methods of inspection, surveillance, and monitoring and of evaluating residual life of structures, systems, and components that will permit compensatory action to counter significant aging effects prior to loss of safety function.

3. Evaluate the effectiveness of maintenance, repair, and replacement practices, current and proposed, in mitigating the effects and diminishing the rate and the extent of degradation caused by aging.

1.2.3 Research Accomplishments in FY 1994

1.2.3.1 Aging Research

Aging affects all nuclear reactor structures, systems, and components. If aging degradation is not detected and corrected, it can increase risks to public health and safety. Failures of safety-related components have occurred in the past because of such age-related degradation processes as corrosion, embrittlement, wear, and fatigue. The objective of aging research is to develop the technical bases for continuous safe operation of nuclear power plants as they progress through their design life; to define the operative aging mechanisms; and to confirm existing and/or developing recommendations for new detection and mitigation methods in order to prevent or mitigate the deleterious effects of the aging process.

The Nuclear Plant Aging Research (NPAR) program continued to study the aging-related degradation of performance of safety-significant components and systems and methods for mitigating and managing the aging of these components and systems in commercial nuclear power plants. During FY 1994, preliminary or comprehensive aging assessments were completed or final reports were issued for the following safety-related components, systems, and associated special topics:

- Chemical and Volume Control System for Pressurized Water Reactors (NUREG/CR-5954)
- Containment Cooling Systems (NUREG/CR-5939)
- Reactor Core Isolation Cooling System (NUREG/CR-5692)
- Selected Fault Testing of Electronic Devices (NUREG/CR-6086)
- Managing Aging in Nuclear Power Plants—Insights from NRC Maintenance Team Inspection Reports (NUREG/CR-6016)
- Accumulators
- Isolation Condenser Systems
- Air-Operated Valves (NUREG/CR-6016)
- Characterization of Check Valve Degradation and Failure Experience (NUREG/CR-5944, Vol. 2)

Aging Effects on Motor-Operated Valve Performance. In 1994, initial research efforts were completed to identify motor-operated valves (MOVs) in typical PWR and BWR plants that are most susceptible to internal environmental corrosion. The NRC concern is whether corrosion of internal valve parts can significantly affect the torque and thrust requirements for operating the MOVs when necessary, particularly to mitigate accident conditions. Friction experiments were conducted on samples of corroded materials typical of certain valves. Although the test results indicated that the friction due to corrosion does increase the thrust requirements, many new questions about the need for simulating actual loadings, temperature, and other parameters must be answered before the magnitudes of the increases in friction can be determined and validated. Subsequent investigations to answer these questions were made, and a better controlled series of friction experiments will start in late 1994 and will be completed in FY 1995.

The information will be used in determining if the MOVs comply with Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

Air-Operated Valves. An evaluation of aging and service wear of air-operated valves was completed and reported in NUREG/CR-6016, "Aging and Service Wear of Air-Operated Valves Used in Safety-Related Systems at Nuclear Power Plants." The evaluation was based on data taken from the Nuclear Plant Reliability Data System (NPRDS) for the period January 1, 1988, to December 31, 1990, which, after removal of inconclusive data, involved reports of 1503 failures of varying degrees. The data were processed in ways to reveal trends and the effectiveness of testing, and it was found that neither were there trends nor was testing especially effective in detecting degradation that led to the failures. The results showed that failures involving complete loss of function were usually the result of failures in the controls or the valve actuator vs. some failure of the valves themselves. While many of the controls and actuator components are known to have high failure rates wherever they are applied, they have not usually exhibited signs of degradation prior to complete failure. It was concluded from this that a basis could be developed for replacing certain components on the basis of aging environment alone in any cases where failures are unacceptable.

Check Valves. The check valve degradation and failure study completed in 1993, covering failures occurring in 1984 to 1990, was expanded to examine and process NPRDS records on failures of check valve internals occurring in 1991. After screening the data base to eliminate unsuitable records, 401 failures remained to be analyzed. As in the past study, a primary goal was to identify any correlations of valve failure rates with plant age, valve size, system of service, manufacturer, etc. A further goal of the study was to identify any apparent trends in failure rates, failure detection, severity of failures, etc. With the cooperation and assistance of the Nuclear Industry Check Valve Group, additional information was obtained on most of the valves regarding specific valve types, specific design features, valve configuration, valve application, and applicable inspection programs. The latter information was vital to a new activity that will provide the independent source of data to allow NRC to respond to expected industry requests for extension of check valve test and inspection intervals. The results reported in NUREG/CR-5944, Volume 2, "A Characterization of Check Valve Degradation and Failure Experience in the Nuclear Power Industry—1991 Failures," showed some positive trends from those reported in Volume 1. For example, failures detected by abnormal occurrences declined from 19% to 5%, and the percentage of significant failures decreased from 53% to 36%. Also, the most effective means of detecting failures continued to be by programmatic inspections—77%, up from 59% in the earlier study. Thus, it appears that degradation is being detected earlier in the failure process.

In response to the improvement in effectiveness of programmatic tests and inspections and condition monitoring techniques, as well as the improvement in availability and reliability of information on failures, a new project has been initiated to help NRC respond to anticipated requests to extend test and inspection intervals for those check valves that have exhibited low failure rates. Independent evaluations of the available information and inspection and monitoring techniques will be produced to provide the basis for approving the requests. In addition, national consensus standards groups will be supported, and industry groups will be contacted to ensure that the direction and focus of the evaluations are consistent with industry activities. Further, participation in the standards activities may contribute to the acceptability of revised standards expected to be produced in response to improved data and condition monitoring. In addition to NUREG/CR-5944, Volume 2, two other reports have been published related to check valve testing and condition monitoring. They are ORNL/NRC/LTR-93/6, "Review of Monitoring and Diagnostic Methods for Check Valves," and ORNL/NRC/LTR-94/04, "Utility Survey PWR Safety Injection Accumulator Tank Discharge Check Valve Testing."

Aging Assessment and Mitigation of Major LWR Components. Of intrinsic importance to reactor aging research is the assessment and mitigation of aging damage to major components and structures. The objective of this aging assessment task, an element of the NPAR program, is to identify, develop, and evaluate various aging management techniques for the major LWR components and structures. The approach is to gauge the degradation of the major LWR components and structures by the synergistic influences of the various aging mechanisms affecting the performance of these components and structures.

Research completed in this area in FY 1994 focused on completing the development of insights for aging management of selected LWR components and structures in order to ensure continued safe operation. The studies also included an evaluation of advanced inspection and monitoring methods for characterizing the aging damage. The results should prove useful to the NRC in its task of resolving safety issues associated with LWR aging degradation and developing policies and guidelines for operating license renewal. The major components completed in 1994 were the LWR metal containments and the LWR reinforced and prestressed concrete containments. Results are documented in a multivolume report, NUREG/CR-5314. A draft report (NUREG/CR-5824) discussing the identification of advanced monitoring methods for estimating stresses causing fatigue damage has also been completed and is finishing internal NRC review. Publication is scheduled for 1995.

PRA-Based Methodology for Aging Assessments and Priority Assignments. The risk-based methodology for assessment of aging in nuclear power plants and for defining priorities among risk contributions and maintenance activities (published in previous years as NUREG/CR-5587 and NUREG/CR-5510) is subject to uncertainties because of limited available aging data and also because of certain modeling assumptions. Research has focused on developing sensitivity and uncertainty analyses to address data and modeling uncertainties and to validate risk-based methods. This work was documented in draft NUREG/CR-6045 in 1994 and submitted for NRC review.

The application of age-dependent risk methodology requires age-dependent component failure rates. But age-dependent component failure rates are not generally available and need to be estimated from limited recorded plant failure data and plant maintenance logs. A major limitation of the age-dependent methodology has been the lack of recorded component aging data and approaches to develop aging failure rates based on the available information. To address this limitation, an approach was developed to incorporate age dependence in probabilistic risk assessments (PRAs) that does not require absolute age-dependent component failure rates. Instead, the aging of a component is expressed in terms of relative aging rates that are found to be fairly constant across different components and different plants. A draft report (NUREG/CR-6067) was completed on the aging data assessment methodology. Because of the importance of the role of PRAs in future risk-based regulations, NUREG/CR-6067 was extensively reviewed by the NRC staff in FY 1994. Many comments were provided, and it is expected that they will be incorporated in FY 1995.

In a previous year, an important application of the risk-based methods resulted in the development of PRA-based approaches for identifying safety-related MOVs having the most impact on plant risk covered under Generic Letter (GL) 89-10, "Safety-Related MOV Testing and Surveillance." Dynamic tests and surveillance tests, in accordance with GL 89-10, could then be performed on those MOVs with the largest risk impact. Relative risk-importance of single MOVs and the interaction of multiple MOVs can be analyzed using this approach. A draft report documenting the results of this work was issued for NRC review in FY 1994. This work has provided the technical basis for evaluating two different submittals by licensees for ranking their respective MOVs for tests in accordance with GL 89-10. These NRC evaluations resulted in identifying many weaknesses with the submittals, which the licensees are now resolving.

Work continued in FY 1994 to set priorities for environmental stressors associated with advanced digital instrumentation and control (I&C) systems in nuclear power plants, based on their risk significance. Analog I&C systems in nuclear power plants are being replaced by digital systems. Digital I&C systems are vulnerable to common environmental stressors, such as moisture/humidity and temperature, and the effects of such stressors are being identified and measures developed to rank them. The risk-based approaches are being tested for the I&C systems using plant-specific PRAs. A draft report was issued in FY 1994 identifying the approaches that can be used to accomplish this work. This effort has required more time than expected because of the lack of failure data for these components in nuclear plant applications.

Aging of Passive Components. In earlier research efforts, a methodology was developed for including the effects of aging on passive components (pipes, structures, and supports) in a PRA model to determine the resulting impact on plant risk. The methodology is based on probabilistic structural analysis for calculating the failure probability of these components. The methodology was documented in a final draft report (NUREG/CR-5730) and submitted for NRC review in FY 1994. Although the method meets the condition of the contract, the method is not compatible with the method for including aging of active components in a PRA. Since a total capability for determining the effects of aging on plant risk is the ultimate goal of this project, both active components (e.g., pumps, valves) and passive components should be included in the same model. Because of this result, a more practical and simple approach, which considers the results of NUREG/CR-5730, is being completed to meet the goals of the project. The documentation of this integrated method will be completed in FY 1995.

Equipment Operability. For the past 5 years, significant progress has been made by the NRC in advancing the state of the art of MOV technology. Full-flow experiments were conducted in prior years that led to the determination that MOVs are not being calibrated properly to ensure their operation when required. Since that time, evaluations of the vast amount of data from those earlier experiments and from additional smaller-scale tests have provided breakthroughs in understanding this complicated technology. The MOV engineers from nuclear power plants in the United States and Europe consider the NRC MOV researchers as leaders in this field.

The timely and effective transfer of research results to the NRC regulatory staffs has been useful for determining the integrity of MOVs and is a high-priority objective of this project. Specifically, the results that were transferred consisted of technical information for determining whether licensees' MOVs are in compliance with regulatory document GL 89–10. This document requests licensees to develop MOV programs that will ensure MOV operability throughout the lives of the respective nuclear power plants. Since there are an average of 150 safety-related MOVs in each nuclear power plant in the United States, it is imperative that these MOVs are accurately calibrated to ensure their performance as required.

The results of research completed in prior years as well as more recent findings in 1993 and 1994 have also provided the technical basis for issuing several other NRC regulatory documents identifying potential MOV problems of which the licensees must be aware. In addition, numerous supplements to GL 89–10 have been issued for licensee compliance as a result of the NRC research findings as well as other experiences from industry MOV testing.

During FY 1994, efforts continued on developing the technical basis for evaluating efficiencies for AC and DC motor-operators. When completed, this information will also be provided to the NRC regulatory staff for their use in determining whether these devices, which supply the power to valves, are achieving the outputs as claimed by the manufacturer, particularly under degraded voltage and elevated temperature conditions.

All the research results obtained over the past 5 years are being used by NRC in evaluating the Electric Power Research Institute (EPRI) topical report on MOVs. The topical report contains the information obtained from the EPRI MOV research program that the licensees will use in complying with GL 89–10. The NRC evaluation of the topical report started late in FY 1994 and will be completed in FY 1995.

Environmental Qualification Research. Questions concerning the environmental qualification (EQ) of electrical equipment used in commercial nuclear power plants have recently become the subject of significant regulatory interest. Initial questions centered on whether compliance with the EQ requirements for older plants is adequate to support plant operation beyond 40 years. After subsequent investigation, the NRC staff concluded that questions related to the differences in EQ requirements between older and newer plants constitute a potential generic issue that should be evaluated for backfit, independent of license renewal activities.

EQ testing of electric cables was performed by Sandia National Laboratories (SNL) under contract to the NRC in support of license renewal activities. Results showed that some of the environmentally qualified cables either failed or exhibited marginal insulation resistance after a simulated plant life of 20 years during accident simulation. This indicated that the EQ process for some electric cables may not be conservative. These results raised questions regarding the EQ process, including the bases for conclusions about the qualified life of components based on artificial aging prior to testing.

As the first step in developing a research program, RES held a public workshop to obtain technical input from industry representatives, as well as experts in the field. The Environmental Qualification Workshop was held on November 15–16, 1993. The workshop proceedings were issued in May 1994 as NUREG/CP–0135.

The workshop provided a unique opportunity for the open exchange of ideas and information among industry personnel, researchers, equipment manufacturers, and regulators involving EQ issues, descriptions of state-of-the-art activities in condition monitoring and research techniques. The discussions included several recent equipment failures and their causes at operating facilities, and presentations describing current licensee actions related to monitoring normal service conditions, such as on-line temperature monitoring in specific plant locations. Additional discussions centered on the limitations of condition monitoring techniques currently available, qualification testing techniques, and pre-aging techniques. Several participants expressed a concern that any testing could lead to additional regulatory burden on licensees.

1.2.3.2 Engineering Standards Support

The national standards program is coordinated by the American National Standards Institute (ANSI). ANSI provides procedural guidelines to help ensure that participation in the private sector standards development process is sufficiently broad based and that input from individual interests are fairly considered. NRC participation in this process is compatible with Office of Management and Budget Circular A–119, dated October 26, 1993, which provides policies for Federal participation in the development and use of voluntary standards.

The NRC staff is very active on the ASME codes and standards writing committees because portions of the ASME Boiler and Pressure Vessel (BPV) Code have, since 1971, been incorporated by reference into 10 CFR 50.55a, "Codes and Standards," of the NRC regulations in order to establish requirements for the construction, inservice inspection, and inservice testing of nuclear power plant components. Section 50.55a has periodically been amended to update the references to include more recent versions of the ASME BPV Code. In 1994, work continued on changes to the regulations that the NRC staff is considering. These changes, which include actions to address a cost-beneficial licensing action (CBLA) request, would:

1. Eliminate the 120–month update requirements for licensees' inservice inspection (ISI) and inservice testing (IST) programs;

2. Establish baseline regulatory requirements for ISI and IST programs, maintain a specified ASME BPV Code edition for ISI and incorporate, for the first time, by reference the 1990 Edition of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for IST, expand the scope of § 50.55a to include inservice testing and examination of safety-related snubbers to allow licensees an option to delete existing technical specification snubber test requirements and use the ASME OM Code for IST of snubbers, and delete the existing supplementary requirements for IST of containment isolation valves;

3. Allow alternatives to the ASME Codes, which would permit licensees to use later editions or addenda of the ASME Codes as alternatives;

4. Identify safety-significant code changes that the staff has determined are necessary for imposition on licensees, specifically Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of Section XI of the ASME Code, which provides rules for qualification of personnel and equipment used to perform inservice nondestructive examinations on nuclear power plant components;

5. Establish a new regulatory guide that would endorse alternatives to the baseline ASME Code rules, including the use of later editions and addenda of the ASME Code; and

6. Establish a new regulatory guide that would document NRC review and acceptance of OM Code Cases.

These actions would support substantial reductions in the current regulatory burden on licensees as determined by the staff during the review of a CBLA request. Work also continued on a rule-making that would, for the first time, incorporate by reference Subsection IWE and Subsection IWL, Section XI, ASME BPV Code. Subsection IWE provides rules for the inservice inspection of metal containments and the liners of concrete containments. Subsection IWL provides rules for the inservice inspection of concrete containments and their post-tensioning systems. The proposed rule was published for public comments in the *Federal Register* on January 7, 1994 (59 FR 979). Comments were received from 25 separate sources. The comments have been addressed, and the draft final rule is under development.

ASME Code Cases provide alternatives to the rules specified in the ASME BPV Code. Regulatory Guides 1.84, 1.85, and 1.147 identify those Code Cases for design and fabrication, materials, and inservice inspection, respectively, that the NRC has found to be acceptable. These regulatory guides, which are updated on a regular basis, were revised and issued in 1994.

1.2.3.3 Structural Integrity

Concrete structures play a vital role in the safe operation of all light-water reactor plants. In general, the performance of concrete structures in nuclear power plants has been good. However, there have been several instances where the capability of concrete structures to meet future functional and performance requirements has been challenged because of problems arising from either improper material selection, construction and design deficiencies, or environmental effects. Examples of some of the potentially more serious incidents include post-tensioning anchor head failures, leaching of concrete in tendon galleries, voids under vertical tendon-bearing plates, containment dome delaminations, corrosion of steel tendons and rebars, water intrusion through basemat cracks, and leakage of corrosion inhibitor from tendon sheaths. Such incidents indicate that there is a need for improved surveillance, inspection and testing, and maintenance to enhance the technical bases for assurance of continued safe operation of nuclear power plants.

The structural aging (SAG) program is addressing the aging management of safety-related concrete structures in nuclear power plants for the purpose of providing improved technical bases for their continued service. To accomplish program objectives, the SAG program has conducted activities under four major technical task areas: (1) program management, (2) materials property data base, (3) structural component assessment/repair technologies, and (4) quantitative methodology for continued service determinations. The final program report will be completed in mid-1995.

Regulatory applications of this research include: (1) improved predictions of long-term material and structural performance and available safety margins at future times; (2) establishment of limits on exposure to environmental stressors; (3) the ability of NRC to reduce its total reliance of licensing on inspection and surveillance through development of a methodology that will enable the integrity of structures to be assessed (either pre- or post-accident); and (4) improvements in damage inspection methodology through potential incorporation of results into national standards that could be referenced by standard review plans.

1.2.3.4 License Renewal Regulatory Standards

A final rule (10 CFR Part 51) concerning the environmental review for renewal of a nuclear power plant operating license is under development. The proposed rule was published for public comment in September 1991. Over 120 comments were received on the technical analyses and certain procedural aspects of the proposed rule. Concern was expressed that the proposed rule would constrain public comment on environmental issues at the time of license renewal review for an individual nuclear power plant. In FY 1994, four public workshops were held to discuss approaches to resolve specific concerns expressed by the Agreement States over the treatment of need for generating capacity and alternative energy sources. All comments are being considered in developing the final rule, the generic environmental impact statement, and other supporting documents. It is expected that the final rule and supporting documents will be published in FY 1995.

2. STANDARD REACTOR DESIGNS

2.1 Systems Performance of Advanced Reactors

2.1.1 Statement of Problem

The Commission has issued a policy statement on the regulation of advanced nuclear power plants (51 FR 24643) that states that the NRC will review and comment on applications for certification of new design concepts with special emphasis on vendor test programs for confirming the performance of novel safety systems. As part of this program, the NRC will develop, review, and implement advanced reactor safety and policy issues in its review of such advanced reactor concepts and carry out any independent research and analysis determined to be necessary to verify that advanced reactor designs have the potential to provide enhanced margins of safety and that plant safety systems will adequately perform their intended functions.

2.1.2 Program Strategy

Research programs have been initiated to support the certification and licensing of advanced reactor designs being developed by the nuclear industry and the Department of Energy. Several different designs are being considered for certification under 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants." These designs fall into two groups. The first group consists of four evolutionary and passive advanced LWR types: the advanced boiling water reactor (ABWR), the CE System 80+, the AP–600 advanced pressurized water reactor, and the simplified boiling water reactor (SBWR). The second group consists of four more nonconventional advanced reactor types: the Process Inherent Ultimate Safety (PIUS) reactor, the Canada Deuterium Uranium (CANDU 3) heavy-water-cooled reactor, the Advanced Liquid-Metal-Cooled Reactor (ALMR), and the Modular High-Temperature Gas-Cooled Reactor (MHTGR).

The two evolutionary LWR designs (ABWR and System 80+) have been judged to be similar enough to the current generation of LWRs so that little additional research is needed to support their certification or licensing; the four nonconventional reactor design concepts have not reached the point where certification is expected in the near term. Consequently, the current emphasis of the advanced reactor research program is on developing the information needed to support the certification of the AP–600 and SBWR reactor designs and making appropriate modifications of existing regulatory assessment methods to accommodate the review of the unique, passive safety systems of these reactors.

2.1.3 Research Accomplishments in FY 1994

2.1.3.1 Support for AP–600 Design Review

Confirmatory testing and analysis of the Westinghouse AP–600 reactor and plant systems are being performed to provide additional confidence in the NRC's evaluation of the safety of the AP–600 design. The most cost-effective means of performing the desired tests was to modify an existing full-height, full-pressure test facility rather than build a new one. Screening revealed that the best choice was the Rig of Safety Assessment (ROSA) large-scale test facility in the Japan Atomic Energy Research Institute (JAERI). To confirm these initial results and to determine the extent of modification necessary to simulate the AP–600, the Idaho National Engineering Laboratory was contracted to perform a comparative study between ROSA and the AP–600 using the RELAP5 code.

A comparison between the existing ROSA facility and the AP–600 design showed that modifications to the ROSA facility would be needed. Facility modifications were completed in February 1994 by Sumitomo Heavy Industries, which constructed the ROSA facility and has been maintaining and operating it for the past several years as a contractor to JAERI.

As of January 30, 1995, 12 tests have been conducted in ROSA simulating various accident scenarios that would challenge the unique safety systems employed in the AP–600 reactor. Small-break loss-of-coolant accidents in different locations and different sizes in the primary coolant loop were investigated. Test results obtained so far indicate that the reactor will be effectively cooled, as designed, under various accident conditions. However, two issues have been raised by the tests performed to date. One involves the large thermal gradient found in the cold leg where cold water from the passive residual heat removal system enters the primary system. The other is the possibility of having a water hammer in the cold leg or in the upper plenum as a result of direct contact between subcooled water and steam. These issues are being carefully evaluated to determine any safety implications that might affect the acceptability of the AP-600 design.

Two additional tests will be conducted by June 1995, with six more in 1996. The RELAP5/MOD3 computer code is being assessed against the test data from ROSA, and necessary modifications are being made in the code to improve its prediction capability.

2.1.3.2 Support for SBWR Design Review

This program provides confirmatory testing and computer code assessment for the General Electric SBWR. There are three elements in the program. First, a well-scaled, integral SBWR test facility has been designed and will be built at Purdue University. The test facility is called PUMA (Purdue University Multi-Dimensional Integral Test Assembly). Second, tests will be performed in the PUMA facility to produce data for a broad spectrum of loss-of-coolant accidents and transients postulated for the SBWR. Third, the PUMA data will be used to (a) assess the capabilities of the thermal–hydraulic RELAP5 code for SBWR analysis, (b) assess the integral performance of the SBWR-unique safety systems that maintain core and containment cooling, and (c) identify and understand the important phenomena observed in the tests.

PUMA is a low-pressure (150 psig) and reduced-height facility. Its design was completed in 1994, and procurement and fabrication of various components and instrumentation is under way. The PUMA facility will be completed by April 1995 and will be ready for testing by July 1995. A total of approximately 40 tests will be performed by August 1996.

2.1.3.3 Support for CANDU 3 Design Review

Several studies were completed in early FY 1994 in connection with a preapplication review of the CANDU 3 design, and four significant research products were produced from these studies. The first was a summary of Canadian regulation of CANDU reactors that identified some contrasts with NRC regulations. The second identified and classified event sequences (i.e., accident scenarios), plant systems, and operator actions in ways that would facilitate the application of NRC regulations. The third was an assessment of data bases that exist as the basis for CANDU safety analyses. And the fourth was a preliminary analytical study, using Canadian computer codes, of events involving the design's positive coolant-void coefficient of reactivity. Additional work has been planned to support the formal review for design certification, but that work will not be initiated until the bulk of the work on the AP-600 and SBWR is finished.

2.1.3.4 Human Reliability

Efforts are continuing to develop methods for assessing the impact on risk of changes in human performance due to the introduction of advanced digital displays and controls.

Research to establish a technical basis for minimum shift staffing (operations) for advanced control room designs was initiated in FY 1994 at the Halden reactor project. The research is based on workload and task allocation studies conducted on power plant and advanced control room simulators and through task network modeling.

2.2 Engineering Issues for Advanced Reactor Designs

2.2.1 Statement of Problem

The safety evaluation of standard/advanced reactors involves (1) new technologies, (2) new environmental conditions, (3) new systems, (4) new design approaches, and (5) new requirements. New technologies include the expanded use of advanced digital instrumentation and control systems and novel modular construction methods. New environmental conditions are reflected in the low-pressure operation of check valves for a planned 60-year operational period. New systems are found in the passive reactor designs where air-conditioning power is not needed for accident response and the safety classifications differ markedly from present standards, thereby affecting probabilistic risk assessments (PRAs) and the fragility information needed to perform these PRAs. New design approaches incorporate the use of experience data as a method of seismic qualification and the elimination of the operating basis earthquake from design. New requirements deal, for example, with NRC goals for containment performance under severe accidents and the ability to withstand interfacing-system loss-of-coolant accidents. Research is needed to support technical positions in safety evaluation reports prepared by the NRC staff for the Final Design Approval, the Design Certification, and the Combined License for standard/advanced reactors dealing with these issues.

2.2.2 Program Strategy

The objective or strategy of engineering research is to verify the safety and quantify the margins in standard/advanced reactors for those features or criteria substantially different from currently operating plants and to ensure that NRC goals and policies are implemented successfully. To deal with these issues, analyses, evaluations, data collections, and limited testing are planned. In some cases, research is used to independently confirm or modify vendor/utility proposals for design, fabrication, construction, and inspection. In other cases, particularly in the passive designs, research serves to demonstrate acceptable design basis and severe accident response, for example, for structural performance of the AP–600 containment during passive containment cooling. In still other cases, research is used to define acceptable standards for new features and requirements.

2.2.3 Research Accomplishments in FY 1994

2.2.3.1 Review of Vendor Data

Under this program the NRC staff has completed analysis of available vendor data. A listing of design, maintenance, and application deficiencies, including historical weaknesses, which could be useful in evaluating advanced light-water reactor (ALWR) applications has been completed. Research efforts have supported NRR (1) in reviewing valve applications and (2) in providing information on the behavior of different valve designs based on research results and other valve experiences. Based on the research completed, the NRR staff has established a position in the safety evaluation reports for advanced boiling water reactors and CE System 80+ regarding the design requirements for piping systems that interface with the primary system to minimize the potential for interfacing-system loss-of-coolant accidents. The research studies are continuing to define the extent to which the piping systems outside the containment should be designed to these requirements, considering cost-benefit aspects.

2.2.3.2 Experience-Based Approach to Seismic Qualification of Equipment

The ALWR Equipment Qualification Panel has resolved all issues pertaining to the experience-based approach for seismic qualification of Group 1 items. Group 1 equipment has a mature design with little design variability and, in general, has demonstrated characteristics of inherent seismic ruggedness. Items included in Group 1 are (1) horizontal centrifugal pumps with electric motor drivers, (2) vertical centrifugal pumps with electric motor drivers, (3) manual valves/check valves, (4) valve assemblies with electric motor operators (MOVs), (5) thermal element assemblies, and (6) diesel generator units. The ALWR Equipment Qualification Panel met with vendors of batteries and transformers to discuss design variability and failure modes for ALWR designs. In addition, the panel toured a select group of the candidate facilities that had experienced earthquake damage and would be used in the EPRI/ARC data base. A geologist has been added as a consultant to the panel to verify EPRI/ARC ground motion estimates.

2.2.3.3 ASME Section III

The preliminary evaluation of the basis for revisions to ASME Section III design rules for piping published in the 1995 Winter Addenda has been completed. It was used to develop a staff position in the safety evaluation report of the AP–600.

The NRC staff is interacting with ASME working group members to understand the bases for the published ASME design code changes to facilitate NRC endorsement through 10 CFR Part 50.55a.

2.2.3.4 Age Dating of Geologic Materials

Research activities on the age dating of geologic materials included the conducting of an international workshop of 60 renowned geochronologists and paleoseismologists. Field tests were conducted in California, Montana, and South Carolina where new and promising techniques were applied.

2.2.3.5 Modular Construction

Preliminary licensing review criteria for structural modules have been developed. Using the developed criteria, input was provided to NRR licensing staff for the draft safety evaluation report for the AP–600 structural modules. Information necessary to complete sections of the review criteria on design and testing of structural modules has been identified in the United States and Japan. The contractor also participated in a design and calculation review on structural modules for the AP–600 with Westinghouse and Bechtel.

The distinguishing construction features, operational modes, and the applicable building, electrical, and mechanical codes and standards for the advanced reactors have been identified, with Tasks 1 and 2 completed. The applicability of the codes and standards is ongoing for the CE System 80+, the AP–600, and the advanced and simplified BWRs.

2.2.3.6 Equipment Anchorage

Phase I of a three-phase test program of anchorages used to anchor mechanical and component supports to concrete was completed. Preliminary review of results indicate that for some anchor types, mainly expansion anchors, dynamic loads reduce performance.

2.3 Regulatory Application of New Source Terms

2.3.1 Statement of Problem

Potential accidents are evaluated during reactor licensing as part of the Commission's defense-in-depth policy. Certain accidents, referred to as "design basis accidents," are postulated to occur and their consequences must be shown to be acceptable.

The most limiting design basis accident evaluated is required by 10 CFR Part 100 and is derived from the 1962 report TID–14844, "Calculation of Distance Factors for Power and Test Reactor Sites," which postulates the release of the entire core inventory of noble gases and 50% of the core inventory of iodine fission products from the core into the containment. This "TID source term" is used for evaluating the suitability of the reactor design as well as the site. Present regulatory guidance, reflected in Regulatory Guides 1.3 and 1.4, stipulates that this source term is instantaneously available for release from the containment to the environment and specifies that the fission product iodine would primarily be in elemental form. The TID–14844 source term, originally intended for site evaluation purposes, has also been applied to many aspects of plant design. These include requirements for fission product cleanup systems, such as sprays and filters, allowable containment leak rate, control room habitability, equipment qualification, and others.

This TID source term is associated with a severe reactor accident since only major core degradation and melting could result in the release of such large quantities of fission products. However, the present source term formulation, which dates from 1962, while providing a high level of protection for plant systems, does not reflect recent research findings. As a result, a rigid application of the TID source term to the evaluations of new plant designs may not provide the best engineering solutions on some aspects of future plant designs.

2.3.2 Program Strategy

On May 25, 1988, the staff presented to the Commission an "Integration Plan for Closure of Severe Accident Issues," SECY–88–147. This plan discussed major elements relating to closure of severe accident issues, including severe accident research efforts and related activities in siting, emergency planning, and potential changes to existing regulations as a result of severe accident research findings. On October 4, 1990, the staff presented to the Commission a "Staff Study on Source Term Update and Decoupling Siting from Design," SECY–90–341. This plan discussed the staff proposal to decouple the issue of reactor siting from the issue of source term composition and dose calculations and to prepare an update and revision of the source term given in

TID–14844. On April 10, 1992, the staff presented to the Commission its "Revised Accident Source Terms for Light-Water Nuclear Power Plants," SECY–92–127. This paper provided a draft of a revised source term for use in calculating offsite consequences of design base accidents. On June 12, 1992, the staff presented to the Commission its proposed "Revision to 10 CFR Part 100, Revisions to 10 CFR Part 50, New Appendix B to 10 CFR Part 100 and New Appendix S to 10 CFR Part 50," SECY–92–215. This paper presented proposed rules to revise the reactor siting criteria. On July 28, 1992, the Commission announced the availability for comment (57 FR 33374) of a draft report on "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG–1465. On September 28, 1992, the Commission issued an advance notice of proposed rulemaking on "Acceptability of Plant Performance for Severe Accidents; Scope of Consideration in Safety Regulations" (57 FR 44513). Revised accident source terms, as well as other severe accident considerations, are to be incorporated into this rulemaking effort.

2.3.3 Research Accomplishments in FY 1994

The Commission's reactor site criteria (10 CFR Part 100) require, irrespective of likelihood, that an accidental fission product release from the core into containment be assumed to occur and that its offsite radiological consequences be evaluated. The criteria for defining the characteristics of that release into containment is derived from the 1962 report, TID–14844.

Since 1962, a better understanding of the timing and nature of the fission product release has been obtained. As a result, a number of areas of regulatory activities have been identified that may benefit from changes introduced as a result of source term and severe accident research. A revised accident source term document—NUREG–1465, "Accident Source Terms for Light-Water Nuclear Power Plants"—which is intended to replace TID–14844, was issued. In support of this effort, the following documents were issued:

1. NUREG/CR–5901, "A Simplified Model of Aerosol Scrubbing by a Water Pool Overlying Core Debris Interacting With Concrete," dated November 1993.
2. NUREG/CR–5747, "Estimate of Radionuclide Release Characteristics into Containment Under Severe Accident Conditions," dated November 1993.

2.3.3.1 Update of Siting Regulations

In FY 1994, staff efforts continued on updating 10 CFR Part 100, "Reactor Site Criteria." A proposed rule to revise Part 100 was first issued for comment in October 1992. Source term and dose calculations were proposed to be eliminated for reactor siting by simply specifying a minimum exclusion area distance around the reactor site and by stating population density criteria in the rule. An update of the seismic site evaluation criteria proposed to incorporate probabilistic as well as deterministic methods. Extensive comments, both domestic and foreign, were received favoring the continued use of source term and dose calculations for reactor siting. In July 1994, the staff recommended that the proposed rule be withdrawn and that a revised proposed rule be issued that would incorporate basic reactor site criteria and would continue the use of source term and dose calculations for the siting of custom plants. Modifications to standardized plant designs to compensate for poor site characteristics would be discouraged, however. The Commission approved the staff's recommendation in September 1994 to issue a revised proposed rule.

2.3.3.2 Emergency Planning Regulations

In FY 1994, staff efforts continued on emergency planning licensing requirements for independent spent fuel storage facilities and monitored retrievable storage facilities. The public comment period for the proposed rule expired in November 1993. The staff analyzed comments received and is developing the final regulations. In FY 1994, a notice of receipt of petition for rulemaking was published in the *Federal Register* (59 FR 17499) requesting public comment on a petition submitted by VEPCO (PRM–50–60) relating to NRC audits of emergency plans. The public comment period ended June 1994, and the staff is currently analyzing comments received. A petition for rulemaking was also received from VEPCO in December 1992 (PRM–50–58) relating to emergency planning exercises. The petition was published in the *Federal Register* for public comment in March 1993 (41 FR 12341). The staff is currently evaluating public comments and developing a proposed resolution to the petition.

In March 1994, a final rulemaking was published in the *Federal Register* (59 FR 14087) to provide a revised emergency planning regulation that updates and clarifies ambiguities that surfaced in the implementation of the Commission's emergency planning exercise requirements.

PART 2--NUCLEAR SAFETY RESEARCH--REACTOR REGULATION SUPPORT

3. PLANT PERFORMANCE

3.1 Statement of Problem

A wide range of reactor plant design variations exists in the United States, and the safety of these plants must be ensured over a wide range of normal and abnormal operating conditions. The NRC is required to independently assess licensees' safety analyses and performance in designing, constructing, and operating a reactor with respect to the safety of the public for the complete spectrum of credible operating conditions and events.

NRC's task is difficult because straightforward testing of all transients in all plant design variations would not be technically and economically feasible. On the other hand, straightforward and exact theoretical analyses of a reactor's thermal-hydraulic behavior is not possible because mass, energy, and momentum exchanges take place over complicated interfaces between reactor components, water, and steam and because of the moving mechanical interfaces in pumps and the extensive baffle arrangements of steam generators in the primary loops.

As a result, the NRC must use available experimental data to validate analytical models for evaluating (1) design basis accidents, (2) the safety implications of actual events in operating reactors, and (3) hypothetical transient scenarios determined to be major contributors to risk as a result of probabilistic risk assessment studies.

3.2 Program Strategy

A dual analytical and experimental approach is used to achieve a firm technical understanding of the thermal-hydraulic behavior of the reactor. The NRC starts by simulating the actual reactor's continuous flow of heat and fluids with a computerized model consisting of many discrete cells exchanging mass, energy, and momentum at each small, but finite, time step. Physical laws are used when possible to calculate all these exchanges. Empirically derived formulas that are obtained from experiments are used as necessary to account for such complex effects as friction between vapor and liquid. The calculations are made for each time step and for each cell. The reactor models interact in a tightly coupled manner at each time step.

Our reliance on the computer codes to provide predictions of reactor response with acceptable uncertainties depends on three levels of experiments and comparisons of experimental results with code predictions. First are basic experiments used to derive empirical formulas for determining basic phenomenological behavior within each cell. Second are separate-effect experiments used to test the code's predictions for a single, complex component such as a steam generator. Third are integral-system tests that are used to evaluate the code predictions of the performance of complete reactor cooling systems. The results of these tests provide feedback to correct and improve the code and to improve our understanding of operating transients. Such understanding provides the basis for improving plant operations to reduce the likelihood of accidents.

3.3 Research Accomplishments in FY 1994

3.3.1 Reactor Safety Experiments

3.3.1.1 High Burnup Fuel Behavior

By the early 1990s, it had become clear that burnups in commercial power reactors were exceeding the burnup range that had been used to validate NRC's fuel behavior computer codes and related fuel damage criteria. Fuel suppliers were providing data to support the licensing of higher-burnup fuel designs, but the NRC's capability to independently validate such data had not been updated. Accordingly, it was decided to assess the need for (1) fuel performance model changes (e.g., UO_2 thermal conductivity, fission gas release), (2) fuel performance code updates (i.e., the FRAPCON code and resultant effects on fuel stored energy), and (3) changes in the threshold criteria for fuel failure under reactivity transients.

Contracts at three laboratories were placed to respond to this need. One is focusing on phenomenological models, one on the modification of computer codes, and the third on plant transient calculations to estimate the impact on reactor safety. During the year, the NRC became aware of new test results on high-burnup fuel being obtained in France, Japan, and Russia suggesting the criteria used to predict fuel failure may need to be revised. Since no such testing is being performed in the United States, efforts were made to enter into cooperative arrangements with foreign laboratories to obtain these data. Invitations were extended to these laboratories to present preliminary information at the NRC's annual Water Reactor Safety Information Meeting, and such presentations were made on October 26, 1994. More definitive results will be available in 1995. These test results will be used to assess, and perhaps modify, fuel damage criteria used by the NRC in licensing.

3.3.1.2 Thermal-Hydraulic Phenomena

Experiments are being performed at the University of Maryland in a scaled (1:4 in height, with a 1:500 volume) experimental facility that simulates a Babcock and Wilcox reactor. This facility was originally constructed under NRC contract to study small-break loss-of-coolant accidents. Following successful completion of that program, the facility's mission was shifted to the current study of mixing phenomena associated with boron injection events. Recent studies have suggested that dilution of boron (used to control reactivity) may occur that could potentially result in reactivity transients.

3.3.2 Safety Code Development and Maintenance

It is generally not possible to assess the safety performance of reactor and plant systems with tests in full-scale facilities, and an understanding of the thermal-hydraulic behavior of these plants must be established with the use of computer codes. Most of the NRC's independent analyses for the AP-600 and SBWR will be done with the RELAP code, which is being upgraded for application to these designs. Before any of the NRC codes are used for this purpose, or released for use by others, they undergo developmental assessment and peer review. Revised documentation is also provided for these improved codes. The upgraded version of RELAP for use on the new passive plant designs will be released in early 1995.

As part of the code maintenance activities for RELAP and the TRAC code (both PWR and BWR versions), the NRC conducts an international Code Applications and Maintenance Program (CAMP). There are now 17 member countries in CAMP, each of which participates in semi-annual meetings and makes cash contributions to supplement the NRC code development and assessment programs. Members also provide code assessment studies, recommend code improvements, and make other technical contributions to assist in the development and assessment of the codes.

4. HUMAN RELIABILITY

4.1 Statement of Problem

About half of all safety-related events reported at nuclear power plants and among nuclear materials licensees continue to involve human performance. Methods and data are needed to identify, systematically set priorities for, and suggest solutions to human performance issues in nuclear operations during normal, transient, and emergency situations and during plant maintenance.

4.2 Program Strategy

The human reliability research program has three objectives: (1) to broaden NRC's understanding of human performance and to identify causes of human error; (2) to accurately measure human performance for enhancing safer operations and precluding critical errors; and (3) to develop the technical basis for regulatory requirements, recommendations, and guidance related to human performance.

The human reliability research program is divided into three interrelated program elements: (1) personnel performance, (2) human-system interfaces, and (3) reliability assessment. The purpose of the personnel performance element is to develop enhanced methods for collecting and managing personnel performance data and to improve understanding of the effects of personnel performance on the safety of nuclear operations and maintenance. In addition, personnel performance research will broaden the understanding of such factors as staffing, qualifications, and training that influence human performance in the nuclear system and will develop information necessary to reduce the negative impact of these influences on nuclear safety. Research in the human-system interface element will provide the technical basis for guidelines and criteria to evaluate the interface between the system and the human user from the perspective of safe operations and maintenance. Lastly, the reliability assessment element includes multidisciplinary research that will integrate human and hardware considerations for evaluating reliability and risk in NRC licensing, inspection, and regulatory decisions.

4.3 Research Accomplishments in FY 1994

4.3.1 Personnel Performance

Work concluded on the development of a method to assess the effectiveness of training programs at nuclear power plants. Measures and supporting documentation for a training effectiveness evaluation method will be published in a two-volume technical report. Data analyses from a project on the factors that are considered when making decisions on operations staffing and on how staffing relates to safe startup, shutdown, and operation of nuclear power plants are now complete. Results of these analyses have been incorporated into NUREG/CR-6122, "Staffing Decision Processes and Issues." A second product of this study was published as NUREG/CR-6123, "An International Comparison of Commercial Nuclear Power Plant Staffing Regulations and Practice: 1980–1990." Work continues on a study to establish a technical basis for minimum shift staffing for both control room crews and for operational support staff outside the control room at nuclear power plants based on workload and task allocation. A handbook on the effects of environmental factors on human performance for use by nuclear power plant inspectors was published as Volume 1 of NUREG/CR-5680, "The Impact of Environmental Conditions on Human Performance." A critical review of the literature was published as Volume 2 of NUREG/CR-5680. Two reports concerning training for responding to accidents were published as NUREG/CR-6126 and NUREG/CR-6127. These reports describe decisionmaking and stress coping skills that may be needed to respond to an accident situation, as well as potential training approaches for developing those skills.

Research was initiated on communication errors in nuclear power plant events to characterize the root cause(s) of these errors, identify potential corrective actions for each category of communication error, and develop proposed review criteria and guidelines. A study on whether links exist between operator effectiveness and the simulator training received by operators at multi-unit stations as compared to simulator training at single-unit stations was initiated.

4.3.2 Human–System Interfaces

Human–system interface research includes NRC participation in the Organization for Economic Cooperation and Development Halden reactor project, which is a multifaceted program that includes verification and validation of digital systems, man-machine interaction, and surveillance and support systems for advanced control rooms. Information was developed on (1) methods and tools for the development and verification and validation of safety-related software, and (2) experience with development and quality assurance of software systems at the Halden project.

Research continued to evaluate the positive and negative attributes of standards and computer-aided-software engineering tools for use in the certification of high-integrity software for nuclear power plant safety systems. Research will be completed in 1995 on a project co-sponsored by the Electric Power Research Institute on verification and validation guidelines and quality metrics for digital high-integrity systems.

A project was initiated to independently evaluate, test, and improve upon verification and validation guidelines for use in the audit of computer-based safety systems.

The NRC and the National Institute of Standards and Technology issued the proceedings of a jointly sponsored Digital Systems Reliability and Nuclear Safety Workshop (NUREG/CP–0136). The goals achieved by the workshop were (1) providing feedback to the NRC from outside experts regarding potential safety issues, proposed regulatory positions, and research associated with application of digital systems in nuclear power plants, and (2) continuing the indepth exposure of the NRC staff to digital systems design issues related to nuclear safety by discussions with experts in the state of the art and practice of digital systems.

Following the workshop, research was initiated to ensure the completeness of the technical bases for regulatory requirements intended to ensure the integrity of safety-related software.

After internal NRC and independent peer review, proposed guidelines in support of the standard review plan for the review of advanced control room designs were published as NUREG/CR–5908, "Advanced Human-System Interface Design Review Guidelines." These guidelines were built on previously validated guidelines available from other industries, including the aerospace and defense industries, and were prepared in both paper form and computerized (interactive) form.

Work was completed as part of resolving Human Factors Generic Issues 5.1, "Local Control Stations," and 5.2, "Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation," which resulted in the publication of NUREG/CR–6146, "Local Control Stations: Human Engineering Issues and Insights," and NUREG/CR–6105, "Human Factors Engineering Guidance for the Review of Advanced Alarm Systems."

The review guidance published in the three documents described above is presently being integrated with additional material to form Revision 1 to NUREG–0700, "Human System Interface Design Review Guideline," a draft of which will be issued for public comment in FY 1995.

Work was begun on a new project to develop guidance for the review of advanced digital alarm systems.

Following recommendations from the Advisory Committee on Reactor Safeguards and from the Commission, the staff initiated the first phase of a project with the National Academies of Sciences and Engineering to conduct a study and workshop on a coherent and effective approach to the regulation of computer-based (digital) systems in nuclear safety and control systems. The results of the full study and workshop will give advice to the NRC on the framework for a coherent and effective regulatory program.

4.3.3 Reliability Assessment

NUREG/CR–4639 revisions were issued to complete research on collecting, cataloguing, and storing, in a computerized library, estimates of probabilities of operator error and hardware failure. Research continued to develop alternative quantification methods for incorporating the influence of organizational factors into probabilistic risk assessment (PRA). NUREG/CR-6208, describing the results of data collected at nuclear power plant simulators to reduce uncertainties associated with operator performance in cognitively demanding simulated emergencies, was published.

Research continued to analyze information from the simulator portion of the NRC-administered operator requalification examinations. Estimates from this source may provide valid error rates for use in a nuclear power plant PRA.

For several years the NRC has been developing reliability and risk analysis tools to evaluate the risk impact of changes of selected requirements in technical specifications. The methods were completed, and both detailed technical reports and a handbook to guide NRC reviewers through the use of these methods will be issued in FY 1995 as NUREG/CR-6141.

The Oak Ridge National Laboratory (ORNL) is conducting a study to identify functional and operating environmental issues arising from the application of new technologies in instrumentation and control (I&C) systems for both current and next-generation nuclear power plants. The goal of this program is to establish the technical basis for augmenting the equipment qualification process to accommodate "advanced" instrumentation. Initial studies have been documented in NUREG/CR-5094, "Functional Issues and Environmental Qualification of Digital Protection Systems of Advanced Light-Water Nuclear Reactors," where the likely effects of environmental stressors on safety system components and interfaces are examined. A methodology for identifying the need for accelerated aging in qualifying new I&C systems for placement in benign environments was also proposed. Current research involves an experimental investigation into the functional behavior and failure modes that result for a microprocessor-based safety system under the application of environmental stressors, such as the presence of smoke, electromagnetic interference, radio-frequency interference, temperature, and humidity. The prototypic safety channel implemented for this study employs technologies representative of those proposed for use in advanced light-water reactors. Environmental tests should reveal any potential system vulnerabilities and help determine the expected effect of a stressor on advanced I&C system components. This information supports a clearer definition of what (and to what level) stressor equipment should be qualified to withstand and thus builds the technical basis for supplementing current qualification guidelines.

5. REACTOR ACCIDENT ANALYSIS

5.1 Reactor Risk Analysis

5.1.1 Statement of Problem

Probabilistic risk analysis (PRA) has been shown to be a systematic and comprehensive method for identifying and evaluating the effectiveness of safety improvements proposed to reduce the likelihood and consequences of nuclear power plant accidents. PRA is used by the NRC staff in a number of ways, including for evaluating the level of safety at selected operating plants; for assessing the margins of safety in current requirements in light of the Commission's safety goal policy; for monitoring plant performance; and for identifying potential improvements in equipment or operator reliability.

5.1.2 Program Strategy

The reactor accident risk analysis research effort is applied in four ways: (1) providing expert review of severe accident PRAs to assess, for example, the risk implications of accident management strategies in order to minimize the release of radioactive material to the environment during severe reactor accidents; (2) developing, verifying, demonstrating, and maintaining methods for analyzing the consequences of in-plant and offsite severe accident physical processes for use in risk assessment and developing and demonstrating methods for quantifying the uncertainty in risk estimates and the relative contributions of specific issue uncertainty to the overall uncertainty; (3) reassessing periodically the frequencies, consequences, and risk of severe accidents in nuclear power plants and performing peer review of methods used and results obtained; and (4) developing risk-based management tools capable of determining the incremental risk reduction associated with proposed plant design and operational modifications and assisting in the setting of priorities for efforts in licensing and research activities.

5.1.3 Research Accomplishments in FY 1994

5.1.3.1 Issue-Oriented Projects

Analysis of Low-Power and Shutdown Accident Risks. As a result of the Chernobyl accident and other precursor events, an extensive two-phased project was initiated in 1989 to examine the potential risks of accidents initiated during low-power and shutdown modes of operation. Phase 1, completed at the end of 1991, was a coarse screening analysis of all operational modes (other than full power) for one BWR and one PWR to provide timely support for the Office of Nuclear Reactor Regulation's (NRR) regulatory analysis and to guide the Phase 2 effort. A significant finding was that the traditional concept of technical specification modes of operation does not adequately delineate plant operating boundary conditions needed for risk analyses. The Phase 2 effort concentrated on a specific operating state for each of the two plants, selecting the potentially highest risk operating state, based on the Phase 1 results. In addition, a simplified analysis of potential in-plant and offsite accident progression and health consequences of such accidents has been performed and provided to NRR in support of their regulatory activities as documented in NUREG–1449. The complete results of Phase 2 are being published as NUREG/CR–6143 for the BWR and NUREG/CR–6144 for the PWR.

Human Reliability Analysis. As part of an NRC-sponsored program evolving from an assessment of human reliability issues in low-power and shutdown operations in nuclear power plants, an improved approach to human reliability analysis (HRA) is currently being developed. It is intended to be fully integrated with PRA methodology and to enable a better assessment of the human contribution to plant risk, both during low-power, shutdown and at-power operations.

In FYs 1992 and 1993, a Human Action Classification Scheme for categorizing human actions and associated influences in actual low-power and shutdown events was developed and implemented. These accomplishments were documented in NUREG/CR–6093, "An Analysis of Operational Experience During Low Power and Shutdown and a Plan for Addressing Human Reliability Assessment Issues."

During FYs 1993 and 1994, work continued on (1) the development of a multidisciplinary framework for integrating HRA with PRA; (2) the characterization of errors of commission (EOCs) and human dependencies, including general guidance for their identification and representation in PRAs; and (3) the recognition of data base improvement needs, including a better characterization of human actions and their associated performance context (e.g., plant conditions, performance shaping factors, and dependencies), as well as a better description of an event timeline. These accomplishments are currently being documented.

This framework provided the capability to identify factors that influence humans to perform unsafe actions and thereby created a systematic basis for evaluating the significance and characteristics of EOCs and dependency from operational events. Thus, the framework has enabled important aspects of EOCs and dependency to be considered in the development of an improved HRA methodology and has clarified the requirements for their more realistic inclusion in PRA models. By the framework's provision of a single language and common structure for relating the different dimensions of human-system interactions, the evaluations of EOCs and dependencies has been demonstrated to be both tractable and tenable. Considering the importance of these issues in nuclear power plant safety, this change is an important advance. These EOC and dependency capabilities will be refined and expanded upon in subsequent tasks pertaining to the development phase.

The primary product of the current workscope (FYs 1994 and 1995) will be a workable HRA quantification process that includes the following: how to identify and incorporate human failure events in the logic models used in PRAs, what information is required for probabilities to be assigned to these failure events, how this information is used to estimate the probabilities, and how the probabilities are incorporated into the PRA quantification process. It is anticipated that the final phase (FYs 1995 and 1996) of the project will demonstrate the usefulness and acceptability of the developed methodology's implementation guidelines using selected parts of the low-power and shutdown Level 1 PRAs.

South Texas Risk Analysis. In 1992, the staff completed a review of the South Texas Project risk analysis and documented the results and findings (NUREG/CR-5606). The licensee estimated the overall mean core damage frequency to be 2 E–4 per reactor-year, which is found to be within the range of core damage frequency estimates provided for similar Westinghouse PWR facilities. The licensee has subsequently requested modifications to its plant technical specifications based, in part, on its risk analysis. The RES staff provided a draft safety evaluation report to NRR, which became a part of the basis for their regulatory decision.

5.1.3.2 Methods Development Projects

SAPHIRE Computer Tools. The set of computer codes called SAPHIRE (System Analysis Programs for Hands-on Integrated Reliability Evaluation) has been updated to version 5.0. This set of codes is to be used in performing probabilistic risk analyses and permit an analyst to perform many of the functions necessary to create, quantify, and evaluate the accident risks of nuclear power plants. The codes were used extensively to perform the low-power and shutdown risk analyses previously described and are currently being used for analysis and resolution of generic safety issues and for evaluating the safety aspects of concept plant designs. During 1994, PRA data from four more licensed nuclear power plants were added to the SAPHIRE data base and most of the data from previous plant loads were updated to version 5.0. This brings the data base total to 17 plants, two of which are advanced concept plants added to support the agency's design certification reviews. Courses continued to be provided to both the NRC staff and contractors using these codes. The documentation for version 5.0 has been published as NUREG/CR-6116, and the new codes and user manuals have been sent to the Energy Science and Technology Software Center at ORNL and made available for U.S. distribution. The previous version, SAPHIRE 4.17, has been made available to foreign countries that do not have cooperative agreements with the NRC.

Consequence Code Benchmark. The NRC has successfully completed the work with the Commission of the European Communities (CEC) and the Organization for Economic Cooperation and Development (OECD) to carry out an intercomparison exercise on probabilistic accident consequence codes. The six codes being evaluated were MACCS (United States), COSYMA (CEC), CONDOR (UK), OSCAAR (Japan), LENA (Finland), and ARANO (Sweden). The intercomparison exercise used six radioactive accident source terms and calculated dose consequences for such measures as whole body dose and fatal cancers. The results of these comparisons were published in FY 1994 by the CEC and OECD and demonstrated that the reactor accident consequence predictions calculated by these new codes all agreed within a reasonable range.

Offsite Consequence Uncertainty Analysis. The NRC has completed a pilot probabilistic consequence uncertainty analysis in cooperation with the CEC. Sixteen internationally renowned atmospheric dispersion and deposition experts participated in a NUREG–1150-based formal expert judgment elicitation and evaluation process in which the information needed for the pilot uncertainty study was elicited from the experts. The individual expert assessments were aggregated to form probabilistic distributions for dry deposition velocity, wet deposition parameters, and Gaussian plume dispersion parameters, respectively. The CEC will use the methods formulated jointly during the pilot study, with the NRC providing limited key technical support to the CEC, in obtaining other relevant information to complete a probabilistic consequence uncertainty study. The final data base will be shared by the two organizations for performing probabilistic consequence uncertainty analyses.

5.1.3.3 Risk-Related Training and Guidance Development

Guidance for Staff Use of Risk Analysis. In a July 1991 letter, the NRC's Advisory Committee on Reactor Safeguards identified a number of concerns with the staff's uses of risk analysis. In response, the NRC's Executive Director for Operations formed a working group of staff management to "consider what improvements in methods and data analysis are possible and needed, the role of uncertainty analysis in different staff uses of PSA, ..." This working group was organized in early 1992 with the following objectives:

- To develop guidance on consistent and appropriate uses of PRA within the NRC;
- To identify skills and experience necessary for each category of staff use; and
- To identify improvements in PRA techniques and associated data necessary for each category of staff use.

The group published NUREG–1489 in March 1994, which included initial guidance to the staff on the use of PRA in screening and analyzing reactor operational events and on basic terms and methods used in PRA. The report also contains a number of recommendations for additional guidance development, improvements to the NRC's PRA training program, and improvements in PRA tools and data bases used by the staff. A draft Commission policy statement on the use of PRA as well as an implementation plan has been developed to guide staff response to the recommendations.

5.2 Containment Performance

5.2.1 Statement of Problem

Core–melt accidents that proceed to vessel failure have the potential to produce high pressures and temperatures that might challenge containment integrity. It is known from previous risk studies, and from the experiences at Chernobyl and Three Mile Island, that containment survival or even delayed failure has an all-important effect on minimizing the release of radioactivity to the environment in the event of a core-melt accident. If realistic assessments of the consequences of core-melt accidents, which so strongly depend on whether or when containments might fail in the course of the accident, are to be made, then an understanding of the phenomena that occur in containment in the latter stages of the accident that could lead to containment failure is essential.

5.2.2 Program Strategy

NRC's research efforts in this program element focus directly on the phenomena believed to be most likely to produce high pressures and temperatures that might challenge the containment integrity: high-pressure ejection from the reactor vessel of finely divided particles of molten core debris; generation of noncondensable and flammable gases from the decomposition of concrete by hot core debris; the direct thermal and chemical attack by molten core debris of structures and engineered safety features; and the burning or detonation of hydrogen and other gases produced in the course of the accident.

NRC's research program dealing with reactor containment safety consists of five areas of research. These five research activities address: (1) the interaction of molten core debris with structural concrete, including the ablation of concrete structures, heat transfer to structures in the containment and to overlying water, the generation of noncondensable and flammable gases, and fission products containing aerosols; (2) direct containment heating by molten debris particles ejected from the vessel at high pressure and hydrogen production resulting from steam oxidation of the metallic component of that debris; (3) the combustion of hydrogen in the containment, including the potential for detonation; (4) the development, validation, maintenance, and application of various computer codes that are capable of describing the multiple phenomena that occur in severe accident sequences of interest; and (5) assessment of severe accident phenomena, including containment performance, for advanced reactors (see Section 2.1).

The overall goals of the research are to develop technical bases for assessing containment performance over the range of risk-significant core-melt events, to develop an improved understanding of the range of phenomena expected during severe reactor accidents, and to develop improved methods for assessing fission product behavior. With these kinds of data, the NRC is better able to confirm the adequacy of its requirements for the design and reliability of the systems that may be used for mitigating the effects of severe accidents.

5.2.3 Research Accomplishments in FY 1994

In order to ensure that existing regulations adequately protect the public from the consequences of severe accidents, the NRC conducts research in several areas, among them source term release and transport, core-melt progression, fuel-coolant interactions, direct containment heating, hydrogen combustion, and melt-concrete interactions. The overall goals of the research are to develop (1) technical bases for assessing containment performance over the range of risk-significant core-melt events, (2) an improved understanding of the range of phenomena expected during severe reactor accidents, and (3) improved methods for assessing fission product behavior. With these kinds of data, the NRC is better able to confirm the adequacy of its requirements for the design and reliability of the systems that may be used for mitigating the consequences of severe accidents.

5.2.3.1 High–Pressure Melt Ejection—Direct Containment Heating

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. A molten core left uncooled will drain and relocate to the bottom of the reactor vessel. If the reactor vessel fails, the core melt will be ejected into the containment cavity under pressure. If the material subsequently should be ejected from the reactor cavity into the surrounding containment volumes in the form of fine particles, thermal energy can be quickly transferred to the containment atmosphere pressurizing it. The metallic components of the ejected core debris can further oxidize in air or in steam and can generate a large quantity of hydrogen and chemical energy that would further pressurize the containment. The projected process is called direct containment heating (DCH).

As part of the DCH issue resolution plan for PWRs, a study was completed for the Zion (Ill.) plant and documented in NUREG/CR–6075 and 6075 Supplement 1, "The Probability of Containment Failure by Direct Containment Heating in Zion." Both reports have been peer reviewed and will be published in December 1994.

The culmination of extensive experimental and analytical research undertaken principally for the Zion reactor has produced the finding that DCH loads are significantly lower than once estimated and consequently pose no tangible threat to the containment during a severe accident. This is due primarily to the inherent design characteristics of many U.S. reactors. Future efforts will explore our ability to extrapolate these findings to the spectrum of reactor designs.

5.2.3.2 Hydrogen Combustion

Significant information exists on hydrogen combustion to assess the possible threat to containment and safety-related equipment. Some ancillary issues remain related to a better understanding of the likelihood of various modes of combustion at high temperature and in the presence of large quantities of steam.

The largest current NRC program in this topical area comes out of a joint agreement between the NRC and the Ministry of International Trade and Industry (MITI) of Japan, managed by the Nuclear Power Engineering Corporation (NUPEC). Under the agreement, a high- temperature hydrogen combustion program related to high-speed combustion modes, i.e., detonation and deflagration to detonation transition, is under way at the Brookhaven National Laboratory. A small-scale developmental apparatus was constructed and has provided a preliminary set of experimental data and solutions to a number of design and operational problems for a larger-scale high-temperature combustion facility (HTCF). The construction of the HTCF has been completed and high-temperature experiments begun. As a result of the cooperative agreement with Japan, the NRC has access to the ongoing hydrogen research in Japan managed by NUPEC. This research provides a greatly expanded and improved data base for the validation of analytical tools.

A hydrogen research program is also under way to investigate diffusion flame behavior in low-speed hydrogen combustion. Experiments were performed in a small-scale facility to examine the influence of ignition source strength on the lean flammability limits of hydrogen-air mixtures at temperatures of 300K and pressure of one bar. The facility has been redesigned to eliminate diffusion flame interference with the walls. Construction will be completed during FY 1995. The results will be used to help resolve several outstanding issues in severe accident behavior, such as high-temperature combustion phenomena and detonation initiation by high-temperature steam-hydrogen-particle jets.

Experiments have been conducted to determine hydrogen combustion behavior under conditions of rapidly condensing steam from water sprays. The experimental conditions were nearly prototypical of those that would be expected in a severe accident in the ABB Combustion Engineering System 80+ containment. The mixtures were initially nonflammable because of dilution by steam. The mixtures were ignited by thermal glow plugs as the mixtures became flammable after sufficient steam had been removed by condensation from the water sprays. No detonations or accelerated flame propagation were observed in these tests. The combustion mode was characterized by multiple deflagrations with relatively small pressure rises. The thermal glow plugs were effective in burning the hydrogen safely by igniting the gases as the mixtures became marginally flammable.

5.2.3.3 Melt-Concrete Interactions and Debris Coolability

In those severe accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity where it can thermally and chemically interact with structural concrete. The major areas of concern associated with melt-concrete interactions during a severe accident are the penetration of the basemat and failure of the liner, the generation of radioactive aerosols and gases, including combustible gases, and the overpressurization of the containment.

Early experiments on melt-concrete interactions were conducted without an overlying water pool. More recent experiments on melt-concrete interactions, otherwise known as debris coolability experiments, were conducted in the presence of an overlying water pool. It has been postulated that adding water to cover the core debris will effectively quench the molten corium and terminate melt-concrete interactions. The currently active experimental research on debris coolability, called the Melt Attack and Coolability Experiments (MACE) program, was developed as an extension of the Advanced Containment Experiments (ACE) program under the sponsorship of NRC, EPRI, and other mostly governmental agencies in several countries. The MACE program is intended to determine the ability of water to cool prototypic ex-vessel core debris of urania-zirconia composition. Four tests, including a scoping test, were conducted in the MACE program in FYs 1992 and 1993. The latest MACE test, M3, performed at a scale more than two times larger than the previous tests, was conducted in December 1994. This test was designed to provide information on the effect of scale on crust formation, stability, and debris coolability. Analysis of the test data is under way and will guide the planning of future activities.

5.2.3.4 Severe Accident Codes

Because of the difficulty in performing prototypic experiments for a variety of severe accident scenarios, substantial reliance must be placed on the development, verification, and validation of system-level computer codes for analyzing severe accident phenomena. Several codes (MELCOR, SCDAP/RELAP5, CONTAIN) have been developed for various stages in severe accidents, both in-vessel and ex-vessel, for both BWRs and *PWRs*. Additional codes such as COMMIX, VICTORIA, HMS, and IFCI are being developed and maintained to perform specific functions that require more detailed modeling than the system-level codes.

MELCOR is an integrated computer code that models the progression of severe accidents in light-water reactor (LWR) power plants. The code can be used to evaluate the progression of severe accidents from initiation through containment failure and to estimate severe accident source terms as well as their sensitivities and uncertainties in a variety of applications. The NRC has been supporting the MELCOR development and assessment program for a number of years. The focus of the development efforts in FY 1994 has been to improve capabilities to model the phenomena of in-vessel natural circulation, core structure melting and relocation, and ex-vessel core-concrete interactions, and to model the performance of passive safety systems in advanced light-water reactors (ALWRs). These and a few other improvements were made in response to a number of recommendations made by an independent peer review group convened by the NRC. A significant effort was made in FY 1994 to incorporate CORCON-MOD3, a stand-alone core-concrete interaction code, into MELCOR. With this work now completed, the NRC has no further plan to support maintenance of the stand-alone CORCON-MOD3 code.

The assessment of MELCOR by NRC contractors continued during this report period as did the MELCOR Cooperative Assessment Program. The goal of the latter work, initiated in FY 1992, is to create an international forum for information exchange on the applicability, limitations, and operational experiences with MELCOR. The goal of the former work is to broaden the MELCOR assessment data base through work conducted in the national laboratories and in other organizations in the United States. MELCOR has been applied to the analyses of various plant accident transients, and a large number of code assessments have been completed in FY 1994 by several United States and international user organizations.

SCDAP/RELAP5 is a computer code that has the capability to perform detailed analyses of the in-vessel progression of LWR severe accidents as well as detailed experiment analyses. The code has been in world-wide use for several years. A SCDAP/RELAP5 peer review committee completed an extensive review of the code in FY 1993 and identified several areas of modeling and documentation that needed improvement. The improvements completed in FY 1994 include (1) bringing code manuals up to date, (2) making the code more reliable and user friendly, (3) streamlining the code output, (4) establishing a closer link between the code and associated documentation, and (5) making assessment reports available for each version of the released code. Specific SCDAP/RELAP5 activities completed in FY 1994 include (1) release of the new version of the code (MOD3.1) and a five-volume set of user and code manuals, (2) completion of BWR control blade/channel box model testing, (3) analyses of severe accident sequences for the Zion (Ill.) and Surry (Va.) plants to support resolution of the DCH issue, (4) completion of nodalization studies for a station blackout accident scenario (high-pressure case) for the Surry plant, (5) completion of a TMI-2 accident sequence study using MOD3.1, and (6) addition of debris oxidation and eutectic interaction models for fuel rod cladding and PWR control rod materials. Ongoing work includes (1) continuing to provide code maintenance, (2) performing time and spatial nodalization studies, (3) performing more assessment studies against experimental data, and (4) continuing to improve high-priority modeling deficiency items as recommended by the peer review committee.

CONTAIN is a detailed code for the integrated analysis of containment phenomena. The code provides the capability to predict the physical, chemical, and radiological conditions inside a reactor containment in the event of a severe accident. One issue currently under investigation is DCH and pressurization of the reactor containment atmosphere by molten core materials ejected following the lower vessel head failure under pressure. Assessment of the DCH models in CONTAIN against experimental data continued in FY 1994. Another development is related to containment analyses for ALWR designs. The industry is developing containment designs for ALWRs that incorporate passive cooling and decay heat removal features for protection against long-term containment overpressure in severe accidents. The CONTAIN code was modified in selected areas to model these passive ALWR safety features. Finally, a comprehensive peer review of the code was completed in which code modeling and validation were assessed for the code's intended applications.

COMMIX is a three-dimensional transient single-phase computer code for thermal-hydraulic analysis of single and multicomponent engineering systems. The code solves a system of time-dependent and multidimensional conservation for mass, momentum, energy, and transport equations. A number of phenomena encountered in postulated severe accidents in ALWRs are inherently multidimensional in nature. The COMMIX code is being developed to address issues such as natural circulation, flow stratification, and the effect of noncondensable gas distribution on local condensation and evaporation for the AP–600 plant.

VICTORIA is a computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe accident. The code provides detailed predictions of the fission product release from the fuel and the transport in the RCS of radionuclides and nonradioactive materials during core degradation. During FY 1994, assessment and validation of models used in the VICTORIA computer code against existing data bases and against new data from several experimental test facilities (e.g., FALCON VI, ST) were carried out. An improved fission product chemistry model was implemented in VICTORIA. In FY 1994, the code was used for a full plant station blackout analysis for the Surry nuclear power plant.

Battelle Columbus Laboratory has performed a large number of experiments on boric acid and its interaction with a variety of chemical species that are expected in the RCS under severe accident conditions (e.g., cesium hydroxide, cesium iodide). The experimental results will enhance the chemistry models already in VICTORIA in simulating severe reactor accidents. Battelle and the NRC staff have modeled the retention of cesium on stainless steel in this chemical system for future implementation in the VICTORIA code.

HMS is a best-estimate, three-dimensional, transient code for analyzing the transport, mixing, and burning of hydrogen. The code can model geometrically complex structures with multiple compartments and can simulate the effects of condensation, heat transfer to walls and internal structures, chemical kinetics, and fluid turbulence. During FY 1994, an HMS user's manual was developed to provide the basic information for setting up and running problems with the code. Also, HMS was converted from a main frame computer code to a workstation code.

IFCI, an Integrated Fuel-Coolant Interactions computer code, provides a numerical tool for calculating and predicting the consequences of fuel–coolant interactions, including the breakup of melt streams, the expansion work, and the dynamic pressure loads on in-vessel and ex-vessel structures. Models in the IFCI code are presently being validated against experimental data. During FY 1994, operational assessment of the IFCI code was completed and a user's manual was published (NUREG/CR–6211).

5.3 Severe Accident Phenomenology

5.3.1 Statement of Problem

Major uncertainties in estimating the probability of early containment failure, and the associated radioactive release, in the event of a severe accident appear to be significantly related to uncertainties in the in-vessel progression of the accident while the fuel material remains in the reactor pressure vessel. Better understanding is being gained of the entire sequence of severe accident phenomena, including core-melt progression, fission product release, fuel-coolant interactions, hydrogen generation, and response of the RCS to fuel melting and relocation. Containment failure probabilities and related source terms can now be estimated with less conservatism than in previous analysis to ensure adequate margin.

5.3.2 Program Strategy

In order to better understand just what happens during a core-melt accident, and thereby reduce the uncertainties in both accident behavior and the potential release of radioactivity, the NRC is pursuing a program of research addressing (1) the heatup and meltdown of the core; (2) hydrogen generation; (3) fission product release, transport, and deposition within the RCS; (4) energetic fuel-coolant interactions (FCIs) that may occur as molten debris falls into the water-filled lower head or as water is added to molten debris; (5) the mass composition and temperature of the core debris at the time of vessel (or RCS) failure; and (6) the mode of vessel failure. The overall program is divided into three main activities: (1) the behavior and chemistry of fission products released during core melt, (2) in-vessel core-melt progression, and (3) fuel-coolant interactions. The in-vessel core-melt progression and hydrogen generation work has included in-reactor experiments, out-of-reactor experiments, examination of specimens from TMI-2, and analytical model development. The research on the amount, the chemical form, and the behavior of the fission products released from the fuel in the course of a severe accident requires experiment at high fuel temperatures. Here the core geometry is changing and fission product chemistry and its effect on the retention of fission products within the RCS are significant. The FCI work is focused on the development and validation of appropriate phenomenological and analytical models addressing the fundamental aspects of FCI, namely, melt quenching and FCI energetics.

5.3.3 Research Accomplishments in FY 1994

5.3.3.1 Source Terms

"Source terms" refer to the magnitudes of the radioactive materials released from a nuclear reactor core to the containment atmosphere, taking into account the timing of the postulated releases and other information needed to calculate offsite consequences of a hypothetical severe accident. NRC research in this area is reflected in the updated version of TID–14844, which has been in use for three decades in connection with plant siting assessments. An extensive review of the update of TID–14844 published in draft NUREG–1465, "Accident Source Terms for Light Water Nuclear Power Plants," has been completed, and the final NUREG–1465 report will be issued in January 1995.

The NRC has entered into an agreement with the Commissariat a L'Energie Atomique (CEA) of France to participate in the PHEBUS–FP (fission product) program. The program is sponsored by the CEA and the Commission of the European Communities and is aimed at studying, under sufficiently prototypic conditions in an in-pile facility, those phenomena governing the transport, retention, and chemistry of fission products under severe accident conditions in LWRs. Phenomena to be studied are those occurring in the core, in the primary reactor coolant circuit, and in the containment. This agreement is of significant benefit to the NRC because, at a relatively modest cost, the NRC can participate in the PHEBUS–FP project over its lifetime. The NRC will be able to obtain integral experimental data to further validate its analytical models for fission product transport in the reactor coolant system and containment and for iodine chemistry in the containment. The experimental data from PHEBUS–FP is confirmatory in nature and will be used to assess the revised source term assumptions used in NUREG–1465.

The first PHEBUS–FP test, FPT–0, was successfully conducted in December 1993. The interpretation of FPT–0 is continuing; lessons learned from FPT–0 will be taken into account in planning for the next test, FPT–1, which is scheduled for June 1995.

5.3.3.2 Core-Melt Progression

"In-vessel core–melt progression" describes the state of an LWR core from core uncovery up to reactor vessel melt-through in unrecovered accidents or through stabilization of the temperatures and the core geometry in accidents recovered by core reflooding. Melt progression provides the initial conditions for assessing the loads that may threaten the integrity of the reactor vessel and the containment. Significant results of melt progression are the melt mass, composition, temperature (superheat), and the rate of release of the melt from the core and later from the reactor vessel if vessel failure occurs. Melt progression research also provides information about the in-vessel hydrogen generation, the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system, and the core conditions for assessing accident management strategies.

Current NRC research on melt progression is focused on two major issues. The first issue is determining whether there are any accident conditions for BWRs (and possibly PWRs) in which a metallic core blockage similar to that at Three Mile Island Unit 2 (TMI–2; Pa.) would not be formed. In this case, the metallic melt, and later the ceramic (fuel) melt, would drain when formed from the core into the water in the lower plenum of the reactor vessel. This would lead to release of a mostly metallic (Zircaloy), much lower temperature melt after the reactor vessel boils dry and fails. The second issue concerns the conditions for melt-through of the growing pool of ceramic melt above the metallic blockage. The melt-through threshold and location determine the mass of the melt released from the core and later, potentially, from the reactor vessel.

On the issue of blockage of the core by metallic melt, TMI–2 and the results of in-reactor tests and laboratory experiments have indicated that, for "wet core" conditions (with water in the bottom of the core), the relocating molten metallic Zircaloy in the core freezes to block the lower core. All but one of the previous experiments for both PWRs and BWRs were performed for these wet core conditions, and this one experiment did not address the blockage or drainage question. The emergency operating procedures for U.S. BWRs, however, call for reactor depressurization, which lowers the water level below the reactor core by flashing so that core heatup occurs with very low steam flow through a "dry core." Analysis of these conditions indicates that the molten core metal (and later molten ceramic fuel) might possibly drain from the core rather than forming a blocked core as at TMI–2 with subsequent ceramic melt pool growth and melt-through.

A series of new ex-reactor (laboratory) experiments to address the question of metallic melt drainage or core plate blockage under BWR dry core accident conditions has been started at the Sandia National Laboratories (SNL). The experimental test assemblies are a mockup at full radial scale of a cross section of the lower quarter of a BWR core (and core plate region) where such blockages might occur, and the test assemblies have prototypic reactor fuel rods, structure, and temperature distributions. Separate melts of metallic Zircaloy and control-blade materials are poured into a test assembly at prototypic rates (dribbles), and the melt relocation and blockage behavior are determined.

Last year two developmental XR1 tests of the experimental system were successfully performed. In FY 1994, preparations were made for a series of four XR2 experiments under closely prototypic conditions to provide definitive data on the question of metallic melt drainage or core blockage under BWR dry core accident conditions. A major part of the FY 1994 effort was the development of a new melt delivery system to furnish the required separate pours (dribbles) of control blade and Zircaloy melt at prototypic times, rates, temperatures, composition, and location at the entrance to the test assembly. If technically feasible, and subject to program review, the XR2 experiments are to be performed in FY 1995.

5.3.3.3 Reactor Vessel Integrity

During the late phase of a severe accident, a significant amount of core material may relocate downward into the lower head of the reactor vessel. When this molten core material is relocated into the lower head of the reactor pressure vessel, a molten pool forms and can impose a significant heat load on the reactor vessel lower head. Knowledge of in-vessel and ex-vessel heat transfer phenomena to the lower head is needed to assess the ability of the reactor pressure vessel to maintain its integrity during a severe accident. When a molten pool forms on the lower head, a solid crust of material forms around the periphery of the pool, but internal heat generation resulting from radioactive decay of fission products ensures that most of the debris remains molten and, in fact, undergoes significant internal natural convection in the pool. Detailed understanding of this natural convection process provides information on the local heat flux distribution around the inside surface of the crust. This distribution, in conjunction with the thermal boundary conditions imposed on the outer crust surface, determines the fraction of the total heat dissipation that is transferred through the upper crust to the inside of the reactor vessel by radiative heat exchange and the fraction that must be conducted through the wall of the reactor vessel lower head.

In August 1994, the NRC, in cooperation with 13 countries and under the auspices of the

Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency (NEA), undertook an investigation of melt-vessel interactions to provide data on the internal natural convection flow and local heat flux distribution inside the lower head of the reactor pressure vessel for various melt compositions. This program involves large-scale integral experiments using molten UO_2 and ZrO_2 in representative reactor lower head geometries; analytical studies; and a number of small-scale separate effects experiments. This program, named OECD RASPLAV, is being performed at the Russian Research Center.

In order to remove the fraction of heat conducted through the vessel lower head, the concept of flooding the reactor cavity to externally cool the reactor pressure vessel lower head and prevent its failure is being investigated. One major uncertainty involved in the external cooling of the lower head is the critical heat flux distribution on the bottom curved surface of the reactor vessel. An experimental program is under way at the Pennsylvania State University to address ex-vessel flooding of the reactor cavity to prevent vessel failure. The program investigates boiling heat transfer on downward-facing surfaces in hemispherical and toroidal geometries. The results of this study include data on the critical heat flux (CHF) and the development of an analytical model for the CHF on downward-facing surfaces. The experimental apparatus was designed and built during FY 1994 and a series of transient experiments performed. Further experiments, analyses, and CHF model development will continue in FY 1995.

5.3.3.4 Fuel-Coolant Interactions

There are several aspects of the interaction of ceramic (fuel) melts and of metallic melts with water coolant in the reactor vessel and also ex-vessel in a flooded reactor cavity that are significant in reactor safety assessment. The first of these aspects is the non-explosive breakup and cooling of the melts in water with both steam generation, and, for metallic melts, oxidation and hydrogen generation. The cooling of the melt is significant both in-vessel, for reactor vessel integrity, and ex-vessel. Explosive melt-coolant interactions (steam explosions) have reactor safety significance both for the expansion work and for impulsive shock failure of reactor structures.

Since the quantification of the probability of a steam explosion-induced missile from the expansion work as a possible mode of containment failure (alpha mode) in the reactor safety study, WASH-1400, significant progress has been made in understanding the limitations on the formation of such potential missiles by an in-vessel steam explosion. Alpha-mode failure was not a dominant contributor to early containment failure in NUREG-1150. The emphasis prior to NUREG-1150 in fuel-coolant interaction (FCI) research was on the alpha containment failure mode process of in-vessel molten fuel pouring into lower plenum water and the probability of causing missile generation and containment failure by an energetic interaction (steam explosion). Current emphasis in steam explosion research has shifted to impulsive shock loading of ex-vessel structures. For application of the experimental results on FCIs, an Integrated Fuel-Coolant Interactions (IFCI) code has been developed by SNL.

The NRC and the Safety Technology Institute of the Joint Research Center (JRC) of the Commission of the European Communities at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of experiments in the FARO facility at Ispra on melt breakup and cooling in water. In this facility, a large mass of reactor fuel (and other prototypic reactor core melt materials) is melted and poured into different depths of water at a high pressure that suppresses steam explosion triggering. In the JRC KROTOS facility, steam explosion energetics (including shock impulse) with prototypic melts are also under investigation. Four melt cooling tests have been performed in FARO, one of which, in FY 1994, used 150 kg of UO_2–ZrO_2 melt with 3% zirconium. Steam generation and the melt cooling characteristics have been measured in all these tests. In the KROTOS steam explosion experiments at one atmosphere, a series of tests with tin and with Al_2O_3 and UO_2–ZrO_2 ceramic melts have been performed. Seven of these tests have used UO_2–ZrO_2 melts, five of which were performed in FY 1994. The results are currently being analyzed.

RES has an ongoing program of FCI research at the University of Wisconsin. This research includes (1) simulant material experiments on the mechanisms of both non-explosive and explosive FCIs; (2) technical participation in and analysis of the results of the FARO and KROTOS FCI experiments; and (3) assessment of the IFCI code against the FARO and KROTOS results. During FY 1994, an initial series of experiments with molten tin was completed, and the results were analyzed and interpreted. A principal result was the importance of the fraction of the melt mass that actually interacts with the water in a steam explosion.

An experimental program has been started at the Argonne National Laboratory to determine whether chemical augmentation of the energetics can occur in Zircaloy melt-water steam explosions. Such chemical augmentation can occur in aluminum melt-water steam explosions and has increased the energetics by a factor of up to five. This possible chemical augmentation of the energetics is of particular importance in assessing impulsive shock loads to structures. In FY 1994, detailed experiment planning and construction of the apparatus were performed. The experiments are to be performed in FY 1995.

5.3.3.5 In-House Severe Accident Analysis Capability

Growth in the capability of workstation-level computers provides an opportunity for running severe accident codes on other than main frame computers. In FY 1994, RES purchased workstations to enhance the in-house analysis capability at NRC. Reactor plant descriptions, or decks, for analyses using the MELCOR, SCDAP/RELAP5, CORCON, CONTAIN, and VICTORIA codes have been installed on the workstations. Typical uses of this in-house capability have been to review input decks developed by NRC contractors, using these decks to extend previous analyses. In-house analyses have been used to check new models in the codes and to do bounding calculations to determine the appropriateness of the new models.

5.4 Reactor Containment Structural Integrity

5.4.1 Statement of Problem

The major source of risk to the public from the operation of nuclear power plants stems from accidents that lead to a containment failure. The regulatory concern is that the failure modes and associated load levels for containment structures cannot be predicted with any real confidence by the methods used for design. This is especially so if the contemplated failure mode is localized leakage. Both assessments of the risk posed by loads outside the design basis and estimates of the effectiveness of proposed mitigative steps require an ability to predict the way in which a containment will fail.

5.4.2 Program Strategy

Research on containment failure modes is based on the observation that excessive leakage can occur, basically, from the following sources:

- Failure of the shell, either the containment shell itself (in the case of steel containments) or the liner (in the case of concrete containments);
- Leakage at large penetrations as a result of the inelastic deformations and/or degradation of seals and gaskets;
- Leakage at electrical penetrations due to degradation of materials under the high temperatures associated with accident scenarios; and
- Leakage through valves due to pressure and temperature effects.

Research related to shell failure or deformations of penetrations rests on analyses of and experiments on model tests of actual containment designs. These tests involve pressurization up to failure levels under ambient temperatures. Since seal and gasket materials are adversely affected by the temperatures associated with severe accidents, separate tests focusing on the development of leakage are performed under pressure and temperature conditions, usually at full scale. Examining the possibility of developing leakage through electrical penetration assemblies and valves also requires experiments under temperature and pressure conditions at full scale.

5.4.3 Research Accomplishments in FY 1994

5.4.3.1 Containment Studies

The major undertaking in this program for the next few years will be a cooperative one with the Ministry of International Trade and Industry (MITI) of Japan. Two areas of cooperation have been identified—one dealing with steel containments used in both the United States and Japan for BWR designs, the other relating to prestressed concrete containments. The current generation of Japanese PWR containments are of prestressed concrete designs. In the United States, there are 41 prestressed concrete containments compared to 20 reinforced concrete containments.

A reinforced concrete model was chosen for the NRC-sponsored testing at SNL that was performed in 1987. Subsequent analyses of the results of that model test have shed light on how potential failure modes develop in concrete containments. Some of the results are felt to be applicable to prestressed concrete containments as well. However, there are two main reasons for performing an additional prestressed containment model test:

- Prestressed designs are the most common concrete PWR containment type in the United States as stated above.

- The margin between the ultimate capacity and the design pressure for prestressed concrete containments is now thought to be somewhat lower than that for reinforced concrete or steel containments; hence, it is important to have accurate predictions of the ultimate behavior of prestressed concrete containments.

The steel containment vessel test specimen is a scale model representing some features of an improved BWR Mark II containment vessel in Japan. A scale of 1:10 is used for the overall geometry of the model with 1:4-scaling of the wall thickness. This selection of scales allows the model to be small enough for transportation from Japan to SNL while being thick enough to ensure quality construction.

The model fabrication, under way at Hitachi Works, Hitachi, Ltd., in Japan, was completed in November 1994, and the model will be transported to SNL in January 1995.

The prestressed concrete containment vessel (PCCV) model will be a scaled representation of an actual PCCV in Japan, which was designed in accordance with the Japanese Concrete Containment Vessel Design Code. The actual PCCV consists of a hemispherical dome, a cylindrical wall, and a basemat. Two buttresses are used to anchor the horizontal or "hoop" tendons. In the vertical direction, a "hairpin" tendon layout is employed. The vertical tendons extend from the basemat up through the cylinder wall, over the dome, and back to the basemat on the opposite side of the containment. They are anchored in a tendon gallery that is inside the basemat. A liner plate, which is made of carbon steel, is placed on the inner surface of the concrete wall, dome, and basemat and forms the containment pressure boundary in these areas.

The basic design of the PCCV model will be completed by the end of 1994. Construction drawings will be prepared soon for construction activities at SNL that are scheduled for 1995–1997. Instrumentation of the model will be conducted in 1997–1998, partly in parallel with the onsite model construction. Testing of the PCCV model will then take place late in 1998.

5.4.3.2 Containment Corrosion Studies

Recent experience suggests the possibility that corrosion effects may significantly degrade the margin that containments have to accommodate accidents beyond their design basis. Evidence of corrosion has been found in both Mark I BWR containments and in ice condenser PWR containments. The robustness of containments, as verified in the tests performed at SNL, showing their capacity to sustain loads well beyond design level is a major support for the Commission's Severe Accident Policy Statement. Thus, we need to understand the significant factors relating to occurrence of corrosion, efficacy of inspection, and capacity reduction so as to be able to formulate regulatory requirements that will ensure the continued availability of sufficient margins.

Comparison of remaining thickness against minimum ASME Boiler and Pressure Vessel Code requirements is the obvious first line of assessment. If the remaining thickness exceeds the limit, a decision on adequacy of margin is easy. However, degradation beyond that limit at localized locations does not, by itself, suggest loss of containment capacity. It may be that, unless the degradation is especially severe, failure at some other location will still control. However, the elastic analysis methods used for design cannot be extrapolated to provide estimates of actual failure. Methods, using the results of research on actual failure modes of containments, are being sought that can relate containment capacity to amount and location of degradation. If this effort is successful, a basis can be found for judging the seriousness of a given degree of degradation at a particular location. The Oak Ridge National Laboratory initiated a program during FY 1994 to assess state-of-the-art nondestructive testing techniques for examining steel containments and the liners of concrete containments. As part of this program, statistically based sampling plans will be developed to provide confidence limits on detection of corrosion occurrence. SNL initiated a program during FY 1994 to investigate and develop analytical methods to account for the effects of corrosion on the capability of steel containments to withstand static internal overpressurization loads associated with severe accident conditions.

5.4.3.3 Rulemaking

In order to improve the state of practice in inspection of containments to reduce the chances of having significant undetected degradation due to corrosion, work continued in 1994 on the rulemaking to incorporate by reference Subsections IWE and IWL of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code into 10 CFR 50.55a. Subsection IWE provides rules for the inservice inspection of metal containments and the liners of concrete containments. Subsection IWL provides rules for the inservice inspection of the reinforced concrete and the post-tensioning systems of concrete containments. As written, Subsection IWE and Subsection IWL address only the accessible areas of containments. A provision was included in the proposed rule to address inspection of inaccessible areas in containments. Some of the instances of containment degradation suggest the possibility that degradation may have occurred in inaccessible areas. As noted in a NUMARC (now NEI) report on PWR containments, the state of practice for inspection of inaccessible areas will have to be improved before a resolution of this issue is achieved.

5.5 Severe Accident Policy Implementation

5.5.1 Statement of Problem

A severe accident in a nuclear power plant is an event in which the core is damaged and there is a potential for release of large amounts of fission products. Significant research has been performed on the likelihood, progression, and consequences of a severe accident as discussed earlier. Much of this work has concentrated on the performance of the containment during a severe accident, including potential containment failure mechanisms and the ability of the containment to mitigate the consequences of a severe accident.

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985, the Commission concluded that existing plants pose no undue risk to the public health and safety and that there is no immediate need for generic rulemaking related to severe accidents. However, based on NRC and industry experience with plant-specific probabilistic risk assessments, the Commission recognized the need for a systematic examination of each existing plant to identify any plant-specific vulnerabilities to severe accidents. The policy statement indicated the intent of the Commission to take all reasonable steps to reduce the probability of a severe accident and, should a severe accident occur, to mitigate its consequences to the extent possible. As part of the implementation of the Commission's Severe Accident Policy Statement, the staff has required individual plant examinations (IPEs) of all existing plants to identify any plant-specific vulnerabilities to severe accidents.

Much of the work performed to implement the Severe Accident Policy Statement has focused on research into phenomena that would occur during severe accidents and methods to systematically discover vulnerabilities for severe accidents. This work has shown that the causes and consequences of severe accidents can be greatly influenced by nuclear power plant operators and that many vulnerabilities to severe accidents can potentially be eliminated by proper operator actions. The TMI-2 accident and other abnormal occurrences in nuclear power plants have shown that operators do not stand idle but actively intervene in attempts to control the event. If operators are provided with proper guidance and training to take beneficial actions when needed and, most importantly, refrain from actions that can have adverse effects, the consequences of a severe accident can potentially be significantly reduced. Since many accident management strategies do not involve significant plant design changes, substantial safety benefits can be quickly achieved by ensuring proper operator actions. Thus, the initiation of accident management programs at operating plants is a logical result of the IPE process.

This program element provides for the implementation of the Commission's Severe Accident Policy Statement and the application of the results of severe accident research directly to the regulatory process. Modification of the Commission's rules or policies regarding siting, emergency planning, and containment design are examples of areas in which the results of severe accident research may affect future changes.

5.5.2 Program Strategy

RES has been given the responsibility for the implementation of the IPE. This implementation has involved developing guidance for performance of the IPE, preparing a generic letter to plant operators requesting the IPE, and developing review plans and reviewing the results of the IPE submittals. The requirement to correct any identified plant-specific vulnerabilities not voluntarily corrected will be determined by the backfit rule. Accident management is not required as part of the IPE process but was highlighted in the IPE generic letter as a future requirement that will make use of the results of the IPE process. Severe accident vulnerabilities due to external hazards (e.g., earthquakes, floods, fires) are being considered under the IPE for External Events (IPEEE) program.

5.5.3 Research Accomplishments in FY 1994

In the 14 years since the Three Mile Island (Pa.) accident, the NRC has sponsored an active program in research on severe accidents at nuclear power plants as part of a multifaceted approach to the assurance of safety in this context. Other elements of this approach include improved plant operations, human factor considerations, and probabilistic risk assessments.

The IPE process involves two different efforts. The first is an examination of existing plants for vulnerabilities to severe accidents resulting from events occurring within the plant (e.g., equipment failures, pipe breaks). The second effort is to consider severe accident vulnerabilities from external hazards (e.g., earthquakes, floods, winds). This activity is referred to as the IPEEE.

Fifteen submittals for internal events were received from licensees in FY 1994, making an overall total of 76 submittals received to date. Staff evaluations were issued for FitzPatrick (N.Y.), Surry 1 & 2 (Va.), Millstone (Conn.), Monticello (Minn.), Palo Verde 1, 2, & 3 (Ariz.), Perry 1 (Ohio), Nine Mile 2 (N.Y.), Oyster Creek (N.J.), H. B. Robinson 2 (N.C.), Browns Ferry 2 (Tenn.), McGuire 1 & 2 (N.C.), Catawba 1 & 2 (S.C.),and Haddam Neck (Conn.), and draft staff evaluations were completed for Sequoyah (Tenn.) and Watts Bar 1 (Tenn.). It is expected that all IPE submittals will be received and reviewed by the end of calendar year 1996.

The approach for the review of the IPEEE will follow closely that developed for reviewing the internal-event IPE submittals. The staff completed the procurement process to obtain contractual assistance for the IPEEE reviews. Sixteen complete and four partial IPEEE submittals have been received of which four are being evaluated.

Studies began of the IPE results to gain more generic insights. Issues such as the plant-to-plant variability in estimated core damage frequency results and the reasons for this variability are being examined.

6. SAFETY ISSUE RESOLUTION AND REGULATION IMPROVEMENTS

6.1 Earth Sciences

6.1.1 Statement of Problem

Earthquakes are among the most severe of the natural hazards faced by nuclear power plants. Very large earthquakes would simultaneously challenge the ability of all plant safety systems to function and, coupled with the likely loss of offsite power and dependent safety systems, could pose a unique threat to public safety. As with many potentially severe conditions, there is much uncertainty associated with the design and evaluation of nuclear plants for earthquakes. Seismic hazard in the Eastern and Central United States remains an issue that is not likely to be easily resolved. These regions contain the highest percentage of nuclear power plants in the United States.

Historically, the largest earthquake in the United States has occurred at New Madrid, Missouri. The geology of the central and eastern regions makes it difficult to estimate earthquake magnitudes or seismic parameters for specific locations or to ensure a proper design basis for individual power plants.

The publication of seismic hazard curves in 1989 by both the NRC (NUREG/CR-5250) and the Electric Power Research Institute (EPRI) (NP-6395) marks the end of major efforts to characterize the seismic hazard at U.S. nuclear reactor sites. Although the best information and procedures available were used, they revealed that large uncertainties still remain in seismic hazard estimates. Also, recent full-scope probabilistic risk assessments, performed as part of the NUREG-1150 effort, continue to show that seismic hazard uncertainties contribute significantly to the overall uncertainty in nuclear reactor risk estimates. These large uncertainties make it difficult to place the contribution of seismic risk into its proper perspective, e.g., in the development of individual plant examination guidelines.

Recent successes in the geological, geophysical, and seismological studies sponsored by RES show that it is possible to answer the basic scientific questions that underlie these seismic hazard uncertainties. It is the goal of the NRC earth science research program to significantly reduce the uncertainty in seismic hazard estimation in the next decade through emphasizing this type of research.

6.1.2 Program Strategy

The strategy to resolve the seismic problem involves research to develop the methods and data that will support the necessary seismic criteria development and provide the evaluation tools. The research is focused on (1) improving estimates of earthquake hazards by identifying potential earthquake sources and determining the propagation of seismic energy with distance, (2) estimating the possible range and likelihood of seismic ground motions at nuclear plant sites, and (3) assessing the effect of these ground motions on soil, structures, equipment, and systems of the plants. The integrated results of this research will be used to quantify the risk to nuclear plants from earthquakes, to assess the seismic safety margins inherent in current or future plant design, and to help identify and set priorities for what improvements are needed in plant designs or what parts of seismic design criteria may be relaxed.

A major focus of the NRC research programs in geology, seismology, and geophysics continues to be identifying and defining potential earthquake sources or source zones in the Eastern United States and using that information in assessing seismic hazards with respect to nuclear power plants. Many unknowns exist regarding these issues, including a strong basis for seismic zonation, source mechanisms, characteristics of ground motions, and site-specific response. The NRC is addressing these uncertainties through research that encompasses sustained seismic monitoring, geologic and tectonic studies, neotectonic investigations, exploring the earth's crust at hypocentral depths, and conducting ground motion studies.

The backbone of the NRC program in the Eastern United States has been the seismographic networks deployed throughout the Eastern and Central United States. The NRC is currently funding seismographic networks in the following regions: Northeastern United States; Virginia; Charleston, South Carolina; the southern

Appalachian region; the New Madrid (Missouri) region; Ohio and Indiana; eastern Kansas; and Oklahoma. An agreement was reached in 1986 between the United States Geological Survey (USGS) and the NRC to jointly support the establishment of the eastern portion of a national seismographic network. The eastern portion of the national network is now fully in place.

6.1.3 Research Accomplishments in FY 1994

The objective of NRC research in earth sciences, as related to reactor regulation, is to define potential earthquake ground motions at nuclear power plant sites. This information provides a basis for evaluating the effects of earthquakes on the plants and their safety systems.

Seismic hazards contribute significantly to overall plant hazards and, because of inherent difficulties in defining the seismic hazards, they form an even more significant portion of the overall uncertainty in estimating plant hazards. In order to reduce these uncertainties, research into the causes and distribution of seismicity is continuing. Research is also progressing on improved methods of applying earth science information to estimates of ground motion levels for use in plant design.

6.1.3.1 Seismographic Networks

The new National Seismographic Network (NSN) was established through a cooperative agreement between the NRC and the USGS. Including cooperative stations, the NSN operates 32 broadband three-component stations and satellite telemetry, thus providing data on significant earthquakes within minutes. During FY 1994, a broad agency announcement was issued soliciting research proposals to analyze NSN data and other available seismological, geological, and geophysical data. A number of proposals were received, resulting in five research contracts. This research will continue the type of investigations previously carried out by the universities operating regional networks. The high-quality, broadband, and three-component data of the NSN will lead to new insights into the causes and distribution of seismicity and the ground motion propagation characteristics of the earth's crust, particularly in the Eastern and Central United States.

6.1.3.2 Southeastern Tectonics

A continued search for possible liquefaction features and investigations of landslides and cave deposits in the southern Appalachian area did not produce any clear evidence of prehistoric earthquakes. This activity concluded the search for evidence of prehistoric earthquakes in the Southeast outside of the Charleston, S.C., area. While the search did not result in positive evidence, the fact that clearly liquefiable deposits outside the coastal area did not show evidence of liquefaction reinforces the impression that the Charleston area is unique with respect to its seismicity.

In an earlier investigation of paleoliquefaction features on the Atlantic Coastal Plain, based on evidence for a prehistoric event that occurred about 70 kilometers north of Charleston, S.C., about 1800 years ago and on the absence of evidence for that event in the Charleston meizoseismal area, investigations have been started to determine whether there is more than one seismic source for a large earthquake in South Carolina.

6.1.3.3 Paleoseismicity of Southern Illinois and Indiana

Previous liquefaction studies have found indications that a large earthquake, centered near Vincennes, Ind., occurred between 2,500 and 7,500 years ago. This earthquake may have been larger than the 1886 Charleston earthquake (magnitude = approximately 7.0) but smaller than the 1811–1812 New Madrid earthquakes (magnitude = approximately 8.0). During this report period, investigations of the Wabash drainage system were extended farther into Illinois and Indiana and also into the Anna, Ohio, seismic area. Evidence for another, smaller prehistoric earthquake was found in the Vincennes region, but no evidence for prehistoric events large enough to cause liquefaction was identified in the Anna, Ohio, area. The results thus far confirm the conservatism of past licensing decisions made regarding the seismic hazard at nuclear power plant sites in this region.

6.1.3.4 New Madrid Seismic Zone

Paleoliquefaction studies are being conducted at several locations in the New Madrid seismic zone, particularly near the Missouri-Arkansas state line, to determine the ages and extent of prehistoric earthquakes. Evidence, both geological and archaeological, indicates the occurrence of at least two prehistoric events.

Seismic reflection investigations are being partially supported by the NRC in the New Madrid seismic zone in the area where waterfalls appeared in the Mississippi River during the 1811–1812 earthquakes. The studies are investigating the possibility that these short-lived features were caused by faulting associated with those earthquakes.

The investigations are part of the ongoing effort to estimate the recurrence of the large-to-great earthquakes (magnitudes 6 to 8) in the New Madrid seismic zone and to define the causative faults.

6.1.3.5 West–Central United States

Three suggested Quaternary faults are being investigated, namely the Cheraw and Fowler faults on the Colorado Piedmont and the Harlan County fault in Nebraska. Preliminary results indicate that the Fowler feature is not a fault but appears to be one because of the current setting of the Quaternary stratigraphy, geomorphic features, and jointing in the underlying Pierre Shale. The Cheraw and Harlan County faults appear to be Quaternary tectonic faults and will be investigated further. The investigations are part of an ongoing effort, which began with the discovery of late Holocene displacement on the Meers fault, to identify other Quaternary faults in the Central United States.

6.1.3.6 Fault Segmentation Studies

During the past 2 years, the surface rupture that occurred during the 1992 Landers earthquake was studied in detail, and during the last half of FY 1994 the surface deformation that occurred during the 1994 Northridge earthquake was investigated. Geological evidence from faults that ruptured the ground surface at Landers indicates that from two to four prehistoric earthquakes occurred in this area. These events are estimated to have been about the same size as the 1992 event, suggesting that the Landers earthquake fault segments behaved in a similar manner in the past. These findings confirm the validity of using fault segmentation to estimate future fault behavior and earthquake magnitude, which was one of the methods used to estimate the seismic hazard of the Diablo Canyon nuclear power plant site.

Preliminary results indicate that the surface deformation at Northridge is most likely the result of strong ground motions and secondary faulting. Evidence pointing to prehistoric events of a similar kind was also found.

6.1.3.7 Strong Ground Motion Studies

The NRC supports several cost-sharing ground motion programs in cooperation with the USGS. Among these are studies estimating high-frequency ground motions in the Eastern and Central United States, using data from the Landers, Petrolia, and Northridge earthquakes; attenuation and source parameter studies for the Eastern and Central United States, using data from the NSN; and strong ground motion studies of large intraplate earthquakes, using data from teleseismic and regional recordings. The results of these studies will be used to make more realistic hazard determinations and to reduce uncertainties in the probabilistic estimates.

6.1.3.8 Geochronological Studies

The NRC is supporting a research program to assemble all state-of-the-art information on methods for determining the age of geological materials. Geochronological analysis of faults, paleoliquefaction features, and other paleoseismic features are an important part in determining the seismic and geological hazard of a site. A limited field research project to validate new dating methods is under way as a part of this project. The goal of this project is to develop a regulatory guide to assist applicants and the regulatory staff in evaluating potential nuclear sites.

6.1.3.9 Crustal Strain Measurements

A 45–station crustal strain network for the Eastern and Central United States was established in 1987 and measured for the third time during FY 1993. After this strain network was established, it became the backbone of a new geodetic network for the United States based on Global Positioning System (GPS) measurements. In the meantime, high-precision GPS networks have been established in many states, and within the next few years all of the United States will be covered with detailed high-precision GPS networks, including continuously operated GPS stations. From preliminary strain analyses, it appears that strains in the Eastern and Central United States are in the range of 10^{-8} per year. Because strain rates in this region are so low, many years may be needed to arrive at meaningful strain determinations. However, with the many high-precision GPS stations now available, it should eventually be possible to get a very detailed picture of strain distribution. Information on strain distribution and strain rates will then provide a basis for refinements in seismic hazard determinations.

6.1.3.10 Probabilistic Seismic Hazard Assessments

During FYs 1993 and 1994, a panel of scientists, assembled under the sponsorship of the NRC and the Department of Energy (DOE) and with input by the Electric Power Research Institute (EPRI), conducted a study of probabilistic seismic hazard analysis (PSHA) methodologies that is now nearing its end. The study has the goal of analyzing existing methodologies, namely those developed by Lawrence Livermore National Laboratory and by EPRI under the NRC and nuclear utility sponsorship, respectively, and of deriving an improved methodology that will be scientifically balanced and usable for regulatory decisions over the next decade. A draft of the final report was completed in November 1994, and a preliminary review meeting was held with the National Academy of Sciences review panel in December. The study is being peer reviewed by a panel appointed by the National Academy of Sciences to ensure impartiality and objectivity. Considerable weight has been placed on methods of eliciting expert opinions, which are of fundamental importance in probabilistic hazard estimates. After the final report for this study is issued, the new PSHA methodology will be verified with a limited amount of site testing.

6.2 Plant Response to Seismic and Other External Events

6.2.1 Statement of Problem

In the 1970s and before, our interest in nuclear plant seismic design was mainly limited to response at design levels (e.g., operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE)) and our knowledge of this was primarily based on analytical techniques and assumptions.

In the 1980s, a considerable effort was made to better predict the potential response of nuclear plants to earthquakes greater than those considered in design. Our understanding has been increased greatly by the testing to failure of equipment and structures, by the gathering and synthesis of earthquake experience data from non-nuclear facilities, and by the large number of seismic probabilistic risk assessments (PRAs) that have been made.

This research has generally found that the seismic capacity of important nuclear plant structures and equipment (when properly anchored) is high. But there remain specific capacity concerns that need to be resolved, such as how to address the potentially harmful effects of relay chatter. The importance of plant-specific walkdown reviews to find nongeneric vulnerabilities has been noted in recent seismic margin studies.

6.2.2 Program Strategy

In recent years, the NRC has supported seismic testing and the collection of earthquake experience data in order to improve and gain confidence in the use of seismic PRAs and seismic margin studies. These data are also being used to support proposed improvements to seismic design criteria. The earthquake resistance of structures, equipment, and piping has been found, in general, to be higher than previously thought. Major efforts in this area were completed in 1990, and the results are being successfully used in licensing actions. Relay chatter is the one remaining seismic capacity issue that will require additional testing to resolve.

Upcoming individual plant examinations and USI A-46 seismic reviews will use the recent results of NRC seismic research.

6.2.3 Research Accomplishments in FY 1994

6.2.3.1 Revision of Appendix A to 10 CFR Part 100

In August 1994 the Commission approved the staff's recommendation to issue a second proposed revision of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," for public comment. This revision reflects new information and research results available since the first proposed revision to the regulations was issued and comments were received from the public on that proposed revision. The second proposed revision to the regulations was published in the *Federal Register* for a 120-day comment period on October 17, 1994 (59 FR 52255). Draft regulatory guides and standard review plan sections providing methods acceptable to the NRC staff for implementing the proposed regulations were issued for public comment in February 1995.

6.2.3.2 Seismic Analysis of Piping

Activities on the program to review significant changes being proposed on portions of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, continued in FY 1994. The workscope includes the evaluation of completed and ongoing initiatives; for instance, results of the EPRI/NRC piping program, Advanced Reactor Corporation and IPIRG activities, NRC staff positions associated with advanced light-water reactor reviews and elimination of the OBE response analysis, and interaction with a peer review panel of nationally recognized experts.

6.2.3.3 Cooperative International Seismic Programs

The NRC's participation in international seismic test programs is beneficial both for the sharing of research resources and for gaining different perspectives on seismic design issues. The pooling of resources allows the development of larger-scale tests, an important element in the validation of methods for predicting the seismic response behavior of nuclear plant systems.

The Large-Scale Seismic Test (LSST) facility is one of the largest in the world for soil/structure interaction (SSI) research. The construction of a 1/4-scale model of a reinforced concrete containment, 10.5 meters in diameter and 16.5 meters high (11.1 meters above the ground), was completed in March 1993 and a formal dedication ceremony was held in Hualien, Taiwan, in April 1993.

The LSST program was initiated in January 1990, and it is expected that it will continue for 5 years. The goal of this program is to collect real earthquake-induced SSI data in order to evaluate computer codes used in SSI analyses of nuclear power plant structures. Observations will be made on the motions of the reactor building model and the surrounding ground during large-scale earthquakes. The expectation is that the test model will be shaken by numerous earthquakes in this seismically active area of Taiwan.

To date there have been several earthquakes recorded at the LSST site—one on September 16, 1993 (4.1 magnitude) and one on January 20, 1994 (5.7 magnitude). Instrumentation located on the scale model and in the field along a three-dimensional strong ground motion array recorded the recent earthquake data. The LSST program at Hualien, Taiwan, is a follow-on to the SSI experiments at Lotung, Taiwan.

EPRI has organized the Hualien LSST experiment and coordinated participation with the Taiwan Power Company (Taipower), the NRC, the Central Research Institute of Electric Power Industry (CRIEPI), the Tokyo Electric Power Company (TEPCO), the Commissariat a l'Energie Atomique (CEA), Electricite de France (EdF), Framatome, the Korea Power Engineering Co. (KOPEC), and Korea Electric Power Corp.

During this report period, a collaborative effort involving exchange of technical information was established with the Ministry of International Trade and Industry (MITI) and Nuclear Power Engineering Corporation (NUPEC) of Japan. In this effort, NUPEC is carrying out a seismic proving test program for a main steam line typical of the PWR plants and a feedwater system typical of the BWR plants. Preliminary tests have begun at the shake table of Tadotsu Engineering Laboratory and will continue in 1995. Tests will be conducted at several levels of seismic excitation, using energy absorber supports for the piping systems. The NRC in this collaborative effort will carry out pre- and post-test analyses to assess the applicability of currently available analytical models. Data are also being obtained from NUPEC for seismic proving tests of a computer system and a reactor shutdown cooling system.

6.2.3.4 Northridge Earthquake

On January 17, 1994, a magnitude 6.7 earthquake occurred in the San Fernando Valley near the town of Northridge, California. This is the same general area affected by the magnitude 6.5 San

Fernando earthquake in 1971. Representatives from the NRC Offices of Nuclear Regulatory Research and Nuclear Reactor Regulation and from an NRC contractor, Lawrence Livermore National Laboratory, toured the damaged area. The NRC, other government agencies, and the nuclear industry continue to study the effects of such earthquakes to improve knowledge of the causes, frequency, and severity of earthquakes, seismic wave transmission, local site amplification, seismically caused soil failure, and performance of structures, equipment, and piping similar to that found in nuclear power plants. Although there were many failures associated with this earthquake, many structures, systems, and components, even those close to the epicentral region, had little significant damage and could be occupied or were functional after the earthquake. In general, well-engineered structures and equipment that may have experienced ground motion far in excess of their design remained functional. Components made of brittle materials, such as ceramic insulators and cast iron components, received significant damage consistent with that observed after other earthquakes.

6.2.3.5 Shear Wall Ultimate Drift Limits

The ultimate drift limit is defined as the lateral displacement at the top of the wall relative to its base normalized by the height of the wall. A research program with the objectives of establishing appropriate values of ultimate drift limits and obtaining the statistics to define this parameter in a probabilistic sense was completed this year. The final report, NUREG/CR-6104, "Shear Wall Ultimate Drift Limits," was published in March 1994. The information in this report will be useful in the seismic PRA or seismic margins analyses done to identify seismic vulnerabilities to severe accidents (in compliance with Generic Letter 88–20, Supplement 4).

6.3 Generic Safety Issue Resolution

6.3.1 Statement of Problem

In order to ensure the timely resolution of important safety concerns raised by the staff and outside sources, the Commission directed the NRC staff to prepare a priority list of all generic safety issues, including TMI-related issues. The list was to be based on the potential safety significance and cost of implementation of each issue. In December 1983, the original listing and procedures were approved by the Commission. This guidance is reflected in the NRC Policy and Planning Guidance, the NRC Strategic Plan, and the NRC Five-Year Plan.

6.3.2 Program Strategy

A generic safety issue (GSI) is one that involves a safety concern that may affect the design, construction, or operation of all, several, or a class of reactors or facilities and may have a potential for safety improvements and issuance of new or revised requirements or guidance. Timely resolution of these issues is a major NRC concern. A prioritization and management process has been established for identifying important issues for immediate action, for eliminating non-safety-related or non-cost-effective and duplicate issues from further consideration, and for keeping the Commission and the public informed of the resolution of these issues. Strategies for this program are to provide timely prioritization of proposed new GSIs, eliminate the backlog of proposed issues (as resources permit), and issue periodic updates on the status and progress toward resolution of GSIs.

6.3.3 Research Accomplishments in FY 1994

6.3.3.1 Priorities of Generic Safety Issues

The NRC continued to use risk and cost data in implementing its methodology set out in the 1982 NRC Annual Report for determining the priority of GSIs. In December 1983, a comprehensive list of the issues was published in "A Prioritization of Generic Safety Issues" (NUREG–0933), and this list has been generally updated semi-annually with supplements in June and December. The results of the NRC's continuing effort to identify, prioritize, and resolve GSIs will be included in future supplements to NUREG–0933.

During FY 1994, the NRC identified no new GSIs, established priorities for three issues (Table 6.1), and resolved five GSIs (Table 6.2). Table 6.3 contains the schedules for resolution of the 14 unresolved GSIs at the end of FY 1994.

Table 6.1 Generic Safety Issues Prioritized in FY 1994

Number	Title	Priority
158	Performance of Power-Operated Valves Under Design Basis Conditions	MEDIUM
165	Spring-Actuated Safety and Relief Valve Reliability	HIGH
167	Hydrogen Storage Facility Separation	LOW

Table 6.2 Generic Safety Issues Resolved in FY 1994

Number	Title
57	Effects of Fire Protection System Actuation on Safety-Related Equipment
106	Piping and Use of Highly Combustible Gases in Vital Areas
B–64	Decommissioning of Nuclear Reactors
I.D.5(3)	On-Line Reactor Surveillance Systems
II.H.2	Obtain Technical Data on the Conditions Inside the TMI–2 Containment Structure

Table 6.3 Generic Safety Issues Scheduled for Resolution

Issue Number	Title	Priority	Scheduled Resolution Date
15	Radiation Effects on Reactor Vessel Supports	HIGH	TDB
23	Reactor Coolant Pump Seal Failures	HIGH	12/95
165	Spring-Actuated Safety and Relief Valve Reliability	HIGH	06/98
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	09/95
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM	06/95
158	Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions	MEDIUM	04/96
B–17	Criteria for Safety-Related Operator Actions	MEDIUM	06/95
B–55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	08/97
B–61	Allowable ECCS Equipment Outage Periods	MEDIUM	12/94
83	Control Room Habitability	NEARLY RESOLVED	06/95
145	Improve Surveillance and Startup Testing Programs	NEARLY RESOLVED	04/95
155.1	More Realistic Source Term Assumptions	NEARLY RESOLVED	02/95
166	Adequacy of Fatigue Life of Metal Components	NEARLY RESOLVED	TBD
168	Environmental Qualification of Electrical Equipment	NEARLY RESOLVED	TBD

6.3.3.2 Progress on GSI Resolution

Important information has been developed this year on the factors contributing to blockage of ECCS strainers in BWR suppression pools. Although this issue had been resolved in 1985 as A–43, more recent events at Barseback Unit 2 in Sweden and at the Perry (Ohio) nuclear plant showed that LOCA-caused fibrous insulation debris coupled with sludge and foreign materials in the drywell could block strainers more rapidly than previously thought. Restudy of this issue in the United States is considered prudent because of the changeout of mirror insulation to fibrous insulation by many utilities since the A–43 resolution, which was based on the presence of mirror insulation.

The potential for BWR ECCS strainer blockage due to LOCA-generated debris was studied in detail using a BWR6/MK1 reference plant to estimate the probability of occurrence and attendant impacts on net pump head suction (NPHS) margin. The results, reported in NUREG/CR–6224, revealed that severe strainer blockage and loss of NPHS margin could occur within the first 30 minutes of a loss-of-coolant accident (LOCA) if other materials or particulates such as suppression pool sludge are present in addition to LOCA-generated debris. Studies have also been started at Alden Research Laboratory to develop a broader experimental data base of strainer blockage, particulate transport, and settling phenomena under simulated post-LOCA suppression pool turbulence conditions for a variety of fibrous debris and sludge configurations for broader applicability to all BWRs. The NRC is also participating in an international working group sponsored by the OECD/NEA–CSNI—Principal Working Group 1—whose charter is to develop an internationally agreed-upon knowledge base for assessing the reliability of ECCS recirculation systems, especially as related to strainer blockage.

6.4 Reactor Regulatory Standards

6.4.1 Statement of Problem

RES has the primary responsibility to manage, coordinate reviews of, and control all NRC reactor-related rulemaking activities and to monitor scheduling of such rulemaking to ensure that rules are developed in a timely manner. In addition, RES provides support for preparation of the regulatory impact analyses (RIAs) that accompany all rulemaking through the development of generic methodology and guidance. Technical reviews of all RIAs are performed upon request. The NRC Regulatory Agenda Report and other management information systems associated with rulemaking activities are maintained.

Needed reactor-related regulatory products, e.g., regulations and regulatory guides, are developed. Rulemaking is proposed or initiated, as appropriate, and complex rulemakings that span the technical or organizational responsibilities of several groups or that involve novel or complex questions of regulatory policy are managed. Petitions for rulemaking are investigated.

6.4.2 Program Strategy

The purpose of this program is to ensure that nuclear reactor facilities are designed, constructed, and operated in a safe manner. Therefore, a continuing need exists to revise rules and guides and to develop new ones. The strategies of this program are to (1) review the effectiveness of LWR regulatory requirements and guidance and make recommendations for revisions; (2) develop screening methodology to systematically review requirements and guidance; (3) coordinate and review proposed changes to the IAEA safety standards; (4) develop or assist the development of rules and regulatory guides; and (5) continue to develop and maintain management information systems for rulemaking.

6.4.3 Research Accomplishments in FY 1994

6.4.3.1 Elimination of Requirements Marginal to Safety

The NRC has institutionalized an ongoing effort to eliminate requirements marginal to safety and reduce regulatory burden by permanently integrating that activity into the regulatory process. This will satisfy the requirement for a periodic review of existing regulations in Section 5 of Executive Order 12866, "Regulatory Planning and Review." The regulatory improvement (RI) program is intended to implement the principle adopted by the Commission that all regulatory burdens must be justified and that NRC's regulatory process must be efficient. The reasons for seeking to remove regulations and license conditions marginal to safety are to eliminate or modify requirements where burdens are not commensurate with their safety significance and thus to free up licensee and NRC resources and improve the focus and effectiveness of the body of regulations. The activities in this program should result in enhanced regulatory focus in safety-significant areas. As a result, an overall net increase in safety is expected from the program. Specific policies, framework for rulemakings, and procedures for the program have been instituted.

As a major action for the RI program, the NRC will propose a revision to its regulations in Appendix J to 10 CFR Part 50 concerning containment leakage testing. Consistent with the policies and framework established for the RI program, the proposed rule is formulated to adopt performance-oriented and risk-based approaches, is less prescriptive, and allows licensees flexibility for cost-effective implementation of the safety objectives in the regulation. The revision would permit greater intervals between required tests, provided that satisfactory performance is achieved on preceding tests. The nuclear industry has supported this activity through the collection of data at nuclear power plants and has developed a guideline for implementation of the rule. This rule revision is expected to result in greater focus on safety-significant activities and a significant burden reduction to the industry.

The NRC has also initiated action and studies for revising its regulations for fire protection of power reactors under Appendix R to 10 CFR Part 50. The NRC is currently conducting a review of initiatives for performance-oriented fire protection regulation in other industries in the United States and abroad and in the nuclear industry in other countries. The NRC is also developing the application of PRA for determining the significance of fire protection features and enhanced focus on fire protection design activities. The nuclear industry is playing a major role in this rule revision and is expected to submit a petition for rulemaking to adopt performance-oriented approaches in the fire protection area.

The NRC is also in the process of revising 10 CFR 2.802, "Petition for Rulemaking," to implement another aspect of regulatory improvement. The proposed revision would offer an alternative beyond the minimum threshold information required by the current 10 CFR 2.802(c) to encourage any petitioner to submit more detailed information and analyses to support the petition. This information would be of the same type as that currently required to be developed by the NRC staff for rulemaking. The revision is expected to expand the use of the petition process by reducing or eliminating requirements that impose a regulatory burden with no commensurate safety benefit and to result in faster disposition of the petitions, as well as more efficient use of NRC staff and industry resources.

6.4.3.2 Other Rulemaking

The Commission issued a final rule on February 9, 1994 (59 FR 5934) on requalification requirements for licensed operators for renewal of licenses (10 CFR Part 55). The amendment deletes the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's 6-year license as a prerequisite for license renewal. Forty-two comments were received, the majority of which supported the proposed amendments.

The Commission issued a proposed rule for public comment on October 24, 1994 (59 FR 53372) on procurement of commercial-grade items by nuclear power plant licensees (10 CFR Part 21). It is expected that the final rule will be published in FY 1995. The proposed amendments would clarify and add flexibility to the process of procuring commercial-grade items for safety-related service by nuclear power plant licensees. The proposed rule responds to a petition for rulemaking (PRM–21–02) submitted by the Nuclear Management and Resources Council (NUMARC), which is now incorporated into the Nuclear Energy Institute (NEI).

The Commission issued a proposed rule for public comment on November 2, 1994 (59 FR 54843) on reduction of reporting requirements imposed on NRC licensees (10 CFR Parts 50, 55, and 73). The amendments would reduce reporting requirements currently imposed on water-cooled nuclear power reactor, research and test reactor, and nuclear material licensees. This action would reduce the regulatory burden on NRC licensees and implements an NRC initiative to review its current regulations with the intent to revise or eliminate duplicative or unnecessary reporting requirements. It is expected that the final rulemaking will be issued late in FY 1995.

The Commission issued an advance notice of proposed rulemaking (ANPR) on November 3, 1993 (58 FR 58664) on standard design certification for evolutionary light-water reactors (10 CFR Part 52). The ANPR requested public comment on the form and content of rules that would certify these designs. The Commission anticipates that two applications for design certification may be ready for such rulemakings in FY 1995. An applicant for a combined license under 10 CFR Part 52 can use these certified designs without further indepth review by the NRC.

The Commission in SECY-94-042 approved withdrawal of six NRC policy statements that have been superseded by subsequent NRC rulemaking actions. The decision for their withdrawal would not change reporting requirements on licensees or in any way reduce the protection of the public health and safety. The policy statements to be withdrawn are: (1) Nuclear Power Plant Access Authorization Program, March 9, 1988 (53 FR 7534); (2) Training and Qualification of Nuclear Power Plant Personnel, March 20, 1985 (50 FR 11147); (3) Fitness-for-Duty of Nuclear Power Plant Personnel, August 4, 1986 (51 FR 27921); (4) Maintenance of Nuclear Power Plants, December 8, 1989 (54 FR 50611); (5) Information Flow, July 20, 1982 (47 FR 31482); and (6) Planning Basis for Emergency Responses to Nuclear Power Reactor Accidents, October 23, 1979 (44 FR 61123). A notice of withdrawal of these policy statements will be published in the *Federal Register* early in FY 1995.

6.4.3.3 Regulatory Analysis

The NRC continued its development of the regulatory analysis guidelines (NUREG/BR-0058, Rev. 2) and the regulatory analysis technical evaluation handbook (NUREG/BR-0184). The guidelines represent the NRC's policy-setting document with respect to regulatory impact analyses (RIAs). The document contains a number of policy decisions for the preparation of RIAs performed to support NRC actions affecting reactor and nonreactor licensees. The accompanying handbook provides methodological guidance to regulatory analysts, promotes preparation of high-quality RIAs, and implements the policies of the guidelines. During this report period, the guidelines were revised in response to public comments, and the handbook was modified as a result of internal NRC reviews. In addition, the NRC continued its re-evaluation of the current $1000 per person-rem conversion factor, which is integral to the value-impact assessment portion of the RIA. This paper has been subject to a number of NRC internal reviews.

Also to aid analysts in preparing RIAs, the NRC published NUREG/CR-6080, "Replacement Energy, Capacity, and Reliability Costs for Permanent Nuclear Reactor Shutdowns," and NUREG/CR-5344, "Replacement Energy Cost Analysis Package (RECAP): User's Guide." The cost estimates available from these studies allow the NRC to estimate the costs associated with the temporary shutdown of a nuclear power reactor in order to make safety modifications or its permanent loss due to an accident.

During this report period, the development or review of about 18 safety-related RIAs was completed or initiated to justify specific regulatory actions for reactor and nonreactor licensees.

6.5 Radiation Protection and Health Effects

6.5.1 Statement of Problem

The NRC must provide reactor-related radiation protection standards and guidance that ensure that workers and members of the general public are adequately protected from the adverse consequences of exposure to ionizing radiation from licensed activities. RES reactor-related activities needed to support the program include developing radiation protection standards; developing guidelines for implementing these standards; and planning, developing, and directing safety research to provide the information necessary for licensing decisions, inspection and enforcement activities, and the standards development process. This includes analyzing available scientific evidence to evaluate the relationship between human exposure to ionizing radiation and radioactive material and the potential occurrence of both late and early radiogenic health effects, including the radiation risk to workers and the public, and estimates of the probability of increased incidence of cancer and genetic effects. These analyses are used to provide bases for severe accident consequence analysis, probabilistic risk assessment (PRA), the development of safety goals and emergency plans, the identification of radiation protection problems, the allocation of priorities for regulatory action, and environmental impact assessments. Recommendations of such organizations as the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP), Presidential guidance to Federal agencies, consensus standards, licensee performance indicators, cost and feasibility data, and available technical information also provide bases for developing regulatory and technical documents related to radiation protection for workers and the public.

6.5.2 Program Strategy

The Commission's regulatory process requires that safety enhancements to reactor rules and guidance be systematically screened to ensure that there is substantial increase in public protection and that based on analysis the costs are justified. Realistic values of the dollar-per-person-rem criterion are needed for analysis to justify changes, but gaps in knowledge associated with radiation health effects cause uncertainties in these analyses. The strategies of this program are to identify and compensate for uncertainties in radiation risk coefficients used for health effect estimates in PRAs and regulatory decisions.

When the Commission approved the whole body dosimetry accreditation rule, they directed the NRC staff to extend the rulemaking to include extremity dosimetry. Therefore, the strategies of this program are to (1) improve regulatory performance for radiation protection by establishing measurement performance criteria and accreditation programs in the areas of extremity dosimetry, bioassay, and air sampling; (2) investigate effective new measurement techniques for these areas; (3) establish the data base required for regulations; and (4) monitor specific indicators to detect improving and declining licensee performance.

Federal guidance was approved by the President on occupational radiation protection. Further, the ICRP has published new recommendations for radiological protection. As a result of this new guidance, NRC reactor regulations and regulatory guides will have to be revised. The strategies of this program are to (1) modify radiation protection guidance and standards to be consistent with Presidential guidance on radiation protection requirements and (2) continue to monitor licensee performance indicators by using the Radiation Exposure Information Reporting System program.

6.5.3 Research Accomplishments in FY 1994

The NRC maintains a program of research and standards development in radiation protection and health effects intended to ensure continued protection of workers and members of the public from radiation and radioactive materials in connection with reactor licensed activities. The program is currently focused on improvements in health physics measurements, identification and dissemination of cost-effective dose reduction techniques, assessing health effects consequences of postulated reactor accidents, and monitoring health effects research.

6.5.3.1 Revision of Part 20 Radiation Protection Standards

Staff efforts to facilitate the mandatory implementation of the new rules continued through FY 1994. These efforts included development of training courses, publication of questions and answers on Part 20, and publication of regulatory guidance. On January 1, 1993, the rule became mandatory for all licensees. Thus, activities have also focused on issues raised by inspections. In February 1994, the staff published Revision 1 to NUREG/CR-5569, "Health Physics Positions Data Base," which updated a number of positions to correspond to the revision of Part 20. This data base is also now available on diskette. Several minor corrective rulemakings were completed, and a proposed rule (10 CFR Parts 19 and 20) was issued in February 1994 dealing with more substantial issues regarding use of "controlled areas," the definition of occupational and public exposure, and training requirements (59 FR 5132).

In February 1994, the staff published NUREG/CR-6112, "Impact of Reduced Dose Limits on NRC Licensed Activities: Major Issues in the Implementation of ICRP/NCRP Dose Limit Recommendations," as a draft report for comment. A critical ongoing issue has been how the agency should respond to the recent recommendations of the ICRP on occupational dose limits. The report provided the information currently available to assess impacts of several alternative approaches.

In an ongoing effort to reduce regulatory burdens where such reductions would not reduce health and safety, the staff published a proposed rule (10 CFR Part 20) in September 1993 on frequency of medical examinations for the use of respiratory protection equipment. The proposed rule would remove the requirement for an annual medical examination and allow for alternative timeframes based upon the determination of a physician. The rulemaking comments will be considered and final action taken in FY 1995.

6.5.3.2 Brookhaven National Laboratory ALARA Center

The Brookhaven National Laboratory (BNL) ALARA Center, funded by the NRC, continued its surveillance and dissemination of DOE and industry dose reduction and ALARA research. BNL continued work that abstracts national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination. In May 1994, Volume 5 of NUREG/CR-4409, "Data Base on Dose Reduction Research Projects for Nuclear Power Plants," was published. The report provides a summary of projects that have been completed or are currently under way to reduce doses. This information is particularly important to power reactor facilities in the planning stages of activities. BNL also continued publication of the newsletter, "ALARA Notes," on about a quarterly schedule. In 1994, BNL focused on making the data base more easily accessible through an on-line fax system, adding information from overseas contacts, and also continued development of an ALARA handbook. In May 1994, BNL hosted the third ALARA international workshop, which was well attended by representatives from the United States and other countries. The proceedings of that conference will be published in FY 1995. The center provided information and advice on dose reduction to NRC staff and licensees.

6.5.3.3 Occupational Exposure Data System

The NRC continued to collect and process data in the computerized data system called the Radiation Exposure Information Reporting System (REIRS). REIRS provides a permanent record of worker exposures for reactors and several other categories of licensees. A report on 1992 exposures, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 1992," was issued (NUREG-0713, Volume 14; December 1993). Compilation of the statistical reports indicated that approximately 200,000 individuals were monitored and half received a measurable dose. The average measurable dose dropped from 0.31 rem (cSv) in 1990 to 0.30 in 1992. The collective dose obtained from summing all the individual doses was 32,000 person-rems (person-cSv). The data base also includes exposure data on individuals who have terminated employment with certain licensees. Data on some 687,000 persons are in the system, most of whom worked at nuclear power plants. NRC continued to respond to requests for individual exposure data from the system. The data also assist in the examination of the doses incurred by transient workers as they move from plant to plant.

In September 1994, the staff published Generic Letter 94-04, "Voluntary Reporting of Additional Occupational Radiation Exposure Data," as a mechanism to complete the data available in the REIRS data system on occupational exposure. With the revision to Part 20, licensees are required to submit only data on the present year's activities. Previously data were collected at the time a person terminated employment. Thus, in order to complete the data base, data were requested for persons that were employed as of January 1, 1994, that were not already covered by termination reports.

6.5.3.4 National Institute of Standards Technology

Interagency Agreement RES-93-01 between the NRC and the National Institute of Standards and Technology (NIST) involves an ongoing study aimed at establishing traceability between NIST and the Pacific Northwest Laboratories (PNL) for neutron irradiations. PNL provides the neutron irradiation to NIST/NVLAP as part of its duties as the testing laboratory for dosimeter processor accreditation run under the NVLAP.

6.5.3.5 Electronic Personnel Dosimeters

PNL is presently involved in developing a set of performance tests and implementing procedures that would permit electronic personnel dosimeters (EPDs) to be used in place of film or thermoluminescent dosimeters (TLDs) to establish radiation doses for radiation workers. The product of this effort is to be a report that could be used by the NRC to evaluate EPDs until such time as an appropriate ANSI standard for EPDs becomes available. This report would be used as the basis for a possible future certification program to qualify EPDs for use in radiation measurements.

In December 1993, NUREG/CR-6062, "Performance of Portable Radiation Survey Instruments," was published. This report evaluated the current status of performance in the portable instrument area and is part of an ongoing activity to examine performance to determine whether new or modified regulatory standards are necessary.

6.5.3.6 Gamma Dose Spectrometer

Work is being carried out under a Small Business Innovative Research Phase II contract that involves the development of a gamma-ray dosimeter/spectrometer that will measure the gamma-ray spectrum over a wide range of energies. From this information and the electronic signal retrieved from the dosimeter, it will be possible to calculate, through the use of appropriate algorithms, the dose delivered to the skin, the eye, and the whole body. To date, an Active Differential Absorption Spectrometer has been designed, developed, and tested.

6.5.3.7 Spent Fuel Heat Removal

The Oak Ridge National Laboratory, funded by the NRC, is also continuing to improve the data base in the guide for BWR and PWR fuel decay heat generation by including analysis of recent data to provide a basis for evaluating the adequacy of the storage system heat removal capability to limit fuel rod temperatures.

6.6 Small Business Innovation Research

Pursuant to the Small Business Research and Development Enhancement Act of 1992, Public Law 102–564, the NRC supports the Small Business Innovation Research (SBIR) program, which stimulates technological innovation by small businesses, strengthens the role of small business in meeting Federal research and development needs, increases the commercial application of NRC-supported research results, and improves the return on investment from Federally funded research for economic and social benefits to the nation. The NRC has participated in the program since its inception in FY 1982, promoting high quality, "cutting-edge" research of relevance and potential importance to the NRC mission. One goal of the program is to couple this research with follow-on private funding, pursuant to possible commercial application. As of FY 1994, the NRC was supporting 17 SBIR projects-in-progress.

PART 3--SAFEGUARDS REGULATION PROGRAM

7. NUCLEAR MATERIALS RESEARCH

7.1 Statement of Problem

RES has the primary responsibility to manage, coordinate reviews of, and control all NRC materials-related rulemaking activities and to monitor scheduling of such rulemaking to ensure that rules are developed in a timely manner. In addition, RES provides support for preparation of the regulatory impact analyses (RIAs) that accompany all rulemaking through the development of generic methodology and guidance. Technical reviews of all RIAs are performed upon request. The NRC Regulatory Agenda Report and other management information systems associated with rulemaking activities are maintained.

Needed materials-related regulatory products, e.g., regulations and regulatory guides, are developed. Rulemaking is proposed or initiated, as appropriate, and complex rulemakings that span the technical or organizational responsibilities of several groups or that involve novel or complex questions of regulatory policy are managed. Petitions for rulemaking are investigated.

The NRC must provide materials-related radiation protection standards and guidance that ensure that workers and members of the general public are adequately protected from the adverse consequences of exposure to ionizing radiation from licensed activities. RES materials-related activities needed to support the program include developing radiation protection standards; developing guidelines for implementing these standards; and planning, developing, and directing safety research to provide the information necessary for licensing decisions, inspection and enforcement activities, and the standards development process. This includes analyzing available scientific evidence to evaluate the relationship between human exposure to ionizing radiation and radioactive material and the potential occurrence of both late and early radiogenic health effects, including the radiation risk to workers and the public, and estimates of the probability of increased incidence of cancer and genetic effects. These analyses are used to provide bases for severe accident consequence analysis, probabilistic risk assessment (PRA), the development of safety goals and emergency plans, the identification of radiation protection problems, the allocation of priorities for regulatory action, and environmental impact assessments. Recommendations of such organizations as the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP), Presidential guidance to Federal agencies, consensus standards, licensee performance indicators, cost and feasibility data, and available technical information also provide bases for developing regulatory and technical documents related to radiation protection for workers and the public.

7.2 Program Strategy

The purpose of the NRC materials regulatory program is to ensure that nuclear materials facilities are designed, constructed, and operated in a safe manner. Therefore, a continuing need exists to revise rules and guides and to develop new ones. The strategies of this program are to (1) develop screening methodology to systematically review requirements and guidance; (2) coordinate and review proposed changes to the IAEA safety standards; (3) develop or assist the development of rules and regulatory guides; and (4) continue to develop and maintain management information systems for rulemaking.

The Commission's regulatory process requires that safety enhancements to materials rules and guidance be systematically screened to ensure that there is substantial increase in public protection and that based on analysis the costs are justified. Realistic values of the dollar-per-person-rem criterion are needed for analysis to justify changes, but gaps in knowledge associated with radiation health effects cause uncertainties in these analyses. The strategies of this program are to identify and compensate for uncertainties in radiation risk coefficients used for health effect estimates in PRAs and regulatory decisions.

When the Commission approved the whole body dosimetry accreditation rule, they directed the NRC staff to extend the rulemaking to include extremity dosimetry. Therefore, the strategies of this program are to (1) improve regulatory performance for radiation protection by establishing measurement performance criteria and accreditation programs in the areas of extremity dosimetry, bioassay, and air sampling; (2) investigate effective new measurement techniques for these areas; (3) establish the data base required for regulations; and (4) monitor specific indicators to detect improving and declining licensee performance.

Federal guidance was approved by the President on occupational radiation protection. Further, the ICRP has published new recommendations for radiological protection. As a result of this new guidance, NRC materials regulations and regulatory guides will have to be revised. The strategies of this program are to (1) modify radiation protection guidance and standards to be consistent with Presidential guidance on radiation protection requirements and (2) continue to monitor licensee performance indicators by using the Radiation Exposure Information Reporting System program.

7.3 Research Accomplishments in FY 1994

7.3.1 Materials Licensee Performance

Through its human factors regulatory research program, the NRC seeks to improve its understanding and to maintain its requirements concerning the effect of human performance on the safety procedures involving the medical and industrial use of nuclear materials.

Reports are being prepared on the results of comprehensive human factor evaluations of the teletherapy and remote after-loading brachytherapy systems. The first volume for each set of evaluation results includes identification of human factor problems within each system, alternative approaches to solving those problems, and an assessment of those approaches with respect to their relative ability to solve system human factor problems. The remaining volumes for each system evaluation will contain support for the findings described in the first volume; specifically, the results of job and task analyses, as well as indepth studies of human-system interface, procedures, training, and organizational practices and policies for each of the systems.

7.3.2 Materials Regulatory Standards

The Commission issued a final rule (10 CFR Parts 30, 40, 50, 70, and 72) allowing self-guarantee as an additional mechanism for financial assurance for decommissioning on December 29, 1993 (58 FR 68726). This rulemaking is in response to a petition for rulemaking (PRM–30–59) submitted by the General Electric Company and Westinghouse Electric Corporation. The final rule allows certain financially strong, non-electric utility licensees to use a self-guarantee as financial assurance for decommissioning funding. It would not apply to electric utility licensees.

A final rule (10 CFR Part 73) to require a physical fitness program for security personnel at Category I facilities was published on July 28, 1994 (59 FR 38347). The rule adds new requirements for a physical fitness program and annual performance testing or a quarterly site-specific content-based performance test.

A proposed rule (10 CFR 72.214) adding a standardized HUHOMS cask to the list of approved spent fuel storage casks was published for public comment on June 2, 1994 (59 FR 28496). The rule will increase the number of NRC-certified spent fuel storage casks from which the holders of power reactor operating licenses can choose to store spent fuel under a general license. It is expected that the final rule will be issued early in FY 1995.

A proposed rule (10 CFR Parts 30, 32, and 35) on the medical use of byproduct material was published for public comment in July 1993 (58 FR 33396). This action was taken in response to a petition for rulemaking (PRM–35–9). The final rulemaking sent for EDO/Commission approval in September 1994 is intended to provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians more discretion to prepare radioactive drugs containing byproduct material for medical use. The proposed rule would also allow research involving human subjects using byproduct material and the medical use of radiolabeled biologics. It is expected that the final rulemaking will be completed early in FY 1995.

A proposed rule (10 CFR 72.214) that would amend the regulations to allow cask VSC–24 to store spent fuel with control components in the storage casks is being developed. The holders of power reactor operating licenses can use approved spent fuel storage casks under a general license to store spent fuel at the reactor site.

A proposed rule (10 CFR Part 73) to update nuclear power reactor physical protection requirements is being developed. It is expected that the proposed rule will be published for comment in FY 1995.

A proposed rule (10 CFR Part 70) on domestic licensing of special nuclear materials is being developed. The proposed rewrite of Part 70 would amend the Commission's regulations to provide performance-based rather than prescriptive-based regulations for special nuclear material licensees. The rewrite will also develop regulations that are granted according to risk and clarify existing requirements. As an additional requirement, licensees with large quantities of special nuclear material would have their safety programs based on an integrated safety analysis.

A petition for rulemaking from Advanced Medical Systems, Inc. (PRM-32-3) was denied on April 12, 1994 (59 FR 17286). The petition requested the Commission to amend its regulations because the petitioner believed the requirements in Part 32, which are applicable to original manufacturers and suppliers, were not equally applicable to manufacturers and suppliers of replacement parts. The petition was denied because the existing NRC regulations apply equally to manufacturers and suppliers of both original and replacement parts, thereby ensuring the integrity of these parts.

7.3.3 Materials Radiation Protection and Health Effects

7.3.3.1 Irradiator Rulemaking

On February 9, 1993, the NRC published (58 FR 7715) a final rule on "Licenses and Radiation Safety Requirements for Irradiators." The rule established a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators. Irradiators use gamma radiation to irradiate products to change their characteristics in some way. The safety requirements apply to panoramic irradiators (those in which the material being irradiated is in air in a room that is accessible to personnel when the source is shielded) and underwater irradiators in which the source always remains shielded under water and the product is irradiated under water. Draft Regulatory Guide DG-0003, "Guide for the Preparation of Applications for Licenses for Non-Self-Contained Irradiators," was published for comment in January 1994. The guide is related to the irradiator rulemaking and describes the information that an applicant should submit for a new license application or renewal license application.

7.3.3.2 Sewer Disposal

In February 1994, the staff published an advance notice of proposed rulemaking (ANPR) on disposal of radioactive material by release into sanitary sewer systems (59 FR 9146). Regulations in 10 CFR Part 20 currently permit disposal into a sanitary sewer of specified quantities of soluble material with the additional constraint of meeting concentration values in Table 3 of Appendix B to the regulation. This rule will also respond to a petition for rulemaking (PRM-20-22) submitted by the Northeast Ohio Regional Sewer District. The ANPR requested comments on the appropriateness of current NRC regulations and solicited comments on a number of possible alternative approaches to the form and content of the regulations. Interest in this area continued with the publication of a GAO report, and the NRC staff is currently contracting with the Pacific Northwest Laboratories (PNL) for additional information related to sewer chemistry to determine what types of regulatory changes may be appropriate.

7.3.3.3 Radiography

In February 1994 (59 FR 9429), the staff published a proposed rule for 10 CFR Part 34, "Licenses for Radiography and Radiation Safety Requirements for Radiographic Operations." This portion of the Commission's regulations covers the conduct of radiography using sealed sources and has not been the subject of a complete revision for a number of years. The proposed rule would respond to a petition for rulemaking (PRM-34-04) submitted by the International Union of Operating Engineers, Local No. 2, and represented a complete revision to this part of the Commission's regulations, including proposals for certification of radiographers and implementation of a two-person rule for work with radioactive sources. The proposals took into account recent regulatory approaches of a number of Agreement States and the Conference of Radiation Control Program Directors. Interest of the Agreement States has been significant, and the NRC staff plans to hold a workshop early in FY 1995 with the States to discuss the issues and possible resolutions.

In related activities, NUREG/CR-4833, "Large Area Self-Powered Gamma Ray Detector: Phase II Development of a Source Position Monitor for Use on Industrial Radiographic Units," was published. This work resulted from a Small Business Innovative Research contract and examined the feasibility for a source position monitor as an additional safeguard to preventing overexposures resulting from disconnected sources during radiographic operations.

7.3.3.4 Uranium Mill Tailings

In November 1993, the staff published a proposed rule (10 CFR Part 40, Appendix A) on uranium mill tailings to conform the NRC regulations to Environmental Protection Agency (EPA) regulations under the Clean Air Act and support recision of certain EPA Clean Air Act requirements as outlined under a memorandum of understanding and a settlement agreement between EPA, several States, and environmental organizations. The final rule was published in June 1994 (59 FR 28220) and EPA published its recision at the end of June.

7.3.3.5 Patient Release Criteria

In June 1994 (59 FR 30724), a proposed rule was published on criteria for the release of patients administered radioactive material. At the same time, Draft Regulatory Guide DG-8015, "Release of Patients Administered Radioactive Materials," was published for comment. Criteria for release of patients is currently contained in 10 CFR 35.75 and is specified in terms of a quantity of material (30 mCi) in the patient. This rulemaking action addressed the requests of three petitions for rulemaking: PRM-20-20 from Dr. Carol S. Marcus and PRM-35-10/10a from the American College of Nuclear Medicine. The petitioners requested that the Commission adopt a dose limit of 5 mSv (0.5 rem) for individuals exposed to patients who have been administered radioactive material rather than the activity limit in the present regulation. The Commission plans to analyze comments and consider final rulemaking action in FY 1995.

7.3.3.6 Improvement of Health Effects Models

Revision 2, Part I to NUREG/CR-4214, "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis," published in October 1993, contains an introduction, integration, and summary of health effects models and risk coefficients intended for use in severe accident analyses, probabilistic risk assessments, emergency response planning, and safety goal and cost/benefit analyses. Leading to modification of the models presented in NUREG/CR-4214 are the reports of the United States Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 1988), the National Academy of Sciences/National Research Council BEIR V Committee (NAS/NRC, 1990), and other revised recommendations of ICRP-60 (ICRP 1991).

7.3.3.7 Embryo/Fetal Dose from Maternal Intake

A study to improve understanding of the contribution of maternal radionuclide burdens to prenatal radiation exposure was continued in FY 1994 with significant progress. In October 1993, NUREG/CR-5631, Revision 1, Addendum 1, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses: Relationships Between Annual Limits on Intake and Prenatal Doses," was published. The report provides an expansion of the methodology presented earlier by examining the relationship between published Annual Limits on Intake in 10 CFR Part 20 and the dose to an embryo/fetus. Research that will permit inclusion of additional radionuclides, such as technetium, molybdenum, and additional transuranic elements, began in FY 1993 and continued in FY 1994. The methods and data developed under this project have been used by the NRC in preparing Regulatory Guide 8.36, "Radiation Dose to Embryo/Fetus," which describes acceptable methods of compliance with 20.1208 of 10 CFR Part 20. The guide might be revised to incorporate the information presented in the addendum. The methods developed under this project are also useful in calculating doses in cases of accidental releases of radioactive materials.

In December 1993, the NRC placed a Letter Report from PNL (PNL-8977), "Dose to the

Embryo/Fetus from Selected Radiopharmaceuticals—Preliminary Recommendations," in the Public Document Room in support of ongoing rulemaking activities related to establishing limits for dose to the embryo/fetus as a result of medical treatments. This rulemaking effort will continue in FY 1995.

7.3.3.8 Criticality and Fuel Cycle Safety

The final Regulatory Guide 3.68, "Nuclear Criticality Safety Training," was published in April 1994. This regulatory guide was developed to provide guidance to licensees on an appropriate nuclear criticality safety training program for the use of special nuclear material, especially the prevention of criticality accidents.

The Los Alamos National Laboratory, funded by the NRC, continued its examination and revision of TID-7016, "Nuclear Safety Guide," for simplification of use, evaluation against new experimental data, and use of current computational codes. The document is a standard guide and reference used by industry and the Office of Nuclear Material Safety and Safeguards (NMSS) staff for initial criticality safety evaluations.

The Oak Ridge National Laboratory (ORNL), funded by the NRC, continued its methods validation of the criticality analytical sequences in SCALE–4 using ENDF/B-V cross-section data. The validation effort will qualify the applicability of SCALE–4 to criticality safety problems covering the range of interest within the NMSS Fuel Cycle Safety Branch. The SCALE code system was developed at ORNL for criticality, shielding, and thermal analysis of nuclear facility and package designs. The system is currently used at ORNL in support of several tasks funded by NMSS. In particular, SCALE–4 is used by ORNL and NRC staff for criticality safety analyses relevant to licensing issues. Valid criticality safety analyses require validation of both methods applied and the user who applies them. The goal of this project is to validate the Criticality Safety Analyses Sequences within the SCALE–4 system by analyzing a large number of benchmark critical experiments whose parameters (enrichment, geometry, fissile fuel/moderator ratio, etc.) cover the range of interest within the NMSS Fuel Cycle Safety Branch. The work will be documented in a report that will include a description of the critical experiments modeled, calculational results, quantification of trends in calculated k-effectives for different types of experiments, and recommended calculational uncertainties to be applied.

7.3.4 Uranium Enrichment

In February 1994, the staff published a proposed rule (10 CFR Part 76) on certification of gaseous diffusion plants (59 FR 6792) to solicit comment on the standards that will be used by the NRC for certification of the operations of the gaseous diffusion enrichment facilities leased by the U.S. Enrichment Corporation from the Department of Energy. The rule covered both the certification process and the standards to be used to judge acceptable performance for certification. Under the enabling legislation, the final rule was to be completed by the end of October 1994. The final rule (10 CFR Parts 19, 20, 21, 26, 51, 70, 71, 73, 74, 76, and 95) was published on September 23, 1994 (59 FR 48944).

8. LOW-LEVEL WASTE DISPOSAL

NRC research in support of regulatory activities for low-level-waste (LLW) disposal facilities is focused on making more realistic assessments of the overall performance of disposal systems. The results of NRC LLW research are also useful to the States regulating LLW disposal and are made available to the States through NRC-sponsored workshops, participation by NRC contractors in forums sponsored by other agencies, as well as the conventional method of publication in journals.

8.1 Statement of Problem

Disposal of LLW involves issues concerning waste form and waste package integrity, transport of radionuclides through the disposal facility environment, and evaluation of long-term doses from releases of radionuclides beyond the disposal facility environment. Research is required to establish regulatory criteria and license application assessment information to permit sound evaluation of proposals for disposal facilities and to ensure that all regulatory requirements, particularly those on radionuclide release limits, will be met. Performing the needed research in a timely manner is made more urgent and complex by two factors. First, the Low-Level Radioactive Waste Policy Amendments Act of 1985 (P.L. 99–240) sets a very tight schedule for establishing facilities within individual States or compacts of States.

Second, the States and compacts of States have chosen to consider alternative disposal methods to shallow land burial. Certain of these alternatives must be critically examined by tightly focused research to determine their acceptability and to give guidance to the States and compacts.

The direction of the LLW research program has responded to legislative action, the changing policy of States now responsible for disposal, and the lessons learned from the history of shallow land burial of wastes at a number of sites for several decades. Vague and differing criteria as to site suitability, waste package design, etc., have been employed and may characterize future efforts.

Disposal criteria for LLW have evolved as experience, knowledge, public awareness, and political controversy have grown. In particular, through the Low-Level Radioactive Waste Policy Amendments Act of 1985, the Congress has required the NRC to provide guidance for regulatory decisionmaking regarding engineered LLW disposal methods. This change has broadened the scope of NRC LLW research.

8.2 Program Strategy

NRC research in support of licensing activities for LLW disposal facilities is examining enhancements and alternatives to shallow land burial, LLW waste forms, infiltration of water, radionuclide migration in the soil, hydrology and contaminant transport, performance assessment, and LLW source term modeling. The NRC's LLW research staff also prepares rulemakings that affect LLW disposal.

The diverse LLW regulatory user community makes the coordination and definition of LLW research and the dissemination of associated products a much more complicated undertaking than similar activities for the high-level waste program. Because many States are licensers of LLW disposal and are looking to the NRC for technical support in their LLW licensing and regulatory programs, NRC's LLW research has to be more prescriptive and developmental than the HLW research program.

8.3 Research Accomplishments in FY 1994

8.3.1 Materials and Engineering

8.3.1.1 Engineered Enhancements and Alternatives to Shallow Land Burial

Many States and State compacts are considering engineered enhancements for the disposal of LLW. Several concepts have been proposed—particularly the use of concrete engineered barriers to contain LLW. NRC research conducted at the National Institute of Standards and Technology (NIST) has investigated the durability of concrete while the Idaho National Engineering Laboratory (INEL) completed their evaluation of concrete barriers in limiting radionuclide transport (NUREG/CR-6070). Three reports of the NIST work are being prepared as NUREG documents that address (1) a new method to determine chloride diffusion coefficients in concrete, (2) the evaluation of stress-induced microcracks on solute transport through concrete, and (3) the evaluation of the effects of stresses caused by sulfate attack in concrete. NIST also has completed a computer program for modeling the degradation of concrete for LLW performance assessment applications. The model incorporates synergistic degradation mechanisms, the effects of cracks and joints, and the precipitation of concrete dissolution products to predict concrete hydraulic properties.

8.3.1.2 LLW Waste Forms

Research conducted at INEL on the stability of nuclear reactor decontamination waste was completed. These studies were aimed at determining radionuclide and chelating agent releases, as well as the compressive strength of the cement solidified waste. Results have been published as NUREG/CR reports; test results are also being summarized in papers that will be published in scientific literature. Field lysimeter studies containing radioactive ion-exchange resins solidified in cement and vinyl ester-styrene are being conducted at the Oak Ridge and Argonne National Laboratories to determine radionuclide release rates under environmental conditions. Studies are being completed at INEL to investigate biodegradation of LLW by microorganisms to ensure that the stability requirements of 10 CFR Part 61 are met. Studies at the Pacific Northwest Laboratories (PNL) to determine scaling factors for assessing hard-to-measure radionuclides in LLW are continuing. Also continuing at PNL are studies to determine the effect of naturally occurring radionuclide-chelating complexes in soils on radionuclide transport.

8.3.1.3 Infiltration of Water

The University of California at Berkeley, in cooperation with the University of Maryland, is continuing to field test a variety of covers for LLW disposal at the Maryland Agricultural Experiment Station in Beltsville, Maryland. These covers are not only applicable to any LLW disposal method that includes an earthen cover, but are applicable to LLW, SDMP (Site Decommissioning Management Plan), UMTRA (Uranium Mill Tailings Remedial Action), and hazardous waste sites as well. Two designs are proving to be particularly effective. One, called bioengineering water management, not only reduced water infiltration to a negligible amount but also dewatered the cells to which it was applied. Hence this cover lends itself to use as a remedial action cover for sites susceptible to subsidence. The New York State Energy Research and Development Administration finished construction in 1993 of a bioengineering water management cover over such a trench at the West Valley LLW disposal facility. A second promising cover consists of a conductive layer barrier placed below a resistive layer barrier. This cover has functioned perfectly since its installation in January 1990.

PNL has developed an infiltration evaluation methodology (NUREG/CR-5523) and has separately modeled infiltration and moisture redistribution using a field experiment data set (NUREG/CR-5998). Various infiltration estimation approaches have also been examined by PNL (NUREG/CR-6114). Future work will focus on applying the infiltration evaluation methodology to an arid site using data from a related cooperative research study with the U.S. Geological Survey.

8.3.2 Hydrology and Geochemistry

8.3.2.1 Radionuclide Migration in Soil

Current models of radionuclide retardation in soils introduce significant conservatisms into current assessments of performance of LLW disposal. This conservatism is necessitated by the quantitative uncertainty as to the degree of retardation in various soil types under various conditions. To reduce this uncertainty, and hence permit more realistic assessment of actual expected performance of an LLW disposal facility, the NRC is developing more realistic retardation models based on filed observations and laboratory experiments. Of particular interest is the role in radionuclide transport played by naturally produced organic complexants. Observations made by PNL at the Hanford, Wash., site (NUREG/CR-3712 and -4030) and at Chalk River Nuclear Laboratory in Canada (NUREG/CR-4879, Vols. 1 and 2) found radionuclide transport through soils at rates faster than predicted by current transport models. This includes radionuclides (e.g., Fe-55, Co-60, Ni-63, Pu, and Am) generally considered unlikely candidates for mobilization based on their presently understood geochemical behavior. Preliminary evidence suggests that naturally produced organic complexes and microparticulates played a significant role in enhancing migration.

8.3.2.2 Hydrology and Contaminant Transport

PNL has evaluated and developed a data set from an earlier field study involving subsurface injection of radioactive tracers in heterogeneous unsaturated porous media at the Hanford site. The data sets reported in NUREG/CR-5996 cover a period of 10 years and will allow confirmatory analyses of existing flow and transport models that may be used in LLW performance assessment. Work has been completed by the Massachusetts Institute of Technology and Princeton University on the application of stochastic methods for simulating flow and transport in heterogeneous soils.

8.3.3 Compliance, Assessment, and Modeling

8.3.3.1 Performance Assessment

Research is continuing work to develop a realistic and computationally tractable performance assessment methodology. The current capabilities and limitations of performance assessment models have been evaluated by the Sandia National Laboratories, and the results have been published in NUREG/CR-5927, Volume 1, which deals with modeling approaches, and Volume 2, which deals with validation needs.

8.3.3.2 LLW Source Term Modeling

During FY 1994, extensions were made to the existing LLW source term code, BLT (breach, leach, and transport), developed by the Brookhaven National Laboratory, to incorporate additional geochemistry and gaseous release. The code is currently being tested and documented.

8.3.4 Low-Level-Waste Regulatory Standards

A proposed rule to amend 10 CFR Parts 20 and 61 to revise low-level-waste shipment manifest information and reporting was published for comment in April 1992. This rule is intended to improve the quality and uniformity of information regarding actual quantities and characteristics of LLW disposed at LLW disposal facilities through the use of standardized NRC forms when the waste is shipped. In turn, the more accurate and complete information on what is actually received at a disposal facility will facilitate more realistic assessments of expected disposal facility performance. It is expected that the final rule will be published in the first quarter of calendar year 1995.

8.3.5 Environmental Policy and Decommissioning

8.3.5.1 Decommissioning Cost Reassessment

In October 1993, Volumes 1 and 2 of NUREG/CR-5884, "Revised Analysis of Decommissioning for the Reference Pressurized Water Reactor Power Station," were published for public comment. This analysis is the first of two documents resulting from an ongoing reassessment of decommissioning costs for commercial nuclear power reactors, using the experience gained in the last 20 years and information available on costs of transport and disposal of waste materials. The draft report indicated that the waste disposal component could be significant depending on the waste site assumed. In support of the revised analysis, the staff published NUREG/CR-6054, "Estimating Pressurized Water Reactor Decommissioning Costs," in October 1993. The report contains the computer program developed by PNL for doing cost assessments. Similar work is under way for BWR facilities, and the reports will be published early in FY 1995.

8.3.5.2 Radiological Criteria for Decommissioning

The NRC continued in FY 1994 with its enhanced participatory rulemaking approach for establishing radiological criteria for decommissioning. In January 1994, NUREG/CR-6156, "Summary of Comments Received from Workshops on Radiological Criteria for Decommissioning," was published to provide the comments received during the seven workshops held across the country on the issues and possible approaches to the rulemaking. In February 1994, the NRC staff published a draft of the rulemaking and support statement for public comment. Numerous comments were received on the draft, and these were used to prepare a formal proposed rulemaking package for Commission consideration. The proposed rule was published for public comment on August 22, 1994 (59 FR 43200). The comment period expires in late December 1994, and the staff anticipates holding a special workshop on issues of public participation and the use of site-specific advisory boards during the comment period.

In support of the proposed rule, a number of additional documents were published, including NUREG-1496, "Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities"; NUREG-1500, "Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment"; and NUREG-1501, "Background as a Residual Radioactivity Criterion for Decommissioning."

8.3.5.3 Decommissioning Funding

In June 1994, the Commission published a proposed rule (10 CFR Parts 30, 40, 70, and 72) on clarification of decommissioning funding requirements (59 FR 32158). The proposed rule was intended to clarify when decommissioning funding assurance was required and to provide that assurance would be available after operations were terminated and decommissioning initiated. The staff will analyze comments and prepare a final rulemaking package during FY 1995.

8.3.5.4 Timeliness

In July 1994 the final rule (10 CFR Parts 2, 30, 40, 70, and 72) on timeliness in decommissioning of materials facilities was published (59 FR 36026). The rule amended the Commission's regulations to establish timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities. The principal effect of these amendments is to formalize and codify the NRC's requirements for timeliness in decommissioning of materials facilities.

8.3.5.5 Safety Issues Related to Permanently Shutdown Reactors

Brookhaven National Laboratory continued its determination of technical and safety criteria that should remain as part of decommissioning regulations under 10 CFR Part 50 when a licensee initiates action to permanently shut down the nuclear reactor in preparation for decommissioning activities. This project will develop a comparison of the safety requirements for a shutdown versus an operating nuclear power reactor after the reactor has permanently shut down. It will also perform financial assurance analysis for offsite liability requirements for shutdown reactors. It will examine the environmental impact of the potential increase in the spent fuel transport and radiological exposure to the public in the event the licensees prefer to ship and store their spent fuel.

PART 4--ASSESSING THE SAFETY OF HIGH-LEVEL WASTE DISPOSAL

9. HIGH-LEVEL WASTE RESEARCH

The Nuclear Waste Policy Act of 1982 requires the Department of Energy (DOE) to dispose of high-level radioactive waste (HLW), which can be spent reactor fuel or the byproduct of reprocessing spent fuel, in a deep geologic repository. The act further requires the DOE to apply for a license from the NRC to dispose of HLW.

The NRC maintains an active HLW research program of theoretical study and laboratory and field experiments directed at understanding the physical processes that control and determine repository performance in the unsaturated volcanic tuff at the Yucca Mountain (Nev.) site currently under consideration by the DOE as directed by the Congress in December 1987. The goal of the NRC's HLW research is to provide models, methods, data, and technical information to support the staff's independent judgments as to the appropriateness and adequacy of DOE's demonstration of compliance of the HLW repository with NRC requirements specified in 10 CFR Part 60 and with the Environmental Protection Agency's HLW standard, incorporated by reference into Part 60. The program is divided into three parts: engineered systems research, which examines issues related to controlled release of radionuclides, containment of waste, and the engineering-geology interface in the repository; geologic systems research, which examines issues related to the hydrology, geochemistry, and geology of the repository site; and performance assessment research, which integrates mathematical models from the other research into NRC's HLW performance assessment methodology. Key technical issues being addressed include methods to assess the long-term performance of the packages containing the HLW, the potential for volcanic and seismic events, and flow and transport mechanisms in unsaturated fractured rocks.

Most NRC HLW research is conducted by the Center for Nuclear Waste Regulatory Analyses (CNWRA), a division of the Southwest Research Institute in San Antonio, Texas. However, a significant portion of NRC HLW research on hydrology is being conducted at the University of Arizona.

9.1 Statement of Problem

The HLW disposal policy for the United States is defined by the Atomic Energy Act, the Energy Reorganization Act, the Nuclear Waste Policy Act, and the Nuclear Waste Policy Amendments Act (NWPAA). The last, signed into law in 1987, provides for the development of a geologic repository for the permanent disposal of high-level radioactive waste in the State of Nevada at Yucca Mountain and assigns responsibility for repository development to the DOE. According to the Federal Government's Reorganization Plan No. 3 of 1970, HLW environmental standards development is the responsibility of the Environmental Protection Agency (EPA), and the Energy Reorganization Act assigns the regulation of HLW disposal to protect public health and safety and the environment to the NRC.

An HLW repository poses problems involving regulatory considerations and uncertainties related to waste emplacement, monitoring, and performance assessment that are unique in the history of the NRC. Much of this uniqueness stems from the type of facility, first-of-its-kind geologic disposal installation, its very long performance time (specified as 10,000 years by the EPA), and the fact that it will be placed in low permeability/low flow geologic systems that have not been investigated previously because of their low economic value. The NRC must have an independent capability to evaluate the DOE safety analyses and decide whether long-term releases predicted by DOE will be within established limits. The NRC research program objective is to provide the technical capability necessary to evaluate DOE's site characterization activities as required by the NWPAA and to assess DOE's license application when it is submitted.

9.2 Program Strategy

The research program has been guided by the need to provide the technical foundation for NRC development of a set of regulations and a licensing process for the review and licensing of the HLW repository. This framework for NRC review will allow the formal licensing activities and the supporting research to be focused on the significant technical issues.

At present, the NRC has active research programs in hydrology, geology, materials science, geochemistry, and several other disciplines related to HLW management. The research combines theoretical study with laboratory and field experiments to identify and quantify the physical processes and phenomena important to waste isolation so that the NRC can assess repository performance and quantify the uncertainties associated with characterization and measurement of these processes. All this work is integrated into an independent HLW performance assessment methodology. Effort is also required to validate many of the models that underlie the methodology. The ultimate goal of the NRC's HLW research program is to provide the technical basis to support the licensing staff's independent review of the appropriateness and adequacy of the DOE's demonstration of compliance with 10 CFR Part 60 and the EPA's HLW standard. In addition, NRC's waste management research seeks to provide technical support to the licensing staff in their interactions with DOE, the State of Nevada, and other participants and interested parties and to develop regulatory standards to support the licensing of the disposal and management of high-level radioactive wastes.

9.3 Research Accomplishments in FY 1994

9.3.1 Engineered Systems Research

9.3.1.1 Controlled Release

10 CFR Part 60 contains a criterion for the maximum rate of release of radioactive material from the repository's engineered barrier system. Research on controlled release is being done at the natural analogue site at Pena Blanca, Mexico, by CNWRA. This site is located in an unsaturated tuff environment similar to that at Yucca Mountain. A uranium ore body is serving as a surrogate for disposed spent fuel, and limits on the expected range of spent fuel behavior in oxidizing chemical environments like those of Yucca Mountain are being developed.

9.3.1.2 Containment

10 CFR Part 60 contains a criterion for the minimum lifetime of HLW containment within waste packages placed in the repository. CNWRA is conducting confirmatory research on the behavior of waste package materials in the expected repository environment. During FY 1994, research was done on stress-corrosion cracking, repassivation potentials for long-term corrosion of stainless steel, corrosion of copper-based waste package container materials, effects of surface conditions on the corrosion of waste package container materials, and crevice corrosion of stainless steel. Work also was initiated on microbial corrosion of waste package container materials.

9.3.1.3 Engineering-Geology Interface

10 CFR Part 60 requires that the repository's engineered and geologic systems function together so as not to compromise repository safety. CNWRA has been conducting two projects on coupled processes deriving from the engineered system's interaction with its surrounding geologic system. One project, on the redistribution of liquid water by emplaced HLW, is using laboratory-based similitude experiments and theoretical simulations to assess models of this redistribution. Work in FY 1994 produced a simplified thermosyphon model of the redistribution process and examined pressure-driven heat flows in unsaturated media. In the other project, on rock-mechanical aspects of repository performance, CNWRA researchers finished a study of the effect of mine seismicity on ground-water hydrology; finished research on rock-joint characteristics; issued an evaluation of the rock mechanics simulator UDEC; and supported NRC's continued participation in DECOVALEX—an international cooperative effort to test the validity of mathematical models of thermal-hydrological-mechanical interactions. NRC also provided financial support to the Swedish Nuclear Power Inspectorate for administration of DECOVALEX.

9.3.2 Geologic Systems Research

9.3.2.1 Hydrology

Because transport by ground water is considered to be the most likely path for radionuclide transport from an HLW facility to the accessible environment, the NRC is actively studying ground-water infiltration, recharge, flow, and transport processes. At an experimental site, the Apache Leap Tuff site operated by the University of Arizona, in partially saturated fractured rock similar to that at the Yucca Mountain site, research continued in FY 1994 on testing hydrologic site characterization methods and on scale effects in fluid flow and radionuclide transport in unsaturated media. Results from theoretical work conducted at both the CNWRA and the University of Arizona suggested that scaling of certain aspects of permeability measurements may be universal and not site specific as previously believed. In FY 1994, the CNWRA also completed a project on stochastic analysis of large-scale flow and transport in unsaturated fractured rock masses. The project developed an efficient method for estimating effective permeabilities measured in unsaturated fractured media and developed a methodology for probabilistic estimation of ground-water travel time. CNWRA is continuing to study hydrology on a regional scale as well as a local scale.

9.3.2.2 Geochemistry

Knowledge and application of the geochemical conditions at Yucca Mountain are important to understanding many aspects of repository performance, including waste package corrosion, radionuclide release and transport, and alteration of ground-water flow paths. During FY 1994, CNWRA finished a project on geochemical effects on mass transport in unsaturated media. In its final phases, the project examined the thermodynamics of ion exchange in the zeolite mineral clinoptilolite, common in tuffs like those of Yucca Mountain. This mineral is expected to play a key role in controlling radionuclide transport in the Yucca Mountain repository.

A significant problem with addressing the geochemistry of radionuclide transport is that the complexity of the chemistry makes calculations difficult and time consuming. Simplified geochemical models that have been developed to make transport calculations tractable oversimplify the chemistry to the point that even so-called bounding calculations may not be truly bounding. For this reason, the NRC asked the CNWRA to determine whether some model could be developed that is sufficiently realistic to retain credibility of the results and yet be calculationally tractable. The CNWRA has subsequently developed and tested in the laboratory a double-layer surface complexation model that meets both objectives.

In a workshop organized and conducted by CNWRA researchers, ways in which natural and archaeological analogues can be used to build confidence in the conceptual and mathematical models used in HLW performance assessment were addressed.

9.3.2.3 Geology

CNWRA has two projects that are investigating (1) techniques to estimate the likelihood of occurrence of volcanos in the Yucca Mountain area that are alternative to the method currently used by the DOE, and (2) possible consequences to HLW disposal of a volcano at Yucca Mountain. During FY 1994, CNWRA found that other methods may suggest a higher likelihood of a volcano at Yucca Mountain than the method currently used by the DOE. The application of seismic tomographic methods to provide insights as to the possible consequences of basaltic volcanism—the type that most likely would occur at Yucca Mountain—also was examined.

9.3.3 Performance Assessment Research

The NRC will assess DOE's demonstration of compliance with both the NRC's requirements for HLW disposal given in 10 CFR Part 60 and EPA's HLW standard. Use of a performance assessment methodology, independent of DOE's performance assessment methodology, is a key element in NRC's strategy to review that demonstration of compliance. To support implementation of that strategy, research is being conducted at the CNWRA on the development of performance assessment tools. The tools are being used in their current state of development in the joint NRC-CNWRA HLW Iterative Performance Assessment (IPA) effort, which is providing insights as to the processes and phenomena that may be critical to repository performance. It is anticipated that as the performance assessment tools become more robust, the IPA effort also will assist in setting priorities for future HLW research.

In FY 1994, the CNWRA reviewed the scenario methodology used in IPA's latest exercises, developed a mathematical model of infiltration that was applied in IPA, examined film flow in fractures, performed laboratory permeability tests on tuff samples from the Pena Blanca analogue, provided training on the flow and transport simulator PORFLOW, and examined ways to increase the efficiency of performance assessment calculations with improved numerical algorithms and massively parallel computation.

APPENDIX

FY 1994 Regulatory Products from the Office of Nuclear Regulatory Research

Date	Regulatory Product	Description
Part 1—NUCLEAR SAFETY RESEARCH—REACTOR LICENSING SUPPORT		
REACTOR AGING AND RENEWAL		
Pressure Vessel Safety and Piping Integrity		
September 1994	Draft Regulatory Guide	DG–1028, "Periodic Testing of Electric Power and Protection Systems."
October 1994	Regulatory Guide	Revision 30 to Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability—ASME Section III, Division 1."
October 1994	Regulatory Guide	Revision 30 to Regulatory Guide 1.85, "Materials Code Case Acceptability—ASME Section III, Division 1."
October 1994	Regulatory Guide	Revision 11 to Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1."
October 1994	Draft Regulatory Guide	DG–1027, "Format and Content of Application for Approval for Thermal Annealing of Reactor Pressure Vessels."
October 1994	Proposed Rule	Fracture toughness requirements for light-water reactor pressure vessels.
STANDARD REACTOR DESIGNS		
Regulatory Application of New Source Terms		
January 1994	SECY–94–017	"Options with Regard to Revising 10 CFR Part 100."
April 1994	EDO Memo to Commission	Review of MIT Ph.D Thesis on Chernobyl Source Term
July 1994	SECY–94–194	"Proposed Revisions to 10 CFR Part 100."
Part 2—NUCLEAR SAFETY RESEARCH—REACTOR REGULATION SUPPORT		
REACTOR ACCIDENT ANALYSIS		
Severe Accident Policy Implementation		
May 1994	SECY–94–134	"Status of the Individual Plant Examinations and the Individual Plant Examination of External Events Insights Program."

Date	Regulatory Product	Description
SAFETY ISSUE RESOLUTION AND REGULATION IMPROVEMENTS		
Earth Sciences		
October 1994	Proposed Rule	Proposed § 100.23, "Geologic and Seismic Siting Factors," issued for public comment.
Generic Safety Issue Resolution		
October 1993–September 1994	Generic Safety Issues	For generic safety issues prioritized and resolved in FY 1994, see Tables 6.1 and 6.2.
Reactor Regulatory Standards		
November 1993	Advance Notice of Proposed Rulemaking	An advance notice of proposed rulemaking, 10 CFR Part 52, concerning standard design certification for evolutionary light-water reactors. The contemplated rulemaking would define the form and content of rules that would certify the designs.
December 1993	Volume 14 of NUREG–0713	A report on 1992 exposures, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1992." It provides a compilation of the statistical reports of individual exposures.
February 1994	Final Rule	The NRC regulation, 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses. The rule deletes the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test during the 6-year license term as prerequisite for license renewal.
February 1994	Proposed Rule	The proposed rule, 10 CFR Parts 19 and 20, regarding use of "controlled areas," the definition of occupational and public exposure, and training requirements.
September 1994	Proposed Rule	The NRC regulation, 10 CFR Part 20, on frequency of medical examinations for use of respiratory protection equipment. The proposed amendment would remove the requirement for an annual medical examination and allow for alternative timeframes.

Date	Regulatory Product	Description
October 1994	Proposed Rule	The NRC regulation, 10 CFR Part 21, on procurement of commercial grade items by reactor licensees. The proposed rule responds to a petition for rulemaking (PRM–21–02) submitted by the Nuclear Management and Resources Council (NUMARC), which is now incorporated into the Nuclear Energy Institute (NEI). The proposed amendment would clarify and add flexibility for procuring items for safety-related service.
November 1994	Proposed Rule	The NRC regulation, 10 CFR Parts 50, 55, and 73, on reduction of reporting requirements imposed on NRC licensees. The proposed amendments would reduce reporting requirements on power reactors, research and test reactors, and nuclear material licensees.

Part 3—SAFEGUARDS REGULATION PROGRAM

Nuclear Materials

Date	Regulatory Product	Description
November 1993	Proposed Rule	The NRC regulation, 10 CFR Part 40, Appendix A, on uranium mill tailings. The proposed rule would conform NRC regulations to Environmental Protection Agency (EPA) regulations under the Clean Air Act and support recision of certain EPA Clean Air Act requirements.
December 1993	Final Rule	A final rule, 10 CFR Parts 30, 40, 50, 70, and 72, to allow self–guarantee as an additional mechanism for financial assurance for decommissioning. This rulemaking is in response to a petition for rulemaking (PRM–30–59) submitted by the General Electric Company and Westinghouse Electric Corporation. The final rule applies to certain financially strong, non-electric utility licensees and allows the use of self-guarantee as financial assurance for decommissioning funding. It would not apply to electric utility licensees.
January 1994	Draft Regulatory Guide	DG–0003, "Guide for the Preparation of Applications for Licenses for

Date	Regulatory Product	Description
		Non-Self-Contained Irradiators." This guide is related to 10 CFR Part 36 and describes the information that an applicant should submit for a new or renewed license application.
February 1994	Advance Notice of Proposed Rulemaking	An advance notice of proposed rulemaking (ANPR), 10 CFR Part 20, on disposal of radioactive material by release in sanitary sewer systems. The ANPR is in response to a petition for rulemaking (PRM–20–22) submitted by the Northeast Ohio Regional Sewer District. The ANPR requested comments on the appropriateness of current NRC regulations and solicited comments on possible alternative approaches.
February 1994	Proposed Rule	The NRC regulation, 10 CFR Part 34, on the conduct of radiography using sealed sources. The proposed rule responds to a petition for rulemaking (PRM–34–04) submitted by the International Union of Operating Engineers, Local No. 2. The proposed rule represents a complete revision to this part of the Commission's regulations, including certification of radiographers and implementation of a two-person rule for work with radioactive sources.
February 1994	Proposed Rule	A proposed rulemaking, 10 CFR Part 76, on certification of gaseous diffusion plants. The proposed rule solicits comments on the standards for certification of the operation of gaseous diffusion enrichment facilities.
June 1994	Proposed Rule	A proposed rule, 10 CFR 72.214, adding a standardized HUHOMS cask to the list of approved spent fuel storage casks. The rule would increase the number of NRC-certified spent fuel storage casks available under a general license.
June 1994	Final Rule	The final rule to amend NRC regulations, 10 CFR Part 40, Appendix A, on uranium mill tailings. See proposed rule above (November 1993).

Date	Regulatory Product	Description
June 1994	Proposed Rule	A proposed rule to amend NRC 10 CFR 35.75, on criteria for release of patients administered radioactive material. The rulemaking action addressed the requests of three petitions for rulemaking: PRM- 20-20 from Dr. Carol S. Marcus and PRM-35-10/10a from the American College of Nuclear Medicine. The proposed amendment would specify a dose limit of 5 mSv (0.5 rem) rather than the limit of 30 mCi currently specified.
June 1994	Draft Regulatory Guide	DG-8015, "Release of Patients Administered Radioactive Materials." The draft guide provides guidance on the proposed rule to amend 10 CFR 35.75 (see above).
July 1994	Final Rule	The NRC regulations, 10 CFR Part 73, on physical fitness programs for security personnel at Category I fuel cycle facilities. The amendment requires physical fitness training programs as well as annual performance testing for specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.
September 1994	Final Rule	The NRC regulations, 10 CFR Parts 19, 20, 21, 26, 51, 70, 71, 73, 74, 76, and 95, for certification of the operations of the gaseous diffusion enrichment facilities leased by the U. S. Enrichment Corporation from the Department of Energy. See proposed rule above (February 1994).
Low-Level Waste Disposal		
June 1994	Proposed Rule	The proposed rule to amend NRC regulations, 10 CFR Parts 30, 40, 70, and 73, on clarification of decommissioning funding requirements. The amendments would clarify when decommissioning funding assurance was required and provide that assurance would be available after operations were terminated and decommissioning initiated.

Date	Regulatory Product	Description
July 1994	Final Rule	The rule on timeliness in decommissioning of materials facilities, 10 CFR Parts 2, 30, 40, 70, and 72. The rule establishes timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities.
August 1994	Proposed Rule	The proposed rule, 10 CFR Parts 20, 30, 40, 50, 51, 70, and 72, to amend the NRC regulations on radiological criteria for decommissioning. The proposed rule is based on comments received from seven workshops and a draft of the rulemaking published in February 1994.
August 1994	NUREG–1496	"Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for Decommissioning of NRC-Licensed Nuclear Facilities." This document supports the rulemaking on this topic.
August 1994	NUREG–1500	"Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment." This document supports the rulemaking on this topic.
August 1994	NUREG–1501	"Background as a Residual Radioactivity Criterion for Decommissioning." This document supports the rulemaking on this topic.

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This report, the tenth in a series of annual reports, was prepared in response to congressional inquiries concerning how nuclear regulatory research is used. It summarizes the accomplishments of the Office of Nuclear Regulatory Research during FY 1994.

The goal of the Office of Nuclear Regulatory Research (RES) is to ensure the availability of sound technical bases for timely rulemaking and related decisions in support of NRC regulatory/licensing/inspection activities. RES also has responsibilities related to the resolution of generic safety issues and to the review of licensee submittals regarding individual plant examinations. It is the responsibility of RES to conduct the NRC's rulemaking process, including the issuance of regulatory guides and rules that govern NRC licensed activities.

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