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NRC Safety Research in Support of Regulation – FY 1991

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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ABSTRACT

This report, the seventh 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 1991.

The goal of this office is to ensure that safety-related research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This research is necessary to make certain that the regulations that are imposed on licensees provide an adequate margin of safety so as to protect the health and safety of the public. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

CONTENTS

TABLES

HIGHLIGHTS

Integrity of Reactor Components

Pressure Vessel Safety

- The pressurized thermal shock rule, 10 CFR 50.61, was amended on May 15, 1991, to incorporate the embrittlement calculation procedure currently contained in Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

- Analysis methods, material property models, and acceptance criteria have been developed that permit evaluation by the regulatory staff of the potential for ductile failures of reactor pressure vessels containing materials whose Charpy upper shelf energy (the indication of reactor vessel toughness) falls below the 50 ft-lb regulatory limit. A regulatory guide will be prepared based on this information. There are 17 operating reactors that may be affected by this issue.

- Thermal annealing of the Novovoronezh Unit 3 nuclear power plant reactor vessel, witnessed by a team of researchers and industry representatives led by RES staff members, demonstrated feasibility for reducing embrittlement due to neutron radiation in U.S. plants.

Piping Integrity

- Dynamic, simulated seismic loading tests were completed on typically sized piping. The results showed that the effects of dynamically imposed strain rates on fracture toughness generally are adequately addressed by current models used by the NRC for leak-before-break analyses of LWR piping systems.

Aging Research

- A report (revision to NUREG/CR-5587) was issued providing the technical bases for identifying the risk-significant components using prioritization procedures. The report includes a description of a procedure for identifying those aged **passive** components with the most impact on plant risk. It described the effect that inspection and maintenance actions can have in controlling aging and the consequential reduction in risk.

- Accelerated aging and accident survival tests of cable products were completed at the Low Intensity Cobalt Array facility at the Sandia National Laboratories. During these tests, cables were aged to the equivalent of 20, 40, and 60 years of operation. The results of these experiments showed that neither tensile strength nor any of the several electrical measurements taken during the experiment exhibited a consistent trend with aging. Most of the cables were found to be functional throughout the 60-year aging and the loss-of-coolant-accident tests that followed aging.

Reactor Equipment Qualification

- A modified valve thrust formula for bounding closing thrust requirements for gate valves was developed. The formula reflects the effects of friction, temperatures, pressure, and fluid conditions that occur in a motor-operated valve during actual operation. The use of the revised equation more accurately predicts the force required to close a gate valve against high flows that will occur if a pipe breaks. This estimate is then used to check whether the proper size of electric motor is installed on the valve to cause the valve to close under potential blowdown conditions.

Earth Sciences

- Field investigations of the surface rupture accompanying the 1989 Ungava, Quebec, earthquake by the Geological Survey of Canada indicated that the main surface rupture was 2 kilometers longer than mapped in 1990. This fault is the only documented historic coseismic surface rupture in eastern North America. This discovery documented the potential for earthquake surface rupture in central and eastern North America and provided a clear indication of the earthquake source characteristics important for the occurrence of surface rupture.

Seismic Engineering Research

- Results were published (NUREG/CR-4867, January 1991) on seismic testing of relays associated with Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors," Unresolved Safety Issue (USI) A-46. The issue of relay chatter during an earthquake has been a difficult issue to resolve because there are many different types of relays. As a result of this work, a list of less seismically rugged relays has been developed and will be used in conjunction with the implementation of USI A-46 and the individual plant examinations for external events.

- Agreement has been reached to build an experimental structure in Hualien, Taiwan, for the purpose of using actual earthquake data to check analytical predictions. The Hualien experiment is a joint venture involving entities from the United States, France, Taiwan, Japan, and Korea. This agreement represents a major milestone in the program to validate the soil-structure-interaction assumptions used currently in plant licensing and issue resolution.

Preventing Damage To Reactor Cores

- Personnel performance research was used to develop the Human Performance Investigation Process to provide a standardized method to identify the causes of human errors. The process was pilot-tested by Region I inspectors. The feasibility of an initial method for observing certain organizational factors was evaluated in cooperative tests with two utilities.

Reactor Containment Performance

- The technical evaluation to support a revision to the offsite radiological assessment criteria for radioiodine (TID-14844) was completed. The results indicate that, with proper pH control of the containment water inventory, elemental iodine inventories can be maintained at low levels.
- Integral testing of the direct containment heating phenomenon, one of the dominant potential early containment failure mechanisms postulated for severe accidents, was initiated.
- Experimental research on debris coolability was initiated through two programs: the WETCOR program conducted at the Sandia National Laboratories and the cooperative ACE/MACE program conducted at the Argonne National Laboratory. These programs are intended to provide the data concerning the coolability of core debris, under severe accident conditions, for advanced reactors.

Reactor Accident Risk Analysis

- Reviews were completed of PRAs for the South Texas plant, Diablo Canyon plant, and the GE Advanced BWR.
- The final version of NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," was issued in December 1990.

Confirming Safety of Nuclear Waste Disposal

- Final amendments to 10 CFR Part 40 on licensing procedures for the custody and long-term care of uranium and thorium mill tailings disposal sites were issued in October 1990.

Resolving Safety Issues and Developing Regulations

Generic Safety Issues

- Final technical resolution for four generic issues was completed (see Table 5.2).
- Prioritization of 12 generic issues was completed (see Table 5.1).

Developing and Improving Regulations

- The Commission has approved a complete revision to the NRC regulations for radiation protection in 10 CFR Part 20. The final rule was issued in May 1991.
- The final rule, 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was issued on July 10, 1991.
- The final rule, 10 CFR Part 54, "Nuclear Power Plant License Renewal," was issued on December 13, 1991.
- A proposed rule, 10 CFR Part 51, "Environmental Review for Renewal of Operating Licenses," was issued on September 17, 1991. This rule reflects the draft Generic Environmental Impact Statement, NUREG-1437.
- A final rule, 10 CFR 50.73, on access authorization at nuclear power plants and an accompanying regulatory guide were issued in April 1991.
- A final rule that amended the regulations in 10 CFR Parts 20, 30, 40, and 70 to revise licensee reporting requirements regarding notifications of incidents related to radiation safety was issued in August 1991.
- A final rule that amended the regulations in 10 CFR Part 50 to require the licensee to implement the NRC-approved Emergency Response Data System (ERDS) at all nuclear power plants was issued in August 1991.
- A final rule to amend the 10 CFR Part 35 regulations that apply to the medical use of byproduct material was issued in July 1991.
- A final rule that would recognize a third-party certification program of the American Society for Nondestructive Testing (ASNT) was issued in March 1991.

Severe Accident Implementation

- Individual plant examination submittals were received and reviews initiated for Oconee, Seabrook, and Turkey Point. The draft safety evaluation report was completed for the Yankee Rowe submittal, and the review of the Seabrook submittal was completed.

Radiation Protection and Health Effects

- A draft report (NUREG/CR-5631), "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses," which provides a methodology for calculating internal doses to the embryo/fetus and a data base for selected radionuclides, was issued for comment.

- Models on health effects of ionizing radiation (presented in NUREG/CR-4214) were modified based on 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 the revised recommendations of ICRP-60 (ICRP 1991).

EXECUTIVE SUMMARY

NRC safety research is vital for implementing a large number of the agency's programs. Research provides the bases for timely rulemaking and related licensing and inspection activities that are based on NRC's longstanding philosophy of defense in depth. This philosophy provides a clear and logical structure for the safety research mission area, which consists of five major programs: Integrity of Reactor Components, Preventing Damage to Reactor Cores, Reactor Containment Performance, Confirming Safety of Nuclear Waste Disposal, and Resolving Safety Issues and Developing Regulations.

Provided herein are the findings, results, and accomplishments of the safety research mission areas that (1) have led to, or are being incorporated in, **specific Commission and staff actions** to ensure or **enhance the level of safety** in activities or facilities being regulated; (2) demonstrate a **need for change** in regulations or regulatory approach; or (3) **confirm** or support **the regulations** or regulatory approach. A summary of the results in the five safety research programs is provided below. Regulatory products emerging from these programs are listed in the appendix to this report.

A. Integrity of Reactor Components

1. Specific Actions to Enhance Level of Safety

a. Embrittlement of reactor pressure vessel materials is the most important issue for pressure vessel safety, and pressurized thermal shock (PTS) is the most serious postulated transient for these vessels. The regulation concerning the permissible level of embrittlement to guard against vessel failure due to PTS is 10 CFR 50.61. The procedure for calculating embrittlement in the rule, however, has not been in agreement with Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." The rule was therefore amended on May 15, 1991, to incorporate the current regulatory guide formulation.

b. Some elbows and pump and valve bodies in light-water reactors (LWRs), as well as the primary system coolant pipes of many pressurized water reactors (PWRs), are made of cast stainless steels. These materials, however, based on their composition and processing, may gradually lose toughness simply by exposure to normal operating temperatures; too much of a loss could eliminate the safety margins. A methodology has now been developed and reported in NUREG/CR-4513 (June 1991) that allows increasingly more accurate estimations of the fracture toughness and of the kinetics of toughness loss for components in service based on the material specification, composition, and other knowledge about the specific materials. Relatively conservative predictions can be made if only the material chemistry is known. However, more accurate predictions can be made if more is known about the material characteristics. Thus, much improved judgments can be made about the long-term serviceability of the component.

c. Continuous online acoustic emission (AE) monitoring techniques are now available to evaluate crack initiation and growth in reactor components. ASME Section XI Code Case N-471 allows use of this method to monitor crack growth in operating reactor components. The technology is currently being used to monitor and evaluate the growth of an acceptably small intergranular stress corrosion crack in a safe-end weld at the Limerick Unit 1 reactor. Monitoring during the May 1989 to September 1990 fuel cycle has been completed and the results analyzed. Comparisons were made between the crack growth indicated by AE and that indicated by followup ultrasonic testing. The AE monitoring at Limerick Unit 1 was deemed sufficiently useful so as to be continued for a second fuel cycle. This application will provide further validation of the technology.

d. Weld sections have been obtained from the reactor pressure vessel of the canceled Midland Unit 1. These welds are typical of other welds fabricated by Babcock and Wilcox using materials that are highly sensitive to neutron radiation embrittlement and have a low resistance to ductile tearing. While similar welds have been fabricated in the laboratory, this is the first opportunity to study these welds in detail using as-fabricated materials. The studies concentrated on fully evaluating the unirradiated material characteristics and the flaw density in the as-fabricated weld. Additional work is under way to study the embrittlement trends for these materials, providing a detailed study of what may be the "life-limiting" factor for as many as 17 nuclear power plants.

e. Thermal annealing has been identified as one means to recover the properties of highly embrittled reactor pressure vessels. Several research efforts have been conducted in the past to evaluate the material response to thermal annealing, although there has been only limited evaluation of the engineering feasibility of high-temperature thermal annealing. Building on cooperation developed under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety, NRC staff members led a team of researchers and industry representatives to witness the annealing of the pressure vessel of the Novovoronezh Nuclear Power Plant, Unit 3, in the USSR (currently known as CIS). While several differences between the Soviet and U.S. designs and conditions were identified, it was generally concluded that the Soviet experience demonstrated the feasibility of annealing a U.S. reactor pressure vessel. Additional work is under way to improve our ability to predict annealing recovery and reembrittlement rates.

f. Seismic testing of electrical and mechanical equipment involves (1) qualification testing to [illegible] prescribed level to ensure integrity of the [illegible] and (2) testing beyond design basis con[illegible] to determine margins and failure [illegible]. This latter is termed fragility testing and is necessary for seismic probabilistic risk assessments (PRAs) and margin studies. Seismic fragility information is much rarer than qualification information but is necessary for quantifying margins and estimating risks. As a result of a 5-year data collection and analysis activity involving more than 15 source organizations, a comprehensive list of seismic fragilities became available for electrical and mechanical equipment. Both the median and uncertainties are quantified, and caveats associated with the use of the data are given. This information has been used for USI A-46, "Seismic Qualification of Equipment in Operating Plants," and is also needed for the seismic portion of the individual plant examinations for external events (IPEEE) and for use in seismic PRAs on future plants.

g. The Trojan plant has reactor pressure vessel (RPV) supports located in the belt-line region that are subjected to high tensile stresses. It cannot be ensured that during design basis transients the supports can maintain their integrity with the postulated low temperature-low flux embrittlement. A consequence evaluation, however, has demonstrated that RPV support failure will not lead to unacceptable consequences because (1) the inlet and outlet piping can transfer loads during postulated earthquakes and pipe ruptures to the steam generator and coolant pump supports; (2) the steam generator and coolant pump supports can carry the additional loads; (3) control rods can still insert during the most unfavorable tilting due to support failure; (4) instrument tubes can accommodate the most unfavorable motions associated with support failure; (5) pump coastdown is not adversely affected to a serious degree; and (6) branch lines do not rupture because of support failure. Unfortunately, these results cannot be generalized to other reactors. A Task Action Plan (TAP) for resolution of Generic Safety Issue 15, "Radiation Effects on Reactor Vessel Supports," was developed. The TAP addressed the underlying causes of support embrittlement, the consequences of support failure, and possible methods to mitigate the effects of support embrittlement.

h. Many precursors to intersystem LOCA (ISLOCA) have occurred leading to concerns that this scenario would dominate early fatality risks. To address these concerns, a major research effort involving engineering, PRAs, and human factors was initiated. One aspect of this work deals with estimating the pressure capacity of low-pressure piping to withstand RCS pressures and temperatures. A milestone effort to develop methods for determining median pressure capacities and uncertainties for piping, flanges, pumps, valves, and various tanks was completed. The method was exercised for three PWRs to allow the staff to prioritize the factors that contribute to ISLOCA risk.

i. Many uncertainties exist about the causes of earthquakes in the Eastern and Central United States and about the nature of earthquake ground motions. As part of its overall plan to reduce these unknowns, the NRC is supporting the establishment of the National Seismographic Network (NSN) by the U.S. Geological Survey. This NSN will replace the regional seismographic networks currently funded by the NRC by FY 1993. The NRC is also supporting many ground motion research projects.

j. Information from the present networks and analyses of ground motion data has substantially reduced these uncertainties over the past decade. Information that will be acquired by the high precision instruments of the NSN, combined with state-of-the-art analytical methodologies, will greatly accelerate the resolution of many of the remaining unknowns in the future.

k. Geologic and seismic unknowns that lead to a wide range of uncertainties in seismic hazard analyses include causes of seismicity in the Eastern and Central United States; the magnitudes, frequencies of occurrences, and levels of ground motions of earthquakes that occurred prior to the historic record; and the nature of seismic sources in both the Eastern and Western United States. As part of the overall effort to resolve these uncertainties, the NRC began, within the last decade, to focus research on identifying and evaluating geological evidence of prehistoric earthquakes. The results of these studies along with more and better earthquake records and modern analytic techniques have greatly improved the accuracy of estimating potential earthquakes and vibratory ground motions. Among these paleoseismic studies are paleoliquefactions, neotectonics, geodesy, fault sedimentation, Meers and Criner fault analysis, and paleoseismic investigations in the Pacific Northwest.

l. Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," is currently being revised to eliminate its major shortcomings. Among these shortcomings are the inflexibility of Appendix A so that difficulties arise in modifying it to incorporate new information and the fact that its text is too prescriptive. The revised regulation is called Appendix B. Several supportive regulatory guides are being developed simultaneously. The regulatory guides for Appendix B are being written to include present-day geoscience and engineering practices and are being made flexible so that new information and changes in the state of the art can be easily incorporated in them. The regulation contains brief general requirements, whereas the regulatory guides describe the requirements and methodologies in detail. The NRC staff is being assisted by the Lawrence Livermore National Laboratory (LLNL) and a panel of earth scientists and engineers.

m. During the past year, the staff performed a careful review of the Executive Order 12699, "Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction," and NRC requirements for the design and construction of buildings associated with nuclear power reactors and other activities. The "other activities" included Class 104 licenses for medical therapy and research and development facilities, processing of uranium ores in milling operations, high-level-waste repository licensing, onsite spent fuel storage, licensing of plutonium processing and fuel fabrication plants, and license application reviews for uranium enrichment facilities. It was concluded that the NRC's current practice meets the requirements of the Executive Order and no regulatory action is necessary.

2. Need for Change

a. Fatigue design of reactor components is governed by fatigue life curves in Section III of the ASME Code. These curves were produced from polished specimens tested at room temperature in air. But tests that have now been performed in typical reactor coolant environments and at typical operating temperatures show a shorter fatigue life. Although this has implications for license renewal applications, it is also important for current regulation. Contractor testing efforts have been expanded, and cooperation is under way with industry groups and the international community to produce appropriate data and methods so that a consensus can be reached on modifications to the Code curves and methodology, if necessary, for more accurate fatigue life predictions and design for future applications. In the short term, the international body of available data is being reviewed and evaluated to develop interim fatigue curves for the NRC's use to ensure the continued safe operation of today's plants.

b. Results from test reactor irradiations suggest that the current approach to shifting the fracture toughness curves to account for irradiation damage in reactor vessel materials may not completely account for that damage. It appears that the procedure may be underpredicting the actual shift in the fracture toughness curves, decreasing the anticipated margin of safety in some of the regulatory analyses. At the same time, other aspects of the overall pressure vessel integrity analyses are known to be extremely conservative so that the final evaluations are still quite conservative. These results have been presented to the cognizant ASME Section XI groups for their consideration. Additionally, test reactor irradiations are under way to provide the technical basis for any needed revisions.

c. It is necessary at this time to use the most advanced equipment methodologies in the process of acquiring and analyzing earthquake information to increase our knowledge about seismic sources and vibratory ground motion. Predicting vibratory ground motion through probabilistic seismic hazard assessments is becoming more important. Methodologies for such assessments must be studied in greater detail in order to arrive at more robust results.

d. The extent of the uncertainties regarding the nature, locations, and sizes of seismic sources; estimating the magnitudes of earthquakes on those sources; and developing ground motion models became apparent after completion of the LLNL and Electric Power Research Institute (EPRI) seismic hazard studies. To obtain more empirical information to improve seismic hazard study results, the NRC began supporting the research listed above under "Specific Actions to Enhance Level of Safety," item k.

3. Confirmation of Regulations

a. A modification to General Design Criterion 4 several years ago allowed utilities to eliminate many pipe whip restraints through establishment of leak before break (LBB) approach in piping systems of LWRs. Permission to employ LBB requires proof of appropriate material, loading, and environmental characteristics. Dynamic and simulated seismic loading tests have now been completed on typical size piping in the first International Piping Integrity Research Group (IPIRG-1) program. This work demonstrated that dynamically imposed strain rate effects on fracture toughness of the pipe generally are acceptably handled by the analysis methods currently being used by the NRC and the industry for LBB in LWR piping systems. Additional work is expected to be performed as part of a follow-on international group (IPIRG-2) to further evaluate loading rate and crack size effects and to evaluate the fracture of fittings containing cracks.

b. The potential for ductile failure of reactor pressure vessels is addressed in Appendix G to 10 CFR Part 50 by requiring that pressure vessel materials maintain a Charpy upper shelf energy of at least 50 ft-lb. If this regulatory limit is exceeded, an analysis must be performed to demonstrate an adequate margin against failure. Research results have contributed to the development of acceptance criteria for these analyses. The acceptance criteria and general analysis method are being considered for inclusion in Section XI of the ASME Boiler and Pressure Vessel Code. Additionally, models have been developed that permit estimation of the material properties needed in these analyses. Regulatory guidance will be prepared that will build on the ASME Code efforts and that will incorporate the material property models, providing a complete analysis methodology.

c. A draft regulatory guide issued for public comment was developed to advise utilities planning to renew their licenses of the necessary technical information needed to support their applications. The document provides guidance criteria for the selection of the key plant structures, systems, and components needing aging management programs; guidance on evaluation of operational and environmental factors that may contribute to age-related degradation of these key structures, systems, and components; aging mechanisms and their locations; and guidance on the mitigation and management of the resulting aging degradation. The necessary information used to develop this guide is a product of the Nuclear Plant Aging Research program.

d. Satisfying NRC seismic requirements (Appendix A to 10 CFR Part 100) has been accomplished in the past by basing conclusions on limited seismic and geologic information. Acquisition of data by the current regional seismographic networks has indicated that most of these conclusions were relatively accurate although untested by a large earthquake. The regulation as it is currently being revised will require even more empirical information. The greater precision of the NSN instruments and the results of the ground motion studies will add to the seismic data base and either continue to confirm previous conclusions based on the regulations or show where modifications are necessary.

e. Appendix A to 10 CFR Part 100 requires that maximum earthquakes be associated with either tectonic structures or tectonic provinces. Because of the paucity of geologic and seismic data, the boundaries of these structures and provinces were highly controversial. It is being proposed that the revised Appendix A (now Appendix B) to 10 CFR Part 100 require a deterministic analysis similar to that in the present Appendix A but also require a probabilistic analysis to take into account the uncertainties. Paleoseismic research will continue to provide information that can be used to validate the seismic hazard analyses, both the deterministic and the probabilistic ones.

B. Preventing Damage To Reactor Cores

1. Specific Actions to Enhance Level of Safety

a. Personnel performance research was used to develop the Human Performance Investigation Process to provide a standardized method to identify the causes of human errors. Both Region I and NRC headquarters personnel have been trained in the use of the process; it has also been pilot-tested by Region I inspectors. A study of operator performance during 8-hour versus 12-hour night shifts was conducted and the data are currently being analyzed. A method derived from the DeGroot memory paradigm was tested as a potential measurement method for evaluating human-computer interface. In the area of methods to support PRA, the feasibility of an initial method for observing the effects of certain organizational factors on safety was evaluated in cooperative tests with two utilities. Alternative methods are being explored for field-testing in 1992. In the area of risk-based performance indicators, a method has been developed for monitoring the availability of three selected safety systems.

b. The March 9, 1989 instability at the LaSalle-2 power plant led to a reevaluation of BWR stability. More than 90 related BWR transients were simulated on the Brookhaven National Laboratory Engineering Plant Analyzer to determine the causes and the potential magnitude of power, flow, and temperature oscillations as well as the possible consequences from postulated scenarios on suppression pool conditions. The results of these tests were instrumental in bringing about improvements to BWR ATWS emergency operating procedures.

2. Confirmation of Regulations

a. A research project applied code scaling, applicability, and uncertainty (CSAU) analyses using RELAP5/MOD3 to a hypothetical PWR (B&W) small-break LOCA. The uncertainty in predicting the safety parameter of reactor vessel water inventory was determined. This study showed that this type of accident can be evaluated using these methods with excellent accuracy. The study also showed that, while the large-break LOCA scenario follows a rather standard path, the risk of small-break LOCA scenarios is dominated by operator action decisions. This information is important in evaluating licensees' accident management plans. These decisions were chosen in advance, were appropriate and consistent, and were the same for all calculations. Thus, for example, the same break scenario with primary coolant pump starts by the operator would follow a different path and the resulting minimum coolant level would likely be different. However, it is believed that the uncertainty would be similar.

b. Research was carried out to evaluate the potential loss of required shutdown margin during refueling operations. A report, NUREG/CR-5771, analyzing the operation was issued and showed that the probability of inadvertent criticality during reloading is acceptably low and risk to the public is acceptably small. A bulletin issued to alert utilities to the possible loss of shutdown margin during refueling has been shown to be sufficient. NUREG/CR-5771 will provide further insights to utilities in addressing the problem.

C. Reactor Containment Performance

1. Specific Actions to Enhance Level of Safety

a. In February 1987, the NRC issued the draft version of the "Reactor Risk Reference Document" (NUREG-1150), as well as a series of supporting contractor reports, for public comment. The draft report assessed the risks from possible core damage accidents in five U.S. nuclear power plants—Surry (Virginia), Zion (Illinois), Sequoyah (Tennessee), Peach Bottom (Pennsylvania), and Grand Gulf (Mississippi). The report discussed the implications of the five analyses on regulatory issues such as implementation of the Commission's Safety Goal and Severe Accident Policy Statements. Two NRC-funded reviews of the draft report were obtained and published as NUREG/CR-5000 and NUREG/CR-5113. In addition, the American Nuclear Society sponsored and published a review of the draft report.

The NRC staff and supporting contractors updated the five risk analyses. The updates, which were quite extensive, were intended to reflect comments received, to reflect the present plant design and operating characteristics, to improve the methods used, and to incorporate new experimental data on severe accidents resulting from the research programs of NRC and others.

The completed new version of NUREG-1150 was delivered to the Commission in April 1989 and published as a second draft for peer review in June 1989. A peer review panel, organized under the Federal Advisory Committee Act, completed its formal review of the document and provided generally positive findings. The final version of the report (NUREG-1150) was issued in December 1990.

b. Probabilistic risk analysis (PRA) is used by the NRC staff to support the resolution of a wide spectrum of regulatory issues. For licensed plants, PRAs are sometimes voluntarily submitted by licensees to support their specific proposed means for resolving such issues. For advanced plants of the future, applicants are required to perform and submit PRAs as part of their overall license applications. The following reviews were performed:

South Texas (Texas). This PRA was a voluntary submittal by the licensee, who plans to use the document as a reference in future technical discussions on regulatory issues.

Diablo Canyon (California). This PRA was developed and submitted as part of a long-term seismic program.

GE Advanced BWR. A PRA was submitted as part of the licensing application for this advanced BWR. In May 1991, a draft safety evaluation report was completed and transmitted to NRR. This was subsequently transmitted to General Electric.

c. In support of the advanced reactor design certification process, EPRI has developed a set of requirements to guide the design of such reactors. One part of this guidance relates to the performance and use of PRA methodologies. A review of this guidance was made with a draft evaluation transmitted to NRR in August 1991. This evaluation was reviewed by NRR and transmitted to EPRI in October 1991.

2. Confirmation of Regulations

a. Examination of the vessel samples from the TMI-2 lower head indicated that a small region of the lower head (approximately 2 feet in diameter) experienced inner surface temperatures of about 1350 K. The examination also indicated that the temperature 2 inches into the wall was about 100 K lower than the inner surface temperature. (The lower head thickness was 5 inches.)

Lower head failure maps have been developed for local penetration failure and for local and global creep rupture failure as a function of the characteristics of the lower head debris and of the lower head structure. The analytical basis for these maps is supported by the results of the metallurgical analyses of the samples in the TMI-2 lower head program. A NUREG/CR report on this analysis is undergoing peer review and will be issued in FY 1992.

b. A comprehensive draft "Research Plan for Melt Progression Issue Resolution" was prepared and subjected to expert peer review. The plan and comments of the reviewers will be used as a basis to update the Severe Accident Research Plan in the melt progression area.

c. Technical evaluations to support the revision to current methods for evaluating the chemistry of the iodine component of the source term entering the containment during a severe accident (TID-14844) were completed. The Oak Ridge National Laboratory performed analyses using a chemical kinetic model to determine the equilibrium distribution of the iodine, cesium, hydrogen, and steam species entering the containment. The results indicate that iodine entering the containment is at least 95 percent CsI (Cesium Iodide) and 5 percent as I or HI (elemental iodine or hydrogen iodide).

Once within the containment CsI is expected to deposit onto interior surfaces and dissolve in water pools, forming I^- (iodide) in solution. Subsequently, iodine behavior within the containment depends on the t[illegible] and pH of the water solution. If pH control is available and maintained, very little of the dissolved iodine will be converted to elemental iodine.

d. A severe accident scaling methodology (SASM) was developed to guide the formulation of experimental programs and analytical methods. Documentation of the SASM and application of the methods to the direct containment heating (DCH) issue were addressed in NUREG/CR-5809, published in November 1991.

Integral testing was initiated to investigate the containment loadings resulting from DCH. The experimental program will explore integral DCH phenomena at different scales for representative reactor designs.

e. A joint agreement was reached for a cooperative program with the Ministry of International Trade and Industry of Japan and the Nuclear Power Engineering Center. Under this program high-temperature high-speed hydrogen combustion research will be conducted for 5 years. Two combustion vessels will be used for this research program at the Brookhaven National Laboratory. The conceptual design report that describes the program was completed.

f. An interim version (CORCON MOD3) of the core-concrete interaction code CORCON was developed. In addition to improved thermal-hydraulic modeling, CORCON MOD3 incorporates the VANESA model of aerosol generation and radionuclide release during core-concrete interactions. Benchmark validation of CORCON MOD3 is currently under way.

Two experimental research programs addressing the debris coolability issue were initiated. The Advanced Containment Experiment (ACE) program, conducted at the Argonne National Laboratory, is an internationally sponsored program with NRC participation, one phase of which deals with melt cooling issues, the Melt Attack and Debris Coolability Experiment (MACE). One MACE test was performed under the ACE program. The WETCOR program, conducted at the Sandia National Laboratories, is an NRC-sponsored program that supplements the MACE program and that deals with the morphological behavior of core debris during the cooling process. The first of a series of three WETCOR tests was conducted.

g. Completion of the initial study of BWR Mark I containment shell failure was documented by issuance of NUREG/CR-5423 in July 1991. Peer review of the study indicated the need for additional evaluation of selected subjects, which is being undertaken in FY 1992.

h. The principal integrated computer codes that support the evaluations of nuclear plant responses to postulated severe accident scenarios are being independently reviewed for their adequacy to meet NRC objectives. The first such comprehensive peer review was completed for the MELCOR code; other codes such as CONTAIN and SCDAP/RELAP5 are being reviewed starting in FY 1992. The results will facilitate an efficient focus on critical needs for completing code development.

D. Confirming Safety of Nuclear Waste Disposal

1. Specific Actions to Enhance Confidence in Waste Management Licensing

a. A petition for rulemaking (PRM-61-1) from the North Carolina chapter of the Sierra Club was completed. The petitioner requested the Commission to adopt a regulation to permit the design and construction of a zero-release low-level radioactive waste disposal facility in a saturated zone. The petitioner stated that the regulation was necessary in order for the General Assembly of North Carolina to consider a waiver of a North Carolina statute that requires that the bottom of a low-level waste facility be at least 7 feet above the season high water table. A denial of petition was published in the *Federal Register* in July 1991 (56 FR 34035).

b. Final amendments to 10 CFR Part 40 that provide licensing for the custody and long-term care of uranium and thorium mill tailings disposal sites were published in the *Federal Register* in October 1990 (55 FR 45591). These procedures will ensure that long-term performance of uranium mill tailings disposal sites is properly monitored and that deterioration requiring restoration will be detected so that appropriate action can be taken.

E. Resolving Safety Issues and Developing Regulations

1. Specific Actions to Enhance Level of Safety

a. Final technical resolution for the following generic issues was completed:

- GI-128, "Electrical Power Reliability"
- GI-130, "Essential Service Water System Failures at Multiplant Sites"
- GI-135, "Steam Generator and Steam Line Overflow"
- GI-II.J.4.1, "Revise Deficiency Report Requirements"

b. Prioritization of the following generic issues was completed:

- GI-24, "Automatic Emergency Core Cooling System Switch to Recirculation"—MEDIUM

- GI-38, "Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris"—DROP

- GI-72, "Control Rod Drive Guide Tube Support Pin Failures"—DROP

- GI-73, "Detached Thermal Sleeves"—NEARLY RESOLVED

- GI-100, "Once-Through Steam Generator Level"—DROP

- GI-120, "On-Line Testability of Protection Systems"—MEDIUM

- GI-143, "Availability of Chilled Water Systems and Room Cooling"—HIGH

- GI-150, "Overpressurization of Containment Penetrations"—DROP

- GI-151, "Reliability of Anticipated Transient Without Scram Recirculation Pump Trip in BWRs"—MEDIUM

- GI-153, "Loss of Essential Service Water in LWRs—HIGH

- GI-A19, "Digital Computer Protection System"—LICENSING ISSUE

- GI-B22, "LWR Fuel"—DROP

c. In March 1988, the Commission issued a Policy Statement on the Maintenance of Nuclear Power Plants. In this statement, the Commission indicated its intention to pursue a rulemaking on maintenance. In developing this proposed rulemaking, the staff had extensive interactions with U.S. industry (airline and nuclear) and studied foreign nuclear maintenance programs and practices. A 3-day public workshop was held in July 1988 to solicit comments on rulemaking options. The information gathered was used in formulating the proposed rule and its supporting regulatory guide. The Commission issued the proposed rule for public comment in November 1988 and the supporting draft regulatory guide in August 1989. In December 1989, the Commission issued a revised policy statement to restate its views with respect to maintenance and to indicate its intention to hold rulemaking in abeyance for a period of 18 months. During the 18-month time interval, the Commission monitored industry initiatives and progress in maintenance improvements and reevaluated the need for issuing a final rulemaking. Based on its evaluation, the Commission concluded that a regulatory framework should be in place to provide a mechanism for evaluating the overall continuing effectiveness of licensee maintenance programs. Accordingly, the Commission issued a final rule, 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," on July 10, 1991.

d. The NRC has been considering what requirements should be placed on nuclear power plants in the event that licenses to operate beyond the 40-year term of the original license should be granted. Public comments on license renewal requirements have been solicited three times through the *Federal Register*—the first time in connection with seven major license renewal issues (published November 6, 1986) and the second as part of an advance notice of proposed rulemaking (published August 29, 1988). The advance notice requested comments on NUREG-1317, "Regulatory Options for Nuclear Plant License Renewal," issued in August 1988. Comments were summarized and analyzed in NUREG/CR-5332, "Survey and Analysis of Public Comments on NUREG-1317: Regulatory Options for Nuclear Plant License Renewal," issued in March 1989. The third time occurred when the NRC published the proposed rule for nuclear power plant license renewal on July 17, 1990 (55 FR 29043). The final rule (10 CFR Part 54) with appropriate supporting documents was published on December 13, 1991 (56 FR 64943).

e. As part of a separate rulemaking, the NRC has undertaken a generic environmental study with the purpose of defining the scope and focus of environmental effects that need to be considered in individual relicensing actions. An advance notice of proposed rulemaking (10 CFR Part 51) was issued on July 23, 1990 (55 FR 29964). Also, a notice of intent to prepare a generic environmental impact statement (GEIS) on the effects of renewing the operating license of individual nuclear power plants was issued (55 FR 29967). The NRC published the proposed rule and draft GEIS for comment on September 17, 1991 (56 FR 47016).

f. The pressurized thermal shock rule, 10 CFR 50.61, was amended in May 1991 to incorporate the embrittlement calculation procedure currently contained in Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials."

g. A final rule, 10 CFR 50.73, on access authorization at nuclear power plants and an accompanying regulatory guide were published in the *Federal Register* in April 1991 (56 FR 18997). The rule requires a nuclear power reactor licensee to have an access authorization program in its site physical security plan. This would provide increased assurance that persons granted unescorted access to protected and vital areas are trustworthy and do not pose a threat to commit radiological sabotage.

h. A proposed rulemaking, 10 CFR Part 74, on the material control and accounting requirements for uranium enrichment plants and an associated regulatory guide were published for public comment in the *Federal Register* in December 1990 (55 FR 51726). The rulemaking is following an accelerated schedule because a license application has been filed by Louisiana Energy Services for the construction and operation of a gas centrifuge plant that would produce low-enriched uranium for the commercial market. The rule will facilitate the licensing of such a facility. We expect the final rulemaking will be published in the *Federal Register* early in FY 1992.

i. In a program initiated in 1985 and continued through 1991, the NRC staff undertook to evaluate existing regulatory requirements in terms of their risk effectiveness and to eliminate or modify requirements with only a marginal safety importance. Final recommendations were forwarded to the Commission in July 1991. Implementations of final recommendations will begin in FY 1992.

j. A final rule that amended the regulations in 10 CFR Parts 20, 30, 40, and 70 to revise licensee reporting requirements regarding notifications of incidents related to radiation safety was published in the *Federal Register* in August 1991 (56 FR 40757). This rule will ensure that significant occurrences at facilities operated by material licensees are promptly reported to the NRC. The Commission will be able to determine whether a licensee has taken the actions necessary to protect public health and safety and whether generic safety concerns that may require prompt NRC actions are identified.

k. A final rule that amended the regulations in 10 CFR Part 50 to require the licensee to implement the NRC-approved Emergency Response Data System (ERDS) at all nuclear power plants was published in the *Federal Register* in August 1991 (56 FR 40178). Earlier, the proposed rule was published in the *Federal Register* for public comment in October 1990 (55 FR 41695). The rule would supplement the voice transmission over the existing Emergency Notification System (ENS) and require a direct electronic data link between the licensee's computer and the NRC's Operation Center to be activated by the licensee during an alert or higher emergency condition to transmit timely and accurate updates of critical information on plant conditions. This would allow the NRC to perform its primary role during an emergency at a licensed nuclear power facility, which is one of monitoring the licensee to ensure that appropriate recommendations are made with respect to necessary offsite actions to protect public health and safety.

l. A final rule was published in the *Federal Register* in July 1991 (56 FR 34101) to amend the 10 CFR Part 35 regulations that apply to the medical use of byproduct material. The amendments require medical-use licensees to implement quality management (QM) programs and revise misadministration reporting requirements. Implementation of the new performance-based requirements is supported by the issuance of a regulatory guide that includes specific guidance for QM programs and an approach acceptable to the NRC for meeting the requirements of the final rule. The rule provides a high confidence that byproduct material and radiation from byproduct material will be administered as directed by the authorized user physician. The feasibility of this approach was evaluated during a pilot program involving 70 medical-use licensees and subsequent discussion with professional associations and Agreement States.

m. The Commission issued a denial of a petition for rulemaking (PRM-50-50) from Charles Young for publication in the *Federal Register* in January 1991 (56 FR 1749). The petitioner requested the Commission to amend its regulations to prevent nuclear power plant operators from deviating from license conditions or technical specifications during an emergency. The petitioner believes that nuclear power plants should be operated in accordance with the operating license and appropriate technical specifications and that requiring a senior operator to follow the technical specifications during an emergency enhances plant safety.

n. Approximately 16 safety-related regulatory impact analyses (both initiated and completed) were processed in FY 1991.

o. Major efforts on the individual plant examination (IPE) program involved review of IPE submittals and completion of the procurement process to obtain contractual assistance for the IPE reviews. Three additional IPE submittals were received and reviews initiated. They were the Oconee (South Carolina), Seabrook (New Hampshire), and Turkey Point (Florida) submittals. The draft safety evaluation report was completed for the Yankee Rowe submittal. Also, the review of the Seabrook IPE was completed. Because the Turkey Point IPE submittal was the first submittal not based on a previously reviewed PRA, it is the first submittal selected for a more indepth review.

p. In December 1987, the NRC established an External Event Steering Group (EESG) to make recommendations concerning the individual plant examinations for vulnerabilities to severe accidents initiated by external events (e.g., earthquakes, floods, fires). Recommendations were needed relative to: (1) what external events need consideration in the IPE, (2) what methods can be used in the examination, and (3) how the IPE for external events (IPEEE) can be coordinated with other ongoing regulatory activities involving external events, particularly in the seismic area.

In May 1990 the staff completed work on a draft generic letter and draft guidance document (NUREG-1407) to be sent to licensees, which describes the scope, acceptable methods, and reporting requirements for the IPEEE. The staff revised the generic letter and NUREG-1407 to clarify and incorporate changes resulting from comments received at a workshop and subsequently issued the final generic letter, GL 88-20, Supplement 4, and NUREG-1407 in June 1991. The generic letter required licensees to submit their plans and schedules for performing their IPEEEs in December 1991, with completion of their IPEEEs by June 1994.

q. Consideration of source terms entered the regulatory process because the Commission's reactor site criteria (10 CFR Part 100) require that an accidental fission product release from the core into the containment should be assumed to occur and that its radiological consequences should be evaluated assuming that the containment leaks at its "expected demonstrable leak rate."

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 activity have been identified that may benefit from change as a result of source term and severe accident research. In FY 1991, work continued on a replacement for TID-14844, and the staff initiated rulemaking to decouple siting from plant design. This effort will more directly incorporate requirements related to acceptable site characteristics. A proposed rule is expected to be sent to the Commission in June 1992.

r. The staff initiated rulemaking to add emergency planning requirements to 10 CFR Part 72 for independent storage of spent nuclear fuel and high-level radioactive waste. It is expected that a proposed rule will be sent to the Commission by mid-1992. Also a revision to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," was initiated to revise the approach for the development of Emergency Action Levels. A draft is expected to be issued for public comment in early 1992.

s. A study to improve understanding of the contribution of maternal radionuclide burdens to prenatal radiation exposure was continued with significant progress. The NRC recently published for comment NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses." This NUREG provides methods for calculating internal doses to the embryo/fetus and a data base for selected radionuclides. Work is currently under way to reissue NUREG/CR-5631 with an expanded data base that will include uranium and additional isotopes of elements such as Sr-89, Cs-134, and Pu-238. Research that will permit inclusion of other radionuclides, such as technetium, molybdenum, americium and other transuranics, is planned. The methods and data developed under this project will be used by the NRC in preparation of an interim regulatory guide describing acceptable methods of compliance with Section 20.208 of the revised 10 CFR Part 20. This guide will be revised as new information warrants. The methodology will also be used to calculate doses in cases of accidental releases of radioactive materials.

t. Revision 1 to NUREG/CR-4214, "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis," published in May 1989 contains 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. An addendum entitled "Modification of Models Resulting From Recent Reports of Health Effects of Ionizing Radiation" was published in August 1991. The reports that led to the revision of models presented in the NUREG/CR-4214 are those 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 the revised recommendations of ICRP-60 (ICRP 1991).

u. A proposed rule for large irradiators was published for public comment in the *Federal Register* in December 1990 (55 FR 29043). A 2-day public workshop to discuss the proposed rule was held in Rockville, Maryland, in February 1991. Large irradiators are defined as those capable of delivering a dose of 500 rads in an hour to a person standing 1 meter from the sources. The final rule is scheduled for publication in FY 1992.

v. A final rule that would recognize a third-party certification program of the American Society for Nondestructive Testing (ASNT) was published in the *Federal Register* in March 1991 (56 FR 11504). The rule would give licensees the option of using the ASNT program in lieu of describing their training program to NRC. The certification program is expected to improve training and safety performance in the workplace.

w. A proposed regulatory guide, "Air Sampling in the Workplace," to meet the requirements of the new Part 20 was published in the *Federal Register* for public comment (56 FR 52078) in September 1991. The guide deals with issues such as what should the licensee do to demonstrate that samples are representative of the air inhaled by workers, and what measurements are necessary to be able to adjust derived air concentrations to account for particle size. The guide is accompanied by a technical manual, "Air Sampling in the Workplace," describing how the recommendations in the guide can be met. Both documents are scheduled to be issued in final form in FY 1992.

x. A proposed rule, published in the *Federal Register* (56 FR 46739) for public comment in September 1991, would amend the Commission's regulations concerning the licensing of uranium enrichment facilities to reflect changes made to the Atomic Energy Act of 1954 (the Act), as amended by the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990. The principal effect of these amendments is that uranium enrichment facilities would be licensed subject to the provisions of the Act pertaining to source material and special nuclear material rather than under the provisions pertaining to a production facility. The Commission is currently reviewing a license application by the Louisiana Energy Service Corporation to construct and operate a commercial uranium enrichment facility.

y. A proposed rule was published in the *Federal Register* (56 FR 50524) for public comment in September 1991 to amend the Commission's decommissioning regulations to require holders of a specific license for possession of byproduct material, source material, special nuclear material, and independent storage of spent nuclear fuel and high-level waste to prepare and maintain additional documentation identifying areas where licensed materials and equipment were stored and used. The Commission's intent is to provide both the NRC and the licensee the necessary information to ensure complete decommissioning of licensed facilities. In addition, this action also is consistent with similar requests made at the Synar Committee Hearing on decommissioning and an earlier GAO report.

2. Confirmation of Regulations

Research programs have been initiated to independently evaluate the new design features as well as the adequacy of advanced nuclear reactor designs to withstand severe accidents. These efforts will provide the technical basis for NRC's support of plant design certifications.

1 INTEGRITY OF REACTOR COMPONENTS

This program is conducted to ensure that reactor plant systems and related components perform as designed during both normal operations and accidents in order that their functional integrity and operability can be maintained over the life of the plant. Reactor safety depends on maintaining the reactor system pressure boundary undamaged and leaktight. 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 release of hazardous fission products.

1.1 Reactor Vessel and Piping Integrity

1.1.1 Statement of Problem

The primary system of a light-water reactor (LWR) is the principal boundary enclosing the nuclear fuel core and the water coolant used both to maintain suitably low temperatures on the fuel cladding and to conduct the heat from the fission reaction to heat exchanger (for a pressurized water reactor (PWR)) where it can be converted into steam for electricity generation. The primary system includes the reactor pressure vessel, primary coolant piping, primary pumps, and steam generators. For boiling water reactors (BWRs), the primary system must include the steam line at least 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 heat, either from direct power generation 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 procedures 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 stress 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, guides, etc., that pertain to the structural integrity of LWRs are focused 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 and Piping Integrity element is to ensure that appropriate analytical procedures and inspection methods exist for assessing the safety of components during normal service and accidents and that sufficient critical experiments are conducted to validate those procedures and methods.

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 bombardment by neutrons escaping from the fuel core during normal service. Experiments are thus conducted to develop a base of information on all the factors that will cause this 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. Thus, use is made of large-scale "models" that realistically represent the true components. Similarly, the ability to predict integrity in piping has required testing of full-sized sections of pipe having a variety of cracks that could develop in service to determine if such cracks could cause failure during either normal service or an accident. 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. Thus, many experiments are conducted on a wide variety of pertinent materials under a very wide 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 both detection and sizing. Improvements in methods are proposed and qualification procedures developed that can provide better assurance for not missing flaws in future inspections and for sizing flaws more accurately. Use is made of materials and components removed from actual service to measure the real condition of material properties resulting from years of service, to establish the real corrosion state, and to validate the existence of flaws that have been "called" and estimated in size through nondestructive examination procedures.

1.1.2 Research Accomplishments in FY 1991

1.1.3.1 Pressure Vessel Safety

The NRC's regulations for vessel safety have always been designed to ensure an adequate level of ductility and fracture toughness in the base metal and welds of reactor pressure vessels. However, many of the analysis methods and material characteristics used in evaluating vessel safety were based on engineering practice, which was intended to be very conservative. As plants have aged and as the industry has started to consider license renewal, it has become clear that the safety analysis methods must be accurate, not just arbitrarily conservative, and that the data bases used to deduce material characteristics and properties must reflect actual material behavior. Thus, validation of the analyses and the material data bases is necessary to ensure an adequate margin of safety and to preclude unnecessarily large margins.

The NRC's fracture evaluation research includes both analytical and experimental efforts. During FY 1991, research continued to evaluate the validity and accuracy of reactor pressure vessel fracture analyses, evaluate the effects of parameters that affect the fracture analyses to identify those warranting additional research, develop and refine analysis methods that can be used reliably in predicting reactor pressure vessel fracture, and develop the material property data needed as input to these analyses.

The potential for ductile failure of reactor pressure vessels is addressed in Appendix G to 10 CFR Part 50 by requiring that pressure vessel materials maintain a Charpy upper shelf energy of at least 50 ft-lb. If this regulatory limit is exceeded, an analysis must be performed to demonstrate an adequate margin against failure. During FY 1990, significant emphasis was placed on developing data and analyses that could be used to evaluate the potential for ductile failure of reactor pressure vessels. Research at the U.S. Navy's David Taylor Research Center in Annapolis, Maryland, and at the Oak Ridge National Laboratory (ORNL) was aimed at developing independent analyses for pressure vessel fracture and for evaluating the fracture resistance of the material based on results from small laboratory specimens. During FY 1991, most of this work was completed and contributed to the development of proposed changes to Section XI of the ASME Boiler and Pressure Vessel Code—changes that are being endorsed by the staff as acceptable criteria for evaluating materials with a low resistance to ductile tearing.

Additionally, during FY 1991, work was completed to develop equations for predicting the material properties needed in analyses of pressure vessels containing materials with a low resistance to ductile tearing. Using pattern recognition techniques, researchers at Modelling and Computing Service developed mathematical models capable of predicting the ductile tearing resistance of pressure vessel steels and weldments. This work, funded under a Small Business Innovation Research contract, provides the NRC and the industry a statistically based methodology for estimating the material properties needed in these analyses.

Several years ago, ORNL, as part of the NRC's Heavy Section Steel Technology program, performed several large-scale "benchmark" experiments—the so-called Intermediate Test Vessel tests and the Pressurized Thermal Shock Experiments—for the purpose of validating the fracture analyses used in reactor pressure vessel safety evaluations. More recently, other large-scale experiments have been conducted in the United Kingdom, the Federal Republic of Germany, and Japan. The results of these experiments, combined with those from the earlier ORNL work, provide test cases that cover a wide spectrum of conditions over which the pressure vessel fracture analyses are expected to be valid.

During FY 1990, as part of an effort by the Committee on the Safety of Nuclear Installations (CSNI) Principal Working Group 3, ORNL organized an international workshop to compare predictions of the fracture behavior in the large-scale experiments to the experiment results. The overall results of this effort led the NRC to increase the emphasis on refinements to the fracture analyses. This work was continued during FY 1991 and is expected to continue for at least another year, culminating in another analysis demonstration. The expanded program builds on the ongoing work at ORNL and the David Taylor Research Center and brings expertise at the University of Illinois, the University of Maryland, the University of Tennessee, and the University of Kansas to bear on the problem. The FY 1992 effort is expected to include other private sector research institutions.

As the fracture analysis technology has matured, the emphasis in the NRC's research has moved from broad spectrum scoping research to research aimed at developing analyses and the supporting data that can eliminate some of the very conservative assumptions incorporated in the early regulatory analyses. A significant initiative in the pressure vessel research program is aimed at evaluating the apparent increase in fracture resistance for shallow flaws.

During FY 1990, it was demonstrated that increases in fracture toughness for very shallow cracks could have a significant effect on pressurized thermal shock (PTS) analyses. This led to an analytical and experimental program to quantify the so-called shallow crack effect for reactor pressure vessel steels. During FY 1991, tests were performed to confirm that shallow cracks initiated at higher fracture toughness values than did deep cracks. This led to an expanded program, linked directly to the analysis efforts described above, that seeks to quantify and validate this fracture behavior. This program is expected to continue for 2 to 3 years. However, once completed, the results could have a major impact on pressure vessel safety analyses and may significantly reduce the currently perceived risk due to accidents such as PTS.

In addition to the research efforts, the FY 1991 program included an unusually large effort in support of the Office of Nuclear Reactor Regulation (NRR). A significant effort was devoted to performing independent analyses of the vessel failure frequency due to PTS transients for a particular plant. These efforts drew on expertise in probabilistic fracture mechanics, embrittlement trends, flaw size distributions, and inservice inspection techniques. While the regulatory decisions concerning this plant were made in NRR, the research efforts contributed substantively to the decision process. In addition to the PTS analyses, analyses were performed to evaluate alternative methods for determining pressure-temperature limits and low-temperature overpressure protection setpoints. The results of this work contributed directly to staff deliberations on industry proposals. The overall effort during FY 1991 demonstrated that the results and expertise developed by the research program provide a valuable resource that can be tapped to provide answers for unusual plant-specific applications.

Embrittlement due to neutron radiation is the chief aging mechanism for reactor pressure vessels. The NRC's regulations require surveillance programs, where samples of the pressure vessel materials are irradiated in the reactor pressure vessel during operation, to monitor the level of embrittlement. Based on these surveillance results, it appears that embrittlement is higher in many plants than previously thought. The NRC's regulatory documents are being updated to reflect this realization. Further, research is being performed to examine the factors that control neutron radiation embrittlement and to develop additional data useful in updating the regulatory documents. In a related effort, the effect of low-temperature, low-flux irradiation on the integrity of reactor pressure vessel supports is being evaluated.

The PTS rule, 10 CFR 50.61, was amended on May 15, 1991, to make the methodology for evaluating reference temperature consistent with that in Regulatory Guide 1.99, Revision 2. Specifically, the amended rule uses the same formula used in the regulatory guide and permits the use of "credible" surveillance data in evaluating the RT_{PTS} values for each plant. With the amended rule, the RT_{PTS} value for some plants will increase and may decrease for others. In some cases, the new value may exceed the PTS screening criterion prior to the end of the licensed life for the plant. In these cases, the licensees will be considering analyses and physical changes to the plant to satisfy the regulatory requirements.

The surveillance program is a central part of the overall process to ensure pressure vessel safety. An important aspect of the surveillance program is the prediction of the amount of neutron radiation exposure (neutron fluence) of the vessel. Fluence determinations are made by calculations to compute the fluence, by dosimetry measurements at key surveillance locations, and by a consolidation of the measurements and calculations to reduce uncertainties of predictions at critical locations of the vessel. These predictions must be reasonably accurate in order to ensure that the plant is operating in conformance with the NRC safety regulations.

Dosimetry research has led to improved analytical methods for estimating the neutron flux and the neutron energy spectrum and has provided the experimental validation of these analytical methods. During FY 1991, this research has provided new values for the cross section for inelastic scattering of iron atoms that have been included in the Evaluated Nuclear Data Files—data that are essential in determining the neutron fluence. Reevaluation of experiments using these data files has resulted in dramatic improvements in correlation between calculations and measurements for predicting reactor vessel fluences. Finally, the dosimetry research has provided the basis for improvements in dosimetry measurements, including ex-vessel dosimetry measurements. This work will culminate in a regulatory guide on dosimetry that is expected to be published in 1992.

The early embrittlement research efforts emphasized an empirical approach where pertinent materials were irradiated in test reactors under accelerated conditions. While these programs were useful in developing an understanding of some of the controlling variables, it soon became obvious that the number of variables that could have a significant influence on embrittlement was so large and interrelated that an empirical approach could not completely resolve the issue. Thus, emphasis has been increasing on studying the underlying mechanisms of neutron radiation embrittlement. While this work will not be completed for several years, there has been significant progress in the recent past through the use of high resolution devices such as the field ion atom microprobe and small angle neutron scattering. This progress has improved confidence in interpreting the empirical results and in defining additional test reactor irradiation programs. An international group of experts has been formed to cooperate on these problems.

Recent results from test reactor irradiations suggest that the ASME Code procedure to account for irradiation damage may not completely address the damage. It appears that the Code's procedure may underpredict the actual shift in the fracture toughness curves, eroding the anticipated margin of safety in some regulatory analyses. At the same time, other aspects of the overall pressure vessel integrity analyses are known to be extremely conservative so that the final evaluations are still quite conservative. During FY 1991, these results were presented to the ASME Code groups responsible for the Code's procedure and were factored into a "white paper" on pressure vessel integrity issues. Additional test reactor irradiations and material testing efforts are under way and will provide a firm technical basis for evaluating the Code's procedure. That work is expected to be completed in the next 3 to 5 years.

To assist in predicting margins of safety, research has been initiated to study the toughness properties of reactor vessel weld metal from the canceled Midland Unit 1 nuclear plant, which has a relatively low resistance to ductile tearing. During FY 1991, a significant effort was completed to characterize the unirradiated properties of the weld material and to inspect these materials using advanced nondestructive examination techniques to characterize the distribution of flaws in the as-fabricated welds. The results of these tests and examinations provided unique insights into the distribution of chemical content and mechanical properties for actual production welds and provided support for the density of "initial" flaws assumed in earlier analyses of pressure vessel failure probability. Irradiation tests of the Midland weld materials will be conducted in 1992–1993 and should provide an enhanced basis for safety analysis of a number of operating reactor vessels having similar weld metal.

The mechanisms research conducted during FY 1991 made significant progress in identifying mechanisms that seem to control the embrittlement process, partially clearing the way for developing a predictive model that can replace the empirical approach currently used in evaluating irradiation damage. The predictive model is the ultimate goal of this research. While the results of the past research have contributed significantly to achieving this goal, they also have identified many interactions that must be understood before a comprehensive predictive model can be completed. This research has demonstrated that the dominant irradiation embrittlement mechanism for pressure vessel steels is the accelerated formation of extremely small (1–2 nanometers) copper-rich precipitates in the microstructure of the steel. Secondary microstructural changes have also been shown to contribute to the irradiation embrittlement of steels.

The NRC's regulations and regulatory guidance contemplate that there may be some pressure vessels that become so embrittled that continued operation is unacceptable. The embrittlement research program has provided initial data to demonstrate the effectiveness of thermal annealing in recovering degradation in mechanical properties due to irradiation damage. The results of this research have identified optimum annealing temperatures and annealing periods, at least for the relatively limited number of materials studied to date.

The results of the annealing work have been supported by industry efforts and by results from research performed in the USSR and exchanged under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety. The combined results of these efforts provide reasonable assurance that thermal annealing is a practical method for mitigating the effects of irradiation damage. During FY 1991, NRC staff members led a team of researchers and industry representatives to witness the annealing of the reactor vessel at the Novovoronezh Nuclear Power Plant Unit 3 in the USSR (currently known as CIS). Several differences between Russian and U.S. designs and conditions were identified, but it was generally concluded that annealing of U.S. reactor pressure vessels was viable. Additional work is in place to improve our ability to predict annealing recovery and reembrittlement rates. While much more work is needed to provide appropriate regulatory guidance, the principle has been demonstrated.

The embrittlement validation research using decommissioned reactor pressure vessels is a relatively new initiative. The only decommissioned pressure vessel material that has been examined to date is material obtained from the Gundremmingen plant in the Federal Republic of Germany. During FY 1991, several other decommissioned reactors were considered for study, and preliminary plans for pursuing these studies were developed.

Finally, results from an investigation of the embrittlement of the decommissioned Shippingport shield tank have shed some light on the embrittlement of the reactor pressure vessel supports. The results of this research indicate that the Shippingport shield tank was not as embrittled as would be expected based on the trends observed at the High-Flux Isotope Reactor. Since the Shippingport shield tank was exposed to a more typical irradiation, these results raise new questions concerning the presumed severity of pressure vessel support embrittlement. During FY 1991, efforts were initiated to examine the embrittlement trends of typical pressure vessel support steels and to evaluate the neutron flux in the biological shield wall of a typical plant. These efforts will contribute to the final resolution of Generic Safety Issue 15, "Radiation Effects on Reactor Vessel Supports."

1.1.3.2 Piping Integrity

The pipework of reactor primary systems is not subject to degradation by irradiation effects, but rather is more influenced by fatigue, corrosion, and thermal aging. The NRC's regulations and regulatory guidance concerning piping integrity are designed to limit damage due to normal operations—fatigue and corrosion, for example—yet include leak detection and fracture analyses to ensure the integrity of the piping under normal and accident conditions in the event that some form of damage has occurred. These are explicit applications of the defense-in-depth concept and have led to highly reliable piping systems in spite of unanticipated damage such as intergranular stress corrosion cracking in BWRs.

Residual life assessment reviews for LWRs indicate that low-cycle fatigue is a potentially significant degradation mechanism in piping systems. Current procedures for estimating fatigue life are based on the ASME Code Section III and its fatigue design curves. These curves, developed approximately 20 years ago, were obtained by adding a safety factor to a mean data curve that was based on tests of smooth, polished specimens tested in a room-temperature air environment. The safety factor was intended to account for several effects, including the effect of loading rate, the effect of the water coolant environment, the effect of surface roughness, and normal material variability.

Based on results obtained during FY 1991, as well as results obtained from earlier work in the United States and abroad, it is now clear that the margins in the Code are smaller than intended for some situations. Since no consensus fatigue life estimation procedure is available, data from ongoing tests and from the literature and programs in Europe and Japan are being evaluated to develop interim procedures that adequately account for the effects of operating temperature and the water coolant on fatigue life. Also, the NRC is participating in an international effort to develop and execute a longer term research program for developing appropriate fatigue design curves and life prediction methodology for updating the ASME code for future applications.

Cast duplex austenitic-ferritic stainless steels are used extensively in the nuclear industry in pump casings and valve bodies and in primary coolant piping in PWRs. Recent investigations suggest that embrittlement in these steels may occur after 10 to 20 years at reactor operating temperatures. This could adversely affect the structural integrity of pressure boundary components during high strain-rate loading typical of seismic events if those components contained a crack. The embrittlement is of most concern in PWRs where slightly higher temperatures are typical—resulting in greater embrittlement—and cast stainless steel piping is widely used.

Research on this subject has been ongoing since 1982. During FY 1991, procedures and correlations for estimating fracture toughness and tensile properties for these materials have been optimized and validated with experimental data from materials removed from a decommissioned nuclear power plant and from materials exposed to accelerated aging in the laboratory. Conservative estimates of fracture toughness can be made for cast stainless steels of unknown chemical composition; progressively more accurate estimates can be made based on the information that is known about the material. These procedures and correlations, published in June 1991 in NUREG/CR-4513, provide an experimentally validated engineering tool that can be used to estimate the service-induced degradation in properties for cast stainless steels. The research is continuing to examine potential degradation in properties for stainless steel welds.

With the discovery of inservice cracking of nuclear reactor piping came an increased interest in how such "service-degraded" pipe would behave under postulated accident conditions: would it leak or break? The matter of the leak-or-break alternatives had been addressed for years without the emergence of a strong consensus. The NRC and the industry have pursued parallel research efforts in evaluating pipe fracture behavior. The industry's effort has focused on the behavior of stress corrosion cracks, and the NRC has explored the broader questions regarding "leak-before-break" phenomena for all piping.

The NRC has funded research into several aspects of pipe fracture, including analysis of material properties and full-scale pipe fracture experiments. The NRC's primary piping fracture research program had been the Degraded Piping Program, conducted by Battelle in Columbus, Ohio. This program, initiated in 1984, was completed in 1988, and the final report was issued in 1989. The Degraded Piping Program has, among its many contributions to an understanding of piping fracture technology, identified several areas that call for deeper study. Three particularly important areas are the effects of anisotropic material properties, the effects of short cracks, and the effects of seismic or dynamic loading.

During FY 1991 a study of several piping-related issues was continued at Battelle. This experimental and analytical program studies the effect of short cracks (in depth and length) on the fracture behavior of typical nuclear-grade piping materials. Prior experimental and analytical efforts examining the fracture behavior of piping that contains flaws have addressed crack depths and lengths greater than those encountered in service and greater than those of interest in leak-before-break analyses. Therefore, this study will provide experimental data for validating and improving pipe fracture analysis methods. Other efforts in this study examine the fracture behavior of bimetallic welds and the significance of material property variability. The study will continue through FY 1995.

During FY 1991, the NRC completed the first International Piping Integrity Research Group (IPIRG-1) program to evaluate the effects of seismic and dynamic loads and other piping integrity issues. The IPIRG-1 was a consortium of nine government and industrial organizations that jointly funded this research. The work involved performing fracture experiments on a typical piping loop constructed with 16 inch diameter pipe that was 1-inch thick. Intentionally cracked test sections were welded into the loop at a high stress location. The tests were performed at typical PWR pressures and temperatures (2250

, si and 550°F), and the loading was intended to simulate seismic events.

In general, the results support the NRC's pipe fracture analysis approach used in leak-before-break analyses and the fracture analysis approach used by Section XI of the ASME Boiler and Pressure Vessel Code in developing flaw evaluation procedures. However, issues were identified that warrant further study.

The success of the IPIRG-1 program, and the progress made by the IPIRG participants toward an international consensus on pipe fracture technology, led the participants to form a second jointly funded program, the IPIRG-2 program. That work, scheduled for completion in about 3 years, will consider more representative seismic loading histories and will include short cracks and cracks in fittings.

1.1.3.3 Inspection Procedures and Technology

This program includes studies of improved methods for the reliable detection and accurate sizing of flaws during inservice inspection of carbon steel and wrought and cast stainless steel piping and pressure vessels. It also includes studies of online continuous monitoring techniques, using acoustic emission, for crack growth and leak detection, and studies of eddy current inspection techniques for steam generator tubes.

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 flaws. In December 1990, the SAFT system was used to inspect a reactor pressure vessel as part of an international study assessing the effectiveness of advanced ultrasonic technologies. Results from these latest tests to show the accuracy of the SAFT-UT inspection are not available because these "blind test" round robins have not yet been completed and evaluated by the international study. In FY 1991, the SAFT technology was being transferred to General Electric for incorporation into their next generation reactor pressure vessel inspection system. Discussions were also held with other major nuclear industry vendors concerning the transfer of the SAFT technology to them. NRR is considering the use of the SAFT technology to provide an independent review of the inservice inspections being planned for the Yankee Rowe reactor pressure vessel examination.

NRC-funded research at the Pacific Northwest Laboratory (PNL) has produced technology in support of the application of acoustic emission (AE) monitoring to continuously detect the initiation and growth of cracks in nuclear reactor components as it might occur during reactor operation. Similar technology also provides a very sensitive coolant leak detection capability, developed under NRC sponsorship, at Argonne National Laboratory (ANL). Results from this research to monitor the initiation and growth of cracks are documented in the NRC report NUREG/CR-5645. The benefits expected include increased safety through detection and evaluation of crack growth as it occurs, improved capability to detect and locate coolant leaks as they initiate, and reduced personnel exposure to radiation through reduced need for manual inspection of reactor components. The program has produced AE monitoring technology and methodology proven in off-reactor tests, as well as application guidance used in ASTM Standard E 1139 and ASME Code Case N-471. Field validation is currently being performed in cooperation with Philadelphia Electric Company (PECO) by the monitoring of an intergranular stress corrosion crack in a nozzle-to-safe-end weld at the Limerick Unit 1 reactor.

Safety analysis had indicated that the flaw indication in the Limerick weld could remain in place during another fuel cycle without compromising safety. PECO, however, elected to apply AE monitoring on a test basis and a crack-arrest-verification specimen (CAVs) technique to give added assurance that the crack would not grow during operation without detection. AE monitoring at Limerick Unit 1 during the May 1989 to September 1990 fuel cycle has been completed and the results analyzed in FY 1991. A relationship developed earlier in the AE program to relate AE data to crack growth rate was used to interpret the AE data in terms of estimated crack growth. There was partial correlation between the crack growth indicated by AE and that indicated by followup UT performed at the end of the fuel cycle, but the AE also indicated crack growth in locations not indicated by the UT. This is not necessarily inconsistent when examined in light of the nominal detection threshold of about 20 percent of wall thickness for UT. The crack growth indicated by AE was small in most locations, and inspection of the weld by UT is particularly difficult in this case because of the geometry of the weld. Comparing the three surveillance methods on a common basis of **maximum crack growth rate per year**, UT and AE agreed within about 25 percent, while the CAVs prediction was about an order-of-magnitude lower. AE monitoring of the weld at Limerick Unit 1 has been continued for a second fuel cycle to maximize the reliability of the field validation. This effort started in December 1990 and will be completed about May 1992.

The NRC research activities support NRC regional and headquarters staff responsibilities by assisting in training the staff in understanding the new and developing technologies that are being applied to inservice inspection. During FY 1991, the effort focused on selecting various computer-based ultrasonic inspection systems for detailed review and evaluation, conducting a seminar for NRC staff entitled "An Introduction to Computer-Based Inservice Inspection," fabricating test blocks for NRC staff, developing a waveform library from the test blocks, developing guidelines for reviewing ultrasonic field procedures, and constructing a steam generator tube bundle mockup (discussed below). A draft report was prepared that describes the general functions of computer-based ultrasonic equipment and provides a review of selected systems.

The inspection procedures and technology research includes an effort to evaluate the integrity of steam generator tubes. Results from a recently completed NRC research [illegible] the r[illegible] of eddy current (EC) inspe[illegible] and characterize steam [illegible] indicated a need for improve[illegible] process. To address this need, [illegible] research at PNL to develop EC per[illegible] [illegible]ration qualification requirements.

[illegible] work focused on participation in the ASME Section XI Special Working Group on ET Examination (SWGET) for development of generic performance demonstration qualification requirements. Toward the end of FY 1990 a draft appendix was prepared and submitted to the SWGET for consideration during FY 1991. An important addition was the adoption of regression methods for grading EC system performance on probability of detection tests.

The NRC has also been active in the international Program for the Inspection of Steel Components (PISC) task to conduct an international round robin on the effectiveness of steam generator tube inspection techniques in which seven U.S. teams are involved.

Other activities in cooperation with the Electric Power Research Institute (EPRI) during FY 1991 included the development of a mathematical model of steam generator tube inspections to evaluate the effect of probability of plugging, initial sample size, tube population size, and distribution of degradation on sampling plan effectiveness for identifying defective tubes. In addition, Monte Carlo simulations of a sampling plan similar to one proposed by EPRI were performed.

Finally, in support of NRC regional activities, a steam generator tube bundle mockup is being prepared. The design for the mockup was completed during FY 1991. Fabrication was started on mockup structural elements as well as on wastage and fatigue-crack-degraded tube samples. Additional efforts are under way to contract for fabrication of chemically degraded tube samples, including intergranular attack, stress corrosion cracking, and pitting, and for characterization of cracked tube samples by computed tomography.

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.

Recently, the nuclear industry has initiated a significant effort aimed at extending the life of existing plants beyond their original term of 40 years. According to a 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 life extension for current plants. If a 40-year life extension is judged feasible, the benefit is even larger. The benefit reflects both the lower fuel cost of the nuclear plants and reduced outlays for replacement of generating capacity. Utilities are currently planning to apply for license renewals and have a tentative schedule for several steps in the process. The first submittal to the NRC is expected in 1992, with a large number of additional submittals to follow shortly thereafter. To keep pace with these industry plans, the NRC will need to devote effort over the next several years to license renewal. 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. Review of these applications at an early stage will provide an indication to the industry of the viability of the life extension option in sufficient time to elect an alternative option if necessary.

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 storage, 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 1991

1.2.3.1 Aging Rese[illegible]

Based on the re[illegible] experience, including the available data [illegible] interactions with codes and standards committees, Phase 1 aging assessments were completed on the following special topics and safety-related components and systems:

1. Instrument and Control Systems,

2. Reactor Core Internals,

3. Control Rod Drive System (PWR),

4. Reviews of Industry Reports, and

5. Degradation Modeling of Components.

Reports were issued on the above-mentioned Phase 1 aging assessments to identify degradation sites within the component and system boundary, aging mechanisms, and aging concerns. The reports, which also made recommendations for maintenance and aging mitigation, were reviewed by industry as well as professional society groups.

Phase 2 aging assessments of components involve some combination of (1) tests of naturally aged equipment or equipment with simulated aging degradation; (2) laboratory or in-plant verification of methods for inspection, monitoring, and surveillance; (3) development of recommendations for inspection or monitoring techniques; (4) verification of methods for evaluating residual service lifetime; (5) identification of effective maintenance practices; (6) in situ examination and data gathering for operating equipment; and (7) verification of failure causes, using results from in situ and post-service examinations. During 1991, Phase 2 aging assessments were completed on the following components and systems:

1. Solenoid-Operated Valves,

2. Auxiliary Feedwater System,

3. Auxiliary Feedwater Pumps,

4. Diesel Generators,

5. Circuit Breakers and Relays, and

6. Reactor Protection System.

1.2.3.2 Residual Life Assessment of Major LWR Components

Intrinsic to the general exploration of reactor aging is the residual life assessment (RLA) of major components and structures. The capability to predict the residual operational lives of major LWR components and structures can be of great benefit to resolving technical issues associated with plant aging and license renewal. The objective of the RLA, as an element of the Nuclear Plant Aging Research (NPAR) program, is to develop technical bases and criteria to assess methods of mitigating the effects of aging on major components and structures when considering possible license renewal. The approach is to gauge the degradation of the major LWR components and structures by the synergistic influences of radiation embrittlement, thermal fatigue, corrosion, environmental attack, metallurgical changes, microbiologically and otherwise induced corrosion, moisture, intrusion, erosion, and so forth.

As of the end of 1991, the RLAs of 21 components and structures important to plant safety have been completed. The components are reactor pressure vessel supports, reactor coolant pumps, PWR pressure vessels, PWR containment structures, PWR coolant piping, PWR steam generators, PWR pressurizers, PWR pressure surge and spray lines, PWR reactor cooling system charging and safety injection nozzles, PWR feedwater lines, PWR control rod drive mechanisms and reactor internals, BWR containments, BWR feedwater and main steam lines, BWR control rod drive mechanisms and reactor internals, electrical cables, and emergency diesel generators. In these assessments, the degradation sites, degradation mechanisms, stressors, and failure modes have been identified for each component and structure under study. The assessments also include a review of the current methods for inspection and surveillance of these components and structures. The results of this effort have been documented in NUREG/CR-4731, Volumes 1 and 2, and in NUREG/CR-5314.

The work completed in FY 1991 focused on developing models and procedures for estimating aging damage in specific LWR components for continued safe operation. The work included the evaluation of advanced inspection, surveillance, and monitoring methods for characterizing the aging damage. The results will be useful for NRC licensing to establish policies and guidelines for making license renewal decisions. The components that were assessed or that are currently being assessed are LWR reinforced concrete containments, PWR pressure vessels, LWR metal containments, PWR steam generator tubes, and cast stainless steel components. Results with respect to PWR pressure vessels, LWR metal containment, PWR steam generator tubes, and cast stainless steel components have been documented in NUREG/CR-5314, Volume 1 (draft), Volume 5 (draft), and Volumes 3 and 4, respectively.

1.2.3.3 Technical Bases for License Renewal

A license renewal rule, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," was issued in December 1991. Besides a final rule, more detailed regulatory guidance addressing the technical safety issues related to aging is needed, both to implement the rule and to advise licensees on license renewal application requirements.

A draft regulatory guide, developed on the standard format and content of technical information for applications to renew nuclear power plant operating licenses, was issued for public comment. The regulatory guide is being revised to reflect the changes in the final rule and to accommodate the public comments. The purpose of the regulatory guide is to establish a uniform format and content acceptable to the NRC staff for structuring and presenting the technical information to be compiled by an applicant for a renewed nuclear power plant operating license and submitted by the applicant as part of an application for a renewed license. The regulatory guide identifies the content of, and provides technical criteria for, the compiled technical information.

1.2.3.4 PRA-Based Prioritization of Risk Contributions and Maintenance

A second report (revision to NUREG/CR-5587) was issued on setting priorities among aged active components according to their risk contributions and maintenance importance. The second report was necessary to resolve numerous comments and suggestions that had been made as the result of an extensive internal NRC review of the first draft report. The format and content of the original report have been changed to include the technical bases for identifying the risk-significant components using the prioritization procedures. For completeness, the second report also describes various approaches for transforming a baseline probabilistic risk assessment (PRA) into an age-dependent PRA, and it provides answers to the questions that are likely to arise when applying an age-dependent PRA. In addition, the report incorporates the work developed under a different task for including the effects of aging on passive components in the baseline PRA.

A report (draft NUREG/CR-5730) was issued on the development of a methodology to include the effects of aging on passive components (pipes, structures, and supports) and the resulting impact on plant risk. The methodology is based on probabilistic structural analysis for calculating the failure probability of these components when subjected to stresses caused by the loading on the components. The failure calculations can be substituted in a PRA for the plant that will calculate the effects of this failure on plant risk. The method is demonstrated in the report for the application to a pipe weld in which a crack occurs. As the crack grows because of the pipe loadings (and stress corrosion), the crack will eventually reach a depth that is unsatisfactory for ensuring continued safety. At this point the pipe can be considered failed, and the effect on plant risk is determined. A computer program has been developed for aiding in these calculations. A procedure was also developed for identifying those aged passive components having the most impact on plant risk. Effective inspection and maintenance actions can be taken on these components that will control the effects of the aging and reduce the risk.

Demonstrations of this work show that aging effects, if unmitigated, cause increases in plant risk; cause component priorities to change as compared to the baseline PRA; and result in significant impacts on plant risk and component priorities from the interactions of multiple aged components. Methods of setting priorities according to risk impacts and maintenance importance of active components were demonstrated, and listings of priorities for components and systems were developed. Approaches to effective maintenance practices, when focused on the components having risk and maintenance importance, were demonstrated to control the impacts of aging on plant risk.

1.2.3.5 Regulatory Instrument Review: Management of Aging of LWR Major Safety-Related Components

Eight selected regulatory instruments, e.g., NRC regulatory guides and the Code of Federal Regulations, were reviewed for safety-related information on three additional major LWR components: cables, containment, and basemat. The focus of the review was on 25 NPAR-defined, safety-related aging issues, including examination, inspection, and maintenance and repair; excessive/harsh testing; and irradiation and thermal embrittlement. It was concluded that safety-related regulatory instruments do provide implicit guidance for aging management but that there is room for improvement with regard to explicit guidance.

1.2.3.6 Inspection Integration

The NPAR program has the potential to support the ongoing inspection effort conducted by the regions in accordance with the NRC inspection program. One objective of the inspection effort is to ensure that safety systems and safety-related components have not degraded as a result of any cause, including aging.

A review of NRC inspection procedures suggests that the information requirements of the inspectors are vast and that the NPAR aging data base can assist the inspectors in focusing their activities on those components and systems most likely to affect the plant safety as the plant ages. Further, the NPAR-developed data and research results can provide the inspector with criteria for judging the validity of findings and the completeness of the licensee's responses.

In light of the information needs identified by the inspectors, NPAR reports for selected components and systems were reviewed, and information was extracted that could be of potential use to inspection activities was excerpted and published in two documents—an "aging report summary" and an "aging inspection guide." The summary for each equipment type and system studied in the NPAR program includes the identification of aging-related problems, highlights of the operating experience, solutions to aging problems, and references. This information will be available as NUREG-1323, "Insights Gained from Aging Research." This report contains visual inspection techniques for detecting aging degradation, external and internal indicators, and important operating parameters.

1.2.3.7 Degradation Modeling of Components

Efforts in relation to the development of component degradation modeling approaches to study aging and maintenance effects on components are continuing. Application degradation modeling approaches to residual heat removal (RHR) pumps, service water pumps, and instrument air compressors have demonstrated that the inclusion of degradation states in reliability modeling provides a better understanding of aging and maintenance effects. The model provides a quantitative means of characterizing aging effects, evaluating maintenance effectiveness, and assessing component reliability. The model being developed is applicable to both standby and continuously operating components.

Since age-related failures pass through a degradation state first, the degradation rate serves as a precursor of the failure rate. Increasing aging trends in the degradation rate can signal future increasing aging trends in the failure rate. In the case of compressors, the failure rate, which is significantly lower than the degradation rate in the first three years, increases faster in the later years, reaching approximately the same values as the degradation rate at the end of 10 years of operation. This behavior indicates the ineffectiveness of maintenance in preventing degradation from transforming into failures as the air compressors age. Another finding relating to RHR pumps indicates a time-lag of 2 years for degradation to affect failure occurrences. The model is being extended to explicitly show the reliability effects of different maintenance and test intervals, different maintenance and test efficiencies, and different repair times. Further developments will include time-dependent Markov approaches, multiple degradation studies to model progression of degradation, subcomponent level modeling, and the potential for using the degradation modeling approaches in PRA models of the plant to further define the core damage frequency as a function of aging.

1.2.3.8 Components, Syst ıs, and Facilities

Component Cooling Water Systems. The component cooling water (CCW) system has been identified as one of the support systems that is important to plant safety. An increase in CCW unavailability can adversely impact plant risk, as discussed in studies such as NUREG-1150 and the TIRGALEX study. NRC Generic Safety Issue 65 relates to the high probability of core melt due to CCW system failures. Because of its importance, an aging assessment of this system was completed this fiscal year. The objectives were to identify and characterize the aging degradation mechanisms relevant to the system, to assess their impact on system unavailability, and to provide recommendations on the available methods for the detection and mitigation of aging in the CCW system. Aging degradation contributes to over 70 percent of the failures, with the most common aging mechanism being "wear." Fifty percent of the failures resulted in degraded performance of the system, while 27 percent caused a loss of redundancy. Component failure rates can increase with time due to the effects of aging, which can lead to an increase in system unavailability as the plants age. To prevent this, good detection and mitigation practices are required. Basic in ection, surveillance, monitoring, and maintenance (ISM&M) practices are good, but they are not comprehensive enough to completely manage the effects of aging. These basic methods include ways to detect incipient aging degradation before failures occur, as well as maintenance practices to mitigate the effects of aging. However, they typically do not address all aging mechanisms. Supplemental activities are available that can improve the system reliability. These were identified based on manufacturers' recommendations or past plant experience. Some of these supplemental activities include thermography examination of pumps to detect hot spots and eddy current testing of heat exchanger tubes to detect cracks or flaws. Each of the supplemental activities was correlated with the aging mechanism it helps to detect or mitigate. The study recommends that various supplemental practices be added to the basic practices to formulate an effective ISM&M program for detecting and mitigating aging.

Control Rod Drive Systems for PWR Plants. The PWR control rod drive (CRD) system positions the control rods within the ontrol reactivi changes encountered during op and to provide a sufficient source of negative reactivity to ensure a rapid reactor shutdown. The aging study of this system examines the design, construction, operation, and maintenance of the system to assess its potential for degradation as the plant ages. The extent to which aging can affect the safety objectives of the system is also included in this study. One such consideration includes component failures in the CRD system resulting in plant transients, which unnecessarily challenge other safety systems. The Westinghouse and Combustion Engineering (CE) plants use similar magnetic jack type mechanisms, with the exception of Palisades and Fort Calhoun, which use a rack and pinion drive mechanism. Babcock and Wilcox (B&W) plants use a roller nut type mechanism. The magnetic jack and roller nut mechanisms are actuated by externally mounted stator coils while an electric motor is used to drive the rack and pinion mechanism. All the mechanisms use similar magnetically actuated feed switches to provide actual rod position indication. Forced air cooling systems are used by Westinghouse and CE, while B&W uses a water-cooled system. Inspections of design differences, consideration of certain design modifications and improved maintenance techniques are being provided as they become evident from their operating experience. Some of the most significant problems identified for the Westinghouse CRD include unexpected wear of control rod cladding surfaces, the susceptibility of certain cast drive mechanism pressure housings to leakage due to embrittlement, the vulnerability of electronic components to elevated ambient temperatures, corrosion and wear of operating coil stack connectors, and a potential generic concern related to inaccurate control rod position information. With regard to CE and B&W designs, corrosion due to primary coolant leakage (seal degradation, housing cracks, and vent valve leaks) and failures of power and control system components (power supplies) were the most prevalent aging degradation mechanisms. Because of the inherent design features in the system and careful maintenance activities performed by the utilities, this system has not exhibited any system failure as a result of component degradation.

A wide variation exists between the preventive and predictive maintenance programs of the utilities; this has had an effect on their ability to identify and mitigate aging. This study has recommended an increased emphasis on inspections and root cause analysis. Some of these activities include the use of advanced monitoring techniques such as infrared thermography for electronic components, motor current signature analysis to detect proper control rod operation, and Electronic Characterization and Diagnostics as a possible alternative to meggering for assessing electrical integrity.

Control Rod Drive Systems for BWR Plants. A study was made by ORNL under the aging research program to collect and evaluate data on the past performance failure mechanisms and aging of BWR CRD systems. A workshop was organized and attended by BWR utility personnel to review and collect information on CRD maintenance. A substantial number of system problems were found to result from the failure of hydraulic control unit components, including the accumulator and various scram valves. The leading causes of CRD mechanism degradation were found to be embrittlement and fatigue fracture of the Graphiton seals. Recommendations for improved maintenance practices were provided to minimize some of the failures observed. The results are documented in NUREG/CR-5699.

Heat Exchangers. Heat exchangers are vital components of nuclear power plants, serving as interfaces between both safety-related and non-safety-related systems and components and are the ultimate heat sink to provide for safe operation and to mitigate the effects of accidents. A review of nuclear plant operating experience by ORNL indicated that interfluid leakage caused by corrosion or erosion of tubing is the most commonly identified problem, accounting for approximately 40 percent of the total reported failures. External leaks, usually from tube erosion or corrosion in space air coolers or from gasket failures, accounted for about 35 percent of the total. In most cases, interfluid or external leakage is more of a nuisance condition than a threat to the ability to bring the plant to a safe shutdown condition. Of more serious consequence is the degradation of the ability of a safety-related heat exchanger to provide design-basis cooling. In this category, tube blockage, most often by bivalves or their shells, accounted for approximately 22 percent of the total. These types of problems may not be recognized because the exchangers normally operate at thermal loads that are only a fraction of design loads, and requirements for inservice testing that would indicate degradation have been minimal.

The NRC's Generic Letter 89-13 requires development of plant-specific inservice testing programs by licensees. In addition, the Operation and Maintenance (O&M) Committee of the American Society of Mechanical Engineers (ASME) standard for inservice testing of heat exchangers, now under development and to which the study provides data, should provide definitive guidance in detecting degraded capability.

Friction in Motor-Operated Valves. A report (draft NUREG/CR-5735) on the effects of aging on internal valve components was issued. The intent of this work is to determine whether moving internal valve components can be affected by corrosion buildup due to fluid conditions during normal plant operations. In the valve experiments reported under Section 1.3, "Reactor Equipment Qualification," it was demonstrated that these internal friction forces were underestimated. For the valves to be able to operate as they should, it is necessary that the effects of friction be understood. Therefore, the information contained in the above draft report identifies the main aging mechanisms (corrosion, deposition, and erosion) that can influence the friction values. The followup results of nuclear plant reactor trips where valves were disassembled for inspection are also reported and show that stainless steel valves are likely to be less affected than either carbon or low alloy valves. Although only a small number of valves were observed, this important work is being supplemented with friction experiments to account for other metal-fluid environment interactions.

Cables. The NRC is currently sponsoring research at the Sandia National Laboratories (SNL) to investigate cable condition monitoring methods and cable aging degradation over a 60-year period of plant performance. Accelerated aging and accident survival tests of cable products have been completed at the SNL Low Intensity Cobalt Array facility during which cables were aged to the equivalent of 20, 40, and 60 years of operation. During the aging process, the condition of the cables was monitored using both electrical and mechanical measurements, including insulation resistance, polarization index at three different voltages, capacitance and dissipation factors over a range of frequencies, elongation profiling, cable indenter modulus measurements, and hardness and density measurements. The most effective of the condition monitoring methods was elongation at break. Hardness, indenter modulus, and density also correlated with aging for some cable insulation and jacket materials. Neither tensile strength nor any of the electrical measurements exhibited a consistent trend with aging. Most of the cables were found to be functional throughout the 60-year aging and the loss-of-coolant-accident (LOCA) tests that followed aging.

Snubbers. The research results provide information relevant to recent operating experience for both hydraulic and mechanical snubbers, particularly in regard to aging-related influences. Methods were identified that are useful in monitoring the service life of snubbers. Recommendations are being developed for the Subsection ISTD of the ASME-O&M Code. The principal findings of this research are:

- The primary environments that contribute to aging degradation in snubbers are temperature, vibration, moisture, and dynamic transients.
- Based on the eight nuclear power plants investigated, approximately 47 percent were mechanical functional test failures and 52 percent of test failures were service related.
- Hydraulic snubber seal life is primarily a function of operating temperature. Seal life limits originally proposed by snubber manufacturers are generally conservative.

Service Water Systems. The objective of the service water system aging study was to identify and characterize the principal aging degradation mechanisms relevant to the service water system to assess their impact on operational readiness and to provide methods for the mitigation of aging in the service water system. The following regulatory applications evolved from the aging assessment of the service water system:

- Technical basis for the implementation of Generic Letter 89-13,
- Support for NRR in the modification of the Standard Technical Specifications addressing service water systems, and
- The development of a draft Research Information Letter (RIL) on service water systems.

To satisfy the need for a formal procedure to identify the cause of age-related degradation of service water systems, a root-cause method of analysis was developed. A positive outgrowth of the service water system aging assessment was the transfer of the methodology related to root-cause analysis and artificial intelligence to a Department of Defense facility.

Low Flow Operation of Safety-Related Pumps. Bulletin 88-04 was issued by the NRC requiring utilities to examine their safety-related pump operation to determine the potential for dead-heading pumps in parallel operation and the adequacy of the minimum flow rate. ORNL evaluated the industry responses under the aging research program and made several site visits to review the utilities' detailed calculations. It was found that low flow operation can degrade pumps and there are no generic guidelines for determining acceptable pump operation in all modes. The minimum low flow was found to be inadequately addressed at some nuclear plants. It was recommended that pump qualification criteria and new diagnostic techniques providing more meaningful information on pump degradation be developed. Parallel pump dead-heading problems were identified in the residual heat removal systems at some plants where the pump discharge miniflow line originates downstream of the pump discharge check valve. Results and recommendations are provided in NUREG/CR-5706.

Fire Safety. The NRC is currently sponsoring a research program at Sandia on the "Vulnerability of Aged Electrical Components to Fire" and is also participating in a large-scale cable fire test program sponsored by Germany at the HDR reactor facility located in Kahl, Germany. Two separate studies on the impact of aging on the performance of electric cables in a fire were completed at Sandia during 1991. The effect of cable thermal aging on material flammability and on cable vulnerability to fire-induced electric failure was studied. It was found that cable material flammability was significantly reduced as a result of aging. Cable thermal vulnerability to fire-induced electric failure was only slightly increased by thermal aging. Thus, cable aging does not appear to significantly increase fire risk.

The NRC is participating in the performance of fire tests in the decommissioned German HDR reactor facility. Recommendations were provided on the test arrangements for the cable fire test to be run in December 1991 involving a large-scale cable tray fire in a lower elevation room in the containment building. The NRC is providing electric cables and electric relays for installation in the fire room to investigate the effectiveness of cable spatial separation in preventing fire damage and the thermal vulnerability of electrical components to heat and smoke from a fire. Efforts under this program also include participation in an international fire computer code validation comparison using the HDR fire test data. The fire computer code models to be evaluated include those frequently used in the fire risk assessment for U.S. nuclear power plants such as COMPBRN.

Structural Components. During fiscal year 1991, three reports were generated by the structural aging research program. The first, an annual report, updated the progress and status of the overall program, as well as giving projections and details of work still to be done. The second report was a single-volume sample of a future four-volume set of data on how structural materials change as they age. These data will eventually also be accessible electronically via personal computers. The third report used three different, but typical, U.S. nuclear power plant types to develop an aging assessment methodology. This methodology uses relative weighting factors to rank concrete structures in nuclear power plants by the importance of their structural elements, safety significance, environmental exposure, and the influence of degradation factors.

Mechanisms for strength degradation due to corrosion of steel reinforcement and detensioning of prestressing tendons were incorporated into the reliability analyses being developed for reinforced concrete structures. A considerable amount of concrete aging data was acquired for input to the structural materials aging data base.

1.3 Reactor Equipment Qualification

1.3.1 Statement of Problem

As a result of the Three Mile Island (TMI) accident, concerns and questions were raised regarding the operability and structural integrity of components during earthquake and LOCA environments. Although design criteria and loading definitions have changed over the years to improve the integrity of these components, the concerns and questions dealt directly with the adequacy of the component qualifications. Therefore, those items that were identified as high priority were given immediate research attention and action. It was also intended that the results of the research would be incorporated into standards.

Subsequent to the TMI research activity, other safety issues were identified and, where these affected equipment qualifications, research effort was proposed to develop the data base to aid in the resolution of these high-priority safety problems. Current effort is addressing one of these generic safety issues. Another effort is providing guidelines for improving valve qualification standards.

1.3.2 Program Strategy

One main NRC staff effort in the equipment qualification program is involved with the completion of the development of a technical data base and analytical procedures for addressing the high-priority generic safety issue (GSI-87) related to the operability of motor-operated valves (MOVs). This same effort is also providing important information and guidance for evaluating some of the utility responses to Generic Letter 89-10, which is related to GSI II.E.6.1. This work is resulting in understanding the capabilities of diagnostic equipment to predict whether thrust measurements can be extrapolated from typical in situ test conditions to accident level conditions. Past and current results will continue to be incorporated in the ASME valve qualification standard and in the operations and maintenance (O&M) standards. These upgraded standards will clarify some of the areas that the staff believes may be contributing to MOV problems.

Future efforts in the equipment qualification program will be devoted to the continuance of the evaluation of data to understand the behavior of other typical valves in high-energy environments and in other piping systems when subjected to operational flows. Other efforts will address the integration of the test data for regulatory applications and for resolving new problems and safety issues consistent with safety and licensing needs. Since industry is becoming more involved in solving some of the pressing valve safety problems, some NRC effort will be devoted to following this work and evaluating the results with regard to licensing applications. Another research effort, which has been delayed because of other high-priority GSI work, will be devoted to understanding the effects of large earthquake loads on the operability of an aged gate valve. The effects of the large dynamic loads on piping, supports, snubbers, and anchors will also be studied. The results from current and future efforts will be incorporated in appropriate ASME qualification and O&M standards to provide the basis for ensuring safer components.

1.3.3 Research Accomplishments in FY 1991

Experiments were completed in early FY 1990 to determine whether valves in high-energy pipes will close as they should to prevent leakage during a pipe-break accident outside the containment. The resulting high-velocity flows that develop in the pipe and in the valves must be stopped by the valves. The leakage, if unchecked and if the valves do not close, can have serious consequences, not only because of steam release outside containment, but also because other emergency equipment may be exposed to the harsh water and steam environment and may fail.

A total of six different valves were tested, three each having six-inch and 10-inch diameters. The six-inch-diameter valves are typical of those installed in high-energy hot water pipes, while the 10-inch-diameter valves are typical of those installed in high-energy steam pipes. All hardware and fluid environments—flow velocity, pressure, temperature—were selected to simulate actual conditions that would occur in the event of a postulated pipe break accident at some operating nuclear power plant.

The main findings from the valve experiments were that one 6-inch valve did not fully close because it had undergone significant damage to internal parts during closure and the five other valves were capable of stopping the high-velocity flows in all of the closing experiments. However, one 10-inch valve also experienced significant damage to its internal parts but the damage did not prevent the valve from closing. (It should be noted that all valve actuators that provide power to close the valves had been set to deliver larger thrusts than would normally be the case for in-plant operation to ensure closure during the experiments.) Another important finding from the experiments showed that the valve internal friction forces that must be overcome by the actuators are underestimated. This latter finding leads to underpredicting the required closing thrusts and ultimately may lead to undersizing actuators for these valve applications. All the above findings and all the test results have been made available to the nuclear industry for their use in improving the overall reliability of valves.

The results reported above support concerns at the NRC about the capabilities of similar in-plant valves to accomplish their intended safety functions when necessary. The NRC had previously notified the nuclear power plant licensees that a valve evaluation program should be instituted at each plant to ensure valve operability over the remaining life of each plant or valve as the case may be. Most plants have already started to comply with the NRC request, and NRC inspectors have been auditing plants over the past several months to evaluate the respective valve programs. The inspectors have required training, technical information, and criteria for performing their evaluations. Thus, during FY 1991, most of the research effort has been devoted to analyzing the test data obtained from the valve experiments (identified earlier) to develop the needed information and criteria and to provide training for the NRC inspectors. Some of the areas where the research effort during FY 1991 has made important inroads in advancing valve technology are:

- A modified valve thrust formula for bounding closing thrust requirements for gate valves was developed. The formula reflects the effects of friction, temperature and pressure, and fluid conditions from the experiments. The formula includes terms that had not been previously identified in the industry standard formula.
- Progress has been made to quantify the effects of corrosion, wear-in, fluid lubrication, and the importance of design parameters on valve operability. Clearances between valve internal parts are important for identifying whether a gate valve will experience damage during high-flow operation.
- A computer program has been developed for the use of NRC inspectors in evaluating valve calculations at nuclear power plants. The program provides a consistent set of criteria and methods the inspectors can use when performing these difficult evaluations. The program has also been made available to parties outside the NRC for their use in predicting valve performance.

The nuclear industry is also contributing to improving valve reliability. An extensive program, including experiments on other valves, has been developed by the Electric Power Research Institute (EPRI). The program started in April 1991 and will continue for approximately 3 years. Some foreign countries currently also have valve programs under way. Although the NRC valve research effort will continue, the level of effort will decrease in FY 1992 and FY 1993. During these years and in subsequent years, the NRC efforts will focus on reviewing, evaluating, and confirming the EPRI program results and, where possible, the results from foreign country programs. These efforts will result in supplying the NRC inspectors with additional technical information for evaluating nuclear power plant valve programs.

1.4 Seismic Safety

1.4.1 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 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 earthquakes in the United States have occurred at New Madrid, Missouri, and at Charleston, South Carolina. The geology of the central and eastern regions makes it difficult to establish 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 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 RES earth science program to significantly reduce the uncertainty in seismic hazard estimation in the next decade through emphasizing this type of research.

In the 1970's and before, our interest in nuclear plant seismic design was mainly limited to response at design levels (e.g., OBE and SSE) and our knowledge of this was primarily based on analytical techniques and assumptions. In the 1980's, 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 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.

1.4.2 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 scheduled to be fully in place by the end of FY 1992. In the meantime, the currently NRC-funded networks in the Eastern and Central United States will be gradually phased out.

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.

1.4.3 Research Accomplishments in FY 1991

1.4.3.1 Earth Sciences

Seismic hazards contribute a sizable proportion of overall plant hazards and, because of inherent difficulties in defining them, they form an even more significant portion of the uncertainty in estimating plant hazards. Although recent NRC (NUREG/CR-5250) and EPRI (NP-6395) studies have advanced the methodology for characterizing seismic hazards at nuclear reactor sites, further seismic hazard research will be needed. The goal of the RES earth science program is to reduce uncertainties in hazard estimates by continued research into the causes and distribution of seismicity. Successes of past research programs together with applications of newly developed methods promise to significantly reduce uncertainties in seismic hazard estimates within the next decade.

The National Seismographic Network (NSN), established through a cooperative agreement with the USGS, has progressed to the operational stage and was officially dedicated on April 3, 1991. The NRC is providing the funds for the stations of this network in the Eastern and Central United States and for the satallite receiving station and associated equipment for data processing and storage at Golden, Colorado. In return, the USGS will operate the network and provide seismographic data to the NRC. At the dedication ceremony, this agreement and the way it was put into effect were highlighted as an example of unusually effective cooperation between two government agencies.

At present, a few stations are operational, together with the satellite link and processing facilities in Golden. However, with the details of the instrumentation having been worked out, installation of new stations will progress rapidly, and a substantial portion of the network is expected to be in place by the end of FY 1992. Completion of the full (eastern and western portions) network is expected to occur in FY 1994.

With its dual range, 3-component seismometers, this network will carry out the functions of both a microseismic and a strong motion recording network. The NSN is designed for fully error-corrected digital data transmission, making data available for rigorous analysis within minutes of the occurrence of an earthquake. The network has the flexibility to incorporate additional stations and regional networks operated by universities and other government agencies. Most of the network components are based on commercially available products, thus minimizing costs and maintenance problems.

NRC support for regional seismographic networks in the Eastern and Central United States was continued during this fiscal year, again at a funding level that was somewhat reduced from the previous year. This support function will be continued until the end of FY 1992. At that time sufficient data will be available from the NSN, and all regional network support from the NRC will end.

Additional investigations were conducted by the Geological Survey of Canada in the area of ground rupture during the December 25, 1989 magnitude 6 Ungava, Quebec, earthquake, and strong motion recorders deployed last fall were retrieved during July 8 through 22, 1991. The investigations revealed that the main rupture extended 2 kilometers farther north than reported last year and many other secondary ruptures were identified. Sub-bottom acoustic profiling was conducted to detect seismically induced deformation in lake sediment in several lakes in the area. The results revealed isolated pockets of disturbed soil. The aftershock activity was not large enough to trigger the strong motion recorders installed last year.

Paleoseismic studies in the epicentral areas of moderate historic earthquakes in the northeastern United States and adjacent Canada have found paleoliquefaction features induced by prehistoric earthquakes and have provided data on seismically induced soil deformation structures.

Last year investigations at Ferland, Quebec, the site of the November 25, 1988 Saguenay earthquake and in the vicinity of the 1727 Cape Ann earthquake at Newburg, Massachusetts, identified liquefaction features induced by these earthquakes and also paleoliquefaction features caused by prehistoric earthquakes. In FY 1991, following up on these studies, new investigations were started at Newbury and Moodus, Connecticut, and Ossippi, New Hampshire—sites of historic and ongoing seismicity. These studies are focusing on identifying geological evidence for prehistoric earthquakes, including seismically induced liquefaction features as occur in fluvial soils, glacial outwash deposits and lacustrine sediments, and landslides, rockfalls, and slumps. This research will also identify similar features that were caused by phenomena other than tectonic and will compare their characteristics with those that were tectonically induced.

A seismic reflection survey that was completed in 1987 included a traverse along the Roanoke River from Bedfor to Brookneal in a generally nonseismic zone and several lines in the central Virginia area near Richmond. Final interpretation of this survey, together with reprocessing and reinterpretation of the I-64 seismic traverse acquired by the USGS and new geological data, has led to new conclusions concerning the structure and seismicity of this area. Most significant is the interpretation that the Piedmont and Blue Ridge region of the central and southern Appalachians contains only one terrain boundary, namely the Taconic suture. The Taconic suture is repeated at the surface by faulting and folding, and it passes through the lower crust and lithosphere somewhere east of Richmond. The rupture is spatially associated with seismicity in the central Virginia seismic zone, but it is not comfortable with earthquake focal planes and seems to have little causal relation to them.

In central Virginia, the metamorphic Piedmont and Blue Ridge plate is 9 kilometers thick, whereas its thickness in the aseismic area of the Roanoke River traverse is only 3 kilometers. However, the plate may be more extensively broken by high angle normal faults in the central Virginia seismic zone. Thus, greater infiltration by ground water may reduce the strength of the fault planes present and lead to a higher rate of seismicity.

The 1886 Charleston, South Carolina, earthquake is the largest seismic event to have occurred on the Atlantic seaboard of the United States. This earthquake caused extensive liquefaction in its meizoseismal region. Research in this area also revealed geologic evidence that there have been at least five prehistoric earthquakes in the Charleston earthquake area. Deposits susceptible to liquefaction are present near the coast from Florida to New Jersey. More than 1,000 sites were investigated within this region and no seismically induced liquefaction was identified outside of South Carolina except for one feature in North Carolina just north of the State line. An event occurred 1800 $\pm$ 200 years ago for which evidence was found north of the Charleston meizoseismal area but not near Charleston. Either this represents another seismic source or the evidence at Charleston has not been found as yet. The evidence for large prehistoric earthquakes in coastal South Carolina and the lack of it elsewhere on the Atlantic coast is consistent with a unique source in the Charleston area.

During FY 1991, field and laboratory work was conducted for a study extending the search for paleoliquefaction features from the coastal areas of South Carolina inland and into the southern Appalachians. Work concentrated on fluvial and lacustrine deposits along the Savannah and Edisto Rivers, on Carolina bays in the coastal plain, and on the Bowman and Union County areas. Some preliminary work in the seismic zones of the southern Appalachian area of Tennessee and Giles County, Virginia, was also carried out.

To date, no clear neotectonic features that are not related to the Charleston source area have been found. Preliminary results suggest that the Bowman area is not connected with the Charleston source area and has not been the source of earthquakes with a magnitude greater than 6 for the past several thousand years. Progress was made in establishing valid criteria for recognizing neotectonic features in these noncoastal areas.

Union County, located in the South Carolina Piedmont, was the site of an intensity VII earthquake in 1913. Liquefiable fluvial sands have been identified, but no liquefaction features have been found so far. The investigation of this area will continue in winter when lower water levels will expose additional outcrops. The Appalachian area has very few deposits that are liquefiable. Those that are available will be investigated, but additional neotectonic methods are needed to obtain information on possible paleoseismic events. Possible seismically induced landslides, ground fissures, and cave deposits have been identified as possible sources of information. While a single line of evidence in this area may not be conclusive, it is expected that multiple lines of evidence may permit firmer conclusions.

The New Madrid (Missouri) area experienced an earthquake sequence in 1811–1812 that included the most severe shocks ever generated in historic times east of the Rocky Mountains. Today the area is still the source of considerable earthquake activity. The source of the seismicity has been identified as reactivated faults within a rift in the crystalline basement. The presence and extent of this ancient rift structure has been defined by geological and particularly by geophysical means. Plots of epicenters in this area also clearly reveal the trends of the underlying rift structure.

Evidence obtained from an array of 3-component seismometers (the PANDA array of Memphis State University) deployed in the area has led to new insights into the structure of the New Madrid area and has proved the value of 3-component digital recording methods. The new technology has made it possible to compute single-event focal mechanisms compared to the composite mechanisms used in the past. As a result it was determined that, in the central segment of the seismic zone, earthquakes define a narrow, 30-kilometer-long fault zone that dips at 33°–52° to the southwest. It was also found that a thin low-velocity layer exists at upper crustal depths in the area. This is interpreted to be a clastic sediment layer.

Tree-ring analyses were performed on drilled cores from bald cypress trees in Reelfoot Lake, Tennessee. Reelfoot Lake is reported to have been formed during the earthquakes of 1811–1812, and growth responses in the bald cypress confirm the coseismic subsidence of the Reelfoot Lake basin. While most of the hardwood trees in the area of the lake were killed, the bald cypress tolerates inundation well and many specimens up to 800 years in age remain. The bald cypress shows a large growth surge after 1811 because of the altered hydrologic regime. In comparison to other areas, Reelfoot Lake is the only area in the midcontinent that shows a growth surge in bald cypress at this time. A second growth anomaly in the Reelfoot Lake area is a sharp decline in the density of latewood that has lasted from 1812 until now. This also is ascribed to altered hydrologic conditions. A third characteristic of the Reelfoot cores is the presence of numerous cracks in pre-1812 portions that may indicate physical damage sustained by the trees during the earthquakes.

During FY 1991, the Bootheel lineament, a 113-kilometer-long lineament identified by satellite photographic analysis in the New Madrid seismic area, was investigated by geologic mapping, seismic reflection profiling, and trenching. The investigations revealed that the lineament is a complex zone of strike-slip deformation consisting of multiple flower structures and fractured rock with deformation at least as young as the base of Quaternary.

The northwest-striking Meers-Duncan-Criner fault zone lies along the northeastern border of a structural trough, the southern Oklahoma aulocogen separating it from a series of crustal uplifts to the southwest such as expressed by the Wichita Mountains. The fault zone, which is aseismic, consists of at least five segments. Two of the segments, the Meers and Criner faults, show evidence of recent (at Quaternary) activity.

Paleoseismic studies along the Meers fault were completed in October 1989. Detailed trench logging and geologic mapping indicate left-lateral oblique slip, down to the southwest, on a steeply northwest dipping to nearly vertical fault. Analyses of these data and radiocarbon dating show that there have been at least two surface faulting events during the past 3,200 years that were probably associated with earthquakes ranging from magnitude 6-3/4 to 7-1/4. The latest displacement occurred about 1,500 years ago. Analysis of faulted alluvial terraces along the Meers fault suggests that a period of quiescence lasting many tens of thousands of years preceded the faulting events.

After completion of the Meers fault investigation, a study of the Criner fault, another member of the Wichita frontal fault system, was begun. Geological reconnaissance and studies of aerial photographs suggest that the Criner fault may also have experienced late Quaternary displacement. It is downdropped to the southwest and is located about 80 kilometers southeast of the Meers fault. Morphologic evidence suggests young displacement and crosscutting relationships between the fault, and late Quaternary terrace deposits at one location suggest that the last displacement occurred between 10,000 and 20,000 years ago. Detailed investigations similar to those carried out on the Meers fault are being conducted to assess the seismic hazard potential of the Criner fault after a long delay in obtaining access to properties containing critical exposures of the fault.

Southeastern Illinois has had seven significant events during the 200-year historical record. There has been considerable debate on the geological structures responsible for this seismic activity, but none has been identified with confidence. For example, the continuity of the seismicity belt, along with geophysical evidence, has led to the interpretation that the fault system in southernmost Illinois and Indiana is a northeastern extension of the New Madrid seismic zone. An alternative explanation is that the earthquakes originate in a complex transition zone connecting two tectonic regimes. Partly because of this lack of knowledge about a causative mechanism of the earthquakes, an investigation was begun in FY 1991 to identify and analyze paleoseismic evidence along the Wabash River and its tributaries.

Mapping and analysis of large dikes and lateral spreads exposed along the banks of the Wabash and White Rivers and other drainages suggest that a large earthquake centered near Vincennes, Indiana, occurred between 2,500 and 7,500 years ago. Comparing the sizes, distribution, and other characteristics of these features with seismically induced features at Charleston, South Carolina, and New Madrid, Missouri, suggests that this earthquake was larger than the 1886 Charleston earthquake (magnitude of about 7.0) but smaller than the 1811–1812 New Madrid earthquakes (magnitude of about 8.0).

The Pacific Northwest is underlain by the Cascadia subduction zone, in which the oceanic Juan de Fuca plate is being subducted beneath the North American plate. This region is an enigma in that the geological and geophysical evidence indicates active subduction, but there have been no historic large-thrust earthquakes along the plate interface, a phenomenon observed in other subduction zones around the rim of the Pacific Ocean.

The USGS is conducting a major study of the geology and tectonics of this region. The NRC is partially funding two neotectonic research projects of this program, one in southwestern Washington and the other in central Oregon. These projects are continuations of investigations that revealed geologic evidence suggesting the occurrence of several prehistoric and Holocene large earthquakes. The evidence lies in buried marsh and shallow marine sediments, which indicate several cycles of normal stratigraphic deposition abruptly terminated by catastrophic events. These events are interpreted by most researchers to indicate large subduction zone earthquakes. At least five events are in evidence in southern Washington. The ongoing research is to better define the ages of these events, determine their regional extent, and estimate their recurrence intervals using precise radiocarbon dating techniques of subsidence-killed Sitka spruce trees to reduce the errors inherent in the conventional technique of dating. Another study to accomplish this is the analysis of diatom fossils and sand found on top of several buried peat layers. This analysis is expected to determine whether these materials were deposited by tsunamis following the earthquakes.

In conjunction with these studies, an investigation is under way to identify and define seismically induced paleoliquefaction features in this region. Thus far, reconnaissances along the Chelahis River and other nearby drainages have not identified evidence of seismically induced deformation features. One paleoliquefaction feature was identified near the Copalis River, and detailed investigations were conducted. The age of the feature, which was determined to have been formed 11,000 years ago, does not coincide with the ages of any of the subsidence events, however. Additional river reconnaissances will be carried out, and lake bed sediment will be examined for earthquake-induced deformation.

The size of the maximum or characteristic earthquake that a fault can produce and the location of that earthquake along the length of a fault are major means of estimating design ground motions. Fault rupture length is a key parameter for constraining the size of future earthquakes on a fault. This constraint can be obtained from studies of the rupture length versus magnitude or energy release of previous earthquakes. During the past decade, fault segmentation has emerged as a field of earthquake research that has important implications and applications for evaluating seismic hazard. It is based on the common observation that fault zones, especially long ones, do not rupture over their entire length during a single earthquake. A variety of structural and paleoseismic studies and investigations of historical earthquakes clearly indicate that the location of rupture is not random, that there are physical controls in a fault zone that define the extent of rupture and divide a fault into segments, and that segments can persist through many seismic cycles. Inherent in the concept of segmentation is the idea of persistent barriers that control rupture propagation. The recognition and identification of rupture segments have the potential to provide new insights into characterizing seismic sources and understanding controls of rupture initiation and termination.

Segmentation for selected faults is being evaluated using paleoseismic recurrence data and information on slip per event and slip rate. The data base is small, but there are a number of faults that have the potential to yield information on long-term segmentation. The data collection and analysis currently under way include: (1) timing of the most recent and prior events along the length of the fault; (2) slip distribution during the historical event and slip during paleoearthquakes on the same segment (repeated similar slips would imply fixed segment lengths; variable slips would indicate variability in segmentation); (3) slip rates at different locations on the fault; and (4) structural geology and geophysics of the fault zone. Although some data are available in the published literature, much is being obtained from unpublished files and paleoseismic studies that are in progress. Part of the data collection has involved field visits for onsite evaluation of published information and, in some cases, development of new data such as slip per event for paleoearthquakes on segments that have had historical ruptures.

Studies during FY 1991 have focused primarily on the Rodgers Creek-Hayward fault zone, the segment of the San Andreas fault that ruptured during the 1989 Loma Prieta earthquake, and the Wasatch fault zone. Possible segmentation boundaries have been identified along the Rodgers Creek and Hayward faults and are being investigated. The two faults are separated by a 6-kilometer-wide releasing bend. The last rupture on the Rodgers Creek fault occurred in 1868, and there is no current seismicity or creep occurring. The Hayward fault, on the other hand, is experiencing creep and seismicity.

Geomorphic reconnaissance and trenching in the Santa Cruz Mountains showed that, although the Loma Prieta rupture did not reach ground surface, repeated surface faulting has occurred on this segment of the San Andreas fault through time. This indicates that variable modes of rupture may occur on this segment.

Intraplate segmentation and its application to hazard focused on comparing the relationship between long-term slip rates, slip per event, and paleoseismically determined timing of earthquakes by using the geologic history over the last 10,000 years of the Wasatch fault zone.

The NRC supports several strong ground motion studies relating to both the Eastern United States and California. A study of soil dynamics at Garner Valley (near Anza), California, employs an array of wide-band strong motion seismometers placed at various depths in boreholes to gain information on soil dynamics and amplification of earthquake motion. This study, performed by the University of California at Santa Barbara, is one of the many research programs that demonstrate effective cost-sharing between the NRC and other agencies. The study is being funded in cooperation with the USGS and the U.S. Army Corps of Engineers and with support from the Commissariat a l'Energie Atomique of France.

Although many theoretical and laboratory studies have investigated the effect of near-surface soil layers on amplification of ground motion, very few direct measurements are available to confirm predictions made by the more theoretical methods. The Garner Valley array is located between the San Jacinto and San Andreas faults in an area of high seismicity. The site is underlain by soil and weathered granite over a granitic basement at about a 45-meter depth. Seismometers were emplaced at the surface and in boreholes at various depths ranging to 220 meters. In less than a year, 125 earthquakes were recorded with magnitudes ranging from 1.2 to 4.7. Analysis of the data shows that, at low frequencies, amplification from bedrock to surface is a factor of six over a wide range of magnitudes. The lower frequencies are also those that have the highest damage potential for engineered structures.

During the summer of 1990, the EPRI elected to join this experiment by supporting the University of California at Santa Barbara in deploying a surface array and an additional borehole instrument at the Garner Valley site.

During FY 1991, records from 17 earthquakes within 20 kilometers of the instrument array with magnitudes of 2 or greater were selected to examine the amplification as a function of frequency. Then, in order to examine the effect of layering the acceleration amplitude spectrum for S waves at various depths for two earthquakes, magnitudes 4.2 and 2.5 were considered. The results showed that the spectrum for relatively unweathered granite bedrock at 220 meters depth has a substantially greater amount of high-frequency energy than materials at shallower depths. Weathered granite amplified low frequencies but attenuated high frequencies, and soil amplified the overall spectrum.

Other ground motion research supported by the NRC is being performed by the USGS and includes the analysis of strong ground motion teleseismic records of large intraplate earthquakes and estimating high-frequency ground motions for earthquakes in the Eastern United States.

In FY 1987, the NRC and the National Geodetic Survey established a portion of a National Crustal Strain Network covering the eastern two-thirds of the United States and consisting of 45 stations whose positions are measured accurately every 2 years via the Global Positioning System (GPS). The purpose of this network is to provide a different set of data to help analyze the causes of seismicity. In addition to its original purpose, the network now forms the backbone of a new GPS survey system for the nation. Because of this, many statewide high-precision networks are being tied to it, providing a much greater density of data points without extra cost to the NRC.

The strain network was measured for a second time in FY 1990. With improved measurement procedures and a better satellite constellation, the accuracy of the measurements was improved and made more consistent across the network. Accuracies were achieved that translate to errors of a few centimeters over baselines of thousands of kilometers. After at least three sets of measurement are available, these data will be analyzed to arrive at preliminary conclusions on crustal strain in the United States, including at least an upper limit for strain. Directional effects will also be investigated because they may indicate non-uniform strain. However, detailed answers on crustal strain are expected only after a time span of about 10 years.

The original set of GPS measurements for the 45-station crustal strain network covering the Eastern and Central United States, which was measured during the winter of 1987–1988, was recomputed in FY 1991 using newer software for improved accuracy. This set of measurements forms the baseline for future measurements. Experience gained, particularly from this first set of GPS measurements, has led to better survey procedures. This, together with improvements in instrumentation, software, and satellite availability, is expected to lead to further improvements in accuracy for the network.

In addition to this national network, the NRC also supports a smaller local strain network in Maine. Analysis of the two sets of position measurements obtained in Maine so far has shown that, within the error margins and considering that the vertical GPS component is the least accurate, a consistent and clear pattern of subsidence has not emerged. However, differences between the two sets of data do show indications of subsidence in the eastern coastal portion of Maine. It also appears that subsidence rates are probably in the range of 5 millimeters per year or less. Earlier postulated values of up to 9 millimeters per year are thus not substantiated.

Probabilistic seismic hazard assessments (PSHAs) began about a decade ago, and they have become an increasingly important aspect of site evaluations for nuclear power plants and other facilities. The revision of Appendix A to 10 CFR Part 100, which is in progress, will put substantial emphasis on PSHAs as part of the investigations required for nuclear power plant sites. PSHAs are of particular interest in the Eastern and Central United States where the uncertainties created by a lack of detailed knowledge of the seismicity make it difficult for a deterministic evaluation to arrive at a balanced estimate of seismic hazards.

Two large-scale PSHA studies are available for the Eastern and Central United States. One was performed by Lawrence Livermore National Laboratory (LLNL), and one was performed by EPRI and sponsored by utilities in the Seismicity Owners Group. These two studies resulted in similar methodologies and produce hazard curves with similar characteristics; they also produce consistent relative hazard rankings for plant sites in this region. A difficulty arises, however, from the fact that absolute hazard levels may differ by as much as two orders of magnitude at certain sites.

Because more consistent absolute hazard levels will be needed in the future to resolve questions of power plant design and licensing, a plan was formulated for a study that will analyze the two existing methodologies to determine the sources of discrepancies and attempt to mitigate the differences between the LLNL and EPRI approaches. From previous analyses, it is known that certain input parameters, such as seismic parameters and ground motion models, cause some of the differences. It appears that the computer codes used to do the calculations will give approximately the same results for a given input although the validity of this statement also needs to be more fully verified. The planned study is aimed at a consolidated methodology that will provide more uniform results and can be used as a basis for PSHAs for the next decade or so. A peer review by a panel appointed by the National Academy of Sciences is planned to ensure the impartiality and objectivity of the study.

1.4.3.2 Seismic Engineering Research

In addition to the earth science research discussed above, the NRC seismic research program includes several engineering-oriented programs to determine the effect of earthquakes on nuclear plant structures and safety systems.

Implementation of Executive Order 12699. Executive Order 12699, "Seismic Safety of Federal and Federally Assisted or Regulated New Building Construction," was issued January 5, 1990, by the President to implement certain provisions of the Earthquake Hazards Reduction Act of 1977. The Executive Order applies to all Federal agencies that (1) are responsible for the design and construction of new Federally owned buildings, (2) are responsible for construction and lease of new buildings for Federal use, (3) assist in the financing, through grants or loans, of newly constructed buildings, (4) guarantee the financing, through loan or mortgage insurance programs, of newly constructed buildings, or (5) are responsible for regulating structural safety of new buildings. Agencies responsible for construction projects of the first two types listed must demonstrate compliance for all projects for which development of detailed plans and specifications is initiated subsequent to the date of the order. Agencies administering the other types of programs have 3 years from the date of the order to establish an appropriate seismic hazard reduction program.

During the past year, the staff performed a careful review of the Executive Order and NRC requirements for the design and construction of buildings associated with nuclear power reactors and other activities. The "other activities" included: Class 104 licenses for medical therapy and research and development facilities, processing of uranium ores in milling operations, high-level-waste repository licensing, onsite spent fuel storage, licensing of plutonium processing and fuel fabrication plants, and license application reviews for uranium enrichment facilities. It was concluded that the NRC's current practice meets the requirements of the Executive Order, and no regulatory action is necessary.

Individual Plant Examinations for Seismic Events. A major activity in the seismic engineering area concluded this year when the NRC published final guidance for conducting the individual plant examinations for seismic events to implement the Commission's Severe Accident Policy Statement. A number of changes were made to the draft guidance documents as a result of an NRC-sponsored workshop and written responses from the public. Further clarifications of staff guidance in the seismic area were made during a question and answer session that was part of the Nuclear Management and Resources Council (NUMARC) workshop.

During the past 2 years, the development of guidance for conducting the individual plant examinations for seismic events has been achieved through the formation of the External Events Steering Group (EESG) consisting of senior level NRC management. The EESG in turn established a seismic subcommittee (among other subcommittees), consisting of NRC staff members with expertise in earth sciences, structural/mechanical engineering, systems analysis, and risk and reliability analysis. The seismic subcommittee issued a report in March 1990. The purpose of the report was to recommend to the EESG: (1) the objectives for the seismic portion of the individual plant examinations for external events (IPEEE), and (2) guidance or guidelines for conducting the seismic portion of the IPEEE. Results and insights from various seismic research programs (e.g., seismic margins method and component fragility testing programs) are incorporated into this guidance. The staff resolved the comments on the draft guidance and the final guidance was issued in FY 1991, as described above. (For further accomplishments, see "External Events" under Severe Accident Implementation in Chapter 5.)

The Commission-approved draft guidance document for individual plant examinations of external events has endorsed the use of the seismic component fragilities developed by the Brookhaven National Laboratory. The components included the motor control centers, switchgears (low and medium voltage), panelboards, switchboards, power supplies, instrumentation and control panels, transmitters, indicators, switches, transformers, batteries, battery chargers, inverters, motors, and electrical penetration assemblies. The seismic fragilities are expressed in terms of medium and standard deviations of spectral acceleration capacities to allow the NRC staff to understand the conservatisms associated with the estimates of seismic fragility. Attention has shifted toward understanding how relay chatter can influence circuit breaker tripping. This has been identified as a safety concern because such a scenario can cause failure to start emergency power during and after an earthquake, leading to station blackout. Test plan development and equipment procurement is under way to address this issue.

Seismic and Geologic Siting Criteria. Starting early in 1991, the staff began a major rulemaking effort—the revision of the seismic and geologic siting criteria for nuclear power plants, Appendix A to 10 CFR Part 100. This activity is intended to (1) benefit from the experience gained in applying the existing regulation; (2) resolve interpretative questions; (3) provide needed regulatory flexibility to incorporate state-of-the-art improvements in the geosciences and earthquake engineering; (4) simplify the language to a more "plain English" text; and (5) acknowledge various internal staff and industry comments. Criteria not associated with the selection of the site or establishment of the safe shutdown earthquake ground motion will be placed in Part 50. This action is consistent with the location of other design requirements in Part 50.

Several issues are being addressed by the staff in conjunction with the revision of the regulations. In the geoscience area, the emphasis on deterministic and probabilistic assessments, along with guidance on how the two methods should be merged if applicable, is being evaluated. In the earthquake engineering area, the proper role of the operating basis earthquake in future plant design is being assessed.

During the development of the proposed regulations, the staff had two public meetings (March and April 1991) with interested industry groups, principally NUMARC and EPRI.

The revision of the geologic, seismic, and earthquake engineering criteria is being performed in conjunction with the revision of the reactor siting criteria, 10 CFR Part 100. The Commissioners have requested that the regulations should be revised as soon as possible to facilitate an early site review. The rulemaking package was submitted to the Advisory Committee on Reactor Safeguards in October 1991 to facilitate an early 1992 release of the proposed revision of the regulations to encourage participation from the public and other organizations in the development of the regulations, as planned.

International Seismic Test 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 bigger, more complex 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) program at Hualien, Taiwan, follows the soil-structure interaction (SSI) experiments at Lotung, Taiwan. In Lotung, at a soft soil site, two scaled cylindrical reinforced-concrete models (1/4-scale and 1/12-scale of typical full-scale reactor containments) were constructed. The two models and their surrounding soil were fully instrumented to monitor ground motion and structural response. Since the test facility completion in October 1985, more than 30 earthquakes ranging from magnitude 4.5 to 7.0 were recorded. The Lotung experiments were completed in 1987 and the data have been subsequently used to check the validity of calculational models. The Lotung experiment was successful in that soil amplification of the earthquake motions was sufficient to challenge the predictive methods. However, bending modes of the structures were not excited due to the softness of the soil. The planned SSI studies at Hualien, Taiwan, will be performed at a stiff soil site that historically has had, on average, larger magnitude earthquakes than Lotung. 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, and new members Korea Power Engineering Company (KPIEC) and Korea Electric Power Corporation (KEPCO).

The test model designed for the Hualien LSST program is almost a replica of the Lotung model, except that, for the purpose of lowering the soil-structure-system frequency to 3 to 5 Hertz (the expected dominant frequency at the site), the roof slab is enlarged and thickened to increase the mass. This similarity will enhance comparison of results between the two experiments. Because of the stiffer foundation, the Hualien model is expected to experience more SSI than the Lotung models. The facility is scheduled for full operation in the fourth quarter of 1992.

2 PREVENTING DAMAGE TO REACTOR CORES

This program encompasses research pertaining to the operations of the reactor as a system, including controlling power level, maintaining core cooling and heat removal, and maintaining proper coolant temperatures and pressures under both normal and abnormal conditions of operation. The program also includes consideration of operator actions as an integral part of the reactor system. A complete understanding of the reactor operating as a system makes it possible to define the conditions of operation that prevent core damage and also actions to minimize the consequences of a core damage event, should one occur. This research program emphasizes severe accident prevention and mitigation by enhancing the understanding of both plant and human behavior related to accidents and transients. This information is used to ensure that regulatory requirements exist that suitably ensure that plant equipment, procedures, and personnel can deal with operating events and prevent serious accidents or can mitigate the consequences of an accident, should one occur.

2.1 Plant Performance

2.1.1 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 energy, mass, 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 surfaces of steam generators in the primary loops.

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

2.1.2 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 heat, liquid, vapor, kinetic energy, and momentum at each small but finite time step. Physical laws are used when possible to calculate all these exchanges. Empirically derived formulas, 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 every 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 to derive empirical formulas for determining phenomena within each cell. Second are separate-effect experiments 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 a complete reactor. The results of these tests provide feedback to correct the code and our understanding of the transients.

2.1.3 Research Accomplishments in FY 1991

2.1.3.1 Plant Transient Analysis

Modeling. As a result of the March 20, 1990 event at the Vogtle nuclear power plant in Georgia, the staff initiated a comprehensive evaluation of safety during shutdown and low-power operations. In particular, RES was asked by the Office of Nuclear Reactor Regulation (NRR) to evaluate the effectiveness of alternate decay heat removal methods after the loss of residual heat removal (RHR). A report, EGG-EAST-9337, Revision 1, "Thermal-Hydraulic Processes Involved in Loss of RHR During Reduced Inventory Operation," was completed on March 1, 1991. Of the two cooling methods identified in NUREG-1410 (a staff document on this subject), i.e., gravity feed from outside sources such as the refueling water storage tank and reflux cooling, the RES report EGG-EAST-9337 concluded that gravity feed phenomena were well understood but reflux cooling at reduced pressures in the presence of air was not well understood.

In order for reflux cooling to work in a system partially filled with air, the pressure must increase enough to expose a condensing surface in the steam generator. The concern was that such a pressure rise may lead to blowing out the nozzle dams or the instrument thimble tube replacements, leading to a possible loss of coolant. The design pressure of the thimble tube replacement is about 50 psi, while the low-temperature overpressure protection setpoint is well above this, at about 100 psi.

Bounding analyses were performed on U-tube steam generators showing that the pressure increase required to expose a condensing surface was only 15 psi, well below the 50 psi design pressure of the thimble tube replacements. In order to confirm these results with test data, initial commitments were obtained from the French and Japanese to perform relevant tests in the BETHSY and ROSA facilities, respectively.

The pressure rise needed to expose a similar condensing surface in Babcock and Wilcox reactors with once-through steam generators may be higher. Special tests were performed at the University of Maryland facility to obtain data on the magnitude of this pressure rise.

A request was received to evaluate fire software for use in calculating heat and smoke propagation in a multicompartment structure to determine the inadvertent actuation of fire protection systems. The results were to be used in a probabilistic risk assessment (PRA) study associated with the resolution of Generic Issue 57, "Inadvertent Actuation of Fire Suppression Systems." A survey concluded that the National Institute of Science and Technology (NIST) produced candidate software. The CCFM code was evaluated, found to be appropriate, and then used to provide the analyses needed for the PRA study. Additional modeling of vertical plumes and horizontal ceiling sets was identified, and a more comprehensive NIST code was then used to provide results with these models to corroborate the previous CCFM results.

Thermal mixing in relation to pressurized thermal shock has been examined experimentally throughout the world in a variety of scales. These include the CREARE-1/5, the IVO/IVO(NRC)-2/5, the PURDUE(UCSB)-1/2, the CREARE-1/2, the HDR-1/1, and the UPTF-1/1 test facilities. The regional mixing model and the associated computer programs REMIX and NEWMIX are used to interpret these data in a comprehensive fashion. These interpretations indicate that cooldown transients and degree of stratification can be predicted with confidence.

Fifteen countries participated in the International Code Assessment and Application Program (ICAP). These countries are Belgium, Finland, France, Germany, Italy, Japan, Korea, Netherlands, Spain, Sweden, Switzerland, Taiwan, United Kingdom, USSR, and Yugoslavia. Representatives of these countries, Idaho National Engineering Laboratory, Los Alamos National Laboratory, and NRC cooperated on reporting on assessment results of NRC thermal-hydraulic codes (RELAP5-MOD3, TRAC-PF1/MOD2, and TRAC-BF1). This was a 7-year effort carried out from 1985 to 1991. With the mission of ICAP completed, a successor program will take effect to maintain a code user's group focused on (1) code applications for plant safety analysis and (2) code maintenance.

Operating Reactor Assessments. In order to extend the length of fuel cycles, utilities operating PWRs have been increasing the enrichment of reload fuel. The fresh reload assemblies may be highly reactive when they do not contain control element assemblies or many burnable poison rods. If they are placed in certain loading configurations, this could lead to a loss of shutdown margin below the required 5 percent or, in the extreme, to an inadvertent criticality.

The NRC alerted PWR owners to this potential problem and requested that licensees ensure (via specific actions) that any intermediate fuel assembly configuration maintains the required shutdown margin. NRR also requested that RES use the most appropriate deterministic and probabilistic techniques to analyze different refueling configurations in order to understand the potential for losing shutdown margin and for inadvertent criticality.

The research results showed that a cluster of at least four fresh fuel assemblies is needed to lose the required shutdown margin, and the frequency of this event is expected to be 1.0E-6/year. If the conservative assumption is made that it only takes five fresh assemblies to cause an inadvertent criticality, then the frequency is 4.8E-9/year, which is acceptably low. If such a criticality event were to occur, there would be radioactivity released from the fuel but no dose to the public if the containment isolation systems were effective.

The March 9, 1988 oscillation event at the LaSalle nuclear power plant in Illinois brought into question the ability of current analytic techniques to predict the instability boundary in BWRs as well as the magnitude of the power and flow oscillations that might occur when the reactor became unstable. This past year the NRC determined that its analytic tools can predict such oscillatory behavior in BWRs. One major result found was that it is important to model major systems in the balance of plant, especially the feedwater temperature and flow, in order to correctly predict the oscillations in the reactor vessel. It should be noted that General Electric has now improved their modeling capability to calculate similar oscillatory magnitudes.

Research results this past year showed that suppression pool heatup after an anticipated transient without scram (ATWS) with unstable oscillations is well below any safety margin. However, it was found that there could be a large increase in fuel rod temperature. Two potential improvements to operating guidelines were investigated. First, it was found that significant oscillations can start about 2 or 3 minutes after an ATWS and thus the operator acti[illegible] boron injection through the standby liquid control sy[illegible] which may take about 10 minutes to influence the power oscillations, will not be immediately effective. Second, it was found that early tripping of the feedwater pumps is very effective in suppressing oscillations.

The RES team reviewing the Yankee Rowe submittal on evaluation of their pressure vessel raised a question on the calculation of the fluid temperature in the downcomer during a postulated small-break loss of coolant. A detailed review was performed, and it showed that the Yankee Rowe calculation with the REMIX code was a best estimate of the downcomer temperature when only high-pressure injection is considered. This conclusion was based on the assessment of REMIX against data from six scaled test facilities, published in April 1991 as NUREG/CR-5677, "A Unified Interpretation of 1/5 to Full Scale Thermal Mixing Experiments Related to Pressurized Thermal Shock."

An analytic study encountered difficulties in the use of thermal/hydraulic codes to calculate the time to fuel pin failure after a loss-of-coolant accident (LOCA). The study investigated the use of new source term information to increase the allowed technical specification time for containment isolation valve closure following a LOCA. After focusing on the key code issues for the analyses being performed, the study was completed showing a time of about 30 seconds to fuel pin failure.

With the completion of testing and subsequent shutdown of large-scale U.S. thermal/hydraulic test facilities in 1989, the NRC was concerned whether small-scale facilities, such as the University of Maryland at College Park (UMCP) facility, could provide useful data for code assessment and issue resolution. The UMCP program was completed this year and showed that useful data could be obtained after a proper scaling methodology was used to specify the test boundary conditions and to interpret the resulting data. UMCP results agreed with major results of the larger-scale MIST facility after they were scaled to vessel inventory and after appropriate scaling of initial and boundary conditions was determined.

The UMCP scaling report was reviewed by a group of experts, and their conclusions on the usefulness of such a small-scale facility were mixed. To summarize the review, a small-scale facility must be designed and operated with careful attention to scaling principles. There is much less experience with the Ishii scaling used for the UMCP than the power-to-volume, full-height, full-pressure scaling used for MIST and most other large-scale facilities.

A small-scale facility may be more useful for uncovering qualitative surprises in system interactions among components than in providing quantitative data for code assessment. Thus, such a facility can meet scaling goals that are modest but achievable. The UMCP program showed that a small-scale facility can reproduce the key phenomena expected in the full-scale plant; they can be reproduced in the same time sequence, and their quantitative characteristics can be reasonably approximated.

2.1.3.2 Accident Management

NRC research continued its dual role of (1) defining the necessary components of a functioning utility severe accident management plan, and (2) providing the technical basis for evaluating industry-documented products on accident management.

In the first category, two reports were completed and transmitted to industry. The first (NUREG/CR-5543), "A Systematic Process for Developing and Assessing Accident Management Plans," clarifies how the five framework elements (strategies, instrumentation, guidance, decisionmaking, and training) could be integrated into a working accident management plan. The second (NUREG/CR-5691), "Instrumentation Availability for a Pressurized Water Reactor with a Large Dry Containment During Severe Accidents," demonstrates how to assess the potential availability or unavailability of instrumentation that is necessary to follow the course of a severe accident.

In the second category, significant progress was made in identifying and assessing accident management mitigative strategies. Two workshops held at UCLA, one on PWRs and the other on BWRs, aided this progress by focusing on a limited set of specific strategies. The key mitigative strategies identified for detailed assessment include the following: BWR boration, external vessel flooding, PWR primary depressurization, late primary bleed and feed, use of containment sprays, containment venting, hydrogen control, fission product control, and late secondary bleed and feed.

A study of potential recriticality in a BWR following a core damage event (NUREG/CR-5653) showed that about 700 parts per million of B-10 is sufficient to ensure subcriticality for conceivable core configurations, including standing fuel rods and melted control rods. A related strategy of initiating RHR cooling of the suppression pool as quickly as possible was suggested as a way to extend the time available for boration.

The strategy of PWR primary depressurization (NUREG/CR-5447) was shown to be most effective if performed late rather than early. For a loss-of-heat-removal accident, "late" refers to time after the core begins to uncover. A survey was completed showing which plants would respond favorably to this strategy, for which plants the strategy would not work at all; and which plants would need further analysis to confirm the effectiveness of the strategy.

There is a question of whether natural circulation might lead to failure of the pressurizer surge line and consequent unintentional depressurization. In order to address this question in a systematic way, a workshop was held and four processes were identified for inclusion in code modeling: (1) steam generator plenum mixing, (2) surge-line flow, (3) hot-leg/vessel flow behavior, and (4) hydrogen distribution. Models and/or bounding assumptions were developed for all four processes so that analyses can be performed next year to resolve the question by predicting the possibility of inadvertent depressurization.

2.2 Human Performance

2.2.1 Statement of Problem

A large fraction of all safety-related events reported at nuclear power plants continue to involve human performance. Methods and data are needed to identify, systematically set priorities for, and suggest solutions to human performance issues in the operation and maintenance of nuclear power plants during normal, transient, and emergency situations.

2.2.2 Program Strategy

The human factors and reliability assessment 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 requirements, recommendations, and guidance related to human performance.

The human factors regulatory research program is divided into four interrelated program elements: (1) personnel performance, (2) human-system interfaces, (3) organizational factors, and (4) 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 measures for evaluating the interface between the system and the human user from the perspective of safe operations and maintenance. Organizational factors research will result in the development of tools for evaluating organizational issues within the nuclear industry. And, 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.

2.2.3 Research Accomplishments in FY 1991

2.2.3.1 Human Factors Research

An ongoing project in the personnel performance element is the Human Performance Investigation Process (HPIP), an investigative process designed to provide a standardized method to identify the causes of human errors. Both Region I and NRC headquarters personnel have been trained in the use of the process; it has also been pilot-tested by Region I inspectors. Three continuing projects are related to human factors evaluation of processes conducted by materials licensees—industrial radiography, brachytherapy using remote afterloaders, and teletherapy. The objective of these projects is to identify factors contributing to human error in these processes. Research continued on an evaluation of the impact of overtime and shift scheduling on operator and plant performance based on nuclear power plant data. A study of operator performance during 8-hour versus 12-hour night shifts has been conducted, and the data are currently being analyzed. Work continued on the development of a method to evaluate the effectiveness of training programs at nuclear power plants. A report describing the results of a workshop of training evaluation experts in developing a framework on which to base a training effectiveness methodology is being prepared. A research effort addressing what factors are considered when making staffing decisions and how these factors relate to safe startup, shutdown, and operations of nuclear power plants continued. A study continued to establish the applicability of available information to the understanding of the impact of environmental influences on human performance. Based on a comprehensive review of the literature in this area, it has been decided to focus the study on the effects of heat and noise.

A new research project in the personnel performance element addresses training for severe accidents, focusing on cognitive skills development and training in skills for coping with stress. A second related project will bring together a group of experts in a workshop to provide their independent views on other issues related to accident management training and decisionmaking.

Human-system interface research continued with NRC participation in the Halden Reactor Project of the Organization for Economic Cooperation and Development. As a followup to an assessment of the costs and benefits of expanded regulatory guidance on normal and abnormal operating procedures (NUREG/CR-5448), a new project was initiated to develop guidance for the review of procedure upgrade programs. Activity continued toward the resolution of Generic Issue 5.1 with an historical survey of plant incidents caused, at least in part, by inadequate consideration of human factors at local control stations (LCS) and with a series of site visits to document the status of LCS upgrades. Work on developing a guideline for use in performing human factors reviews of advanced control and display technology continued, as did a related effort to evaluate the effects of alarm reduction techniques on operator performance. Work also continued on the identification of the frequency, severity, and nature of procedure violations in U.S. nuclear power plants. Research continued on computer classification, which involves review and evaluation of existing regulatory guidance documents and quality assurance methods relative to their adequacy as applied to computer-based safety systems, and was expanded to focus on Class IE systems. Research continued on developing guidelines for verification and validation of expert systems, which is being jointly funded by the Electric Power Research Institute and the NRC. Work continued on the development of a baseline of operator performance that would serve as a base against which to evaluate changes to operator interfaces. A method derived from the DeGroot memory paradigm was tested as a potential measurement method for evaluating human-computer interfaces. A project on attributes of high-integrity software was initiated. Also, efforts on applications of expert systems, neural networks, and task network modeling were initiated as first-phase feasibility projects.

Additional RES activities in human performance have included direct support for activities such as support for rulemaking on training, development of draft Regulatory Guide 1.8 on qualification and training of nuclear power plant personnel, and plant inspections. In addition, a study was started to examine the feasibility of establishing NRC human factors regulatory research facilities.

2.2.3.2 Organizational Factors

This research is intended to characterize the attributes of nuclear power plant organizational and management functions important to risk and to develop measures, procedures, and criteria for systematically evaluating plant management performance. The results are intended to enhance the NRC inspections and diagnostic evaluations of potential problems at nuclear power plants. Also, this work is intended to develop programmatic (if possible, leading) indicators of plant safety performance. This work will develop methods to integrate into PRA the important reality of organizational influences.

In the area of methods to support PRA, the feasibility of an initial method for observing certain organizational factors was evaluated in cooperative tests with two utilities. Alternative methods are being developed for field testing in 1992.

In the area of programmatic performance indicators, organizational factors that correlate with safety performance in both the chemical industry and the nuclear power industry are being explored.

2.2.3.3 Reliability Assessment Research

This research is intended to develop methods for applying reliability technology to help improve the regulatory program in the following areas.

In the area of technical specification requirements, this work developed methods to analyze the risk impact of allowed outage times and surveillance test intervals. In FY 1991, these methods were enhanced to allow optimization of surveillance test intervals from a risk perspective. Also, methods are being developed to evaluate the risk impact of technical specification requirements regarding surveillance test intervals during plant shutdown, action statements that require plant shutdown under certain conditions, and strategies for scheduling preventive maintenance on standby equipment during plant operation.

In the area of risk-based performance indicators, a method has been developed for monitoring the availability of three selected safety systems.

In the area of cognitive errors, experiments are being conducted using training simulators and operating crews at nuclear power plants. The objective is to understand the thinking processes of operators when they are trying to handle novel and challenging sequences that go beyond the specific guidance of existing training and procedures. This understanding will be incorporated into an artificial-intelligence computer simulation of thinking processes of operators to help the analysis of error-likely events.

3 REACTOR CONTAINMENT PERFORMANCE

The basic criteria for licensing nuclear power plants for construction and operation are judged to have provided a considerable safety margin, affording the public protection from radiation even under severe accident conditions such as those that occurred in 1979 at Three Mile Island. The physical possibility of even more severe accidents than that at TMI is, however, recognized. Considerable progress has been made in recent years in understanding the underlying physical and chemical phenomena that can occur in a severe accident. Such information is essential as a basis for assessing potential safety improvements and for making decisions on whether or not particular improvements are warranted. As pointed out in the Commission's Severe Accident Policy Statement, such decisions should be based on a combination of engineering judgment (i.e., a deterministic method of setting and assessing safety margins) and the application of probabilistic risk assessment techniques based on up-to-date experimental information to evaluate the likelihood of the occurrence of rare events.

In similar fashion, the same underlying science and decision process can be applied to reevaluations of existing safety systems and regulatory requirements to determine if particular conservative assumptions have been warranted in terms of risk reduction.

3.1 Core Melt and Reactor Coolant System Failure

3.1.1 Statement of Problem

Major uncertainties in estimating the probability of early containment failure, and the associated radioactive release, 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. Until a better understanding of core melt, including fission product release, hydrogen generation, and response of the reactor coolant system to fuel melting and relocation, is gained, containment failure probabilities and related source terms will continue to be conservatively biased to ensure an adequate margin of safety.

3.1.2 Program Strategy

In order to better understand just what happens during a core melt accident, and thereby reduce the uncertainty 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 reactor coolant system, (4) the natural circulation of hot gases that might cause early failure of a pipe or steam generator tubes, (5) energetic fuel-coolant interactions that occur as molten debris falls into the water-filled lower head or as water is added to molten debris, (6) the composition, morphology, and temperature of debris at the time of vessel (or reactor coolant system) failure, and (7) the mode of vessel failure. The core melt and reactor coolant system (RCS) failure program is divided into three main activities: (1) core melt progression and hydrogen generation, (2) the behavior and chemistry of fission products released during core melt, and (3) fuel-coolant interaction. The in-vessel core melt progression and hydrogen generation work includes in-reactor experiments, out-of-reactor experiments, examination of specimens from TMI-2, and analytical model development. The research on the behavior and chemical form of fission products released from the fuel in the course of a severe accident is being conducted at high temperatures when core geometry is changing and fission product chemistry and its effect on retention of fission products within the RCS are significant. The fuel-coolant-interaction work is focused on the development and validation of a model for fuel-coolant interactions for use in accident analysis.

3.1.3 Research Accomplishments in FY 1991

3.1.3.1 Core Melt Progression and Hydrogen Generation

In-vessel core melt progression describes the state of an LWR reactor core from core uncovery up to reactor vessel meltthrough in unrecovered accidents or through temperature stabilization in accidents recovered by core reflooding. Melt progression provides the initial conditions for estimating loads that may threaten the integrity of the reactor containment. The significant results of melt progression are the melt mass and the rate of release, composition, and temperature (superheat) of the melt released from the core and later from the reactor vessel at meltthrough. Melt progression provides the in-vessel hydrogen generation and the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system. Melt progression also provides the core conditions for assessing accident management strategies.

A great deal has been learned about the processes involved in severe fuel damage and in the early phase of melt progression that extends through metallic (but not ceramic) material melting and relocation. This information has come from integral tests in the PBF, ACRR, NRU, NSRR, and PHEBUS test reactors, from the LOFT FP-2 test, from tests in the CORA ex-reactor fuel-damage test facility, and from separate-effect experiments on significant phenomena. Most of the available information on late-phase melt progression has come from the post-accident examination of the TMI-2 core. Despite the core reflooding that successfully terminated the TMI-2 accident, the general late-phase melt progression phenomenology of that accident, although not the detailed behavior, appears to be applicable to unrecovered as well as to recovered accidents and possibly to some BWR accidents as well.

The results of these integral tests and the TMI-2 core examination have provided a very consistent picture of melt progression in blocked core accidents like TMI-2. This involves the development of a debris-supporting metallic blockage above the water level in the lower portion of the core during coolant boildown. This blockage is produced by the relocation and freezing of metallic melt formed from unoxidized Zircaloy cladding and control rod material. As shown by the TMI-2 core examination, fission product decay heating produces a growing pool of mostly ceramic UO_2 fuel and oxidized Zircaloy in the particulate debris bed above the metallic core blockage. The growing pool melts downward through the supporting metallic blockage and the secondary ceramic crusts that surround the growing ceramic melt pool and also radially outward. At TMI-2 with a reflooded core, pool meltthrough was out the side of the core. The mass and other characteristics of the ceramic melt that drains from the core into the lower plenum in blocked core accident sequences are largely determined by the location of the point of breakthrough of the supporting crust by molten core material.

During FY 1991, a "Comprehensive Research Plan for Melt Progression Issue Resolution" was prepared and subjected to expert peer review. The plan and the comments of the reviewers will be used as a basis to update the Severe Accident Research Plan in the melt progression area. The plan includes the objectives of melt progression research, the technical background, a description of the current research needs, a description of the current research program, and the anticipated results of this research.

Current NRC research on melt progression is focused on two major uncertainties or issues. The first issue is determination of whether the TMI-2 blocked core accident sequence is applicable to all unrecovered LWR accidents, including the BWR dry core accidents that result from automatic depressurization. The second issue concerns the conditions for the meltthrough of the growing pool of ceramic (fuel) melt that is supported by the metallic blockage. The meltthrough threshold and location largely determine the mass and other characteristics of the melt released from the core and later from the reactor vessel.

A program of experiments and corollary analysis has been started on both these melt progression issues. On the issue of blockage of the core by metallic melt, TMI-2 and the results of the experiments cited above 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, as happened at TMI-2. All previous experiments for both PWRs and BWRs were performed for these wet core conditions. The emergency operating procedures for U.S. BWRs, however, call for reactor depressurization, which would lead to lowering the water level below the reactor core so that core heatup occurs with very low steam flow through a "dry core." Analysis of this case indicates that the molten core metal (and later molten ceramic fuel) might drain from the core rather than forming a blocked core as at TMI-2. This would produce a major difference in the mass and other characteristics of the melt released from the core and later from the vessel at meltthrough. In FY 1991, preparations were made for a series of experiments to resolve this question of core blockage under BWR dry core conditions. The first of these experiments will be performed in FY 1992.

Experiment preparations and analyses were also started to develop an improved understanding of the process of core melt behavior in accident sequences similar to TMI-2. These experiments are to be performed in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories. The results will be used to assess models of the meltthrough process, and these in turn will be used to assess the adequacy of the modeling in the accident analysis codes MELCOR and SCDAP/RELAP5.

As core melt material relocates to the lower head of the reactor vessel, the major concern of severe accident analysis becomes the mode and timing of lower head failure. The research program in this area includes analysis and the examination of samples from the TMI-2 lower head. Failure mode analyses have been conducted to examine failure by the ejection of the vessel penetrations, failure of a penetration outside the vessel shell, and global and local creep-rupture failure of the vessel shell. The failure mechanisms have been evaluated for debris and thermal-hydraulic conditions that have been estimated for current BWR and PWR designs as well as for TMI-2 specific conditions.

Typical severe accident scenarios have been used for each of the reactor types to develop lower head failure maps. The failure maps show the mechanisms by which a vessel is most likely to fail as a function of system pressure and inner vessel wall temperature. The studies have shown that significant parameters include not only the system pressure and vessel temperature but also the effective flow area and wall thickness of the penetrations and the size of the annular gap between the penetration tube and the vessel wall. A report on this analysis will be issued in FY 1992.

Under conditions that result in low heatup of the vessel wall, the most likely mode of failure is ejection of a penetration due to failure of the vessel seal weld. In this case, the friction between the penetration tube and vessel opening is so low that tube ejection can occur without tube rupture or damage. As the temperature of the vessel wall rises sufficiently to restrain the penetration tube by friction, failure may occur by melt material penetration of the vessel through the tube with subsequent meltthrough or rupture of the tube outside the vessel boundary. Two modes of melt penetration have been studied for this case: (1) conduction limited penetration and (2) bulk freezing penetration. Conduction limited penetration has been found to give the greater melt penetration for all reactor types. Molten debris was also found to be more likely to penetrate through a tube with a large effective diameter, such as a BWR control rod guide tube or a BWR drain nozzle, rather than through a smaller diameter PWR instrument tube.

Failure of the lower head of the vessel by creep rupture may be possible when the system pressure and vessel temperature are sufficiently high. The failure mode can be either a global failure of the hemispherical head that is circumferential around the vessel below the debris surface or failure by local bulging and membrane rupture. Vessel failure analyses to date have assumed the debris conditions that drive simple thermal analyses. Dimensionless groups derived from these analyses may now be used in further detailed analyses with severe accident codes such as SCDAP/RELAP5 to provide the debris conditions. An analysis for the local bulging case has also been developed. When the vessel wall heating is localized, the local failure temperature may be significantly higher than the temperature for global failure because of the structural support from the cooler, nonbulged, adjacent vessel wall.

Examination of the vessel samples from the TMI-2 lower head indicated that a small region of the lower head (approximately 2 feet in diameter) experienced inner surface temperatures of about 1350 K. The examination also indicated that the temperature 2 inches into the wall was about 100 K lower than the inner surface temperature. (The lower head thickness was 5 inches.)

3.1.3.2 Fission Product Behavior and Chemical Form

"Source term" refers to obtaining information on the magnitudes of the radioactive materials released from the core to the containment atmosphere. The timing and other release information, as well as containment failure mode and timing, are needed to calculate the offsite consequences following a postulated severe reactor accident. The NRC conducts source term research to help define and focus accident-management concerns, containment performance improvements, and individual plant examinations to seek out potential vulnerabilities previously undetected.

At present, research is under way to develop theoretically based fission product behavior models to predict fission product release and transport in the reactor coolant system (RCS) and the containment. For the RCS, the mechanistic VICTORIA code is being developed to provide the capability to estimate the quantities of fission products and aerosols released from the reactor core, the extent of their transport through the RCS, the inventory of radionuclides available for release after core debris is expelled from the reactor vessel, and the extent of fission product revaporization from the RCS.

A version of the VICTORIA code has been completed and a user manual was published as NUREG/CR-5545. Model development related to those phenomena encountered during late phases of severe accidents (e.g., fission product release during late phases of core degradation, reentrainment of deposited fission products in the RCS) has been completed. For the containment, the TRENDS models have been developed to calculate the partition of iodine between the aqueous phase and the gas phase in the containment, the production of organic iodide species, containment water pool chemistry, and the extent of iodine revaporization and resuspension from containment surfaces and sumps. The models were used to calculate the revolatilization of iodine from the containment water pool and the production of organic iodine in the containment. The calculations were completed and the results documented in NUREG/CR-5732 (Draft Report for Comment), "Iodine Chemical Forms in LWR Severe Accidents." The results will be employed in the revision of the source terms delineated in TID-14844 (1962), which outline a procedural method to calculate the offsite radiation dose from iodine exposure. In FY 1992, the TRENDS models will be incorporated into the CONTAIN code used for the analysis of containment response to severe accident conditions.

In FY 1990, the NRC entered into an international agreement with the Commissariat a l'Energie Atomique (CEA) of France to participate in the PHEBUS-FP program. This program, sponsored by the CEA and the Commission of the European Communities, consists of "in-pile" severe fuel damage experiments and a study of the fission product behavior and transport in the RCS and the containment system. The program consists of six integral tests for five different simulated severe accidents. The first test is scheduled for October 1992. The agreement is of significant benefit to the NRC because, at a relatively modest cost, the NRC can participate in the PHEBUS-FP project over the life of the project. The NRC will be able to obtain integral experimental data to further validate its analytical models for fission product transport in the RCS and containment and for iodine chemistry in the containment. Information on core melt progression will also be obtained to supplement data obtained under the NRC Cooperative Severe Accident Research Program. This information is confirmatory in nature with regard to current efforts to revise the source term assumptions now based on TID-14844 and for other aspects of the NRC's "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147).

3.1.3.3 Fuel-Coolant Interaction

The Integrated Fuel Coolant Interaction (IFCI) computer code is nearing completion at the Sandia National Laboratories. This code treats the major fuel-coolant interactions in an integrated manner. A draft documentation of the IFCI was completed in FY 1990. Data review and validation of the interactive models in IFCI were continued in FY 1991 with the intent of completing the code manual in FY 1992

The NRC, EPRI, and the Safety Technology Institute of the Joint Research Center of the Commission of the European Communities (STI-JRC) at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of fuel-coolant-interaction experiments at the FARO facility located in Ispra. At the STI-JRC FARO facility, large masses of real reactor core material can be melted and can interact with different depths of coolant at different temperatures and pressures. At least five molten fuel-coolant-interaction experiments will be conducted. The data obtained from FARO is considered to reflect more prototypical reactor conditions than earlier experiments and should greatly enhance the existing severe accident data base in the United States. This technical exchange arrangement is of significant benefit because it enables the NRC, at a modest cost, to obtain prototypic integral experimental data to benchmark analytical models used for predictions of molten fuel-coolant interactions in containment. The information will supplement those data obtained under the NRC Cooperative Severe Accident Research Program and will provide confirmatory information to support the source term assumptions and other aspects of NRC's "Integration Plan for Closure of Severe Accident Issues" (SECY-88-147). The first scoping test at FARO was successfully performed on December 2, 1991. The scoping test employed 32 kilograms of a UO_2ZrO_2 molten mixture (80% UO_2, 20% ZrO_2) poured from a 10-centimeter single jet into a 47-centimeter-diameter chamber containing saturated water, 50 bar pressure, and 1 meter depth. Preliminary results indicate that there was no explosion, and the pressure increased by only 14 bar because of melt quenching. The thermocouples in the simulated lower head experienced temperature excursions, which indicates that not all the melt was immediately quenched. This is consistent with the benign pressure increase. Data reduction and analysis are under way. The test results were reviewed, and the test matrix was completed in January 1992. Another scoping test is scheduled for June 1992, and the base case test is scheduled for November 1992.

3.2 Reactor Containment Safety

3.2.1 Statement of Problem

Core melt accidents have the potential to produce high pressures and temperatures that might cause containment failure. 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 imperative.

3.2.2 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 fail the containment: high-pressure ejection from the reactor vessel of finely divided particles of molten core debris; generation of noncondensible 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) 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; (2) the transport, mixing, and combustion of hydrogen in the containment, including the potential for detonation; (3) the interaction of molten core debris with structural concrete, including the ablation of concrete structures, heat transfer to structures in the containment, the generation of flammable and noncondensible gases, and fission products containing aerosols; (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 5.2, "Advanced Reactors").

3.2.3 Research Accomplishments FY 1991

3.2.3.1 Direct Containment Heating

In certain reactor accidents, degradation of the reactor core can take place while the RCS remains pressurized. Left unmitigated, a molten core will slump and collect at the bottom of the reactor vessel. If a breach occurs, the core melt will be ejected under pressure. If the material should be ejected from the reactor cavity into surrounding containment volumes as fine particles, thermal energy would be quickly transferred to the containment atmosphere. The metallic components of the ejected core debris could further oxidize in air or in steam, and that could generate a large quantity of chemical energy and further pressurize the containment. This process is called direct containment heating (DCH).

To help develop a data base to estimate the risk associated with high-pressure core melt accidents, the following two activities were completed in FY 1991: (1) the development of a system level scaling methodology to ensure the applicability of future integral effects tests at different scales to a full- size containment, and (2) the modification of the 1/10th scale facility at the Sandia National Laboratories (SNL) and the 1/40th scale facility at the Argonne National Laboratory (ANL) to conduct companion tests. The development of a severe accident scaling methodology (SASM) was undertaken to guide the development of experimental programs and analytical methods. To assist in the formulation of a SASM, a technical program group was formed that was composed of experts from a cross section of the research community. This group's activities to develop a SASM and to apply the methodology to DCH resulted in the issuance of NUREG/CR-5809 (Draft Report for Comment), "An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution," published in November 1991. In September 1991, the first integral test of DCH phenomena was conducted in a 1/10th scale representation of the Zion nuclear power plant. The Zion design was chosen for simulation since its geometric configuration is representative of a group of operating PWRs. In FY 1992, most of the integral effects tests at SNL (1/10th scale and 1/6th scale) and ANL (1/40th scale) will be conducted. Appropriate analysis will also be performed to assess the extrapolation of computer code models for DCH phenomena to a full-scale containment.

3.2.3.2 Hydrogen Combustion

Hydrogen combustion research seeks to assess the possible threat to containment and safety-related equipment. It is necessary to understand how hydrogen is transported and mixed within containment and to determine the likelihood of various modes of combustion, i.e., deflagrations, diffusion flames, accelerated flames, transition from deflagration to detonations (DDT), and detonations. The NRC has sponsored over the past several years considerable research in the area of hydrogen combustion and transport that has led to the development and confirmation of our regulatory approach. The current research is focused on expanding the experimental data base to include hydrogen combustion and mixing under a wider range of conditions that might exist during severe accidents. Available theory and analyses support regulatory positions for areas in which data are unavailable, principally combustion of gas mixtures at pre-existing high temperatures; the research under way will provide the corroborating information. During the reporting period, several hydrogen research programs were initiated.

The largest program initiated was under a joint agreement between the NRC and the Ministry of International Trade and Industry (MITI) of Japan (managed for Japan by the Nuclear Power Engineering Center). This program is to address high-temperature high-speed hydrogen combustion modes, i.e., detonations and DDT. Two combustion vessels will be used for this research program at the Brookhaven National Laboratory. The design of these two vessels is under way. Another joint agreement between the NRC and Germany involves a program to evaluate data from the KfK/PHDR hydrogen behavior experiments. The University of Maryland is assisting in the evaluation of the data from these experiments.

Also, a hydrogen research program was initiated at the Rensselaer Polytechnic Institute to investigate diffusion flame behavior. The HMS code, a three-dimensional finite difference analysis tool developed at the Los Alamos National Laboratory, is also used to provide more detailed hydrogen transport and mixing calculations. The assessment and documentation of the HMS code has continued.

3.2.3.3 Core-Concrete Interactions/Core Debris Coolability

In order to better characterize the long-term consequences of core melt accidents, the NRC has sponsored research to investigate those phenomena that may occur subsequent to the failure of the reactor pressure vessel at lower pressures, i.e., relocation of the molten core to the concrete basemat and ablation of the concrete and generation of refractory radionuclides. The NRC program has included integral and separate-effect testing as well as development and assessment of an analytical methodology, the CORCON computer code, which is applied to reactor analysis. More recently the NRC research has focused on the issue of whether water addition, i.e., covering the molten core material with an overlying water pool, can quench or cool the debris and thereby terminate concrete ablation and establish stable containment conditions.

In FY 1991, the NRC analytical research program resulted in the interim release of the CORCON MOD3 code. This code release is the culmination of significant development activities to improve the capability of the code to calculate thermo-chemical reactions between molten core debris and the concrete basemat of a reactor containment. In addition to CORCON modeling improvements, the VANESA model of aerosol generation and radionuclide release during core-concrete interactions has been fully integrated into CORCON. CORCON MOD3 application and assessment included analysis of core-concrete interactions associated with accidents in a BWR Mark I reactor that were used to independently assess the BWR Mark I liner early failure evaluation. The code was also successfully used to conduct calculations for the International Standard Problem (ISP 30) of the German Beta Test V5.1, which involved molten steel and zirconium reactions with concrete, and the ACE L6 experiment that was performed at the ANL and involved a prototypic oxidic melt (UO_2, ZrO_2) interacting with concrete.

Experimental research on debris coolability was started through two programs: the WETCOR program conducted at SNL and the MACE tests conducted under the aegis of the Advanced Containment Experiments (ACE) program at ANL. These programs are intended to provide the data concerning the coolability of core debris, under severe accident conditions, for advanced reactors. The ACE program is an internationally sponsored project in which the NRC participates along with EPRI and the DOE. The project includes four phases: (1) phase A deals with large-scale filtration tests, using filter designs from different countries; (2) phase B involves experiments on the physical and chemical behavior of iodine in a containment that includes the presence of hygroscopic aerosols, steam, and water pools; (3) phase C deals with molten core-concrete interaction; and (4) phase D (MACE) deals with melt cooling issues, seeking a determination as to what debris configurations (power level and depth) can be cooled by an overlying pool of water. Phase C testing was completed in FY 1991 and seven integral core-concrete interaction tests that were conducted addressed the effects of various compositions on typical concrete substrates. Analysis and documentation of the phase C tests will continue in FY 1992. The first MACE experiment has been performed under phase D; two large-scale integral tests are scheduled to be completed in FY 1992.

Under the WETCOR program, the first test, WETCOR-1, was conducted in September 1991. This test, involving the induction heating of oxidic simulant material to represent decay heating of molten core material, was performed to investigate cooling of molten debris and termination of concrete ablation by an overlying water pool. Test results indicate the potential for forming a crust on the top surface of the debris that severely limits heat transfer to the water, thereby diminishing the likelihood of establishing long-term coolability. Additional tests using prototypic oxidic material are planned for FY 1992. These tests will focus on morphological behavior of crust formation and will provide a better understanding of the debris coolability issue.

Under the programmatic element of core-concrete interactions/core debris coolability, the NRC has also sponsored research to address resolution of a severe accident issue generic to BWR Mark I reactor containment shell meltthrough, so designated since it addresses the phenomena associated with the spreading of molten core debris across the floor of the containment to the containment shell. If the debris is sufficiently hot upon reaching the shell, the resultant thermal attack could fail the containment boundary. The conclusion from an extensive evaluation of this issue is that the presence of water on top of the core debris would prevent containment shell failure.

Over the past year, the NRC has worked to address the comments provided by the peer review group formed to evaluate the methodology and conclusions of draft NUREG/CR-5423, "The Probability of Liner Failure in a Mark I Containment." The outcome of the peer review process was the establishment of four working groups of experts composed of persons from industry, national laboratories, and academia to address, in detail, four corresponding areas (shell failure criteria, corium spreading, core-concrete interactions, and melt conditions) that warranted additional analytical confirmation. While this additional analysis is not expected to alter the basic conclusions of the report, which was published in final form in July 1991, the staff plans to issue an update documenting the completion and results of these confirmatory activities by December 1992. A final peer review will then be conducted.

3.2.3.4 Integrated Codes and Applications

In order to support the evaluation of nuclear plant responses to postulated severe accident scenarios as well as analyze the results of phenomenological and system tests, the NRC sponsors research to develop, verify, and validate integrated/system computer codes.

The CONTAIN code is a detailed mechanistic code for the integrated analysis of containment phenomena. It has been developed under NRC sponsorship to provide the capability to predict the physical, chemical, and radiological conditions inside a nuclear reactor containment in the event of a severe accident and fission product releases to the environment in the event of containment failure.

Several research programs are ongoing in an attempt to quantify the effects of DCH. This program is closely coupled to the DCH experimental programs at SNL and ANL. Limited code modifications were made to

CONTAIN in order to capture the dominant DCH processes. The CONTAIN code will be assessed against the experimental data, and ultimately full-scale power plants will be analyzed to determine the consequences for postulated DCH scenarios. The NRC continues to update and upgrade code documentation; NUREG/CR-5715, the CONTAIN Code Reference Manual, was published in July 1991. The QA procedures applied to CONTAIN development were documented in NUREG/CR-5518, published in January 1991. The code will undergo peer review in FY 1992.

The MELCOR code was developed to model the progression of severe accidents in LWRs. The code treats the entire spectrum of severe accident phenomena, including reactor coolant system and containment thermal-hydraulic response, core heatup, degradation and relocation, and fission product release and transport for both BWRs and PWRs. During FY 1991, a comprehensive peer review was completed, which had as its objective to assess the adequacy of the code for the NRC. The peer reviewers found the code generally acceptable but had significant comments and recommended that improvements be made in several areas. These included improvements in numerical stability, revising and improving certain phenomenological models, adding some new models, performing more comprehensive comparisons of the code against data, and improving documentation. These recommendations are being dealt with in the FY 1992 continuing code development efforts.

MELCOR code development in FY 1991 produced new models for MELCOR for handling the following situations: (1) in-vessel natural circulation, which previously could not be done because numerical modeling did not allow downflow in the reactor core; (2) integrated containment response to direct containment heating (DCH) resulting from the expulsion of molten core materials through a failure in the reactor pressure vessel; and (3) containment performance with the ice condenser safety feature, specifically to handle the thermal and hydraulic effects and the fission product removal phenomena. The addition of these models into MELCOR addresses both long-range plans of NRC on the code's development and recommendations for new models from the peer reviewers.

MELCOR benchmarking and assessment activities in FY 1991 included the application of MELCOR to the calculation of accident sequences in a Combustion Engineering designed plant (Calvert Cliffs) and a Babcock and Wilcox designed plant (Oconee). Reports were written describing the results of the calculations. Code calculations were also made and compared with data from the NRU Full-Length High-Temperature-4 (FLHT-4) experiment, which showed reasonable agreement between the data and MELCOR results for the in-core heat transfer and fuel rod temperatures. Extensive code assessment calculations were also provided with data from the LWR Aerosol Containment Experiment (LACE) LA-4 test, the FLECHT-SEASET natural circulation tests, and the German HDR V44 testing. Reports were prepared showing the agreement of MELCOR results with data and the effects of parametric variations in MELCOR input choices on the comparisons of output with the data. In certain cases, the results of MELCOR calculations were also compared with results from other codes that had been run to simulate the experiments.

The SCDAP/RELAP5 code has been developed to provide detailed modeling of the reactor core and reactor coolant system during severe accidents. In FY 1991, reports were issued describing the quality assurance program for SCDAP/RELAP5 and the results of independent review of the natural circulation calculations of the code. A peer review for SCDAP/RELAP5 was started in FY 1991 and will be completed in FY 1992.

3.3 Containment Structural Integrity

3.3.1 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.

3.3.2 Program Strategy

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

1. Failure of the shell, either the containment shell itself (in the case of steel containments) or the liner (in the case of concrete containments);

2. Leakage at large penetrations as a result of the inelastic deformations and/or degradation of seals and gaskets;

3. Leakage at electrical penetrations due to degradation of materials under the high temperatures associated with accident scenarios; and

4. 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.

3.3.3 Research Accomplishments in FY 1991

3.3.3.1 Structural Tests

The major effort in this program for the next few years will be a cooperative one with the 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 prestressed concrete designs.

A reinforced concrete model was chosen for the NRC-sponsored test at SNL since it would provide a greater challenge for analytical models. However, there are two main reasons for performing an additional prestressed containment model test:

1. Prestressed designs are the most common concrete PWR containment type in the United States. There are 41 prestressed containments compared to 20 reinforced containments.

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

A test to failure of a model of a steel BWR containment vessel will also be included in the cooperative research program. The vessel would be fabricated in Japan and shipped to SNL in Albuquerque, New Mexico. This test would complement the test to failure of a steel containment model performed by SNL in 1984 under NRC sponsorship. That model was cylindrical in cross section and was representative of PWR ice condenser and BWR Mark III containments. The proposed Japanese model would include the "knuckle regions" that are present in BWR designs in the United States. It is currently presumed that state-of-the-art analytical methods can be relied upon to provide adequate predictions for the response of those designs to severe accident conditions. However, there are no experimental data against which the predictive methods can be checked. The proposed model test would fill that gap in the data base.

3.3.3.2 Equipment Hatch Tests

The final report for the pressure-unseating equipment hatch test program was completed in FY 1991. This report, in addition to presenting the results of the test program, presents analytical procedures to estimate the pressure and temperature conditions at which leakage can be expected by unseating the hatch. The test program has therefore provided information useful in accident scenario analyses and in future hatch designs.

3.4 Reactor Accident Risk Analysis

3.4.1 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.

3.4 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.

3.4.3 Research Accomplishments in FY 1991

3.4.3.1 Review of PRAs

PRA is used by the NRC staff to support the resolution of a wide spectrum of regulatory issues. For licensed plants, PRAs are sometimes voluntarily submitted by licensees to support their specific proposed means for resolving such issues. For advanced plants of the future, applicants are required to perform and submit PRAs as part of their overall license applications. Reviews performed in FY 1991 included the following:

South Texas (Texas). This PRA was a voluntary submittal by the licensee, who plans to use the document as a reference in future technical discussions on regulatory issues. The review was completed in FY 1991.

Diablo Canyon (California). In order to comply with a license condition, the licensee for Diablo Canyon has developed a long-term seismic program. As part of this program, the licensee has performed a PRA for seismic as well as other potential accident initiators. The review was completed near the end of FY 1991.

GE Advanced BWR. A PRA has been submitted as part of the licensing application for this advanced BWR. In May 1991, a draft safety evaluation report was transmitted to NRR. This was subsequently transmitted to General Electric.

3.4.3.2 Completion and Review of Reactor Risk Reference Document

In February 1987, the NRC issued the draft version of the "Reactor Risk Reference Document" (NUREG-1150), as well as a series of supporting contractor reports, for public comment. The draft report assessed the risks from possible core damage accidents in five U.S. nuclear power plants—Surry (Virginia), Zion (Illinois), Sequoyah (Tennessee), Peach Bottom (Pennsylvania), and Grand Gulf (Mississippi). The report discussed the implications of the five analyses on regulatory issues such as implementation of the Commission's Safety Goal and Severe Accident Policy Statements. Two NRC-funded reviews of the draft report were obtained and published as NUREG/CR-5000 and NUREG/CR-5113. In addition, the American Nuclear Society sponsored and published a review of the draft report.

The NRC staff and supporting contractors updated the five risk analyses. The updates, which were quite extensive, were intended to reflect comments received, to reflect the present plant design and operating characteristics, to improve the methods used, and to incorporate new experimental data on severe accidents resulting from the research programs of NRC and others.

The completed new version of NUREG-1150 was published as a second draft for peer review in June 1989. A peer review panel, organized under the Federal Advisory Committee Act, completed its formal review of the document and provided generally positive findings. The final version of the report (NUREG-1150) was issued in December 1990 (Vols. 1 and 2) and January 1991 (Vol. 3).

3.4.3.3 Risk Model Development and Application

Probabilistic risk analysis has become an important tool in the NRC's assessments of safety issues in the design and operation of commercial nuclear power plants. To use this tool well, it is necessary to develop and use state-of-technology methods for performing and reviewing PRA and to develop, maintain, and provide quality assurance for such methods.

Version 1.5 of the MACCS code—a computer code that estimates the post-accident release of radioactive material to the environment and health and economic consequences to the public—was completed and made available to the public in FY 1990. Final benchmarking of the code with international standard problems is under way and is expected to be completed in early FY 1993.

In regulatory decisionmaking, it is necessary to ask what impact a proposed modification to plant hardware or procedures will have in terms of risk. Generally, the most appropriate way in which to answer such a question is to examine existing PRAs, change the affected parameters, perform the analysis again, and observe the resulting change in core damage frequency and public risk. Such calculations are currently employed in setting priorities in the use of agency resources and for regulatory analyses of generic safety issues. Other uses, such as targeting inspection activities, are also emerging.

The System Analysis and Risk Assessment (SARA) system and the Integrated Reliability and Risk Assessment System (IRRAS) were conceived to address the need described above and also to provide the NRC with reliability data that are currently available only on large mainframe computers. The development of high-performance microcomputers has provided greater capacities to interact with extensive data bases for a large number of users. During FY 1991, versions of these codes were used by NRC contractors to perform risk studies of accidents initiated during low-power and shutdown operations (described below), as well as by the staff to assess, for example, the sensitivity of NUREG-1150 results to variations in human error rates and motor-operated valve failure rates and the benefit potentially achievable by the resolution of certain generic issues.

3.4.3.4 EPRI Requirements Document

In support of the advanced reactor design certification process, the Electric Power Research Institute (EPRI) has developed a set of requirements to guide the design of such reactors. One part of this guidance relates to the performance and use of PRA methodologies. A review of this guidance was made with a draft evaluation transmitted to the Office of Nuclear Reactor Regulation (NRR) in August 1991. This evaluation was reviewed by NRR and transmitted to EPRI in October 1991.

3.4.3.5 Analysis of Low-Power and Shutdown Accident Risks

Since 1989 the staff has had under way a study of the risks associated with accidents initiated during low-power and shutdown plant operating conditions. The first phase of this analysis was completed, providing a rough categorization of potential accidents by their frequency and consequences. This information is being used by NRR in their analysis of the need for additional regulation under these operating conditions.

4 CONFIRMING SAFETY OF NUCLEAR WASTE DISPOSAL

The NRC's waste management research seeks to (1) develop and verify methods for predicting and assessing the performance of waste disposal facilities; (2) evaluate the data bases used in such performance assessments; (3) provide technical support to the licensing staff in their interactions with the Department of Energy (DOE) and the States; and (4) develop regulatory standards to support the licensing of facilities and methods for the disposal and management of high-level and low-level radioactive wastes.

4.1 High-Level Waste

4.1.1 Statement of Problem

The high-level waste (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.

4.1.2 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 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.

4.1.3 Research Accomplishments in FY 1991

4.1.3.1 Geohydrology

Since transport by ground water is the most likely path by which most radionuclides from disposed waste might reach the environment, the NRC is actively studying the movement of ground water in the unsaturated fractured media currently under consideration by DOE. An experimental site has been located in unsaturated fractured tuff (the same rock type as the repository host rock) in Arizona, where field and laboratory testing is being conducted by the University of Arizona. The objective of the field study is to determine what types of measurements are needed to characterize the hydrology of fractured rock and how measurement data should be analyzed to model ground-water flow. This work currently entails assessing techniques and methods for measuring rock properties in place and assessing infiltration and movement of water in rock formations. The project is using numerical calculations of flow and transport to assess the importance of site features, appropriateness of fracture models, and theories and measurements of flow-controlling properties and processes.

Investigators at the Center for Nuclear Waste Regulatory Analyses (CNWRA) in San Antonio, Texas, are examining methods to perform stochastic hydrologic analyses for repository-scale systems. And the validity of the models used to describe ground-water flow and radionuclide transport is being appraised in an international project called INTRAVAL. The NRC staff and research contractors from the CNWRA, the University of Arizona, Sandia National Laboratories, Massachusetts Institute of Technology, and the Battelle Pacific Northwest Laboratory are participating in the 13-country validation effort.

Cooperative experiments and data analyses being done under a cooperative agreement between NAGRA (Switzerland) and the NRC, negotiated during FY 1987, continue to augment the field testing program cited above.

4.1.3.2 Stability of Underground Openings

When specifying suitable site conditions for a repository, 10 CFR Part 60 specifically requires consideration of natural phenomena and site conditions that could adversely affect achievement of the prescribed performance objectives. An important phenomenon that could affect both the short- and long-term performance of a repository is ground motion resulting from seismic activity. Similarly, ground motion caused by underground nuclear explosions at the Nevada Test Site needs to be evaluated. Ground motion from either source could cause rock displacement, rise in water tables, etc., which could violate established repository performance objectives.

To investigate the effects of seismicity on the underground openings for an HLW repository, the NRC is sponsoring research at the CNWRA. Initial results from the study indicate that structural damage at depth can occur, and a field site in an existing mine is being instrumented to study these effects.

4.1.3.3 Sealing of Boreholes and Shafts in Tuff

The isolation of nuclear waste in deep geological repositories may require that penetrations in the geological host rock barrier—such as shafts, drifts, ramps, and boreholes in the vicinity of the repository—be sealed to prevent the creation of potential pathways for the movement of radionuclides to the accessible environment.

To evaluate the performance of seals in the unsaturated HLW tuff environment, the NRC has supported research studies at the University of Arizona. Both laboratory and field tests of seals were conducted for a variety of potential seal materials. Characterization testing confirmed that tuff is an extremely non-uniform rock with highly variable properties and extremely low hydraulic conductivity.

4.1.3.4 Geochemistry

Knowledge and application of geochemistry is important to an understanding of many aspects of repository performance, including problems related to waste package corrosion, radionuclide release and transport, and alteration of ground-water flow paths. The NRC has an active research program in geochemistry as it affects the management of HLW. In 1991, the chemistry of the natural waters at Yucca Mountain was evaluated at the CNWRA by geochemical models in the context of the variable compositions of minerals that rapidly react with ground water. The evolution of water as it is heated while moving toward the waste packages was modeled and used as input for waste package performance testing. The state of the art in measuring and modeling physical-chemical processes that retard radionuclide transport was reviewed by the CNWRA. The NRC is participating in an international field study at the Koongarra ore body in northern Australia, observing the actual movement of radionuclides. This study is providing a basis for validating performance assessment models to be used in HLW repository licensing. The fourth year of the study has seen the completion of data collection and the undertaking of indepth hydrologic and geochemical modeling. The results of simple transport models have been compared with site data, and more sophisticated transport modeling is continuing. A study at Johns Hopkins University to develop a coupled thermo-hydrogeochemical transport model has successfully completed model development, and research has been started to test it against data from natural systems such as the Koongarra ore body.

4.1.3.5 Rulemaking

A proposed guide published for public comment in November 1990 provides the information needed by the NRC to review DOE's license application for the HLW repository. The NRC continued to closely monitor EPA's development of a revised high-level radioactive waste standard. The NRC staff commented on EPA working drafts of the standard. In July 1990, a petition was received from the States of Washington and Oregon asking the NRC to undertake rulemaking regarding the classification of some radioactive wastes at DOE facilities at Hanford, Washington. During FY 1991, the NRC completed internal review of this petition, and formal resolution is expected early in FY 1992.

4.2 Low-Level Waste

4.2.1 Statement of Problem

Disposal of low-level waste (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.

4.2.2 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, LLW source term modeling, hydrologic flow and contaminant transport, and performance assessment. The NRC's LLW research staff also prepares rulemakings that affect LLW disposal. The NRC LLW research program is described in NUREG-1380, which was published in November 1989. NUREG-1380 identifies issues, regulatory needs, a strategy, and a schedule for resolving them. It was widely circulated for comment to potentially interested parties, including State regulatory agencies. NRC-funded LLW research is useful not only to the NRC licensing staff but also to the States regulating LLW disposal. In order to make their research results available to the States, NRC research contractors, besides publishing their work, gave presentations at meetings well attended by State representatives such as "Waste Management '91," the Annual DOE LLW Management Conference, and the first annual LLW research review meeting organized by RES.

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 HLW 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.

4.2.3 Research Accomplishments in FY 1991

4.2.3.1 Engineered Enhancements and Alternatives to Shallow Land Burial

There is great interest on the part of States and State compacts in alternatives to shallow land burial for the disposal of low-level radioactive waste. Concrete is expected to play an important role in engineered alternatives to shallow land burial. In 1991, the National Institute of Standards and Technology (NIST) continued investigating, for the NRC, the durability of concrete in engineered alternatives to shallow land burial, while the Idaho National Engineering Laboratory continued to develop a mathematical model describing concrete performance. In NUREG/CR-4269, NIST has reported on modeling transport processes in concrete and the diffusion of chloride ions in concrete.

4.2.3.2 LLW Waste Forms

Low-level radioactive waste collected from operating nuclear power stations and solidified in cement is being tested at the Idaho National Engineering Laboratory. The studies are aimed at ensuring that radionuclide and chemical leaching characteristics, as well as the compressive strength of the solidified waste, are consistent with NRC technical positions and requirements of 10 CFR Part 61 for waste form stability. Under examination is the stability of decontamination waste obtained from nuclear reactors using commercial decontamination processes and solidified in cement. Field studies are being conducted at the Oak Ridge and Argonne National Laboratories to determine whether radionuclides are released from solidified waste forms under certain environmental conditions. A report was issued on the release of radionuclide and chelating agents from cement-solidified LLW (NUREG/CR-5601).

4.2.3.3 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 units at the Maryland Agricultural Experiment Station in Beltsville, Maryland. (Results are reported in NUREG/CR-4918, Volume 3.) 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 two experimental cells. A second cover consists of a resistive layer barrier (compacted clay) over a conductive layer barrier. This second system has functioned perfectly since its completion in January 1990. However, its long-term performance needs to be assessed.

4.2.3.4 Performance Assessment

Research is continuing on a performance assessment methodology. Emphasis is being given to engineered enhancements to shallow land burial. The Sandia National Laboratories are assessing the validity of performance assessment models, and the Pacific Northwest Laboratory (PNL) is exploring mathematical models for radionuclide transport through concrete. The Massachusetts Institute of Technology (MIT) has been investigating the use of stochastic methods for dealing with large-scale non-uniformity of site hydrologic characteristics. The University of Arizona and New Mexico State University are working cooperatively with MIT by providing a field test at Las Cruces, New Mexico, of MIT's theoretical work.

4.2.3.5 LLW Source Term Modeling

Development of the LLW source term code, BLT (breech, leach, transport), continued during FY 1991. The Brookhaven National Laboratory has refined and expanded the transport submodel to consider geochemistry and gas transport. To provide confidence in the model predictions, the BLT code continues to be benchmarked against lysimeter experiments of saltstone waste forms at the Savannah River Laboratory and cement, bitumen, and polymer waste forms at PNL. Results of sensitivity analyses continue to be used to assess radionuclide releases as a function of key parameters. This work represents a first attempt at quantification of source terms for use in performance assessment.

4.2.3.6 Hydrology and Contaminant Transport

The NRC continues to sponsor field tests of flow and transport in unsaturated soils at a New Mexico State University field site near Las Cruces, New Mexico. The program—which includes NRC-sponsored research by PNL, the University of Arizona, and MIT—is intended to confirm the reliability of unsaturated flow and transport models of LLW disposal facilities. This work is a part of the INTRAVAL international study that deals with model validation of ground-water flow and transport models.

4.2.3.7 Rulemaking

Final amendments to 10 CFR Part 40 that provide licensing for the custody and long-term care of uranium and thorium mill tailings disposal sites were published in the *Federal Register* in October 1990 (55 FR 45591).

The resolution of a Petition for Rulemaking (PRM-61-1) from the North Carolina Chapter of the Sierra Club was completed. The petitioner requested the Commission to adopt a regulation to permit the design and construction of a zero-release low-level radioactive waste disposal facility in a saturated zone. The petitioner stated that the regulation was necessary in order for the General Assembly of North Carolina to consider a waiver of a North Carolina statute that requires that the bottom of a low-level waste facility be at least 7 feet above the season high water table. A Denial of Petition was published in the *Federal Register* in July 1991 (56 FR 34035).

5 RESOLVING SAFETY ISSUES AND DEVELOPING REGULATIONS

This program is directed toward the development of the technical basis and related regulatory requirements needed to protect the health and safety of the public from radiation and from the risk resulting from the generation of electricity and the manufacture, use, transport, and storage of nuclear fuel and other radioactive materials. This program also supports efforts to ensure that proposed Commission regulations are adequate and that they are developed in an efficient and timely manner.

5.1 Generic Safety Issues

5.1.1 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.

5.1.2 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.

5.1.3 Research Accomplishments in FY 1991

The NRC has continued to use the 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 the list has been updated semi-annually (supplements in June and December). The list of issues includes TMI Action Plan (NUREG-0660) items. The results of the NRC's continuing effort to identify significant unresolved GSIs will be included in future supplements to NUREG-0933. During FY 1991, the NRC identified 29 new GSIs, established priorities for 12 issues (Table 5.1), and resolved 4 GSIs (Table 5.2). At the end of FY 1991, approximately 30 proposed GSIs were awaiting prioritization.

Table 5.1
Issues Prioritized in FY 1991

Number	Title	Priority
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM
38	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	DROP
72	Control Rod Drive Guide Tube Support Pin Failures	DROP
73	Detached Thermal Sleeves	NEARLY RESOLVED
100	Once-Through Steam Generator Level	DROP
120	On-Line Testability of Protection Systems	MEDIUM
143	Availability of Chilled Water Systems and Room Cooling	HIGH
150	Overpressurization of Containment Penetrations	DROP
151	Reliability of Anticipated Transient Without Scram Recirculation Pump Trip in BWRs	MEDIUM
153	Loss of Essential Service Water in LWRs	HIGH
A-19	Digital Computer Protection System	LICENSING ISSUE
B-22	LWR Fuel	DROP

Table 5.2
Generic Safety Issues Resolved in FY 1991

Number	Title
128	Electrical Power Reliability
130	Essential Service Water System Failures at Multiplant Sites
135	Steam Generator and Steam Line Overflow
II.J.4.1	Revise Deficiency Report Requirements

5.2 Advanced Reactors

5.2.1 Statement of Problem

The Commission has issued a policy statement on the regulation of advanced nuclear power plants (51 FR 24643), which states that the NRC will review and comment on new design concepts with special emphasis on vendor test programs for confirming the performance of novel safety systems. It also encourages early interaction with applicants. As part of this program, the NRC will develop, review, and implement advanced reactor safety and policy issues in the ongoing NRC review of advanced reactor concepts (NUREG-1226). Indepth independent analysis will be performed as necessary to verify that advanced reactor designs have the potential for enhanced margins of safety and that appropriate means will be used to accomplish their safety functions.

5.2.2 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. Eight different designs are being considered for certification under the newly formulated 10 CFR Part 52. These designs fall into two groups. The first group consists of four evolutionary and passive advanced light-water reactor (LWR) types: the Advanced Boiling Water Reactor (ABWR), the C-E System 80+, the AP600 Advanced Pressurized Water Reactor, and the Simplified Boiling Water Reactor (SBWR). The second group consists of four future advanced reactor types: the Process Inherent Ultimate Safety (PIUS) reactor, the Canada Deuterium Uranium heavy-water-cooled (CANDU 3) 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 passive AP600 and SBWR reactor designs and making appropriate modifications of existing regulat[illegible]ments to accommodate the unique features of [illegible]actors. A relatively small amount of research is [illegible]conducted on the evolutionary LWRs, and small research efforts on the nonconventional reactor types have been initiated and will be maintained.

5.2.3 Research Accomplishments in FY 1991

During this period, a review of the preliminary reactor system design of the AP600 and SBWR and the related vendor test program was carried out to identify any potential safety concerns and ensure that the vendors' test and NRC research programs were properly focused on these concerns. In this review, several issues that warrant confirmatory research were identified. Prominent among these were the effectiveness and reliability of natural circulation under gravity flow conditions as a means of removing decay heat, the performance of the boron injection system used to mitigate certain anticipated reactor transients, the effectiveness of proposed containment cooling methods, qualification of advanced digital instrumentation and control systems, containment structural performance during severe accidents, reliability of modular construction, and the adequacy of existing computer codes to accurately predict safety system performance under severe transients and accident conditions. New research efforts were initiated for PIUS and CANDU 3, and ongoing work was redirected for the ALMR and MHTGR designs. Early emphasis is being placed on identifying important accident sequences and safety systems. Systematic screening is being done with risk-assessment techniques.

Future work will be carried out by NRC to provide the independent capability needed to audit the submittals of license applicants for the AP600 and SBWR reactors. This work will include computer code modification and verification and the execution of whatever experiments are needed to confirm safety system performance or to develop required assessment methods. The full scope of these programs will be determined after additional study.

5.3 Developing and Improving Regulations

5.3.1 Statement of Problem

RES has the primary responsibility to manage, coordinate reviews of, and control all NRC 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 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.

5.3.2 Program Strategy

The purpose of the NRC nuclear regulatory program is to ensure that nuclear 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: (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.

5.3.3 Research Accomplishments in FY 1991

5.3.3.1 Develop or Modify Regulations

A final rule, 10 CFR Part 71, on modifying NRC's transportation regulations is awaiting publication until the Department of Transportation is prepared to issue a companion rule. Public comments on the proposed rulemaking have been evaluated, and the final rule is being developed. We expect the final rule to be published in FY 1992. This rule proposes limitations on the shipment of low-specific-activity materials and maximizes compatibility between NRC and International Atomic Energy Agency (IAEA) regulations.

A final rule, 10 CFR 50.73, on access authorization at nuclear power plants and an accompanying regulatory guide were published in the *Federal Register* in April 1991 (56 FR 18997). The rule requires a nuclear power reactor licensee to have an access authorization program in its site physical security plan. This program would provide increased assurance that persons granted unescorted access to protected and vital areas are trustworthy and do not pose a threat to commit radiological sabotage.

A proposed rulemaking, 10 CFR Part 74, on the material control and accounting requirements for uranium enrichment plants and an associated regulatory guide were published for public comment in the *Federal Register* in December 1990 (55 FR 51726). The rulemaking is following an accelerated schedule because Louisiana Energy Services has filed a license application for the construction and operation of a gas centrifuge plant that would produce low-enriched uranium for the commercial market. The rule will facilitate the licensing of such a facility. We expect the final rulemaking will be published in the *Federal Register* early in FY 1992.

In a program initiated in 1985 and continued through 1991, the NRC staff undertook to evaluate existing regulatory requirements in terms of their risk effectiveness and to eliminate or modify requirements with only a marginal safety importance. A three-volume research report (NUREG/CR-4330) provided detailed technical assessments of requirements associated with a limited number of topics. In a follow-on effort, a set of regulatory requirements were identified as candidates for possible elimination or modification. Work was started in 1990 to evaluate the safety significance of these candidate regulations to identify those of marginal safety significance and for which modification or elimination can be proposed. Final recommendations were forwarded to the Commission in July 1991. Implementation of final recommendations will begin in FY 1992.

A final rule amending the regulations in 10 CFR Parts 20, 30, 40, and 70 to revise licensee reporting requirements regarding notifications of incidents related to radiation safety was published in the *Federal Register* in August 1991 (56 FR 40757). This rule will ensure that significant occurrences at facilities operated by material licensees are promptly reported to the NRC. The Commission will be able to determine whether a licensee has taken the actions necessary to protect public health and safety and whether generic safety concerns that may require prompt NRC action are identified.

A final rule amending the regulations in 10 CFR Part 50 to require the licensee to implement the NRC-approved Emergency Response Data System (ERDS) at all nuclear power plants was published in the *Federal Register* in August 1991 (56 FR 40178). (The proposed rule had been published in the *Federal Register* for public comment in October 1990 (55 FR 41695).) The rule would supplement the voice transmission over the existing Emergency Notification System and require a direct electronic data link between the licensee's computer and the NRC's Operation Center to be activated by the licensee during an alert of higher emergency condition to transmit timely and accurate updates of critical information on plant conditions. This measure would allow the NRC to perform its primary role during an emergency at a licensed nuclear power facility, which is one of monitoring the licensee to ensure that appropriate recommendations are made with respect to necessary offsite actions to protect public health and safety.

A final rule was published in the *Federal Register* in July 1991 (56 FR 34101) to amend the 10 CFR Part 35 regulations that apply to the medical use of by-product material. The amendments require medical-use licensees to implement quality management (QM) programs and revise misadministration reporting requirements. Implementation of the new performance-based requirements is supported by the issuance of a regulatory guide that includes specific guidance for QM programs and an approach acceptable to the NRC for meeting the requirements of the final rule. The rule provides a high confidence that byproduct material and radiation from byproduct material will be administered as directed by the authorized user physician. The feasibility of this approach was evaluated during a pilot program involving 70 medical-use licensees and subsequent discussion with professional associations and Agreement States.

A proposed rule, 10 CFR Parts 31 and 32, on requirements for the possession of industrial devices containing byproduct material was submitted to the Commission for approval in August 1991. This rule would require general licensees to provide the NRC with specific information about radioactive material used under the provisions that establish general domestic licenses for byproduct material. The proposed action would reduce the likelihood of unnecessary exposures from radioactive materials by ensuring that generally licensed devices are properly accounted for and disposed of. We expect the proposed rulemaking will be published in the *Federal Register* for public comment early in 1992.

A proposed rulemaking, Appendix H to 10 CFR Part 73, on day-firing qualifications and physical fitness programs for security personnel at category I fuel cycle facilities was submitted to the Commission for approval in September 1991. The proposed rule would amend the Commission's regulations to include day-firing qualification courses in each type of required weapon as well as a standardized physical fitness training course and fitness standards for security personnel. Standardization of day-firing courses to be consistent with those established for night firing is needed to provide a uniform, enforceable program. We expect the proposed rulemaking will be published in the *Federal Register* for public comment early in FY 1992.

The Commission is considering a proposed rulemaking, 10 CFR Part 50, on training and qualification of nuclear power plant personnel. The proposed rule would amend the Commission's regulations to require each applicant and holder of a license to operate a nuclear power plant to establish and use a systems approach in developing training programs for management, supervisory, professional, and technical workers who have an impact on the health and safety of the public. Licensees and applicants would also be required to establish qualification requirements for these personnel. The objectives of the proposed rule are to codify existing industry practices related to personnel training and qualification and to meet the directives contained in Section 306 of the Nuclear Waste Policy Act of 1982 (Pub. L. 97-425). We expect the proposed rulemaking will be published in the *Federal Register* for public comment early in 1992.

The Commission issued a denial of a petition for rulemaking (PRM-50-50) from Charles Young for publication in the *Federal Register* in January 1991 (56 FR 1749). The petitioner requested the Commission to amend its regulations to prevent nuclear power plant operators from deviating from license conditions or technical specifications during an emergency. The petitioner believes nuclear power plants should be operated in accordance with the operating license and appropriate technical specifications and that requiring a senior operator to follow the technical specifications during an emergency enhances plant safety.

A final rule was published in the *Federal Register* in January 1991 amending the 10 CFR Parts 20 and 50 regulations that apply to the Operations Center Area Code telephone numbers. The amendment provides the correct commercial telephone number for licensees to contact the NRC Operations Center.

During FY 1991, 91 rulemaking actions were processed, of which 23 rules were formally published, 16 were terminated/withdrawn, and 52 are ongoing (see Table 5.3). Besides the 52 ongoing rulemaking actions, there are 49 potential rulemaking actions, and it is estimated that in FY 1992 there will be approximately 15 to 20 new rulemaking requests requiring RES review and approval by the Executive Director for Operations.

Table 5.3
Rulemaking Actions Processed During FY 1991

Rulemaking Activities	Number
Final Rulemakings Published	23
Rulemakings Terminated/Withdrawn	16
Ongoing Final Rulemaking Actions	14
Ongoing Proposed Rulemaking Actions	25
Rulemakings on Hold	13
Total Rulemakings	91

5.3.3.2 Regulatory Analysis

The NRC conducts regulatory impact analyses [illegible] support of regulatory [illegible] (e.g., rulemakings, [illegible] generic safety issues, regulatory guides). The NRC is in the process of updating and revising NUREG/BR-0058, Revision 1, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory [illegible]" and [illegible] CR-3568, "Handbook for [illegible]" These documents establish [illegible] and policy [illegible] the [illegible] process. These revisions [illegible] structure of the existing [illegible] to integrate backfit analysis [illegible] elements and safety policy considerations and to update the method[illegible] formation base for performing regulatory impact analyses to reflect experience gained over the past [illegible] years. Also, to aid NRC analysts in preparing RIAs, work has begun on updating replacement energy costs and estimating the long-[illegible] loss of a plant due to an accident. These generic [illegible] methods [illegible] useful in quantifying direct costs [illegible] industry and averted onsite costs, which are both integral components of the value-impact portion of the RIA.

Development of these types of methodologies will continue in an effort to facilitate NRC decisionmaking in evaluating the need for and the effectiveness of a variety of regulatory actions, including rulemaking, standards development, and backfitting safety improvements on nuclear power plants. During this report period, approximately 16 safety-related regulatory impact analyses (both initiated and completed) have been processed.

5.3.3.3 Maintenance of Nuclear Power Plants

In March 1988, the Commission issued a Policy Statement on the Maintenance of Nuclear Power Plants. In this statement, the Commission indicated its intention to pursue a rulemaking on maintenance. In developing this proposed rulemaking, the staff had extensive interactions with U.S. industry (airline and nuclear) and studied foreign nuclear maintenance programs and practices. A 3-day public workshop was held in July 1988 to solicit comments on rulemaking options. The information gathered was used in formulating the proposed rule and its supporting regulatory guide. The Commission issued the proposed rule for public comment in November 1988 and the supporting draft regulatory guide in August 1989. In December 1989, the Commission issued a revised policy statement to restate its views with respect to maintenance and to indicate its intention to hold rulemaking in abeyance for a period of 18 months. During the 18-month time interval, the Commission monitored industry initiatives and progress in maintenance improvements and reevaluated the need for issuing a final rulemaking. Based on its evaluation, the Commission concluded that a regulatory framework should be in place to provide a mechanism for evaluating the overall continuing effectiveness of licensee maintenance programs. Accordingly, the Commission issued a final rule, 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," on July 10, 1991.

The purpose of the maintenance rule is to require commercial nuclear power plant licensees to monitor the effectiveness of maintenance activities for safety-related and certain non-safety-related plant equipment, as defined in 10 CFR 50.65, in order to minimize the likelihood of failures and events caused by the lack of effective maintenance. The rule requires that licensees monitor the performance or condition of certain structures, systems, and components (SSCs) against licensee-established goals in a manner sufficient to provide reasonable assurance that those SSCs will be capable of performing their intended functions. Such monitoring would take into account industry-wide operating experience. Where monitoring proves unnecessary, licensees would be permitted the option of relying upon an appropriate preventive maintenance program.

The NRC staff will prepare a regulatory guide, a draft of which will be released to the public document room at the end of March 1992. The nuclear industry (NUMARC) is producing a consensus guidance document for monitoring the effectiveness of maintenance in nuclear power plants in parallel with the NRC staff effort. It is expected that this document will be available, at least in draft form, for NRC evaluation at the end of March 1992. The NRC will make a preliminary decision at that point whether to proceed with the staff-developed regulatory guide or endorse the industry guidelines. A final decision will be made at the end of July 1992.

5.3.3.4 License Renewal

The NRC has been considering what requirements should be placed on nuclear power plants in the event that licenses to operate beyond the 40-year term of the original license should be granted. Public comments on license renewal requirements have been solicited three times through the *Federal Register*—the first time in connection with seven major license renewal issues (published November 6, 1986) and the second as part of an advance notice of proposed rulemaking (published August 29, 1988). The advance notice requested comments on NUREG-1317, "Regulatory Options for Nuclear Plant License Renewal," issued in August 1988. Comments were summarized and analyzed in NUREG/CR-5332, "Survey and Analysis of Public Comments on NUREG-1317: Regulatory Options for Nuclear Plant License Renewal," issued in March 1989. The third time occurred when the NRC published the proposed rule for nuclear power plant license renewal on July 17, 1990 (55 FR 29043). The final rule (10 CFR Part 54) was published in the *Federal Register* on December 13, 1991 (56 FR

64943). The following supporting documents were issued with the final rule:

- NUREG-1362, "Regulatory Analysis for Final Rule on Nuclear Power Plant License Renewal," December 1991. This regulatory analysis provides the supporting information for the final rule that defines the NRC's requirements for renewing the operating licenses of commercial nuclear power plants.

- NUREG-1398, "Environmental Assessment for Final Rule on Nuclear Power Plant License Renewal," December 1991. This document provides an assessment of the possible environmental effects of promulgating nuclear power plant license renewal standards by the final rule rather than applying requirements in an ad hoc manner in individual licensing actions.

- NUREG-1412, "Foundation for the Adequacy of the Licensing Bases," December 1991. This analysis describes the regulatory bases for the generic finding for all nuclear power plants that the findings of reasonable assurance of adequate protection for issuance of an operating license continue to be true at the time of the renewal application and accordingly need not be made anew at the time of license renewal.

- NUREG-1428, "Analysis of Public Comments on the Proposed Rule on Nuclear Power Plant License Renewal," December 1991. This report provides a summary and analysis of public comments on the proposed license renewal rule and documents the NRC's resolution of the issues raised by the commenters.

- NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," December 1991. This report presents a review of generic safety issues to identify those whose resolution could be affected by an additional operating term of 20 years.

As part of a separate rulemaking, the NRC has undertaken a generic environmental study in order to define the scope and focus of environmental effects that need to be considered in individual relicensing actions. An advance notice of proposed rulemaking (10 CFR Part 51) was issued on July 23, 1990 (55 FR 29964). Also, a notice of intent to prepare a generic environmental impact statement (GEIS) on the effects of renewing the operating license of individual nuclear power plants was issued (55 FR 29967). The NRC published the proposed rule and draft GEIS for comment on September 17, 1991 (56 FR 47016). Also announced at that time was a public workshop to review the technical basis of the proposed rule. This public workshop was held in the fall of 1991. The following support documents were issued with the proposed rule:

- NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Draft for Comment, Volumes I and II, August 1991. This document provides the technical basis for the proposed rule.

- NUREG-1440, "Regulatory Analysis of Proposed Amendments to Regulations Concerning the Environmental Review for Renewal of Nuclear Power Plant Operating Licenses," Draft for Comment, August 1991. This regulatory analysis examines alternatives to the proposed rule and provides information that supports the alternative chosen.

- Draft Regulatory Guide DG-4002, Proposed Supplement 1 to Regulatory Guide 4.2, "Guidance for the Preparation of Supplemental Environmental Reports in Support of an Application to Renew a Nuclear Power Station Operating License," August 1991. The proposed supplement to this guide details the information that should be included in an application for license renewal.

- NUREG-1429, "Environmental Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Draft for Comment, August 1991. This report provides a framework for the NRC staff to determine whether or not environmental issues important to license renewal have been identified and the impacts evaluated and provides acceptance standards to help the reviewers comply with the National Environmental Policy Act.

The final Part 51 rule and GEIS are expected to be published in 1992.

5.3.3.5 Safety Goal Implementation

In 1986, the Commission published its Safety Goal Policy Statement. On June 15, 1990, the Commission directed the staff to routinely consider the safety goals in reviewing and developing regulations and regulatory practices. To accomplish this objective, plans have been established to develop a formal mechanism to ensure that future regulatory initiatives are evaluated for conformity with the safety goal policy requirements. While the Commission recognizes that consideration of the safety goal in assessing regulatory actions will at first vary, the Commission also believes, as detailed guidance is developed and experience gained, this variation will be minimal. In support of the implementation plan for the Safety Goal Policy Statement, a definition of a "large release" is being developed. When completed, this definition will provide a subtier element for use in reviewing and developing regulations and regulatory practices.

5.4 Severe Accident Implementation

5.4.1 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 (50 FR 32138), 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 (PRAs), the Commission is convinced of 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 requested 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.

5.4.2 Program Strategy

The Containment Performance Improvement (CPI) program systematically examines insights gained from severe accident research to identify containment vulnerabilities and to identify potential improvements to correct vulnerabilities.

Because of concerns about Mark I containments, the CPI program initially studied these containments. However, studies of all types of containments are also in progress. If potential improvements that provide significant enhancements to safety are identified and are shown to be cost effective pursuant to 10 CFR 50.109, this program will recommend specific regulatory requirements. The CPI program is closely related and complementary to the individual plant examinations (IPEs) and accident management programs. The CPI program examines containments for vulnerabilities on a generic basis so that utilities do not have to deal with complex and highly uncertain severe accident phenomena on an individual basis. The IPE, on the other hand, deals with plant-specific containment vulnerabilities unique to a particular plant that are not treated under the generic CPI program.

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 eventually reviewing the results of the IPE submittals in cooperation with the Office of Nuclear Reactor Regulation. 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.

On May 25, 1988, the staff presented to the Commission an "Integration Plan for Closure of Severe Accident Issues," SECY-88-147. This plan discusses the relationships among the major elements of the plan, which include (1) the CPI program, (2) IPEs, (3) external hazards, (4) accident management, (5) improved plant operations, and (6) the severe accident research program (see

Sections 3.1 and 3.2). The commission paper also discusses the relationship with related elements such as safety goals, severe accident policy for future plants, and generic safety issues as well as requirements for closure of severe accident issues. This program element includes work to examine the areas of siting, emergency planning, and generic safety issues for potential resolution of issues or changes to existing regulations as a result of severe accident research findings.

5.4.3 Research Accomplishments in FY 1991

5.4.3.1 Individual Plant Examinations

On November 23, 1988, the NRC issued Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR 50.54(f)," to all licensees of nuclear power plants. This letter requested that all licensees perform a plant examination that looks for vulnerabilities to severe accidents and cost-effective safety improvements that reduce or eliminate the important vulnerabilities. The specific objectives for these individual plant examinations (IPEs) are for each utility to (1) develop an overall appreciation of severe accident behavior; (2) understand the most likely severe accident sequences that could occur at its plant; (3) gain a more quantitative understanding of the overall probability of core damage and radioactive material releases; and (4) if necessary, reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents. Upon completion of the examination, the utility would be required to submit a report to the NRC describing the results and conclusions of the examination. This submittal would be reviewed and evaluated by the NRC.

The NRC also issued NUREG-1335, "Individual Plant Examination: Submittal Guidance," as a draft for comment in January 1989 to provide guidance on the conduct of the IPEs. A workshop was held on February 28 and March 1 and 2, 1989, in Fort Worth, Texas, for utilities and interested members of the public to address comments and questions on the IPE process and the guidance document. NUREG-1335 was revised to reflect comments received and issued in final form in August 1989. The issuance of NUREG-1335 formally started the IPE process. Utilities would have 3 years (until September 1, 1992) to complete and submit their IPEs to the NRC.

Major efforts on the IPE in FY 1991 have involved review of IPE submittals and completion of the procurement process to obtain contractual assistance for the IPE reviews. Three additional IPE submittals were received and reviews initiated. They are the Oconee (South Carolina), Seabrook (New Hampshire), and Turkey Point (Florida) submittals. The draft safety evaluation report was completed for the Yankee Rowe submittal. Also, the review of the Seabrook IPE submittal was completed. To support the IPE reviews, three contracts were awarded to allow for a more indepth review of select IPE submittals. Because the Turkey Point IPE submittal was the first one not based on a previously reviewed PRA, it is the first submittal selected for the more indepth review.

5.4.3.2 External Events

In December 1987, the NRC established an External Event Steering Group (EESG) to make recommendations concerning the individual plant examinations for vulnerabilities to severe accidents initiated by external events (e.g., earthquakes, floods, fires). Recommendations were needed relative to: (1) what external events need consideration in the IPE, (2) what methods can be used in the examination, and (3) how the IPE for external events (IPEEE) can be coordinated with other ongoing regulatory activities involving external events, particularly in the seismic area.

Three subcommittees were established in April 1988 to make recommendations in the areas of (1) seismic, (2) fires, and (3) high winds, flood, and others (e.g., man-made hazards such as nearby transportation and military and industrial facilities). During 1989, the three subcommittees completed their studies and made recommendations for the IPEEE to the EESG.

In May 1990, the staff completed work on a draft generic letter and draft guidance document (NUREG-1407) to be sent to licensees, which describes the scope, acceptable methods, and reporting requirements for the IPEEE. The draft documents were issued for public comment on July 25, 1990. In September 1990, the staff conducted a workshop on the draft generic letter and on NUREG-1407 to solicit comments and answer questions concerning their content. Approximately 210 representatives from industry, State agencies, and the public attended the workshop. The staff revised the generic letter and NUREG-1407 to clarify and incorporate changes resulting from feedback received at the workshop and subsequently issued the final generic letter, GL 88-20, Supplement 4, and NUREG-1407 in June 1991. The generic letter requires licensees to submit their plans and schedules for performing their IPEEEs in December 1991, with completion of their IPEEEs by June 1994.

In August 1991, the staff completed its review of an EPRI/NUMARC fire evaluation methodology, "Fire Vulnerability Evaluation Methodology (FIVE)," and issued an evaluation report endorsing the use of FIVE as a viable alternative to fire PRA in the IPEEE process.

The staff is currently developing a review plan for the IPEEE submittals. It is expected that the approach for review of the IPEEE will follow closely the one developed for review of the internal-event IPE submittals.

5.4.3.3 Containment Performance Improvement

Severe accident research has identified a number of insights concerning containment performance during a severe accident. These insights have included both strengths and weaknesses of existing containment designs. In some cases, identified containment weaknesses or uncertainties in containment performance have raised concerns about severe accidents, particularly for BWR Mark I containments. The Containment Performance Improvement (CPI) program systematically examined insights gained from severe accident research to identify containment vulnerabilities and to identify potential improvements to correct vulnerabilities. Because of concerns about Mark I containments, these containments were initially studied under the CPI program. The result was a requirement that BWR Mark I plants backfit a hardened containment vent. However, studies of all other types of containments were also made.

The CPI program is closely related and complementary to the IPE and accident management programs. The CPI program examines containments for vulnerabilities on a generic basis and has resulted in identifying certain features that licensees should evaluate on a plant-specific basis as part of their IPEs.

All major elements of the CPI program have been completed. Generic letters (GLs) have been issued to licensees starting the plant-specific backfit of the hardened vent for all BWR Mark I containments (GL 89–16, dated September 1, 1989) and requ[illegible] that other improvements be considered in the [illegible] supplement 1 to GL 88–20, dated August 29, 1989, for BWR Mark I containments and Supplement 3 to GL 88–20, dated July 6, 1990, for the other containment types). The only remaining activity under this program is to complete and issue for information a series of technical reports documenting the analyses and evaluations done by the staff and its contractors in assessing the various containment types. These reports address the potential vulnerabilities identified (characterization reports), the potential fixes evaluated (enhancement reports), and analyses of the effects of uncertainties (parametrics reports). It is expected that these reports will provide licensees with information they may find useful in assessing their plants as part of the IPE. To date, 11 out of the planned 12 reports have been issued with the remaining report scheduled for issuance by November 1991.

5.4.3.4 Regulatory Application of New Source Terms

Consideration of source terms entered the regulatory process because the Commission's reactor site criteria (10 CFR Part 100) re[illegible]ire that an accidental fission product release from the [illegible] into the containment should be assumed to occur and that its radiological consequences should be evaluated assuming that the containment leaks at its "expected demonstrable leak rate." The criteria for the release into the containment are derived from the 1962 report, TID–14844, which assumed an instantaneous release of fission products. Although this source term is included in the Commission's regulations for siting, it has traditionally affected plant design more than siting.

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 activity have been identified that may benefit from change as a result of source term and severe accident research. In FY 1991, work continued on a replacement to TID–14844. It is expected that a draft report will be sent to the Commission in February 1992.

In FY 1991, the staff initiated rulemaking to decouple siting from plant design. This effort will n[illegible] directly incorporate requirements related to acceptable site characteristics. A proposed rule is expected to be sent to the Commission in February 1992.

The staff initiated rulemaking to add emergency planning requirements to 10 CFR Part 72 for independent storage of spent nuclear fuel and high-level radioactive waste. It is expected that a proposed rule will be sent to the Commission by mid-1992. Also in FY 1991, a revision to Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," was initiated to revise the approach for the development of Emergency Action Levels. A draft is expected to be issued for public comment in early 1992.

5.5 Radiation Protection and Health Effects

5.5.1 Statement of Problem

The NRC must provide radiation protection standards and guidance that ensures that workers and members of the general public are adequately protected from the adverse consequences of exposure to ionizing radiation from licensed activities. RES 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.

Effective regulation associated with environmental policy and decommissioning activities involves the task of planning, developing, and issuing appropriate regulatory positions. Using information generated internally or through narrowly directed research, new positions are developed or existing positions are modified. These positions can take the form of regulatory requirements, policy statements, guidance, or criteria for activities pertaining to decontaminating and decommissioning licensed nuclear facilities, exemption of materials or products from regulatory control, and disposing of low-level radioactive waste streams. Setting priorities for regulatory needs or deficiencies are undertaken to ensure that the problems of greatest significance to the public health and safety or the common defense and security are addressed in an expeditious manner through properly defined regulatory and supporting research programs.

5.5.2 Program Strategy

The Commission's regulatory process requires that safety enhancements to 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 technology 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 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.

The NRC needs to develop a regulatory approach to evaluate future requests involving decommissionings and license terminations. This regulatory approach should define acceptable alternatives, requirements, and criteria for decommissioning before such a request is received. The strategy has two parts: (1) to develop or modify regulatory requirements and guidance to protect workers and the public from radiation risks associated with operations involving the decommissioning of licensed nuclear facilities, and (2) to establish radiological criteria for residual radioactivity.

In the area of "below regulatory concern" (BRC), the Low-Level Radioactive Waste Policy Amendments Act of 1985 requires NRC to establish standards and procedures for expedited action on BRC waste disposal petitions. Federal agencies, including NRC, are currently in the process of establishing and implementing BRC or exemption levels for radioactive waste disposal as well as other areas. The strategy of this program is to carry out Commission directives to develop and implement Commission BRC policy and, in particular, the Commission initiative to establish a consensus on BRC policy issues. This consensus would serve as the framework for specific exemption decisions involving disposal of low-level waste streams, as well as other activities concerning the release or use of radioactive material.

5.5.3 Research Accomplishments in FY 1991

5.5.3.1 Radiation Protection Issues

ALARA Center. The Brookhaven National Laboratory (BNL) ALARA Center, funded by the NRC, continued its surveillance of the Department of Energy and industry dose reduction and ALARA research during the report period. BNL has published a series of reports (NUREG/CR-3469) that abstracts 252 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 1991, BNL focused on providing guidance to high dose worker groups and developing an international dose reduction data base.

The Center is recognized by the nuclear industry and others as a major source of information on new and effective dose reduction techniques, and its publications are standard references for ALARA planning. The BNL staff is available through the Center to the entire NRC organization and to its licensees for information and advice on all aspects of radiation protection and dose reduction. This effort becomes even more important with the implementation of the new Part 20, making ALARA a requirement.

In FY 1991, the BNL ALARA Center worked on an analysis of impacts of implementing new recommendations by the ICRP and the NCRP for dose limits. This work will provide a technical base for future NRC regulatory decisions regarding further changes in worker dose limits.

Accreditation and Testing of Personnel Dosimetry Processors. An ongoing program that requires accreditation of personnel whole body dosimetry processors became effective in February 1988. Accreditation is acquired through the National Voluntary Laboratory Accreditation Program (NVLAP), operated by the National Institute of Standards and Technology (NIST), and reaccreditation of processors is required every 2 years. The program goal is to improve and maintain quality assurance and quality control over all aspects of personnel dosimetry processing by requiring all processors to meet the performance requirements of the national consensus standard for processing, ANSI N13.11–1983.

As of July 1, 1991, 67 laboratories, including one in Taiwan, were accredited for processing whole body dosimeters. These include commercial dosimetry processors, military establishments, commercial shipbuilders, nuclear power companies, and other commercial establishments that use radiation measurement techniques. A draft regulatory guide that will discuss methods of meeting the NVLAP procedures for processor accreditation will be published for comment early in FY 1992.

In the extremity dosimetry areas, a revised standard has been voted on by the Health Physics Society Standards Committee (HPSSC), and it is expected that ANSI acceptance will occur early in 1992. Tests against the revised standard (HPSSC P/N 13.32) have begun, and testing is expected to continue through June 1992. Twenty-four facilities are expected to participate. Should the tests indicate that the revised standard is a suitable criterion for testing, appropriate rulemaking will be initiated to require extremity dosimeters to be processed by processors certified under the NVLAP procedures in use at NIST.

New Skin Dose Computer Code. A new computer code for calculating dose to the skin from radioactive materials on the skin will be published in 1992. This code will replace the VARSKIN code in use since 1986. The new code will be a great deal more flexible than VARSKIN, allowing for self-absorption of radiation within radioactive particles on the skin and backscattering of radiation, and it will permit the calculation of dose from different shapes of particles and particles separated from the skin by clothing. The code will also calculate the dose from both gamma and beta radiations.

Self-Powered Photon Detector. Research to develop a large area self-powered photon detector using a concept similar to that for self-powered neutron detectors (first developed in the Soviet Union in 1961 and improved upon and patented in Canada in 1968) is completed. The contractor has applied for a patent application for the use of this detector. A final report on this research will be published shortly as NUREG/CR-4833.

Tissue Equivalent Thermoluminoscent Dosimeters. Phase II of research to develop a gamma-ray spectrometer/dosimeter has begun. The purpose is to demonstrate the feasibility of developing a differential energy absorption spectrometer coupled to a small microcomputer that would have essentially the same response to radiation as that of human tissue over the energy range of 0.5–10 MeV. Current dosimeters are essentially flat over this range while tissue response varies by a factor of about eight. Phase I research demonstrated feasibility of the concept using a four-detector cadmium telluride assembly but had some detector leakage problems that prevented making low dose measurements. These have now been corrected, and it is projected that the Phase II research will provide adequate measurements to lead to development of a commercial prototype under Phase III.

"Hot Particles" on Clothing Detector. The rapid detection, measurement, and location of small particulate radioactive material on laundered ("clean") protective clothing is the objective of another research contract. Under Phase II of this contract, a prototype of a system for surveying clothing has been successfully demonstrated. It is expected that this system will be marketed for commercial use. The system has potential for reducing radiation exposure of personnel who may wear "clean" protective clothing and be unaware that the clothing bears particulate radioactive material.

5.5.3.2 Health Effects Research

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 1991 with significant progress. The NRC recently published for comment NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Doses." This report provides a methodology for calculating internal doses to the embryo/fetus and a data base for selected radionuclides. Work is currently under way on reissuance of NUREG/CR-5631 with an expanded data base that will include uranium and additional isotopes of previously described elements such as Sr-89, Cs 134, and Pu-238. Research that will permit inclusion of other radionuclides, such as technetium, molybdenum, americium and other transuranics, is planned. The methods and data developed under this project will be used by the NRC in preparation of an interim regulatory guide describing acceptable methods of compliance with Section 20.208 of the revised 10 CFR Part 20. This guide will be revised as new information warrants. The methodology will also be used to calculate doses in cases of accidental releases of radioactive materials.

Improvement of Health Effects Models. Revision 1 to NUREG/CR-4214, "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis," published in May 1989, contains 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. An addendum entitled "Modification of Models Resulting From Recent Reports of Health Effects of Ionizing Radiation" was published in August 1991. The reports that led to the revision of models presented in the 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 the revised recommendations of ICRP-60 (ICRP 1991).

Cellular and Molecular Biology. Based on the discovery of oncogenes, the development techniques of recombinant DNA molecular biology, and the progress that has been made in the characterization of certain human cancers in genetic terms, the NRC sponsored a feasibility study aimed at reduction of uncertainties in risk coefficients. The results of this study were reported in NUREG/CR-5635, "Cellular and Molecular Research to Reduce Uncertainties in Estimates of Health Effects from Low-Level Radiation."

The study concluded that it is feasible to reduce uncertainties of radiation-induced health effects by mounting a program of radiation research directed at the mechanism(s) of radiation-induced cancer with special reference to risk of neoplasia due to protracted, low doses of sparsely ionizing radiation. NUREG/CR-5635 has been distributed to Federal agencies, CIRRPC, National Academy of Science (NAS), Radiation Effects Research Foundation (RERF), and individual scientists.

Chemical Toxicity of Uranium Hexafluoride Compared to Radiation Doses (NUREG-1391). This staff report compared the chemical toxicity of uranium hexafluoride with the acute effects of a radiation dose of 25 rems to the whole body (the value used in Part 100 dealing with reactor siting criteria). The work will be used to support development of licensing requirements for commercial uranium enrichment facilities. The draft report was published for comment in April 1990; a final report is scheduled for publication in 1992.

5.5.3.3 Development of Rules and Regulatory Guides

Occupational Exposure Data Systems. In 1969, the Atomic Energy Commission began requiring certain licensees to submit reports on occupational radiation dose received by workers. These data are collected and computerized in an NRC system called the Radiation Exposure Information Reporting System (REIRS). The system provides a permanent record of the data and permits expeditious analyses of the two kinds of reports required (annual statistical summaries and individual termination reports). Exposures received as a result of medical procedures are not required to be reported.

A preliminary compilation of summaries of the annual statistical reports for 1989 revealed that about 203,000 persons were monitored of whom about 53 percent received measurable doses. The workers received a collective dose of approximately 36,200 person-rems or an average annual dose of about 0.33 rem per worker among those receiving a measurable dose. These figures are about 10–15 percent lower than those found for 1987. Of the persons monitored, 90 percent worked in nuclear power plants, and they incurred about 90 percent of the total annual collective dose. After declining for several years, the annual collective dose incurred by nuclear power plant workers appears to have leveled off. Preliminary compilations of the exposure data reported by nuclear power plants for calendar year 1990 are not significantly changed. One additional reactor was reported during this period.

A second kind of exposure report required of certain NRC licensees provides identification and dose data each time a monitored person terminates work at the licensed facility. Such information is now maintained for some 575,000 persons, most of whom worked at nuclear power plants. The computerization of these data enables the NRC staff to respond quickly to requests for individual exposure histories and to analyze the data for trends. The data also assist in the examination of the doses incurred by transient workers as they move from plant to plant. For example, further analysis of the data reported for 75,400 persons terminating employment during 1988 revealed that 3,622 of them had worked at two or more nuclear power facilities and that none of them had received doses in excess of the regulatory limits as a result of their multiple employment.

Revision of Part 20 Radiation Standards. The Commission has approved using a complete revision to the NRC regulations for radiation protection in 10 CFR Part 20. The final rule was published in the *Federal Register* in May 1991 (56 FR 23360). This revision updates the Commission's regulations to incorporate recommendations made by the ICRP, the NCRP, and the revised Federal

Radiation Guidance for Occupational Exposure issued in 1987. The new standards represent a significant change from the methods previously employed to assess and control radiation doses. The new Part 20 will result in reducing the annual occupational dose from a possible 17 rems (3-rem/quarter external + 5-rem annual internal) to a total effective dose of 5 rems per year. The dose limit for members of the general public is reduced from an implicit 0.5 rem per year in the present rule to an explicit value of 0.1 rem per year. The new Part 20 contains appendices that give the radionuclide concentration limits for air, water, and sewerage.

Proposed Rule on Large Irradiators. A proposed rule for large irradiators was published for public comment in the *Federal Register* in December 1990 (55 FR 29043). A 2-day public workshop to discuss the proposed rule was held in Rockville, Maryland, in February 1991. Large irradiators are defined as those capable of delivering a dose of 500 rads in an hour to a person standing 1 meter from the sources. The final rule is scheduled for publication in FY 1992.

Certification of Industrial Radiographers. A final rule that would recognize a third-party certification program of the American Society for Nondestructive Testing (ASNT) was published in the *Federal Register* in March 1991 (56 FR 11504). The rule would give licensees the option of using the ASNT program in lieu of describing their training program to NRC. The certification program is expected to improve training and safety performance in the workplace.

Air Sampling in the Workplace. A proposed regulatory guide, "Air Sampling in the Workplace," to meet the requirements of the new Part 20 was published in the *Federal Register* for public comment (56 FR 52078) in September 1991. The guide deals with issues such as what should the licensee do to demonstrate that samples are representative of the air inhaled by workers, and what measurements are necessary to be able to adjust derived air concentrations to account for particle size. The guide is accompanied by a technical manual, "Air Sampling in the Workplace," describing how the recommendations in the guide can be met. Both documents are scheduled to be issued in final form in FY 1992.

Fuel Cycle. A proposed rule, published in the *Federal Register* (56 FR 46739) for public comment in September 1991, would amend the Commission's regulations concerning the licensing of uranium enrichment facilities to reflect changes made to the Atomic Energy Act of 1954 (the Act), as amended by the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990. The principal effect of these amendments is that uranium enrichment facilities would be licensed subject to the provisions of the Act pertaining to source material and special nuclear material rather than under the provisions pertaining to a production facility. The Commission is currently reviewing a license application by the Louisiana Energy Service Corporation to construct and operate a commercial uranium enrichment facility.

The staff is continuing to follow emerging technologies for uranium enrichment and other fuel cycle facilities for potential radiological, chemical, and criticality safety concerns. NUREG/CR-5768, "Ice Condenser Aerosol Tests," was published and presents the results of an experimental investigation of aerosol particle transport and capture using a full-scale height and reduced-scale cross section test facility based on the design of the ice compartment of a PWR ice-condenser containment system. Results of 38 tests included thermal-hydraulic as well as aerosol particle data.

5.5.3.4 Decommissioning

A proposed rule was published in the *Federal Register* (56 FR 50524) for public comment in September 1991 to amend the Commission's decommissioning regulations to require holders of a specific license for possession of byproduct material, source material, special nuclear material, and independent storage of spent nuclear fuel and high-level waste to prepare and maintain additional documentation identifying areas where licensed materials and equipment were stored and used. The Commission's intent is to provide both the NRC and the licensee the necessary information to ensure complete decommissioning of licensed facilities. In addition, this action also is consistent with similar requests made at the Synar Committee Hearing on decommissioning and an earlier GAO report.

A Notice of Receipt of Petition for Rulemaking was published (56 FR 4845) on a joint petition by the General Electric Company and the Westinghouse Electric Corporation requesting that the Commission amend its decommissioning regulations to provide a means for self-guarantee of decommissioning funding costs by certain NRC non-electric utility reactor licensees who meet stringent financial assurance and related reporting and oversight requirements. As requested by the Commission, the notice also solicits public comments on other self-guarantee criteria, if any, and the basis for the criteria and additional information on self-guarantee.

Four reports associated with reactor decommissioning technology and costs were published. These are NUREG/CR-5343, "Radionuclide Characterization of Reactor Decommissioning Waste and Spent Fuel Assembly Hardware"; NUREG/CR-2601, Addendum 1, "Reevaluation of the Cleanup Cost for the Boiling Water Reactor (BWR) Scenario 3 Accident from NUREG/CR-2601"; NUREG/CR-1307, Revision 2, "Report on Waste Burial Charges"; and NUREG/CR-0672, Addendum 4, "Comparison of Two Decommissioning Cost Estimates Developed for the Same Commercial Nuclear

Reactor Power Station." The final regulatory guides on standard format and content of plans for reactor decommissioning and reactor decommissioning recordkeeping are in progress.

A Commission Paper, SECY-91-164, "Decommissioning Costs," for nuclear power reactors, as requested by the Commission, was completed on May 31, 1991. The staff effort on the development of information on the safety, costs, and wastes related to the decommissioning of LWRs and other nuclear facilities is progressing according to schedule. As stated in SECY-91-164, the staff expects the completion of revised cost estimates for LWRs by October 1993.

APPENDIX

FY 1991 Regulatory Products from the Office of Nuclear Regulatory Research

Date	Regulatory Product	Description
Integrity of Reactor Components		
December 1990	Draft Regulatory Guide DG-1009	Provides regulatory guidance on standard format and content of technical information for applications to renew nuclear power plant operating licenses.
May 1991	Revision of 10 CFR 50.61	Revision to provide consistency in irradiation embrittlement trend curves with Regulatory Guide 1.99, Revision 2.
May 1991	NUREG-1426, Volume 1	Provides a compilation of reports from research supported by the Materials Engineering Branch (1965-1990).
July 1991	NUREG-1377, Revision 2	Provides a listing and summaries of reports issued through June 1991 on the NRC research program on plant aging.
Reactor Containment Performance		
December 1990	NUREG-1150	This final report assessed the risks from possible core damage accidents in five U.S. nuclear power plants—Surry (Va.), Zion (Ill.), Sequoyah (Tenn.), Peach Bottom (Pa.), and Grand Gulf (Miss.).
May 1991	Safety Evaluation Report	Staff evaluation of PRA performed by GE as part of ABWR design certification submittal.
June 1991	NUREG-0675 Supplement 34	Staff evaluation of PRA performed as operating license condition for Diablo Canyon nuclear power station.
August 1991	Safety Evaluation Report	Staff evaluation of PRA guidance developed by EPRI in support of ALWR design certifications.
October 1991	Safety Evaluation Report	Staff evaluation of PRA performed by South Texas plant to support license modifications.
Confirming Safety of Nuclear Waste Disposal		
October 1990	Final Rule	The NRC regulation (10 CFR Part 40) was revised to provide licensing for the custody and long-term care of uranium and thorium mill tailing disposal sites.

Date	Regulatory Product	Description
		These procedures will ensure that long-term performance of uranium mill tailing disposal sites is properly monitored and that deterioration requiring restoration will be detected so that appropriate action can be taken.
November 1990	Draft Regulatory Guide DG-3003	Provides guidance on the information needed to review DOE's license application for an HLW repository.
July 1991	Petition Denial	Denial of petition for rulemaking (PRM-61-1) from the North Carolina chapter of the Sierra Club to permit the design and construction of a zero-release low-level radioactive waste disposal facility in a saturated zone. The petitioner maintained that the regulation was necessary in order for the General Assembly of North Carolina to consider a waiver of a North Carolina statute that requires that the bottom of a low-level waste facility be at least 7 feet above the season high water table.
Resolving Safety Issues and Developing Regulations		
October 1990 – September 1991	Generic Safety Issues	For generic safety issues prioritized and resolved in FY 1991, see Tables 5.1 and 5.2.
December 1990	Proposed Rule/Draft Regulatory Guide DG-5002	The proposed rulemaking would amend the NRC regulation (10 CFR Part 74) on the material control and accounting requirements for uranium enrichment plants. An accompanying draft regulatory guide was issued. The rulemaking and guidance will facilitate licensing of facilities that request construction and operation of a gas centrifuge plant to produce low-enriched uranium for the commercial market.
December 1990	Proposed Rule	The proposed rule would amend NRC regulations to incorporate a new regulation for large irradiators that are capable of delivering a dose of 500 rads in an hour to a person standing 1 meter from the sources.
January 1991	Petition Denial	Denial of a petition for rulemaking (PRM-50-50) from Charles Young to amend NRC regulations to prevent nuclear power plant operators from deviating from license conditions or

Date	Regulatory Product	Description
		technical specifications during an emergency. The petitioner maintains that nuclear power plants should be operated in accordance with the operating license and appropriate technical specifications and that requiring a senior operator to follow the technical specifications during an emergency enhances plant safety.
January 1991	Final Rule	The NRC regulations (10 CFR Parts 20 and 50) that apply to the Operations Center Area Code telephone numbers were revised. The rule provides correct commercial telephone numbers for licensees to contact the NRC Operations Center.
January 1991	Proposed Rule	The proposed rule would amend 10 CFR Part 50.55a, "Codes and Standards," to update references to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections III & XI, through the 1989 Edition.
March 1991	Final Rule	The NRC regulations were revised to recognize a third party certification program of the American Society for Nondestructive Testing (ASNT). The rule would give licensees the option of using the ASNT program instead of describing their training program to NRC. The certification program is expected to improve training and safety performance in the workplace.
April 1991	Final Rule/ Regulatory Guide 5.66	The NRC regulation (10 CFR 50.73) on access authorization at nuclear power plants was revised, and an accompanying regulatory guide was issued. The rule requires a nuclear power reactor licensee to have an access authorization program in its site physical security plan. This would provide increased assurance that persons granted unescorted access to protected and vital areas are trustworthy and do not pose a threat to commit radiological sabotage. The regulatory guide provides associated guidance for implementing the NRC regulation.
April 1991	Draft NUREG-1391	The staff report compared the chemical toxicity of uranium hexafluoride with the acute effects of a radiation dose of 25 rems to the whole body (the value used in

Date	Regulatory Product	Description
		Part 100 dealing with reactor siting criteria). The work will be used to support development of licensing requirements for commercial uranium enrichment facilities.
May 1991	Final Rule	The NRC regulations for radiation protection in 10 CFR Part 20 were revised. This revision updates the Commission's regulations to incorporate recommendations made by the ICRP, the NCRP, and the revised Federal Radiation Guidance for Occupational Exposure issued in 1987. The new standards represent a significant change from the methods previously employed to assess and control radiation doses. The new Part 20 will result in reducing the annual occupational dose from a possible 17 rems (3-rem/quarter external + 5-rem annual internal) to a total effective dose of 5 rems per year. The dose limit for members of the general public is reduced from an implicit 0.5 rem per year in the present rule to an explicit value of 0.1 rem per year. The new Part 20 contains appendices that give the radionuclide concentration limits for air, water, and sewerage.
June 1991	NUREG-1374	"Technical Findings Related to Generic Issue 79: Reactor Vessel Thermal Stress During Natural Convection." Resolution did not require any action by licensees.
June 1991	Generic Letter 88-20, Supplement 4	"Individual Plant Examinations of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f)," which describes the scope, acceptable methods, and reporting requirements for the IPEEE.
June 1991	NUREG-1407	"Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities."
July 1991	Final Rule	10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was issued to provide a regulatory framework for evaluating the overall effectiveness of licensee maintenance programs.
July 1991	Generic Letter 91-11	Provided resolution for GI-128, "Electrical Power Reliability," which found existing requirements were adequate.

Date	Regulatory Product	Description
July 1991	Final Rule	The NRC regulations (10 CFR Part 35) were revised to require medical-use licensees to implement quality management programs and revise misadministration reporting requirements. The rule provides a high confidence that byproduct material and radiation from byproduct material will be administered as directed by the authorized user physician.
August 1991	Final Rule	The NRC regulations (10 CFR Parts 20, 30, [illegible] and 70) on licensee reporting requirements regarding notifications of incidents related to radiation safety were revised. This rule will ensure that significant occurrences at facilities operated by material licensees are promptly reported to the NRC. The Commission will be able to determine whether a licensee has taken the actions necessary to protect public health and safety and whether generic safety concerns that may require prompt NRC actions are identified.
August 1991	Final Rule	The NRC regulations (10 CFR Part 50) were revised to require the licensee to implement the NRC-approved Emergency Response Data System at all nuclear power plants. The rule would supplement the voice transmission over the existing Emergency Notification System and require a direct electronic data link between the licensee's computer and the NRC's Operation Center to be activated by the licensee during an alert of higher emergency condition to transmit timely and accurate updates of critical information on plant conditions. This would allow the NRC to perform its primary role during an emergency at a licensed nuclear power facility, which is one of monitoring the licensee to ensure that appropriate recommendations are made with respect to necessary offsite actions to protect public health and safety.
September 1991	Proposed Revision 1 to Regulatory Guide 8.25	Provide guidance on "Air Sampling in the Workplace," to meet the requirements of the new Part 20. The guide

Date	Regulatory Product	Description
		deals with issues such as what the licensee should do to demonstrate that samples are representative of the air inhaled by workers and what measurements are necessary to be able to adjust derived air concentrations to account for particle size. The guide is accompanied by a technical manual, "Air Sampling in the Workplace," describing how the recommendations in the guide can be met.
September 1991	Proposed Rule	The proposed rule would amend the Commission's regulations concerning the licensing of uranium enrichment facilities to reflect changes made to the Atomic Energy Act of 1954 (the Act), as amended by the Solar, Wind, Waste, and Geothermal Power Production Incentives Act of 1990. The principal effect of these amendments is that uranium enrichment facilities would be licensed subject to the provisions of the Act pertaining to source material and special nuclear material rather than under the provisions pertaining to a production facility.
September 1991	Proposed Rule	The proposed rule will amend decommissioning regulations to require holders of a specific license for possession of byproduct material, source material, special nuclear material, and independent storage of spent nuclear fuel and high-level waste to prepare and maintain additional documentation identifying areas where licensed materials and equipment were stored and used. The Commission's intent is to provide both the NRC and the licensee the necessary information to ensure complete decommissioning of licensed facilities.
September 1991	Proposed Rule	10 CFR Part 51,"Environmental Review for Renewal of Operating Licenses," to define the focus and scope of environmental effects to be considered in relicensing actions.
September 1991	NUREG–1437	Generic Environmental Impact Statement (GEIS) for License Renewal.
September 1991	NUREG–1445	"Regulatory Analysis for the Resolution of Generic Issue 29: Bolting Degradation or Failure in Nuclear Power Plants."

Date	Regulatory Product	Description
September 1991	Safety Evaluation Report	The draft Safety Evaluation Report was completed as part of the IPE for the Yankee Rowe plant.
September 1991	Safety Evaluation Report	The review of the IPE for the Seabrook plant was completed.
December 1991	Final Rule	10 CFR Part 54, "Nuclear Power Plant License Renewal," was issued to describe requirements for licensees to operate beyond the 40-year term of the original license (56 FR 64943).

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11. ABSTRACT (200 words or less)

This report, the seventh 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 1991.

The goal of this office is to ensure that safety-related research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This research is necessary to make certain that the regulations that are imposed on licensees provide an adequate margin of safety so as to protect the health and safety of the public. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

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