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NRC Safety Research in Support of Regulation – FY 1993

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ABSTRACT

This report, the ninth 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 1993. A special emphasis on accomplishments in nuclear power plant aging research reflects recognition that a number of plants are entering the final portion of their original 40-year operating licenses and that, in addition to current aging effects, a

focus on safety considerations for license renewal becomes timely.

The primary purpose of performing regulatory research is to develop and provide the Commission and its staff with sound technical bases for regulatory decisions on the safe operation of licensed nuclear reactors and facilities, to find unknown or unexpected safety problems, and to develop data and related information for the purpose of revising the Commission's rules, regulatory guides, or other guidance.

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HIGHLIGHTS

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

Reactor Aging and License Renewal

Pressure Vessel Safety and Piping Integrity

- A draft regulatory guide was issued that would provide evaluation methods for demonstrating that materials with low resistance to a "ductile tearing" failure mode meet the requirements of Appendix G to 10 CFR Part 50 on Charpy upper-shelf energy (the indication of reactor vessel toughness). Generic analyses using the draft regulatory guide methodology demonstrated that adequate pressure vessel integrity exists for pressure vessels with Charpy upper-shelf energy values well below the values specified in Appendix G.
- A draft regulatory guide was issued that would provide acceptable methods for determining calculational and dosimetry methods for determining pressure vessel neutron fluence for power reactors. In support of this draft guide, new neutron cross-section libraries were developed that apply the latest evaluated nuclear data files.
- A mechanistically based model was developed for predicting steam generator tube failures and leak rates of degraded tubes under normal and main steam line break conditions. This model was used by the NRC Task Group on Interim Plugging Criteria, which was charged with evaluating industry proposals on the subject.
- As part of NRC participation in Working Group 3 (pressure vessel integrity) of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety, an effort continued on test reactor irradiations in the United States and Russia, Russian scientists visited the United States, and a Russian scientist was accepted for a 1-year assignment to Oak Ridge National Laboratory to investigate radiation embrittlement and thermal-annealing effects.
- Interim fatigue design curves were developed and the information provided in NUREG/CR-5999; they more accurately describe fatigue life in the high-temperature water characteristics of LWR coolant systems. Fatigue tests are in progress to validate and/or update the proposed design curves.

- A draft regulation and draft regulatory guide addressing the engineering and metallurgical aspects of thermal annealing for U.S. plants were developed.

Aging of Reactor Components

- The process is continuing in the development of information on, as well as mitigation and managing methods for, the aging process of safety-significant components of commercial nuclear power plants. Final reports on the following subjects were issued: BWR internals; PWR internals; check valves; auxiliary feedwater system (follow-on study); aging, condition monitoring, and loss-of-coolant accident (LOCA) tests of Class 1E electrical cables; record-keeping; HEPA filters and adsorbers; and essential HVAC chillers. The results will be used by the NRC in identifying and developing policies and guidelines for making operating plant management decisions that may safely extend the life of the plants.
- The risk-based methodology for assessing aging in nuclear power plants and for defining priorities among risk contributors and maintenance activities is subject to uncertainties because of limited available aging data and because of various modeling assumptions (e.g., modeling of effects of test and maintenance). Research in 1993 focused on developing sensitivity and uncertainty analyses to define priorities addressing data and modeling uncertainties and was documented in draft NUREG/CR-6405.

Standard Reactor Designs

Engineering Issues for Advanced Reactor Designs

- A probabilistic methodology was developed to estimate the failure probability for low-pressure piping subjected to reactor coolant pressures and temperatures during an intersystem loss-of-coolant accident. The method is applicable to both carbon and stainless steel piping and other piping components, including flanges, valves, pumps, and heat exchanger tubes. The methodology can be used for piping reviews for operating reactors and for Advanced Light-Water Reactor (ALWR) design certification reviews.
- The Oak Ridge National Laboratory is conducting a study to identify functional and environmental issues arising from the application of new technologies to the instrument and control (I&C) systems in both present and next generation of nuclear power plants. Preliminary results were published in draft

NUREG/CR-5904. Heavy emphasis was placed on the applicability of military and industrial standards for the qualification of such systems. Generally such standards were shown to be overly conservative as applied to digital I&C systems for nuclear plants.

Systems Performance of Advanced Reactors

- Modifications to convert the ROSA facility for AP600 testing were defined and the hardware installed.
- A contract to build a loop for confirmatory testing of the Simplified Boiling Water Reactor (SBWR) was awarded to Purdue University.
- A review of the data base supporting the CANDU 3 design has been completed.
- Improvements were made to the models in the RELAP5 code for the AP600 and SBWR designs.

Advanced Reactor Risk Analysis

- The passive reliability project was continued throughout FY 1993. It is presently focusing on the Westinghouse AP600 design and developed a candidate methodology for quantifying the uncertainty distribution in the core damage frequency arising from uncertainty in the modeling of the natural processes. Project documentation will be completed in FY 1994.

Regulatory Application of New Source Terms

- Work continued on a replacement for TID-14844. In support of this effort, four contract NUREG reports were issued.
- Staff efforts continued on revising 10 CFR Part 100, "Reactor Site Criteria." In an SRM dated March 28, 1994, the Commission approved the staff's recommendation provided in SECY-94-017, "Options with Regard to Revising 10 CFR Part 100, Reactor Site Criteria," that the proposed rule issued for comment in October 1992 not be adopted and that Part 100 be revised to emphasize siting aspects by including basic site criteria.
- In May 1993, a proposed rule was published on the emergency planning licensing requirements for independent spent fuel storage facilities and monitored retrievable storage facilities.
- In June 1993, a proposed rule was published on revised emergency planning that would update and clarify ambiguities that have surfaced in the imple-

mentation of the Commission's emergency planning exercise requirements.

- In March 1993, a petition for proposed rulemaking by VEPCO was published. Proposed rulemaking would change the frequency of emergency exercises from annual to biennial.

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

Plant Performance

- In order to share international experience with thermal-hydraulic system codes, over 14 countries have joined the Code Applications and Maintenance Program, thereby providing \$390,000 in financial support.

Human Reliability

- The Human Performance Investigation Process is a systematic method for investigating the root causes of human performance problems during events at nuclear power plants. Field testing of the method was completed and is being used in current event investigations. Future training on the method will be provided by NRC's Technical Training Center.
- The NRC requested and received a report on the lessons learned from the study of over 10 years of experience with test and evaluation methods for computer-based operator support systems at the Halden Reactor Project in Norway.
- The NRC initiated and conducted an international workshop on digital systems reliability and nuclear safety. The workshop provided feedback from experts from nine different countries regarding potential safety issues, proposed regulatory positions, and research.

Reactor Accident Analysis

Reactor Risk Analysis

- A simplified analysis of potential in-plant and offsite accident progression and the health consequences of accidents initiated during low-power and shutdown operating conditions has been performed and provided to the Office of Nuclear Reactor Regulation (NRR) in support of their regulatory activities, as documented in NUREG-1449. The results of Phase 2 of this work will be published as contractor NUREG reports early in FY 1994.

- The South Texas nuclear power plant requested modifications to its plant technical specifications based, in part, on its risk analysis. The RES staff is now working with NRR on the acceptability of the requested modifications and expects to complete this work early in FY 1994.
- PRA data from three more licensed nuclear power plants were added to the SAPHIRE (System Analysis Programs for Hands-on Integrated Reliability Evaluation) data base; most of the data from previous plant loads was updated to Version 5.0.
- The NRC continued working with the Commission of the European Communities and the Organization for Economic Cooperation and Development to perform an intercomparison exercise on probabilistic accident consequence codes.
- A survey and evaluation of aging risk assessment methods and applications is being performed. A draft contractor NUREG report has been received; the final report will be published early in FY 1994.
- The PRA Working Group, organized to provide guidance to the staff on the use of PRA, completed a draft report in October 1993. The report includes initial guidance on the use of PRA in screening and analyzing reactor operational events and on basic terms and methods used in PRA. The report also contains a number of recommendations for additional guidance development, improvements to the NRC's PRA training program, and improvements in PRA tools and data bases used by the staff. It is expected that the final report will be published in March 1994.
- In response to a request from the Office of Analysis and Evaluation of Operational Data (AEOD), RES has developed a new course that is intended to treat reactor safety in a broad sense. Two presentations of the course have been offered at NRC's Technical Training Center in Chattanooga during 1993. With the developmental work completed, responsibility for the conduct of the course will be turned over to AEOD in FY 1994.
- An extensive series of integral effects testing for direct containment heating (DCH) at different scales was completed. Two reports were completed, one to describe the DCH issue resolution process for the Zion and Surry plants and PWRs in general, and the other to document the probability of containment failure by DCH at Zion. These reports are undergoing peer review and will be completed in FY 1994.
- The independent peer review of the SCDAP/RELAP severe accident computer code was completed, and improvements to the code based on peer review recommendations were initiated. An independent peer review of the CONTAIN code began in FY 1993 and will be completed in FY 1994. The peer review of NRC's third major severe accident systems code, MELCOR, was completed and improvements to the code made in FY 1993 to address peer review comments.
- The late-phase core melt progression experiment, MP-2, was completed as were two initial ex-reactor tests of core damage progression in BWR dry core accident conditions. Results of these experiments, as well as plans for future research in core melt progression, will be reviewed early in FY 1994 by an expert peer review group to provide input on the need for additional research in this area.

Severe Accident Policy Implementation

- Twenty-six new IPE (internal-event) submittals were received (a total of 63 to date) and seven were evaluated. It is expected that all IPE submittals will be received and reviewed by the end of CY 1995.
- Four IPEEE (external-event) submittals were received of which two are now being evaluated.

Safety Issue Resolution and Regulation Improvements

Earth Sciences

- Several noteworthy developments occurred in studies related to the strong ground motions. By studying 97 earthquakes recorded by the Eastern Canada Network, a model for attenuation of ground motions was developed, and information was obtained about propagation and source characteristics in the Eastern United States.
- Another ground motion study concerned rupture histories of eastern North American earthquakes. Such large events as Miramichi, Nahanni, Ungava, and Saguenay were included in the study and the time histories of movement on the earthquake failure surface were obtained for these events by

Containment Performance

- The TMI-2 Vessel Investigation Project was completed and final technical reports issued. Examinations of reactor vessel lower-head samples were completed, and the results were used as input to perform calculations of potential reactor vessel failure modes. The conclusions and results of this project were presented at a 3-day public meeting in October 1993 in Boston, Massachusetts.

inverting teleseismic and strong motion recordings. These studies provide improved understanding of the earthquake source mechanism in the Eastern United States and also allow for better source modeling for ground motion predictions.

- In an update of the earlier work, a stochastic ground motion model was extended to include four broad site categories for the ground motion prediction taking into account local site geology. In the previous work, the model was limited to the deep site conditions.

Plant Response to Seismic and Other External Events

- The NRC published for public comment the proposed revision of Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100. Responses were received from approximately 47 domestic and foreign commenters. The staff is reviewing all the comments and will revise the proposed rule as appropriate in FY 1994.
- Seismic testing of relays to determine the influence of relay chatter on circuit breaker tripping was completed in FY 1993. Results obtained indicate that the acceptability of relay chatter depends on the circuit parameters of the specific circuit. These results will facilitate implementation of the resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants," and the IPEEE program.
- During FY 1993, the NRC initiated a program to review significant changes being proposed to portions of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division 1, which deal with seismic design of piping systems. The objectives of this program are to (1) assist the NRC staff in developing regulatory changes and perform supporting research activities as needed, and (2) evaluate the cumulative impact of proposed changes on the overall safety margins of the piping systems. This program will be completed in 1995, allowing the staff to develop its position on the piping design requirements.
- A collaborative effort involving exchange of technical information was established with the Ministry of International Trade and Industry and Nuclear Power Engineering Corporation (NUPEC) of Japan. In this effort, NUPEC is carrying out a seismic proving test program for a main steamline typical of the PWR plants and a feedwater system typical of the BWR plants. The NRC will carry out pre- and post-test analyses to assess the applicability of currently available analytical models. In addition, data will also be obtained from NUPEC for seismic proving tests of a

computer system and a reactor shutdown cooling system.

Generic Safety Issue Resolution

- During FY 1993, the NRC identified five new generic issues, established priorities for 12 issues (see Table 6.1), and resolved 10 issues (see Table 6.2). Table 6.3 contains the schedules for resolution of all unresolved issues.

Reactor Regulatory Standards

- In April 1993, the NRC conducted a public workshop on elimination of requirements marginal to safety. The purpose of the workshop was to provide information on the NRC program, solicit comments from the public and regulated industry on the program, and discuss a number of specific initiatives being considered.
- The NRC issued a draft report, "Proposed Regulatory Analysis Guidelines," (NUREG/BR-0058, Revision 2) for public comment in September 1993. The proposed guidelines represent the NRC's policy-setting document with respect to regulatory impact analyses (RIAs). In addition, a draft report, "Regulatory Analysis Technical Evaluation Handbook" (NUREG/BR-0184) was issued. The purpose of the handbook is to provide guidance to regulatory analysts, to promote preparation of high-quality RIAs, and to implement the policies of the guidelines.
- A final rule, 10 CFR Part 50, on training and qualification of nuclear power plant personnel was issued in April 1993. The rule as revised codifies existing industry practices related to personnel training and qualification and meets the directives of the Nuclear Waste Policy Act of 1982.
- A final rule and six regulatory guides relating to the effective implementation of the new 10 CFR Part 20 rule were issued in FY 1992. Three additional regulatory guides needed for the implementation of the revised Part 20 rule were issued in June and July 1993.
- A proposed rule, 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses was issued in May 1993. The proposed amendment would delete the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's 6-year license as a prerequisite for license renewal.
- A final rule, 10 CFR 50.65, on monitoring the effectiveness of maintenance at nuclear power plants was issued in June 1993. The rule requires that the

licensee monitor the performance or condition of certain structures, systems, and components (SSCs) against licensee-established goals so as to provide reasonable assurance that those SSCs will be capable of performing their intended functions.

- In June 1993, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which endorses a NUMARC guidance document, was issued.

Part 3--NUCLEAR MATERIALS LICENSING AND REGULATION SUPPORT

Nuclear Materials

- A proposed rule, 10 CFR Parts 30, 40, 50, 70, and 72, that would allow self-guarantee as an additional mechanism for financial assurance was published in January 1993. This proposed rule is in response to a petition for rulemaking submitted by the General Electric Company and Westinghouse Electric Corporation.
- A proposed rule, 10 CFR Parts 30, 32, and 35, on the medical use of byproduct material was published in July 1993. This action, taken in response to a petition for rulemaking, is intended to provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians greater discretion to prepare radioactive drugs containing byproduct material for medical use.
- A proposed rule, 10 CFR Part 73, on a physical fitness program for security personnel at Category I facilities was published for comment in October 1993.
- A final rule, Appendix H to 10 CFR Part 73, on day-firing qualifications for security personnel at Category I fuel cycle facilities was published in August 1993.
- A final rule, 10 CFR Part 73, amending the regulations covering the physical protection of special nuclear material at fixed sites, was published in March 1993.
- A final rule, 10 CFR Parts 26, 70, and 73, on fitness for duty for Category I facilities and shipments was published in June 1993.
- A final rule and a proposed rule, 10 CFR 72.214, adding two casks to the list of approved spent fuel storage casks, were published in April 1993.
- A final rule on licenses and radiation safety requirements for irradiators was published in February 1993. The rule established a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators.

Uranium Enrichment

- The Commission is reviewing the proposed rulemaking developed to amend 10 CFR Part 76, "Regulation Governing the Operation of Gaseous Diffusion Facilities." This rulemaking, required by the Energy Policy Act of 1992, would establish both the procedural and technical requirements for certification of the operation of the gaseous diffusion facilities by U.S. Enrichment Corporation.

Low-Level Waste Disposal

- A final rule to amend 10 CFR Part 61 to clarify that the requirements related to the performance of land disposal facilities for low-level waste are applicable to aboveground disposal (i.e., built on the ground without an earthen cover) was published in June 1993.
- A proposed rule on radiological criteria for decommissioning is being pursued under an Enhanced Participatory Rulemaking process. This process was initiated to actively solicit early input from a wide spectrum of interests. Seven public workshops were held across the United States, and these were followed by eight generic environmental impact statement (GEIS) scoping meetings in four cities. The EPA participated in these workshops and meetings and is a cooperating agency in the development of the GEIS.
- A final rule, 10 CFR Part 20, on disposal of waste oil by incineration at nuclear power plants was published in December 1992. The rulemaking action responds to a petition for rulemaking originally filed by Edison Electric Institute and the Utility Nuclear Waste Management Group.
- A final rule (10 CFR Parts 30, 40, 70, and 72) was published in July 1993 to amend the NRC'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.

Highlights

- A proposed rulemaking, 10 CFR Parts 30, 40, 70, and 72, on timeliness in decommissioning a materials facility was published for comment in January 1993.
- The NRC issued, in final form, NUREG/CR-5512, Volume 1, "Residual Radioactive Contamination from Decommissioning: Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," in October 1992.

Office Program

- The NRC supports the Small Business Innovation

Research (SBIR) program to stimulate technological innovation by small businesses, strengthen the role of small business in meeting Federal research and development needs, increase the commercial application of NRC-supported research results, and improve the return on investment from Federally funded research for economic and social benefits to the nation. Participation in this program has continued since the program was established in FY 1982. In FY 1993, the NRC was supporting 20 SBIR projects-in-progress.

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

1. REACTOR AGING AND LICENSING RENEWAL

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

1.1 Reactor Vessel Safety and Piping Integrity

1.1.1 Statement of Problem

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

and that sufficient critical experiments are conducted to validate those procedures and methods.

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

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

1.1.2 Program Strategy

The approach used for this element is to develop analytical procedures for predicting continuing integrity or conditions-for-failure and to ensure that an adequate experimental basis exists to validate those procedures. The most critical facet of pressure vessel integrity is embrittlement of the pressure vessel steel as a result of exposure to neutrons escaping from the fuel core during normal service. Experiments are conducted to develop a base of information on all the factors that will cause embrittlement to increase during service life. Much work is done to establish correlations between small-specimen behavior and thick-section behavior to ensure that the analyses performed to assess structural integrity are valid. Similarly, the ability to predict integrity in piping has required testing of full-sized sections of pipe having a variety of cracks to determine if such cracks could cause failure during either normal service or accident conditions. For both vessels and piping, knowledge of the rate at which cracks grow is very important to ensure that a component will not fail during its forthcoming operational period. Experiments are conducted on a wide variety of pertinent materials under a range of typical and expected exposure conditions to determine the maximum bounding rates of crack growth. Detection and sizing of flaws and cracks in all primary system components are conducted by the industry through periodic inservice inspections at shutdowns. To ensure that the inspections reliably detect and accurately size the flaws, extensive tests are conducted with inspection teams drawn from the industry using

1. Reactor Aging

typical equipment and techniques on samples whose flaw conditions are known. From the results, it is possible to determine which techniques are effective and the magnitude of the error bands for flaw detection and sizing. Improvements in methods are proposed and qualification procedures developed that can provide better assurance of flaw detection in future inspections and for sizing flaws more accurately. Materials and components removed from actual service are used to measure material properties after years of service, to evaluate the extent of corrosion, and to validate the existence of flaws that have previously been identified and had their size estimated.

1.1.3 Research Accomplishments in FY 1993

1.1.3.1 Pressure Vessel Safety

This area of NRC research focuses on ensuring the structural integrity of the reactor system pressure boundary, i.e., keeping it free from damage and leaktight. The underlying concern in ensuring the integrity of the pressure boundary is that failure to do so could compromise the operator's ability to cool the reactor core and could lead to a loss-of-coolant accident (LOCA) accompanied by the release of hazardous fission products.

The research program in this area is a broad-based program, initiated in 1967. The original program was focused solely on the properties and fracture behavior of the reactor pressure vessel—the large, thick-walled steel cylinder that houses and supports the reactor core. As the full challenge of ensuring the integrity of this critical component was realized, the scope of the research program was expanded to include irradiation damage, service-induced cracking mechanisms, and methods for periodically inspecting the pressure vessel.

The pressure vessel safety program is closely tied to the regulatory efforts to ensure pressure vessel integrity. The technical efforts in the research program—fracture analysis and radiation embrittlement—are central to sound regulatory positions addressing the safe operation of the pressure vessel. For example, efforts to revise the basis for determining the allowable operating pressure and temperature to preclude brittle failure of the pressure vessel are drawing on research results from the pressure vessel safety program.

1. *Fracture Analysis.* The part of the pressure vessel safety program that includes fracture analysis methods has had a particularly large role in the overall program during FY 1993. The fracture analysis work involves an ongoing program to develop, and reduce to practice, advanced analysis methods that will improve the ability to predict the allowable pressures and temperatures for the pressure vessel and the ability to evaluate the integrity of the pressure vessel

under design basis and hypothetical accident conditions. Basic work is being performed by researchers at the Oak Ridge National Laboratory (ORNL), with additional efforts being performed by researchers at Brown University, the University of Illinois, Texas A&M University, and the U.S. Navy's Naval Surface Warfare Center (NSWC), Annapolis Detachment. These researchers are developing state-of-the-art analysis methods and evaluating those methods against test data developed as part of this program by ORNL, the National Institute for Standards and Technology (NIST), and the NSWC. The initial work has been very promising, and the program has been continued to permit evaluation of test geometries that are more typical of reactor pressure vessels. Additionally, the researchers are coordinating their work with international efforts in this technical area. Collaborative efforts with a European Community program are expected to provide results from a test that will closely simulate a reactor pressure vessel subjected to accident loads. This will provide a more realistic validation of the revised analysis methods.

During FY 1993, the results of several efforts were put to use in performing generic analyses of reactor pressure vessels fabricated from materials with a low resistance to a "ductile tearing" failure mode. In the early 1970's, the NRC recognized that some pressure vessels were fabricated using steel plates and some weld materials that did not provide the high resistance to this failure mode exhibited by most of the plates, forgings, and welds used in reactor pressure vessels. The NRC issued Appendix G to 10 CFR Part 50 in 1973 to provide explicit requirements on the Charpy upper-shelf energy—a measure of the ductile tearing resistance of these materials—for both new construction and for operating plants. However, it was recognized that some of the early vessels did not meet these requirements. Therefore, Unresolved Safety Issue A-11, "Reactor Vessel Materials Toughness," was established to develop methods to evaluate the integrity of pressure vessels that did not satisfy the Appendix G requirements. That issue was resolved in 1982 with the publication of "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue" (NUREG-0744). A key aspect of the overall resolution was a request to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to recommend criteria that would satisfy the Appendix G requirement to demonstrate margins of safety equivalent to those in the ASME Code. The Code committee responded in February 1991 with a set of evaluation criteria that were reviewed and accepted by the NRC. The Code also developed an analysis method (Code Case N-512) that was similar to the NUREG-0744 recommended method but that had the benefit of several years of review and experience by the Code

committee members and by the NRC staff. Unfortunately, Code Case N-512 (Section XI, Division 1, February 1993) did not address complete details of all the potential loading conditions for reactor pressure vessels, nor did it include guidance on determining appropriate material properties for use in the evaluation method.

The research staff undertook an effort to address the shortcomings of the Code Case during FY 1993. These staff efforts include development of a draft regulatory guide that would expand the Code Case's guidance to include evaluation methods pertinent to all service loading conditions and would provide specific guidance on estimating material properties. In developing this draft guidance, the staff drew on results from past NRC-funded research efforts at ORNL and the Pacific Northwest Laboratories (PNL) to provide a comprehensive fracture analysis methodology. Additionally, results from a Phase II Small Business Innovative Research (SBIR) program were used as an acceptable method for estimating the material properties. The draft regulatory guide was published at the end of September 1993.

In addition to developing this draft regulatory guide, the research staff worked with researchers at ORNL to perform generic analyses using the draft regulatory guide methodology in support of the regulatory staff efforts to evaluate responses to Generic Letter 92-01, "Reactor Vessel Structural Integrity." This work demonstrated that, on a generic basis, adequate pressure vessel integrity was ensured even with Charpy upper-shelf energy values below the value specified in Appendix G to 10 CFR Part 50.

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

During FY 1993, the radiation embrittlement research efforts moved forward on several fronts. Test reactor irradiations were initiated by ORNL, using the University of Michigan test reactor, to evaluate the effects of neutron radiation on weld materials

removed from the cancelled Midland Unit 1 reactor pressure vessel. The materials being irradiated are representative of the so-called "limiting" material in several operating nuclear power plants. Additionally, these materials are being irradiated in the surveillance programs of an operating power plant as part of an NRC-industry coordinated research effort. When the results from each of these programs are available in the late 1990's, they will provide important information about the embrittlement trends for these materials and equally important information about the differences between test reactor and power reactor irradiation conditions and about the mechanisms controlling embrittlement of these materials.

During FY 1993, ORNL processed the updated Evaluated Nuclear Data File, Version B-VI, to develop neutron cross-section libraries that can be used in evaluating the neutron fluence for power reactors. These cross-section libraries are needed to predict the neutron fluence, which is an essential input in estimating the level of radiation embrittlement for reactor pressure vessels. The work will be completed in early 1994, and the updated cross-section libraries will be available for use shortly thereafter. In addition to the cross-section library work, researchers at ORNL have worked with researchers in the Czech Republic, and other East European researchers, in performing calculations to predict the results of carefully controlled "benchmark" experiments conducted by the Czech researchers. This continuing work is providing important data to the NRC's program to validate neutron fluence calculation methods and is providing technology transfer and validation of the methods being used by the different laboratories. This work contributed to the staff's effort to prepare a draft regulatory guide on "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," which was published for public comment in September 1993.

Work continued in FY 1993 to compile and evaluate embrittlement trends using the power reactor pressure vessel material surveillance data. These data are reported to the NRC in accordance with Appendix H to 10 CFR Part 50 and reflect embrittlement trends for reactor pressure vessels irradiated under typical power reactor conditions. The ORNL work to compile these data into a comprehensive data base has provided the basis for work by Modeling and Computing Services to develop statistically based models for predicting radiation embrittlement. Additionally, the ORNL data base has been used by the regulatory staff in both plant-specific and generic evaluations. The ORNL work is a continuing effort while the Modeling and Computing Services work is expected to be completed in late 1994 or early 1995.

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This work will enable the NRC to evaluate the need for further revision to Regulatory Guide 1.99, which provides the methods for estimating radiation embrittlement and is a fundamental part of the NRC's approach to ensuring pressure vessel safety.

The research to understand the mechanisms of radiation embrittlement continued in FY 1993, with significant advances being made by the University of California at Santa Barbara (UCSB) and ORNL, in conjunction with researchers in the United Kingdom, in modeling the complex interactions among the impinging neutrons and the atoms in the pressure vessel steel. This work is closely integrated with the experimental work being done at UCSB, at ORNL, and in Europe. Understanding the controlling mechanisms is essential to confidently extrapolate the empirical models of radiation embrittlement to unique plant-specific operating circumstances. The progress in the mechanisms research is providing assurance that the empirical models are conservative and is helping to define the limits of extrapolation for those models.

The interactions with researchers in Russia under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCCNRS) Working Group 3 were continued in FY 1993. The scope of the continuing efforts has expanded from radiation embrittlement to include the general subject of pressure vessel integrity analysis methods. During FY 1993, coordinated test reactor irradiations were continued in the United States and in Russia, and ORNL hosted a visiting scientist on a 1-year assignment to investigate radiation embrittlement and thermal-annealing effects. The Russian scientist's work at ORNL has provided valuable information and data concerning the ability of thermal annealing to mitigate the effects of radiation embrittlement. Work in this area is continuing.

The interactions with the Russians have provided first-hand examination of their procedures for conducting thermal-annealing treatments of their reactor pressure vessels. Additionally, the NRC started research into the metallurgical effects of thermal annealing many years ago and in the mid-1980's funded work to examine the engineering feasibility of this process for U.S. plant designs and conditions. During FY 1993, the research staff made use of this experience and the results of prior research to draft a regulation and regulatory guide addressing both the engineering and metallurgical aspects of thermal annealing for U.S. plant designs. The performance-based regulation and guidance are undergoing final internal reviews and should be published for comment in early 1994.

1.1.3.2 Steam Generator Tube Integrity

Steam generator tube integrity and identification of degradation continues to be an important area in the research program. The thin-walled steam generator tubes are an integral part of the reactor system pressure boundary. Tube failures could lead to a LOCA and containment bypass resulting in the escape of radioactive fission products directly to the environment. During FY 1993, the research staff worked closely with the regulatory staff in the review and evaluation of industry-proposed interim (or alternative) criteria for plugging those tubes where stress-corrosion-cracking degradation has been detected at the tube-to-tube-support-plate intersections. The research staff developed a mechanistically based model for predicting tube failure and cumulative leak rate of degraded tubes under normal and main steam line break conditions. The model was useful for comparisons to leak rates predicted by the industry's statistically based model and for gaining an understanding of the important parameters that influence failure and leak rates. This model was used by an NRC Task Group on Interim Plugging Criteria, which was charged with evaluating the industry proposal. The research staff and research contractors from the Argonne National Laboratory (ANL), PNL, and ORNL contributed to the task group's effort to develop an NRC consensus position on interim plugging criteria for steam generator tubes, which was published in NUREG-1477. The research activities focusing on steam generator tube integrity are continuing and are focused on validation of inspection results, integrity prediction methods, and leak rate estimation methods.

1.1.3.3 Piping Integrity

The piping integrity research program has been an important part of the overall pressure boundary integrity research program for many years. It took on heightened importance in the early 1980's with the emphasis on intergranular stress corrosion cracking (IGSCC) in BWR piping systems, and the work on environmentally assisted cracking and piping fracture behavior became high-priority activities. Over the last decade, research on IGSCC has largely been completed and has contributed to the NRC's position on this issue. The pipe fracture research is drawing to a close, with the work during FY 1993 being focused on conducting large-scale pipe fracture tests to provide final validation of the methods used by the NRC and by the ASME Code to ensure that piping will not fail under accident conditions. Further, work to evaluate the significance of the reactor operating temperature on the fracture behavior of cast stainless steel piping and components was completed during FY 1993, and the results of the research have been provided to, and are being used by, the regulatory staff in plant-specific evaluations. Also, the research program on environmentally assisted cracking has been developing data on the effects of the water

coolant on the fatigue life of pressure boundary components.

Environmentally Assisted Cracking. Irradiation-assisted stress corrosion cracking (IASCC) of core internal components fabricated from solution-annealed austenitic stainless steels and high-nickel alloys has been observed in both BWRs and PWRs. Failures caused by IASCC have become more numerous as reactors age and core materials accumulate higher fluence. Although many of the affected components can be replaced, some safety-significant components would be difficult or impractical to replace. Additional tests were performed in FY 1993 to compare the susceptibility of conventional commercial purity type 304 stainless steel, which is the material currently used in most reactors, with special high-purity heats of type 304 stainless steel—materials that have been suggested as a replacement for the current materials. Both types of materials were irradiated in commercial power reactors and, contrary to expectations, the susceptibility of high-purity materials to IASCC was higher than those of the commercial purity materials. Microstructural evaluations of the irradiated materials are in progress to determine the underlying mechanisms for this mode of material degradation. Numerous heats of stainless steel with a range of compositions are being irradiated in the Halden reactor in Norway as part of an investigation of material compositions and water chemistries that promote susceptibility to this type of cracking.

Fatigue is a potentially significant damage mechanism in LWR pressure boundary system components. The repetitive nature of the service conditions in nuclear power plant systems, particularly the piping systems, lead to progressively greater damage. This phenomenon has been recognized for many years, and design criteria were included to account for it. The early design Codes provided a less rigorous fatigue analysis than does the current design Code. However, recent data suggest that the effects of the water coolant on the expected fatigue life of typical materials are not adequately addressed by the design Codes—past or present. Since no consensus design procedure is available, data from ongoing tests and from the literature were evaluated during FY 1993 to identify the key environmental variables that influence fatigue life and to define the material and loading conditions for which additional information is needed. During FY 1993, interim design curves were developed and published that more adequately describe fatigue life in the high-temperature water characteristics that could exist in LWR coolant systems. Fatigue tests are in progress to validate and/or update the proposed design curves.

1.1.3.4 Inspection Procedures and Technologies

The NRC's approach to ensuring the integrity of the reactor system pressure boundary builds on the overall defense-in-depth concept. Research activities parallel this

approach and include programs addressing each of the major considerations in providing structural integrity—analysis methods, material properties, and inspection techniques. The research program for inspection procedures and technologies provides an independent basis for evaluating the efficacy and reliability of industry inspection techniques. The program includes studies of improved methods for selecting components for inspection and strategies for setting the sample size and inspection intervals to provide a reliable overall inspection process. The program also includes the inspection technologies and methods necessary to ensure reliable detection and accurate sizing of flaws. Finally, the program includes a focused effort to transfer this technology to practitioners in the NRC regional and headquarters offices.

Cast Stainless Steel Inspectability. Cast stainless steel has been widely used in piping systems and pipe components in both the nuclear and non-nuclear industry. Owing to its unique microstructural characteristics, this material is difficult to inspect reliably. During FY 1993, PNL researchers developed and evaluated a technique that is based on equipment and procedures developed in earlier NRC research, e.g., the use of specially designed low-frequency transducers coupled with the Synthetic Aperture Focusing Technique for Ultrasonic Testing (SAFT-UT). The technique was evaluated in a blind test coordinated by the Electric Power Research Institute (EPRI) in conjunction with Northeast Utilities. The PNL technique showed improved results over those currently being used and ranked in the top group of techniques tested. The majority of cracks in the samples were detected, but the "false call" rate was higher than desirable. Further work is planned to refine the technique to improve crack detection and to lower the false call rate.

Risk-Based Inspections. The NRC has funded research to evaluate methods for establishing an overall inspection program—components to be inspected, sample size, and inspection frequency—based on risk concepts. The successful application of the risk-based methods in a demonstration project using the Surry Unit 1 (in Va.) nuclear power plant led the ASME Code to establish a working group under the subcommittee responsible for Section XI (the inservice inspection section of the Code) to consider changes in the Code's approach.

During FY 1993, the NRC supported the working group's efforts through active participation by the research staff and researchers from PNL. The working group has moved quickly to develop initial proposals to revise Code criteria for the inspection of pressure vessels and piping. The expected benefits of implementing risk-based inservice inspection programs include enhanced plant safety and the elimination of costly inspections, particularly those inspections that involve higher levels of occupational radiation exposures when inspecting "low risk" components. Completion of this work will provide Code revisions that

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promote reliable detection of defects before degradation can compromise the structural integrity of "risk important" components.

Surface Roughness Evaluation. As inspection capabilities have improved, and emphasis has shifted toward reliable detection of flaws, the surface roughness and local geometry conditions of the region being inspected have become more important. However, there are no ASME Code requirements for surface conditions for inservice inspections. The surface conditions for plant components, which were specified by the Code for construction radiography, can vary appreciably. To address this issue, the NRC has entered into a cooperative program with EPRI.

Working under EPRI funding, the Iowa State University Center for Nondestructive Evaluation is developing computer codes for four different inspection modes to describe ultrasound interactions and responses. Working under NRC funding, PNL will experimentally validate and use the computer codes to assess various surface conditions. These studies will form a technical basis for proposing Code requirements for surface roughness and changes in geometry. Two of the four codes were completed and experimentally validated in FY 1993, and evaluations of acceptable surface conditions were started.

International Reliability Studies. The NRC is an active participant and leader in the Program for the Inspection of Steel Components, Phase III (PISC III). This international program, organized in 1986 by the Community of European Countries (CEC) and the Committee for the Safety of Nuclear Installations of the Organization for Economic Cooperation and Development's Nuclear Energy Agency, is assessing the effectiveness of nondestructive testing technologies and procedures for inservice inspection (ISI) of nuclear power plant components. The CEC and participating organizations have invested an estimated \$40 million in the program, including contributions of material, inspection services, and manpower. The products from this program will assist regulators and Code bodies in establishing technical bases for improving ISI requirements.

The focus of the PISC III program is on the nondestructive testing of realistic LWR primary circuit components containing realistic flaws. The experimental portion of the PISC III program was completed during calendar year 1993, and the results will be released during a symposium to be held in the spring of 1994. As the results become available, the NRC will assess the adequacy of current ISI requirements and, as appropriate, will recommend changes to the ASME Code and to NRC regulations and regulatory guides.

1.2 Aging of Reactor Components

1.2.1 Statement of Problem

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

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

1.2.2 Program Strategy

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

1. Identify and characterize aging and service wear effects that, if unmitigated, could cause degradation of structures, systems, and components and thereby impair plant safety.
2. Develop methods of inspection, surveillance, and monitoring and of evaluating residual life of structures, systems, and components that will permit compensatory action to counter significant aging effects prior to loss of safety function.

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

1.2.3 Research Accomplishments in FY 1993

1.2.3.1 Aging Research

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

The Nuclear Plant Aging Research (NPAR) program provides technical information useful in understanding the effects of aging on the safety functions of electrical and mechanical components of commercial nuclear power plants. During FY 1993, preliminary or comprehensive aging assessments were completed or final reports were issued for the following safety-related components, systems, and associated special topics:

- BWR Internals (NUREG/CR-5754)
- PWR Internals (NUREG/CR-6048)
- Check Valves (NUREG/CR-5944)
- Auxiliary Feedwater System (Follow-on Study) (NUREG/CR-5404, Vol. 2)
- Aging, Condition Monitoring, and Loss-of-Coolant Accident (LOCA) Tests of Class 1E Electrical Cables (Ethylene Propylene Rubber Cables—NUREG/CR-5772, Vol. 2, and Miscellaneous Cables—NUREG/CR-5772, Vol. 3)
- Recordkeeping (NUREG/CR-5848)
- HEPA Filters and Adsorbers (NUREG/CR-6029)
- Essential HVAC Chillers (NUREG/CR-6043)
- Fans and Blowers
- Impact of Aging on Accident Precursors
- Standard Technical Specifications Aging Evaluation

Technical Information for License Renewal. License renewal is a high-priority activity for the NRC and for the nuclear power industry. The NPAR program is providing technical

information on equipment aging that may be considered for the implementation of the license renewal rule.

Standard Technical Specifications Aging Assessment. The NPAR program is evaluating the Standard Technical Specifications (STS) for selected nuclear power plant systems and components to determine the effectiveness of current surveillance requirements (SRs) to detect age-related degradation effects. The purpose of these SRs is to ensure the operability and availability of safety-related systems and components by verifying and demonstrating that they are capable of performing their required functions. Aging effects have not always been recognized or addressed explicitly in the SRs; many significant forms of aging degradation may not be detected prior to failure and, in some cases, the test methods and frequency of testing may even contribute to premature degradation.

The STS aging assessments were completed during FY 1993 for check valves, the auxiliary feedwater system, and the reactor protection system. These assessments reviewed the current surveillance and testing requirements to determine their effectiveness in detecting degraded conditions in systems, structures, and components (SSCs) prior to failure; to determine if the current surveillance and test methods and frequency of testing contribute to premature aging degradation; to identify, based on the results of the indepth aging assessments of specific systems and components conducted under the NPAR program, the parameters or indicators useful to monitor the degraded state of SSCs; and to develop recommendations for inspection, surveillance, trending, and condition monitoring methods to be incorporated as part of SRs to evaluate age-related degradation. Work was also initiated to combine the 10 STS aging assessments completed into a NUREG/CR report.

Aging Management NUREG Update. For several years the NPAR program has been developing technical understanding of the processes that, through time-dependent age-related degradation of SSCs, could reduce operational safety margins in operating nuclear power plants below acceptable limits. The results from the NPAR program, and other complementary aging management programs, were compiled and critically reviewed in a draft report, "A Review of Information Useful for Managing Aging" (NUREG/CR-5562). This draft report has been a valuable resource for the NRC staff in such tasks as preparing the draft standard review plan for the review of license renewal applications for nuclear power plants and in the review of aging issues related to license renewal.

NUREG/CR-5562 was extensively revised and updated during FY 1993 to provide additional insights and technical guidance for aging management, including (1) identifying SSCs in which age-related degradation should be managed, (2) understanding aging mechanisms and identifying degradation sites in these SSCs, and (3) managing

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degradation through effective monitoring and maintenance programs or by modifications to operating conditions.

Information Management. The aging assessments conducted by the NPAR program have generated an extensive volume of valuable information on aging processes and effective methods for detecting and mitigating aging degradation in safety significant systems and components. Improved methods are needed to make this information readily available to the NRC regulatory staff, nuclear power plant licensees, other government agencies, and the public. This need was addressed during FY 1993 by developing a demonstration electronic document. Technical and regulatory information can be rapidly and efficiently computer searched for specific words and topics. The electronic document also includes a description and aging overview for the systems and components of interest, a compilation of the aging assessment information, and colored line drawings that illustrate the various types and components where aging mechanisms are operative.

1.2.3.2 Components, Systems, and Facilities

Engineered Safety Features. Engineered safety feature systems are systems to control and mitigate specific occurrences that might challenge the integrity of the reactor and/or adversely affect plant personnel or the general public. Issues related to the aging and service wear of these systems at commercial reactor facilities could impact both public and plant safety. NPAR aging assessments have been conducted for nuclear air treatment and cooling system fans and for the high-efficiency particulate air (HEPA) filters and activated carbon beds that remove radioactive particulates and volatile radionuclides that otherwise could be substantial contributors to public dose. The fan aging assessment revealed that aging degradation appears to be an important factor when considering fan failure and that breakdown can impact both plant and public safety. Details compiled from surveys containing information concerning aging and wear effects suggest that bearings are the component most frequently linked to fan failure. The investigation also indicated that monitoring techniques that will detect irregularities arising from improper lubrication, cooling, alignment, and balance will aid in counteracting many of the aging effects that could impair fan performance. The aging assessment of the HEPA filters and activated carbon adsorbers identified heat, moisture, radiation, airborne particles, and contaminants as key stressors, with resulting aging mechanisms and degradation ranging from particle loading to degraded sealant and gasket properties. This work is documented in "Phase I Assessment of Nuclear Air Treatment System HEPA Filters and Adsorbers" (NUREG/CR-6029).

Chillers. Essential chillers are required in nuclear plants to cool rooms, such as the main reactor control room, that

contain safety-related equipment that is essential to plant safety. Without cooling, control room temperature can rapidly rise leading to operator stress and causing the electronic equipment to give erroneous readings and spurious alarms and to start to fail. The newer digital controls are even more sensitive to high temperatures than the older analog controls. An NPAR aging assessment of these essential chillers is in progress. The initial assessment results are documented in "Aging Assessment of Essential Chillers Used in Nuclear Power Plants" (NUREG/CR-6043). This work shows that chillers are affected by vibration, excessive temperatures and pressures, thermal cycling, chemical attack, and poor quality cooling water. Aging is accelerated by moisture and non-condensable gases such as air, dirt, and other contamination within the refrigerant containment system; by excessive start/stop cycling; and by operating below the rated capacity. The primary cause of chiller failures is a lack of adequate condition monitoring and failure to perform scheduled maintenance. The comprehensive assessment now in progress will identify actions that could help reduce chiller failures and premature aging through effective monitoring and preventive maintenance programs; develop effective procedures for maintaining the reliability of essential chillers used in all modes, including the emergency standby-only mode; and provide guidelines for a safer and more effective transition to the mandated new chiller refrigerants.

Service Water System. The NPAR service water system aging assessment was completed with the publication of "Nuclear Service Water System Aging Degradation Assessment" (NUREG/CR-5379). However, an advanced power plant monitoring and diagnostic system, which was developed as an outgrowth of the NPAR service water system aging assessment, has been successfully tested at a U.S. Marine Corps base.

Emergency Diesel Generators. Although the emergency diesel generator aging assessment was completed earlier, there is continuing work related to the incorporation of the findings and recommendations in relevant codes and standards such as those of the Institute of Electrical and Electronic Engineers (IEEE). Guide document P-1205, "Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations," was published as the result of Working Group 3.4 effort. It included an appendix on diesel generators derived from NPAR information. From Working Group 4.2, IEEE Std. 387, "Standard Criteria for Diesel-Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations," has been approved for final balloting. This new revision of the standard includes the aging results and information from the NPAR research.

Aging Assessment and Mitigation of Major LWR Components. Intrinsic to the general exploration of reactor aging

is the assessment and mitigation of aging damage to major components and structures. The objective of this aging assessment task, an element of the NPAR program, is to identify, develop, and evaluate various aging management techniques for the major LWR components and structures. The approach is to gauge the degradation of the major LWR components and structures by the synergistic influences of radiation embrittlement, thermal fatigue, stress corrosion cracking, thermal embrittlement, erosion corrosion, and so forth.

Research completed in this area in 1992 focused on developing insights for aging management for selected LWR components and structures to ensure continued safe operation. The studies also included the evaluation of advanced inspection and monitoring methods for characterizing the aging damage. The results will be useful to the NRC in identifying and resolving safety issues associated with LWR aging degradation and developing policies and guidelines for making operation plant aging management decisions that may safely extend its operation. The major components assessed in 1993 are the LWR metal containments and the LWR reinforced and prestressed concrete containments. PWR reactor pressure vessels and the PWR coolant piping research will continue. Results of these assessments are being documented in a multivolume report, NUREG/CR-5314.

In addition to the above efforts, a draft report (NUREG/CR-5824) that discusses the identification of advanced monitoring methods for estimating stresses causing fatigue damage was completed.

PRA-Based Methodology for Aging Assessments and Component Prioritization. The risk-based methodology for assessing aging in nuclear power plants and for defining priorities among risk contributions and maintenance activities (published in previous years, NUREG/CR-5587 and NUREG/CR-5510) is subject to uncertainties due to limited available aging data and due to various modeling assumptions (e.g., modeling of effects of test and maintenance). To address the validity of results obtained from the risk-based approach, research in 1993 focused on developing sensitivity and uncertainty analyses to define priorities addressing data and modeling uncertainties. This work was documented in draft NUREG/CR-6045 in 1993.

The application of age-dependent risk methodology requires age-dependent component failure rates; however, these age-dependent component failure rates are not generally available and need to be estimated from limited recorded plant failure data and plant maintenance logs. A major limitation of the age-dependent methodology has been the lack of recorded component aging data and approaches to develop aging failure rates based on the available information. To address this limitation, an approach

was developed in 1993 to incorporate age dependence in PRAs that does not require absolute age-dependent component failure rates. Instead, the aging of a component is expressed in terms of relative aging rates that are found to be fairly constant across different components and different plants. A draft report (NUREG/CR-6067) was completed on the proposed aging data assessment methodology.

Also in 1993, an important application of the risk-based methods resulted in the development of PRA-based approaches for identifying safety-related motor-operated valves (MOVs) having the most impact on plant risk covered under Generic Letter 89-10, "Safety-Related MOV Testing and Surveillance." Dynamic tests and surveillance tests, in accordance with GL 89-10, could then be performed on those MOVs having the largest risk impact. The risk importance of single MOVs and the interaction of multiple MOVs can be analyzed using this approach. A draft NUREG/CR documenting the results of this work is being prepared.

In addition to the above-described effort, work was initiated in 1993 to identify and prioritize, based on their risk significance, environmental stressors associated with advanced digital instrumentation and control (I&C) systems in nuclear power plants. Analog I&C systems in nuclear power plants are becoming obsolete and are being replaced by digital systems. Digital I&C systems are vulnerable to common environmental stressors, e.g., moisture/humidity, temperature. The effects of each environmental stressor are being identified and importance measures are being developed to rank the stressors. These risk-based approaches are being tested using plant-specific PRAs.

Aging of Passive Components. In earlier efforts, a methodology was developed to include the effects of aging on passive components (pipes, structure, 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. The failure calculation can be incorporated in a PRA for the plant that will calculate the effects of this failure on plant risk. During this fiscal year, approaches were investigated that can be applied to a large number of passive components that exist in a nuclear plant. A screening approach can be used to identify those components that age and contribute most to risk. Two approaches were investigated. The first is a simple probabilistic structural analysis approach and the second is an approach called failure attributes. The simple probabilistic structural analysis is an approximation of the large, complex structural probabilistic computer codes. The second approach uses the attributes that have been shown to most affect aging and failure. These approaches, including a screening approach, will be documented in a report in FY 1994.

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The draft report (NUREG/CR-5730) that describes the methodology for including the effects of aging on passive components was reviewed internally by other NRC research staff, and responses to their comments have been developed. A revised NUREG/CR-5730 is almost completed and will be recirculated for general internal NRC review. Calculations were also completed to investigate the effects of passive components on the risk of containment failure. An approach was developed to identify the passive components that most contribute to risk.

Aging Effects on Motor-Operated Valve Performance. In 1993 an investigation was initiated to assess the effects of aging on the operability of MOVs and to identify those safety-related MOVs in typical PWR and BWR plants that are most susceptible to internal or external environmental aging effects. The investigation has drawn from the Nuclear Plant Aging Research reports by ORNL on valve aging and from the reports by the Idaho National Engineering Laboratory on high-pressure injection systems. The current investigation includes the review of the inservice testing programs for four BWR and five PWR plants to identify safety-related MOVs. From this review it has been determined that the majority of the safety-related valves are either butterfly, globe, or gate valves constructed of carbon steel. Internal and external environmental conditions are being combined with valve type and material of construction to identify those most susceptible to aging effects. Final identification will also include the safety significance as well as maintenance and testing routinely performed. A draft contractor NUREG report is being prepared to report the results of the investigation. Also, experiments are under way to compare valve sliding surface friction factors for corroded and uncorroded valve material specimens and measurement methods to detect the changes being evaluated.

Check Valve Failure Data Characterization. A detailed review of historical check valve failure data for operating nuclear plants in the United States was conducted by ORNL. The results were published as NUREG/CR-5944. The source of data for the review was the Nuclear Plant Reliability Data System (NPRDS), operated by the Institute of Nuclear Power Operations (INPO). Narratives describing the failures of check valves during the period of 1984-1990 were initially reviewed to eliminate those failures that only involved external leakage because the purpose of the study was to better understand those failures that affect the internals of the valves. A total of 4,680 failure narratives were reviewed; of this total, 1,227 were identified as involving internal degradation and were further characterized.

The characterization of the failed valves included the following parameters:

- Valve size

- Valve age at failure
- Plant age at failure
- System in which the valve was used
- General operating status of the system
- Valve manufacturer
- Failure mode
- Extent of degradation
- Detection method
- Affected areas

The characterized data were analyzed for relative failure rates for each of the characterized parameters. Cross tabulations of the parametric data were also made. Some notable observations were:

- There was not a strong relationship between valve age and failure rate.
- The largest valves (> 10 inches) experienced about twice the failure rate of smaller valve sizes. The largest valve's failures were also most prone to be the most significant failures.
- The emergency service water, main feedwater, diesel starting air, and main steam systems experienced the highest failure rates.
- Just over half (54%) of the failures were detected by programmatic means, including inservice testing, surveillance testing, and other periodic inspection programs.
- Normally operating systems experienced only slightly greater failure rates than did systems that are used only during shutdown or only when tested.

The results of the study are being used by ASME Code Working Group OM-22 in support of code development activities. Follow-on studies for failures occurring during 1991 and 1992 are planned.

Detection of Pump Degradation. This Phase II Nuclear Plant Aging Research study examines the leading causes of pump degradation and describes the existing methods used in domestic and overseas nuclear facilities to diagnose pump problems. Research results are being published in the report, "Detection of Pump Degradation" (NUREG/CR-6089), which evaluates the criteria mandated by required pump testing at the United States nuclear power plants and compares them to those features that are characteristic of state-of-the-art diagnostic programs and practices currently implemented by other major industries. Degradation that is caused by low-flow pump operation is also discussed along with new analysis techniques that may be used to ascertain unstable flow. In

addition, since many pump operational problems can be attributed to the pump driver, motor current analysis methods are presented that can assist in the determination of specific kinds of motor degradation.

Vibration spectral analysis is widely accepted as a powerful diagnostic tool for determining numerous types of pump degradation, such as misalignment, unbalance, looseness, and various bearing anomalies. Many nuclear plant maintenance departments use vibration spectral analysis to diagnose pump problems. Thermography and lubrication analysis are other important diagnostic technologies that have made significant improvements within the past decade, both in their ease of application and their diagnostic capabilities. The next major thrust of development in diagnostic methodologies is focused on the development of expert systems. Low-flow operation, which was often performed by using minimum flow loops to conduct required ASME Code testing, has been observed to cause pump degradation through destructive low-flow phenomena. Motor power analysis techniques have also been developed that may assist in the determination of the onset of unstable flow conditions as well as enable the pump analyst to determine the most efficient operational ranges of a particular pump system.

Auxiliary Feedwater System. In the Phase I study, a thorough review of system controls and functions was performed, and several limitations of current maintenance and surveillance practice were identified, such as the failure to verify many safety-related control functions by periodic testing and the degradation of auxiliary feedwater (AFW) pumps by testing at low flow. The follow-on study categorizes the limitations in current monitoring/operating practice identified in Phase I and evaluates failure modes and component degradation caused by these practices. The Phase I findings have applicability to all plants in that they point out typical testing omissions or sources of degradation.

Significant conclusions of the study (NUREG/CR-5404, Vol. 2) are as follows: The present method of testing AFW pumps at the minimum flow condition may lead to degradation of the pump and also does not provide an indication of pump condition. The report discusses hydraulic instability at low-flow operation and provides examples from the industry of pump degradation. The report also provides head-capacity curves showing that mini-flow tests are inadequate for assessing pump capability or ability to operate at design basis conditions. An alternative method of testing is proposed that consists of testing at normal operating pressure to eliminate degradation and to verify flow at design conditions. Suggested changes in control logic testing, check valve monitoring, and emergency service water to AFW system line flow path testing are included.

Steam Turbine Drives. Steam turbine drives for safety-related pumps are used at most of the commercial nuclear power plants in the United States. Turbine-driven pumps are used in PWR AFW systems. Turbine-driven pumps are used at BWR plants in the reactor core isolation cooling and high-pressure coolant injection systems. The turbine-driven pumps provide a means of heat removal from the reactor coolant system in the event of station blackout.

Evaluation of failure records indicates that the turbine governor is the component that is most often involved in reported turbine failures. There are multiple sources of governor problems, with dirty or water-contaminated oil and setpoint drift accounting for a large percentage of the failures (33%). Oil problems (18%) can be mitigated by periodic chemical analysis of oil samples. Setpoint drift (15%) can be mitigated by periodic calibration of the governor. Currently there are no specific regulatory requirements for either chemical analysis or governor calibration, and current maintenance practices have not been consistent, although there are indications of improvement in recent years in that regard. The results of the study are being published as NUREG/CR-5857.

Cables. The research program conducted at the Sandia National Laboratories on the aging effects on electrical cables was completed in 1993 and the three-volume final report (NUREG/CR-5772, Vols. 1, 2, and 3) was issued. This program evaluated the capability of cable types commonly found in operating nuclear power plants to meet equipment qualification standards for periods of 20, 40, and 60 years of operation. Several of the cables tested in this program failed at a fairly early age. NRC Information Notice 93-33 summarized the cable test results and alerted the utilities to possible deficiencies. Research on the aging degradation of cable connectors and penetrations is currently under way.

Control Rod Drive Systems. The Babcock & Wilcox (B&W) and Combustion Engineering (CE) control rod drive (CRD) systems consist of mechanical and electrical components that position the control rod assemblies in the core in response to automatic or manual reactivity control signals. Both systems are designed to allow rapid gravity insertion of the control rods upon removal of the ac power that holds the rods. This study examined the design, materials, maintenance, and operation of the system to assess the potential for age degradation.

A detailed operating experience review highlighted age-related component degradation and failures that significantly affected plant operation. These effects include power reductions, reactor shutdowns and scrams, and engineered safety feature actuation. While there have been no system failures, component failures and degradation resulted in increased component stresses and

1. Reactor Aging

unnecessary thermal and pressure cycles that challenged other plant systems.

The majority of component failures in the CE control drive system were caused by the degradation of the control system (61%). Failures of the CRD mechanisms accounted for 60 percent of reported failures of the B&W control drive system. Aging was the direct failure cause for 40 percent of the CE power and control system and 55 percent of the B&W CRD mechanisms. The operating and environmental stresses for the system, and the aging effects resulting from continued exposure to these stresses, were evaluated for the major system components. Detailed failure modes and effects analyses were performed for the subsystems. In addition, a survey was made of the current surveillance, inspection, monitoring, and maintenance practices of utilities.

Effects of Solar Geomagnetically Induced Currents on Plant Electrical Systems. Transient disturbances in the earth's magnetic field caused by auroral currents can induce electrical potential gradients across the earth's surface. These potentials act like quasi-dc voltage sources impressed between the grounded neutrals of transformers at opposite ends of power transmission systems. To study the effects of these geomagnetically induced currents (GICs) on plant equipment, a plant-specific electrical distribution system for a nuclear power plant was modeled using the ElectroMagnetic Transient Program (EMTP). The model simulated on-line equipment and loads from the station transformer in the switchyard to the safety busses at 120 volts, to which all electronic devices are connected for plant monitoring.

The EMTP analysis used the half-cycle saturation of the station transformer (due to GIC) and studied the effects on the voltage harmonic levels that were noted at various electrical busses. The results indicate that the emergency circuits appear to be more susceptible to high harmonics due to normally light load conditions. Protective relays (both electromagnetic and solid state relays) without an harmonic filter, which operate purely as a peak detector, are vulnerable to false high readings with GICs as low as 50 amps present in the system. Based on these results, an input side harmonic filter can be used on undervoltage protective relays, which senses an undervoltage or degraded voltage condition for starting the onsite diesel generators. The report includes other parametric studies on the subject and discusses potential harmonic effects on the uninterruptible power system.

Equipment Operability (Mechanical). In the past, MOVs have had a poor performance history. Because of this poor performance, the NRC in 1989 issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance." The purpose was to ensure that licensees take appropriate steps to ensure all safety-re-

lated MOVs will perform their intended functions over the life of the nuclear power plants.

Earlier NRC research on MOV performance revealed deficiencies in the accepted industry formula for calculating/predicting required MOV thrusts for controlling flows through the MOVs. These results also contributed to the issuance of GL 89-10. However, subsequent NRC research resulted in a modified and more accurate formula for predicting the required MOV thrust values. These latter results and other findings supported the issuance of other regulatory documents by the NRC. In 1993, additional effort was expended on refining the modified formula for the application of extrapolating low-flow thrust requirements to high-flow thrust requirements, which is a process commonly employed by licensees.

Since the issuance of GL 89-10, the NRC has been performing audits and inspections of licensee MOVs to evaluate compliance with GL 89-10. This implementation requires a large effort on the part of both the licensees and the NRC because of the large number of safety-related MOVs in each nuclear power plant. During the implementation and auditing efforts, additional MOV deficiencies in predicting electric motor performance have been identified. Therefore, in 1993 the NRC started conducting experiments and analyses to resolve these newest deficiencies. Specifically, technical data bases are being developed to determine how the motor outputs are affected by increased loadings and by increased temperatures. This information is important since the entire MOV performance is dependent on the efficiencies of these valve power sources.

The licensees are supporting a large program to develop a basis for evaluating in-plant MOV performance without testing the MOVs. During 1993, the NRC exchanged technical information with industry researchers and continued reviewing the applicability and accuracy of their results. In addition, exchanges of technical findings were made with agencies in the United Kingdom and Germany.

The NRC efforts through contractor staff continues to be one of the leaders in MOV diagnostic research. As a result, current and past research results are contributing to improving MOV reliability and plant safety in the United States and abroad.

1.2.3.3 Engineering Standards Support

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

1982, which provides policies for Federal participation in the development and use of voluntary standards.

The NRC staff is particularly active on ASME codes and standards writing committees because portions of the ASME Boiler and Pressure Vessel (B&PV) Code have, since 1971, been incorporated by reference into 10 CFR 50.55a to establish requirements for the construction, in-service inspection, and inservice testing of nuclear power plant components. Section 50.55a is periodically amended to update the references to include more recent versions of the ASME B&PV Code. In 1993, work continued on rulemaking, initiated in 1992, that not only would update the reference to the ASME B&PV Code, but would for the first time incorporate by reference the new ASME Operations & Maintenance (O&M) Code, which provides rules for inservice testing of pumps, valves, and snubbers. The proposed rulemaking would expedite implementation of certain new ASME B&PV Code requirements for qualification of personnel and equipment used to perform inservice nondestructive ultrasonic examinations on nuclear power plant components.

1.2.3.4 Structural Integrity

As with all the structural integrity research, timely and effective transfer of the results to the NRC practitioners is a high-priority activity. For the inspection procedures and technologies research program, an inter-office Technical Advisory Group has been established to provide input to the research plans and to facilitate transfer of the research results.

During FY 1993, the research program provided both resource reports and direct support to the regulatory staff. Specifically, a document on the fundamentals of computer-based ultrasonic systems was prepared to serve as a resource on the technology and to describe the methods for characterizing computer-based ultrasonic systems. Additionally, a review was completed that will be published as a supplement to the resource document to provide a systematic evaluation of the computer-based system called P-SCAN.

Direct support was provided to the regulatory staff by researchers at PNL who assisted in evaluating inspections to detect and characterize cracks that were discovered in the core shroud of a BWR. This application posed difficulties because of severe access limitations, and the ultrasonic system employed was a modification of a computer-based system for inservice inspection of piping. The PNL researchers were able to provide the benefit of their considerable experience in developing and applying unique inspection systems.

Concrete structures play a vital role in the safe operation of all light-water reactor plants. In general, the perform-

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

The structural aging (SAG) program is addressing the aging management of safety-related concrete structures in nuclear power plants for the purpose of providing improved technical bases for their continued service. The SAG program objective is to prepare documentation providing (1) identification and evaluation of the structural degradation processes; (2) issues to be addressed under nuclear power plant continued-service reviews, as well as criteria and their bases, for resolution of these issues; (3) identification and evaluation of relevant inservice inspection or structural assessment programs; and (4) methodologies required to perform current assessments and reliability-based life predictions of safety-related concrete structures. To accomplish this objective, the SAG program has conducted activities under three major technical task areas: (1) materials property data base, (2) structural component assessment/repair technologies, and (3) quantitative methodology for continued service determinations. Nearly 50 reports on these topics have been published since 1991. Those published in 1993 covered surveillance and inspection of concrete structures; tests on age and degraded concrete; data surveys; repairs for damage; and reliability-based life assessment. The final seven reports will be completed in 1994.

Recent experience suggests the possibility that corrosion effects may significantly degrade the margin that containments have to accommodate accidents beyond their design basis. Evidence of corrosion has been found both in BWR Mark I containments and in PWR ice condenser containments. Additional information is necessary to understand the significant factors relating to corrosion, efficacy of inspection, and capacity reduction to be able to formulate future regulatory requirements that will ensure the continued availability of sufficient margins. Research programs were initiated in 1993 to provide that information.

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1.2.3.5 License Renewal Regulatory Standards

A final rule (10 CFR Part 51) concerning the environmental review for renewal of a nuclear power plant operating license is under development. The proposed rule was published for public comment in September 1991. Over 120 comments were received on the technical analyses and certain procedural aspects of the proposed rule. Con-

cern was expressed that the proposed rule would constrain public comment on environmental issues at the time of license renewal review for an individual nuclear power plant. All comments are being considered in developing the final rule, the generic environmental impact statement, and other supporting documents. It is expected that the final rule and supporting documents will be published in 1995.

2. STANDARD REACTOR DESIGNS

2.1 Engineering Issues for Advanced Reactor Designs

2.1.1 Statement of Problem

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

2.1.2 Program Strategy

The objective or strategy of engineering research is to verify the safety and quantify the margins in standard/advanced reactors for those features or criteria substantially different from currently operating plants and to ensure that NRC goals and policies are implemented successfully. To deal with these issues, analyses, evaluations, data collections, and limited testing are planned. In some cases, research is used to independently confirm or modify vendor/utility proposals for design, fabrication, construction, and inspection. In other cases, particularly in the passive designs, research serves to demonstrate acceptable design basis and severe accident response, for example, for structural performance of the AP600 containment during passive containment cooling. In still other cases, research is used to define acceptable standards for new features and requirements.

2.1.3 Research Accomplishments in FY 1993

2.1.3.1 Qualification of Advanced Instrumentation and Control System Hardware

The Oak Ridge National Laboratory (ORNL) is conducting a study to identify functional and environmental issues arising from the application of new technologies to the instrumentation in both present and next-generation nuclear power plants. Specifically, the program seeks an understanding of the technical issues involved in evaluating long-term properties and performance of "advanced" instrumentation and control (I&C) systems intended for plant upgrades and/or proposed for use in advanced light-water reactors (ALWRs). Special emphasis will be given to identifying vulnerabilities and environmental limitations that could be imposed on microprocessor-based systems in nuclear plant environments. Initial studies have focused on protection systems and the I&C employed in engineered safety systems. The environmental and functional issues studied thus far are reported in a draft report (NUREG/CR-5904). In this document evaluation templates are presented. They were developed by assembling reasonably complete configurations of safety channel components for the major ALWR designs. The templates provide for evaluating the impact of environmental stressors affecting these components and interfaces. Functional issues considered in the evaluation include distribution of functions, sources, supply and distribution of electrical power, calibration and testing capabilities, and attributes of failure predictions. The application and acceptance of digital computers in reactor protection systems are also reviewed in light of current industry standards.

2.1.3.2 Technical Basis for Regulatory Guidance on Electromagnetic Interference Issues

ORNL is developing the technical basis for regulatory guidance to address upsets and malfunctions in I&C systems caused by electromagnetic and radio-frequency interference (EMI/RFI) and power surges. These concerns stem from EMI/RFI and power surges affecting digital I&C systems at nuclear power plants and other process industries. The technical basis centers around good engineering practices to ensure that sufficient levels of electromagnetic compatibility (EMC) are maintained between the nuclear power plant's electronic and electromechanical systems. Good EMC design and installation practices are recommended to control interference sources and their impact on nearby circuits and systems in accordance with IEEE Std. 1050-1989, "Guide for Instrumentation and Control Equipment Grounding in Generating Stations." These good EMC practices include circuit layouts, terminations, filtering, grounding, bonding, shielding, and adequate physical separation. In addition, a

2. Standard Reactor Designs

test and validation program is being developed to outline the EMI/RFI and surge withstand capability (SWC) test and the associated test methods to be followed to ensure that I&C systems are capable of performing their intended functions. The test and validation program is being developed around the EMI/RFI test criteria from Military Standard (MIL-STD)-461, "Requirements for the Control of Electromagnetic Interference Emission and Susceptibility," the associated EMI/RFI test methods extracted from MIL-STD-462, "Measurement of Electromagnetic Interference Characteristics," and the SWC guidelines contained in IEEE Std. C62.41-1991, "Recommended Practice on Surge Voltages in Low-Voltage AC Power Circuits." Currently, the electromagnetic environment in a typical commercial nuclear power plant is undefined; hence, electric and magnetic spectral receivers need to be placed at various nuclear power plants. Long-term, unattended measurement data will be collected with these spectral receivers and used to provide the NRC with a realistic assessment of the probable ambient electromagnetic environment in nuclear power plants.

2.1.3.3 National Codes and Standards

Work initiated in FY 1992 on assessing current national codes and standards, such as the ASME Boiler and Pressure Vessel Code, is continuing through FY 1993. The information derived from this work will be integrated into the advanced reactor design approval/licensing process. Also under the guidance of this program, Canadian and other international quality assurance (QA) standards are being evaluated for compliance with Federal QA standards.

2.1.3.4 Evaluation of Low-Pressure Piping for Intersystem LOCA

Development of evaluation criteria for a new design goal for ALWRs was completed in FY 1993. The intent of the new design goal is that low-pressure piping attached to the reactor coolant loop withstand reactor coolant pressures and temperatures. The condition in question, which could follow from multiple valve failures, is important because for certain postulated events it can lead to rapid core damage and the release of radioactivity outside the containment. The potential event is called an intersystem loss-of-coolant accident and is being treated as a severe accident. Because of the low frequency of occurrence of this potential event, the performance goal is to achieve a failure probability in the low-pressure piping of about 10 percent. A probabilistic methodology to estimate failure was applied to both carbon and stainless steel piping and other piping components—including flanges, valves, pumps, and heat exchanger tubes. These results will be used in the Design Certification for the advanced boiling water reactor (ABWR) and the System 80+, and it is

expected that they will be used for the passive designs as well.

2.1.3.5 Experience-Based Seismic Qualification

In its Utility Requirements Document for ALWRs, the Electric Power Research Institute (EPRI) has proposed the use of experience as a method of seismic qualification as an applicable substitute on a case-by-case basis for more traditional tests and evaluations. An expert panel established to assess the viability of the experience-based method has recommended the use of a graded approach in which equipment categories are placed in one of three groups. In the first two groups, the use of experience data is possible but seismic capacities are different, and the attributes for membership in these two groups are also different. In the third group, the use of experience data is not allowed. Equipment categories likely to be placed in the first two groups are horizontal and vertical pumps; motor-operated, manual, and check valves; thermal element assemblies; diesel generators; transformers; and batteries on racks.

2.1.3.6 Containment Performance Goals

In support of the NRC Severe Accident Policy Statement, as it applies to ALWRs, work began on development of performance requirements for containments under severe accident conditions. Deterministic criteria will be established for both steel and concrete containments emphasizing standards for local and global strains and deformations. For these deterministic criteria, probabilistic models will be constructed to allow comparison with the conditional containment failure probability of 0.1 specified by the Commission for implementing safety goals. Particular attention will be given to evaluating the severe accident performance requirements for containments mentioned in the EPRI Utility Requirements Document.

2.2 Systems Performance of Advanced Reactors

2.2.1 Statement of Problem

The Commission has issued a policy statement on the regulation of advanced nuclear power plants (51 FR 24643) that states that the NRC will review and comment on 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. In-depth 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.

2.2.2 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. Several 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 LWR types: the advanced boiling water reactor (ABWR), the CE 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 (CANDU 3) heavy-water-cooled reactor, the Advanced Liquid-Metal-Cooled Reactor (ALMR), and the Modular High-Temperature Gas-Cooled Reactor (MHTGR).

The two evolutionary LWR designs (ABWR and System 80+) have been judged to be similar enough to the current generation of LWRs so that little additional research is needed to support their certification or licensing; the four nonconventional reactor design concepts have not reached the point where certification is expected in the near term. Consequently, the current emphasis of the advanced reactor research program is on developing the information needed to support the certification of the passive AP600 and SBWR reactor designs and making appropriate modifications of existing regulatory assessment methods to accommodate the review of the unique features of these reactors.

2.2.3 Research Accomplishments in FY 1993

2.2.3.1 NRC Confirmatory Safety System Testing in Support of AP600 Design Review

Westinghouse Electric Corporation submitted the Advanced Passive 600 MWe (AP600) nuclear power plant design to the NRC for design certification. RES is proceeding to conduct confirmatory testing of AP600 safety systems to help the NRC staff evaluate the safety of the AP600 reactor systems. NRC confirmatory safety system testing is not required for design certification but would provide additional technical bases for assessing NRC computer codes, which are then used to assist NRC in reaching licensing decisions.

For confirmatory testing, it was determined that the most cost-effective route was to modify an existing full-height, full-pressure test facility rather than build a new one. Thus, all the existing integral effects test facilities, both in the United States and abroad, were screened to select the best candidate. The criteria for the initial screening included the size, facility configuration similarity, availability schedule, willingness to share the cost, and the ability to enter into a confidential agreement with Westinghouse

for handling proprietary information. This screening revealed that the best candidate was the Rig of Safety Assessment (ROSA) Large-scale Test Facility in Japan Atomic Energy Research Institute (JAERI). To confirm these initial results and to determine the extent of modification necessary to simulate the AP600, the Idaho National Engineering Laboratory (INEL) was contracted to perform a comparative study between ROSA and AP600 using the RELAP5 code.

A comparison between the existing ROSA facility and the AP600 design showed that ROSA did not contain the key components important for safety response of the AP600. It was not obvious how much hardware modification to the ROSA facility would be needed to simulate the AP600. The fidelity of simulation must be balanced against the associated cost. The fidelity should be high enough to result in a facility capable of producing data for code assessment covering the major AP600 phenomena in the correct sequence. At the same time, the cost and the schedule have to be affordable. To make an optimum choice, four levels of modification in progressively more extensive stages were considered. The first level of modification was the absolute minimum, and the fourth level was the most inclusive among the four levels.

The comparative analyses of ROSA and AP600 with the RELAP5 code for selected accident scenarios led to the conclusion that the optimum choice was the fourth level. This included two core makeup tanks (CMTs) with appropriate pressure balance lines, a passive residual heat removal system with simulated secondary cooling, automatic depressurization system with stages 1 through 3 on top of the pressurizer and stage 4 on the hot leg, minimization of the pump loop seal lengths, a properly scaled AP600 pressurizer with a surge line, installment of appropriate upper-head flow paths, and an incontainment refueling water storage tank. Since ROSA has only one cold leg in each loop, CMT cold leg pressure balance lines are connected to the same cold leg for most transients when asymmetry between the two CMTs is not expected, but connected to a different cold leg for nonsymmetric transients, such as a break in a pressure balance line or a break in a direct vessel injection line.

The chosen level of facility modification is being implemented by Sumitomo Heavy Industries, which constructed the ROSA facility and has been maintaining and operating it for the past several years as a contractor to JAERI. The facility modification will be completed by February 1994 and a series of tests performed later in 1994.

As a confirmatory testing program, the ROSA/AP600 testing will cover not only design basis accidents but also beyond design basis accidents, which vendors may not be required to address. There will be some counterpart tests among ROSA, SPES-2, and OSU tests. Test data from

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different scale facilities will help predict what will occur in a full-scale AP600 reactor.

The status of the NRC containment response codes (i.e., CONTAIN and COMMIX) that will be used for evaluating the performance of the AP600 passive containment cooling system design are discussed in Section 5.2.3.4, "Severe Accident Codes."

Scaling analyses of the AP600 core makeup tanks and passive residual heat removal system were completed and findings sent to NRR for their use.

2.2.3.2 SBWR Test Facility

Purdue University was awarded a 3-year contract on July 26, 1993, to build a test facility for an advanced reactor design, the Simplified Boiling Water Reactor (SBWR). The objective is to provide data to assess the capabilities of the NRC's computer codes to analyze the SBWR. The Purdue test facility will have all the necessary components and systems scaled from the SBWR design. Included are a vessel with electrically heated fuel rods, upper and lower drywells, suppression pool, gravity-driven cooling system (GDSCS), passive containment cooling system (PCCS), isolation condenser system (ICS), drywell and wetwell sprays, piping and valves, and instrumentation.

The facility is a low-pressure, reduced-height facility. It is designed for 150 psia, which is adequate for investigating the integral performance of the GDSCS and PCCS. The height of the Purdue facility is 1/4 of the SBWR height and its volume is 1/400 of the SBWR volume. The aspect ratio of the facility defined as height scale/diameter scale is 2.5. As a result of the reduced scale, events will occur in the facility on a time scale twice that calculated for the SBWR. The facility is scheduled to be built by December 1994. Approximately 50 tests will be performed by April 1996 to cover a broad spectrum of loss-of-coolant accidents and transients. At least 10 of the tests will be completed prior to the completion of the final safety evaluation report on SBWR, which is scheduled for July 1995.

2.2.3.3 Support for CANDU 3 Preapplication Licensing Review

A review of the regulation of CANDU reactors in Canada was performed by the Argonne National Laboratory. That review summarized Canadian regulations as they are applied to CANDU. It described safety systems in the CANDU designs and their relationship to Canadian regulations. The review also identified major NRC regulatory requirements that have a bearing on unique features of the CANDU design.

A systems analysis of the CANDU 3 design was performed by ORNL. The purpose of this work was to classify event sequences, plant systems, and operator actions in a

way that would assist the NRC with its preapplication review for design certification. The selection of certain transients and postulated accidents for consideration in the safety analysis will be crucial in determining what acceptance criteria have to be demonstrated for CANDU 3; the criteria will be different from those for LWRs. Plant systems that are identified as important to safety must be given early attention to ensure that they receive appropriate seismic and environmental qualification. Identifying the need for operator actions to mitigate accidents will also lead to early decisions regarding automatic versus manual actions.

An assessment of existing and planned data bases for CANDU 3 was performed by the staff with assistance from ORNL, INEL, and Sandia National Laboratories. This assessment provides an early identification of areas requiring long lead times to plan additional test programs that are needed for completeness. The assessment also provides the NRC with a basis for planning its resource requirements for confirmatory research and analysis support for design certification of CANDU 3.

CANDU reactors are somewhat overmoderated and consequently exhibit a positive reactivity coefficient for coolant voiding, unlike LWRs that have been licensed by the NRC. This is a well-known characteristic of the design, and the CANDU reactors are equipped with a second, rapid-acting, independent shutdown system to provide safety margin. Nevertheless, it is the NRC's practice to study postulated accidents of very low probability if they have the potential for serious consequences. Therefore, a series of calculations was performed by the NRC staff and INEL for several transients for which it was assumed that neither of the shutdown systems worked. These calculations provide an indication of the magnitude and time scale of the resulting power excursion that will help the NRC with decisions about the licensability of this design.

2.2.3.4 Human Reliability

A study is under way to develop methods for assessing the impact on risk of changes in human performance prompted by the introduction of advanced digital displays and controls. Research to establish a technical basis for necessary shift staffing to support safe operations for advanced control room designs is being planned for initiation in FY 1994. The research will be based on workload and task analysis studies.

2.3 Advanced Reactor Risk Analysis

2.3.1 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 nu-

clear power plant accidents. PRAs are required as part of the license applications of all advanced reactor designs. RES supports the Office of Nuclear Reactor Regulation (NRR) review of these applications by performing independent assessments of accident-initiating-event frequencies and the reliability of key systems in these designs.

2.3.2 Program Strategy

The research effort in this area is focused on the assessment of the frequencies of challenges to an advanced reactor design and the ability (reliability) of mitigating systems to cope with such challenges and prevent damage to the reactor core. The program accomplishes this by defining the set of potential challenges, estimating the frequencies of these challenges, and estimating the reliability of systems included in the design in response to the challenges.

2.3.3 Research Accomplishments in FY 1993

The advanced passive reactors have engineered safeguard systems that maximize the use of passive devices such as nitrogen-powered accumulators, natural circulation flow, and gravity-driven safety injection. They do not rely on active systems such as ac-powered equipment, although certain valves may require stored energy (e.g., battery power) to change state. These passive designs are expected by the designers to both increase safety and decrease cost as a result of their simplified design. However, because of the lack of actual working experience with the design and because of uncertainties in the modeling of processes such as natural circulation, there are uncertainties in the performance of the engineered safeguard systems.

The passive reliability project, presently focusing on the Westinghouse AP600 design, is developing a candidate methodology for quantifying the uncertainty distribution in the core damage frequency arising from uncertainty in the modeling of the natural processes. The project will be completed in FY 1994.

2.4 Regulatory Application of New Source Terms

2.4.1 Statement of Problem

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

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

This source term is associated with a severe reactor accident, since only major core degradation and melting could result in the release of such large quantities of fission products. However, the present formulation, while providing a high level of protection for plant systems, is now recognized to be incompatible with present research findings. As a result, a rigid application of the TID source term may not permit the best engineering solutions on some aspects of future plant design.

2.4.2 Program Strategy

On May 25, 1988, the staff presented to the Commission an "Integration Plan for Closure of Severe Accident Issues," SECY-88-147. This plan discussed major elements relating to closure of severe accident issues, including severe accident research efforts and related activities in siting, emergency planning, and potential changes to existing regulations as a result of severe accident research findings. On October 4, 1990, the staff presented to the Commission a "Staff Study on Source Term Update and Decoupling Siting from Design," SECY-90-341. This plan discussed the staff proposal to decouple reactor siting from source term and dose calculations and to prepare an update and revision of the source term given in TID-14844. On April 10, 1992, the staff presented to the Commission its "Revised Accident Source Terms for Light-Water Nuclear Power Plants," SECY-92-127. This paper provided a draft of a revised source term. On June 12, 1992, the staff presented to the Commission its proposed "Revision to 10 CFR Part 100, Revisions to 10 CFR Part 50, New Appendix B to 10 CFR Part 100 and New Appendix S to 10 CFR Part 50," SECY-92-215. This paper presented proposed rules to revise the reactor siting criteria. On July 28, 1992, the Commission announced the availability for comment (57 FR 33374) of a draft report on "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465. On September 28,

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1992, the Commission issued an advance notice of proposed rulemaking on "Acceptability of Plant Performance for Severe Accidents; Scope of Consideration in Safety Regulations" (57 FR 44513). Revised accident source terms, as well as other severe accident considerations, are to be incorporated into this rulemaking effort.

2.4.3 Research Accomplishments in FY 1993

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. 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 has long been included in the Commission's regulations for siting, it has traditionally had a greater effect on plant design than on siting.

Since 1962 the NRC has attained a better understanding of the timing and nature of the fission product release and has identified a number of areas subject to regulation that may benefit from changes introduced as a result of source term and severe accident research. In FY 1993, work continued on a replacement to TID-14844. The comment period for a draft report, "Accident Source Terms for Light-Water Nuclear Power Plants" (NUREG-1465), which was issued in July 1992, expired in December 1992. Comments were solicited and received from an internationally recognized group of experts as well as from the public. A final version of NUREG-1465 is in preparation. In addition, in support of this effort, the following documents were issued:

- NUREG/CR-5950, "Iodine Evolution and pH Control," dated December 1992.
- NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," dated June 1993.
- NUREG/CR-5942, "Severe Accident Source Term Characteristics for Selected Peach Bottom Se-

quences Predicted by the MELCOR Code," dated September 1993.

- NUREG/CR-5901, "A Simplified Model of Aerosol Scrubbing by a Water Pool Overlying Core Debris Interacting with Concrete," dated November 1993.

2.4.3.1 Update of Siting Regulations

In FY 1993, staff efforts continued on updating 10 CFR Part 100, "Reactor Site Criteria." A proposed rule to revise Part 100 was issued in the *Federal Register* (57 FR 47802) on October 20, 1992, for a 120-day comment period. The rule proposed to decouple calculation of the exclusion area distance from source term and dose calculations by specifying a minimum acceptable exclusion area distance and by stating population density criteria as well. In addition, an update of seismic considerations proposed to incorporate probabilistic as well as deterministic methods. The comment period, extended twice, expired on June 1, 1993. Extensive comments, both domestic and foreign, were received. The staff is analyzing comments received and is examining further options in this regard.

2.4.3.2 Emergency Planning Regulations

In May 1993, a proposed rule was published in the *Federal Register* (58 FR 29795) on the emergency planning licensing requirements for independent spent fuel storage facilities and monitored retrievable storage facilities.

In June 1993, a proposed rule was published in the *Federal Register* (58 FR 34539) on revised emergency planning that would update and clarify ambiguities that have surfaced in the implementation of the Commission's emergency planning exercise requirements.

In March 1993, a petition for proposed rulemaking by VEPCO was published in the *Federal Register*. In this petition VEPCO proposed that the NRC amend its regulations to change the frequency of emergency exercises at nuclear power plants from annual to biennial.

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

3. PLANT PERFORMANCE

3.1 Statement of Problem

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

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

As a result, the NRC must use available experimental data to validate analytical models for evaluating 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.

3.2 Program Strategy

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

Our reliance on the computer codes to provide predictions of reactor response with acceptable uncertainties depends on three levels of experiments and comparisons of experimental results with code predictions. First are basic experiments 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 genera-

tor. 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.

3.3 Research Accomplishments in FY 1993

3.3.1 Reactor Safety Experiments

Experiments are being conducted on decay heat removal by natural circulation at North Carolina State University in a facility that is a scaled model of a Westinghouse pressurized water reactor. It is 1:9 in height with a 1:700 volume scale and uses freon as the working fluid. The facility includes a secondary system to provide representative plant behavior. Parts of the facility are constructed with plexiglass to allow visual observation of internal conditions. Facility construction was underwritten by several utilities and is used extensively for the training of utility personnel and university students. The NRC is conducting decay heat removal experiments under conditions of reduced coolant inventory in the primary system, using the visual observation features of the experimental facility.

Experiments are also being performed at the University of Maryland in a scaled experimental facility that simulates a Babcock and Wilcox reactor and is 1:4 in height with a 1:500 volume scale. This facility was originally constructed under an NRC contract to study small-break loss-of-coolant accidents and, following successful completion of this program, its mission was shifted to the current study of the natural circulation of steam under severe accident conditions. For these tests, sulfur hexafluoride is being used as the working fluid. The current program will be completed in FY 1994.

3.3.2 Safety Code Development and Maintenance

The third semi-annual international thermal-hydraulic code applications and maintenance program (CAMP) meeting was held during October 20-22, 1993, in Santa Fe, New Mexico. There are now 14 member countries in CAMP, each of which had representatives at this meeting, in addition to representatives from the NRC and its code development contractors.

Cash contributions made by, or on behalf of, these 14 member countries in the amount of \$390,000 annually are used to supplement the code development and assessment programs funded by the NRC. As part of their agreements, the members provide code assessment

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studies or other noncash contributions to assist NRC's assessment of code applicability, scalability, and uncertainty when applied to nuclear power plant safety. The codes covered by the agreement are RELAP5/MOD3, TRAC/PWR, and TRAC/BWR. Papers presented at the meeting discussed work on these programs funded by NRC and CAMP member contributions, CAMP member activities, code assessment performed by CAMP members, and code activities of United States code users.

Improvements were made to RELAP5 in support of the AP600 and SBWR design reviews. The following set was completed in FY 1993: energy balance out the break, condensation in the presence of noncondensables, containment modeling (coupling RELAP with CONTAIN code), code speedup to analyze long-term transients, level tracking, boron transport (moving wave front), mechanis-

tic steam separator, spherical accumulator model, and improved IRWST model.

Improvements were also made to the RAMONA code to perform stability analyses in SBWRs. These improvements include modeling of the isolation condenser, chimney component, boron transport and reactivity feedback, standby liquid control system, and flow-dependent loss coefficients for low flows. In addition, the modeling of all balance-of-plant components is completed. This capability is needed to analyze stability in both SBWRs and existing BWRs. The RAMONA code was made operational on a workstation used at the NRC.

The manual for the TRAC-P code that is being used to analyze large-break LOCA scenarios in the AP600 design has been prepared for publication. The TRAC-P code has been made operational on workstations used at the NRC.

4. HUMAN RELIABILITY

4.1 Statement of Problem

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

4.2 Program Strategy

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

The human reliability research program was 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 technical basis for guidelines and criteria to evaluate the interface between the system and the human user from the perspective of safe operations and maintenance. 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.

4.3 Research Accomplishments in FY 1993

4.3.1 Personnel Performance

Work continued on the development of a method to assess the effectiveness of training programs at nuclear power

plants. Measures and supporting documentation for a training effectiveness evaluation method will be included in a report to be issued in FY 1994. Data analyses are being incorporated into a final report (NUREG/CR-6122) for a project on the factors that are considered when making decisions on operations staffing and on how staffing relates to safe startup, shutdown, and operation of nuclear power plants.

A study to establish a technical basis for minimum shift staffing for operational crews at nuclear power plants based on workload and task analysis was initiated. A handbook on the effects of environmental factors on human performance was completed and will be issued as NUREG/CR-5680 in 1994 for use by nuclear power plant inspectors. Two reports concerning training for responding to accidents are being prepared. These reports describe decisionmaking and stress coping skills that may be needed to respond to an accident situation, as well as potential training approaches for developing those skills.

An effort to identify measures for better characterizing the quality of personnel performance in an operational setting was initiated. These measures and methods could be used for evaluating the effects on safety of changes in system interfaces, particularly changes from analog to digital interfaces.

4.3.2 Human-System Interfaces

Human-system interface research includes NRC participation in the Organization for Economic Cooperation and Development (OECD) Halden Reactor Project, which is a multifaceted program that includes verification and validation of digital systems, man-machine interaction, and surveillance and support systems for advanced control rooms. Specific NRC research needs were identified in the form of lessons-learned reports on (1) methods and tools for the development, verification, and validation of safety-related software, (2) experience with development and quality assurance of software systems at the Halden Project, (3) man-machine interaction with computer-based systems, (4) test and evaluation activities on man-machine interaction with computer-based systems, (5) recommendations on advanced control rooms based upon the Halden Project's experience, (6) coordination and integration of computer-based operator support systems, and (7) reports containing information related to human reliability. These reports will serve as part of the technical bases for NRC guidelines. The lessons-learned report on test and evaluation activities was completed; it summarized over 10 years of experience with various activities on computer-based operator support systems.

Research continued to evaluate the positive and negative attributes of standards and computer-aided software

4. Human Reliability

engineering tools for use in the certification of high-integrity software for nuclear power plant safety systems. Research also continued to develop and test a computer-aided software engineering tool for assessing the degree of functional diversity in software that performs safety functions. A project, cosponsored by EPRI, to develop guidelines for the verification and validation of expert systems is nearing completion. Research continued on a project, also cosponsored by EPRI, on verification and validation guidelines and quality metrics for digital high-integrity systems.

The project to independently evaluate, test, and improve upon verification and validation guidelines for use in the audit of computer-based safety systems completed a field test of the verification guidelines at a developer's site.

The NRC, assisted by the National Institute of Standards and Technology, sponsored a Digital Systems Reliability and Nuclear Safety Workshop. The purposes achieved by the workshop were to (1) provide feedback to the NRC from outside experts regarding potential safety issues, proposed regulatory positions, and research associated with application of digital systems in nuclear power plants and (2) continue the indepth exposure of the NRC staff to digital system design issues related to nuclear safety by discussions with experts in the state of the art and practice of digital systems. Proceedings of the workshop will be issued early in 1994.

Research was initiated to ensure the completeness of the technical bases for regulatory requirements intended to ensure the integrity of safety-related software. These technical bases will be used to develop guidelines that may be used in developing regulatory positions on software.

Draft guidelines prepared for advanced control room design reviews were subjected to internal and peer review. The draft guidelines are built on the validation of guidelines available from other industries, including the aerospace and defense industries. The draft guidelines were prepared in paper form and computerized (interactive) form. The paper form will be issued as NUREG/CR-5908. The interactive version of the guidelines was demonstrated to the NRC staff. The final version is being developed for an application through Windows software.

Two meetings with subject matter experts were held to assess the feasibility of conducting NRC human factors regulatory research at available and potentially available facilities.

4.3.3 Organizational Factors

Research on organizational factors developed draft, integrated research products that could be considered for use in regulatory applications such as risk-based inspection and diagnostic evaluation. Additional research continued to develop alternative quantification methods for incorporating the influence of organizational factors into PRA. It was discovered that an analysis of organizational factors can identify dependent failures across systems in a selected work process at a plant.

The NRC is reassessing the need for continuing research on organizational factors, taking into consideration the remaining obstacles to incorporating organizational factors into risk.

4.3.4 Reliability Assessment

Work is nearly completed on collecting, cataloguing, and storing in a computerized library estimates of probabilities of operator error and hardware failure. Because one of the largest contributors to risk is operator cognitive error, research continued, in order to gather data on cognitive performance, on an effort to validate a computer simulation model of human cognitive tasks during accident sequences. The data were gathered from operating crews responding to two simulated accident scenarios on training simulators.

Research was initiated to analyze information from the simulator portion of the NRC-administered operator requalification examinations. Estimates from this source may provide empirical error rates for use in a nuclear power plant PRA. Research continued to evaluate the risk impact of replacing existing nuclear analog systems with digital systems.

For several years the NRC has been developing reliability analysis tools for use with PRA to improve the technical bases of selected requirements in technical specifications. These tools can evaluate risk technical specification changes such as (1) surveillance test intervals, (2) allowed outage times, both during operation and during shutdown, (3) action statements that require shutdown, (4) technical specification defenses against dependent failures, and (5) scheduling preventive maintenance. Almost all the methods are completed, and both detailed technical reports and a handbook to guide NRC reviewers through the use of these methods are in preparation. A Research Information Letter will be issued early in FY 1994.

5. REACTOR ACCIDENT ANALYSIS

5.1 Reactor Risk Analysis

5.1.1 Statement of Problem

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

5.1.2 Program Strategy

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

5.1.3 Research Accomplishments in FY 1993

Probabilistic risk analysis is used by the NRC staff to support the resolution of a wide spectrum of reactor regulatory issues. In FY 1993, these applications encompassed both specific issue-oriented projects and more general work, including development and demonstration of risk analysis methods and development of risk-related training and guidance for the NRC staff.

5.1.3.1 Issue-Oriented Projects

Analysis of Low-Power and Shutdown Accident Risks. As a result of the Chernobyl accident and other precursor events around the world, an extensive two-phased project

was initiated in 1989 to examine the potential risks of accidents initiated during low-power and shutdown modes of operation. Phase 1, completed at the end of 1991, was a coarse screening analysis of all operational modes (other than full power) for one BWR and one PWR to provide support for the Office of Nuclear Reactor Regulation's (NRR) regulatory analysis and to guide the Phase 2 effort. A significant finding was that the traditional concept of technical specification modes of operation does not adequately delineate plant operating boundary conditions (states) needed for risk analyses. The Phase 2 effort has concentrated on a specific operating state for each of the two plants, selecting the potentially highest risk operating state based on the Phase 1 results. In addition, a simplified analysis of potential in-plant and offsite accident progression and health consequences of such accidents has been performed and provided to NRR in support of their regulatory activities, as documented in NUREG-1449. The results of Phase 2 will be published as NUREG/CR reports early in FY 1994.

South Texas Project Risk Analysis. In 1992, the staff completed a review of the South Texas Project risk analysis and documented the results and findings (NUREG/CR-5606). The licensee estimated the overall mean core damage frequency to be 2×10^{-4} per reactor year, which is found to be within the range of core damage frequency estimates provided for similar Westinghouse PWR facilities. The licensee has subsequently requested modifications to its plant technical specifications based, in part, on its risk analysis. The RES staff is now working with NRR on the acceptability of the requested modifications and expects to complete this work early in FY 1994.

5.1.3.2 Methods Development Projects

SAPHIRE Computer Tools. A suite of computer codes called SAPHIRE (System Analysis Programs for Hands-on Integrated Reliability Evaluation) has been updated to Version 5.0. This set of codes is for use in performing PRA and to permit an analyst to perform many of the functions necessary to create, quantify, and evaluate the accident risks of nuclear power plants. The codes are being used extensively to perform the low-power and shutdown risk analyses described above, to perform analysis for resolving generic safety issues, and to set priorities for inspection activities. During 1993, PRA data from three more licensed nuclear power plants were added to the SAPHIRE data base, and most of the data from previous plant loads was updated to Version 5.0. This brings the data base total to 13 plants, two of which are advanced concept plants that have been added to support the agency's design certification reviews. Courses continued to be provided to both the NRC staff and contractors on the use of these codes. When the documentation for Version 5.0 is completed (scheduled for January 1994), the

5. Reactor Accident Analysis

new codes and user manuals will be sent to the Energy Science and Technology Software Center at Oak Ridge National Laboratory for general distribution.

Consequence Code Benchmark. The NRC is working with the Commission of the European Communities and the Organization for Economic Cooperation and Development to perform an intercomparison exercise on probabilistic accident consequence codes. The six codes being evaluated are MACCS (U.S.), COSYMA (Germany), CONDOR (UK), OSCAAR (Japan), LENA (Finland), and ARANO (Sweden). The comparison exercise uses six radioactive accident source terms and calculates dose consequences for such measures as whole body dose and fatal cancers. The results of these comparisons will be obtained early in FY 1994 and will provide a data base to judge the performance of new codes to predict reactor accident consequences.

Survey and Evaluation of Aging Risk Assessment Methods. A survey and evaluation of aging risk assessment methods and applications is being performed. A draft NUREG/CR on the work has been received from the contractor and the final NUREG/CR will be published early in FY 1994. Major findings of the review include:

- The issue of aging in nuclear power plants cannot be addressed by models that are based solely on the current PRA structure and failure rates. Structures, systems, and components (SSCs) neglected in the basic PRA model may become important because of aging; the basic degradation mechanisms, such as fatigue, embrittlement, and erosion-corrosion, must be considered.
- Probabilistic models for degradation mechanisms would allow the effective use of information regarding the aging of SSCs. Failure-rate-based models cannot accommodate this type of information, which typically does not include a significant number of failures. The use of these probabilistic models for the degradation mechanisms would decrease the uncertainties present in models that are exclusively failure rate based.
- Probabilistic models for the degradation mechanisms would also allow the development of effective risk management strategies.
- The development of a methodology that includes aging mechanisms can build on existing PRA models, appropriately modified, as current external event analyses do.

5.1.3.3 Risk-Related Training and Guidance Development

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

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

In October 1993 the group completed a draft final report that included initial guidance to the staff on the use of PRA in screening and analyzing reactor operational events and on basic terms and methods used in PRA. The report also contains a number of recommendations for additional guidance development, improvements to the NRC's PRA training program, and improvements in PRA tools and data bases used by the staff. It is expected that the final report will be published early in 1994.

Reactor Safety Training Course. In response to a request from the Office of Analysis and Evaluation of Operational Data (AEOD), RES has developed a new course that is intended to treat reactor safety in a broad sense. Topics include a historical overview, design-basis accidents, beyond-design-basis accidents, accident progression in the reactor vessel, accident progression in the containment, radiological releases and consequences, and emergency response. The intended audience includes new agency employees and other NRC staff not generally familiar with these topics. Two presentations of the course have been offered at NRC's Technical Training Center in Chattanooga in 1993. With the developmental work on this course completed, responsibility for its conduct will be turned over to AEOD in FY 1994.

5.2 Containment Performance

5.2.1 Statement of Problem

Core melt accidents that proceed to vessel failure have the potential to produce high pressures and temperatures that might challenge containment integrity. It is known from previous risk studies, and from the experiences at

Chernobyl and Three Mile Island, that containment survival or even delayed failure has an all-important effect on minimizing the release of radioactivity to the environment in the event of a core melt accident. If realistic assessments of the consequences of core melt accidents, which so strongly depend on whether or when containments might fail in the course of the accident, are to be made, then an understanding of the phenomena that occur in containment in the latter stages of the accident that could lead to containment failure is imperative.

5.2.2 Program Strategy

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

NRC's research program dealing with reactor containment safety consists of five areas of research. These five research activities address: (1) the interaction of molten core debris with structural concrete, including the ablation of concrete structures, heat transfer to structures in the containment and to overlying water, the generation of noncondensable and flammable gases, and fission products containing aerosols; (2) direct containment heating by molten debris particles ejected from the vessel at high pressure and hydrogen production resulting from steam oxidation of the metallic component of that debris; (3) the combustion of hydrogen in the containment, including the potential for detonation; (4) the development, validation, maintenance, and application of various computer codes that are capable of describing the multiple phenomena that occur in severe accident sequences of interest; and (5) assessment of severe accident phenomena, including containment performance, for advanced reactors (see Section 2.2).

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

5.2.3 Research Accomplishments in FY 1993

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

5.2.3.1 Melt-Concrete Interactions and Debris Coolability

In those severe accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity where it can thermally and chemically interact with structural concrete. The consequences of these melt-concrete interactions can have a significant effect on containment loading, the potential modes of containment failure, and the radiological source terms. The major areas of concern associated with melt-concrete interactions during a severe accident are the penetration of the basement and failure of the liner, the generation of radioactive aerosols and gases, including combustible gases, and the overpressurization of the containment.

The NRC has conducted an extensive program of analytical and experimental research to obtain improved understanding of melt-concrete interactions. The experimental research is focused on conducting experiments simulating a wide variety of concretes used in nuclear power plants in the United States and consideration of the diverse accident scenarios that may lead to melt-concrete interactions. The analytical research focused on the development of models for studying phenomenological aspects of melt-concrete interactions and included a reassessment of models used to predict aerosol generation and radionuclide release.

Early experiments on melt-concrete interactions were conducted without the presence of an overlying water pool. The experimental data base generated from these experiments is extensive and spans a broad range of melt release conditions as well as concrete types. No additional melt-concrete interaction experiments are planned. More recent experiments on melt-concrete interactions were conducted in the presence of an overlying water pool. In FY 1991, the NRC initiated the WETCOR program, also called the debris coolability program, to address two specific issues: (1) the comparative coolability of oxidic and

5. Reactor Accident Analysis

metallic debris and (2) the effects of boundary conditions on coolability, i.e., crust formation and stability. A report describing the WETCOR-1 test (NUREG/CR-5907), the only integral test conducted under this program, was published in FY 1993.

The second experimental program on debris coolability, called the Melt Attack and Coolability Experiments (MACE) program, was developed as an extension of the Advanced Containment Experiments (ACE) program under the sponsorship of NRC, EPRI, and other largely governmental agencies in several countries. The MACE program is intended to determine the ability of water to cool prototypic core debris (urania-zirconia composition). Three tests were conducted under the MACE program in FYs 1992 and 1993. Data from these tests were analyzed in FY 1993. Except for the last test that was terminated prematurely, the results from the MACE tests generally support the concept of crust formation at the melt-coolant interface with periodic access of water to the melt and partial melt cooling. A fourth MACE test at a scale larger than the previous tests is planned in FY 1994 to study the effect of scale on crust formation, stability, and debris coolability.

A topic related to melt-concrete interactions, particularly in connection with the BWR Mark I containments, is that of melt-structure interactions, leading to early containment failure attributable to liner (shell) meltthrough. The NRC research over the past few years has addressed key phenomena associated with the liner meltthrough issue. Integration of the research into the assessment of the conditional probability of liner failure—both with and without an overlying water pool in the drywell—given a core melt accident that proceeds to vessel failure, was completed in FY 1991 and documented in NUREG/CR-5423. The overall conclusion from the assessment was that the presence of water on top of the core debris would prevent containment shell failure. The report was peer reviewed, and, as a result of the review process, four areas were identified that needed further resolution—liner failure criteria, melt spreading, melt-concrete interactions, and melt release conditions. Additional work was performed in these areas in FYs 1992 and 1993. The results from the additional research confirm the conclusions in the original study. NUREG/CR-6025, which included updates from the additional research efforts, was issued in FY 1993.

5.2.3.2 High-Pressure Melt Ejection—Direct Containment Heating

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system is projected to remain pressurized. A molten core, if left unmitigated, will slump and collect at the bottom of the reactor vessel. If a breach occurs, the core melt will be ejected under pressure, and if the material should be

ejected from the reactor cavity into surrounding containment volumes in the form of fine particles, thermal energy could be quickly transferred to the containment atmosphere. The metallic components of the ejected core debris could further oxidize in air or in steam and could generate a large quantity of hydrogen and chemical energy that would further pressurize the containment. This process is called direct containment heating (DCH).

To help develop a data base by which to estimate the risk associated with high-pressure core melt accidents, the NRC has, in FY 1993, completed four DCH integral effects tests for a containment configuration simulating that of the Surry PWR plant, three in the 1/6th-scale Containment Technology Testing facility and one in the 1/10th-scale SURTSEY facility at the Sandia National Laboratories. Analysis of the test results is near completion. In addition, in the 1/40th-scale COREXIT facility at the Argonne National Laboratory, three integral effects tests were conducted using reactor materials (urania-based melt) instead of an iron-alumina melt simulant. Separate-effects testing is being performed at Purdue University in a 1/10th-scale model for a containment configuration simulating that of the Zion PWR plant to study the DCH phenomena in detail and will be completed in FY 1994.

As part of the DCH issue resolution plan for PWRs, a study was completed and documented in "Integrated Report on DCH Issue Resolution for PWRs" (NUREG/CR-6109). This report outlines the DCH issue resolution process, demonstrates the process for two specific plants, Zion and Surry, and describes the approach for resolution for the remaining PWRs. Supporting documentation and studies related to this report have also been completed. One of these supporting documents is a study of DCH for the Zion plant, "The Probability of Containment Failure by Direct Containment Heating in Zion" (NUREG/CR-6075). Both reports are now undergoing peer review, and it is expected that they will be completed in FY 1994.

5.2.3.3 Hydrogen Combustion

Significant information exists on hydrogen combustion, and it is sufficient to assess the possible threat to containment and safety-related equipment. Some residual issues remain that are related to understanding the likelihood of various modes of combustion at high temperature and in the presence of large quantities of steam, i.e., deflagrations, diffusion flames, accelerated flames, transition from deflagration to detonations (DDT), and detonations.

The largest current program in this topical area comes out of a joint agreement between the NRC and the Ministry of International Trade and Industry (MITI) of Japan (managed by the Nuclear Power Engineering Corporation). Under this agreement, a high-temperature hydrogen combustion program related to high-speed combustion

modes, i.e., detonations and DDT, has been developed and is under way at Brookhaven National Laboratory. A small-scale developmental apparatus (SSDA) was constructed. The SSDA has provided a preliminary set of experimental data to characterize the effect of high temperatures on the ability of hydrogen-air-steam mixtures to undergo detonation. Equally important, the SSDA was used to support the design of the larger-scale high-temperature combustion facility (HTCF) by providing solutions to a number of design and operational problems at high temperatures. The construction of the HTCF is completed, and high-temperature experiments will begin during FY 1994. As a result of the cooperative agreement with Japan, the NRC has access to ongoing hydrogen mixing and distribution testing in the Tadotsu facility and the combustion testing in the Takasoga facility. This research provides a greatly expanded and improved data base for the validation of analytical tools.

A hydrogen research program is also under way to investigate diffusion flame behavior in low-speed hydrogen combustion. Experiments were performed in a small-scale facility to examine the influence of ignition source strength on lean flammability limits of hydrogen-air mixtures at a temperature of 300K and pressure of 1 bar. A report was issued describing the facility and the first series of experiments. The experimental facility will be modified to eliminate jet interference with the sidewalls and to establish stable jets. The results from this research program will be used to help resolve outstanding issues in severe accidents, i.e., hydrogen combustion aspects of DCH, high-temperature combustion phenomena, and detonation initiation by high-temperature steam-hydrogen-particle jets.

5.2.3.4 Severe Accident Codes

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

MELCOR is an integrated computer code that models the progression of severe accidents in LWR power plants. The code can be used to evaluate the progression of severe reactor accidents from initiation through containment failure and to estimate severe accident source terms and their sensitivities and uncertainties in a variety of applications. 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—is treated in MELCOR in a unified framework for both BWRs and PWRs.

MELCOR has been applied to the analyses of various plant accident transients, and assessment efforts have been completed by a number of United States and international user organizations. The focus of the development efforts in FY 1993 has been to improve capabilities to handle the phenomena of natural circulation, external heat transfer, and lower-head failure and to model a few specific features of the advanced light-water reactor (ALWR). The efforts have also addressed a number of suggestions for improvement of the code made by an independent peer review group convened at the NRC's request. Further, a MELCOR Cooperative Assessment Program effort, started last year, has continued in FY 1993. The goal of this effort is to create an international forum for information exchange on the applicability, limitations, and operational experience of MELCOR while providing detailed MELCOR assessment calculations and reports to the NRC and to the MELCOR user community.

CONTAIN is a detailed code for the integrated analysis of containment phenomena. The code provides the capability to predict the physical, chemical, and radiological conditions inside a nuclear reactor containment in the event of a severe accident. The code also provides the capability to predict fission product releases to the environment in the event of containment failure. Among the models included in CONTAIN are heat and mass transfer, aerosols and fission product behavior, flammable gas combustion, core-concrete interactions, and direct containment heating. The code is capable of analyzing a wide variety of LWR plants, including their engineered safety systems and many kinds of accident scenarios.

One issue currently under investigation is DCH and pressurization of the reactor containment atmosphere by molten core materials ejected following the lower-head failure of the vessel under pressure. A program to incorporate selected DCH models into the CONTAIN code, including the assessment against available experimental data, will be completed in FY 1994. Also, plant cases were run with the updated CONTAIN code to determine the impact of DCH on the containment. Another development effort is related to containment analyses for ALWR designs. The industry is developing containment designs for ALWRs that incorporate passive cooling and decay heat removal features for protection against long-term containment overpressure in accident situations. The CONTAIN code was modified in selected areas in this regard; it is planned to use the code to evaluate the experimental data generated by the industry's research.

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SCDAP/RELAP5 is a computer code that has the capability to perform detailed analyses of in-vessel core melt progression phenomena during various severe accident conditions. The code has been used for severe accident analyses and for the evaluation of the effect of different accident management strategies, including natural circulation studies and the analysis of lower-plenum debris and lower-head heatup. The systematic assessment of SCDAP/RELAP5 that began in 1991 identified several areas of modeling improvements. Work on these improvements that was completed in FY 1993 included: (1) improvement of code modeling and elimination of numerical errors (such as converting SCDAP fixed input format to free format, expanding input error checking and diagnostic printout, identifying and resolving programming errors causing the code to fail, and correcting the code failure due to axial nodalization limit); (2) enhancement of code reliability (such as correcting numerical oscillations, code failures, excessive run time, code failure or results changed upon restart); and (3) improvement in late-phase molten pool formation and melt slumping modeling (such as modifying changes in junction and volume flow area to account for actual debris-bed height, independent of node size, and adding user input to allow radial spreading of liquefied material into adjacent channels). These modeling improvements have significantly reduced uncertainties in the code calculation of core melt progression.

Other SCDAP/RELAP5 research activities accomplished in FY 1993 include the completion of an independent peer review of the code and initiation of model extensions to the code in order to address ALWR issues. Further model improvement and code documentation based upon the recommendations of the peer review committee are being considered. To ensure that SCDAP/RELAP5 meets design objectives and targeted applications, model assessment and validation efforts will continue.

COMMIX is a three-dimensional transient single-phase computer code for thermal-hydraulic analysis of single and multicomponent engineering systems. The code solves a system of time-dependent and multidimensional conservation of mass, momentum, energy, and transport equations. A number of phenomena encountered in postulated severe accidents in ALWRs are inherently multidimensional in nature. The COMMIX code is being developed to address issues such as natural circulation, flow stratification, and the effect of noncondensable gas distribution on local condensation and evaporation for the AP600 plant. Code upgrades that were completed in FYs 1992 and 1993 include the implementation of multicomponent capability, development of the liquid film tracking model, and incorporation of heat and mass transfer models. Code validation effort has been initiated using the 1/8th-scale test results for the Westinghouse AP600 passive containment cooling system. After the validation is

completed, COMMIX can serve as a benchmarking tool for portions of the CONTAIN code.

The **CORCON** code was developed as a best-estimate computational tool to calculate the thermal hydraulics and chemistry involving the progression of high-temperature core debris as it erodes concrete in the reactor cavity. A significant update of the code, designated **CORCON-MOD3**, was completed in FY 1993. This update involves improved axial and radial heat transfer models; inclusion of a condensed phase chemistry model for oxide-metal reactions; improved coolant heat transfer models, including the effects of subcooling and gas injection on film boiling; addition of models for interphase mixing and stratification, improvement of models for bubble behavior (e.g., bubble size, bubble rise velocity, and void fraction); and incorporation of the VANESA model for aerosol generation and radionuclide release. A topical report has been prepared to describe the phenomenological models and correlations incorporated in the code and to identify acceptable limits of validity for the models and correlations. Extensive validation of the code was performed in FY 1993 to determine its capabilities and limitations. The CORCON-MOD3 is now incorporated into CONTAIN and MELCOR.

VICTORIA is a computer code designed to analyze fission product behavior within the reactor coolant system (RCS) during a severe accident. The code provides detailed predictions of the fission product release from the fuel and transport in the RCS of radionuclides and nonradioactive materials during core degradation. During FY 1993, assessment and validation of models used in the VICTORIA computer code against existing data bases and against new data from various experimental test facilities (e.g., FALCON VI, ST) were carried out.

Battelle Columbus Laboratory is performing (for NRC) additional experiments and analyses on the deposition and revaporization of certain radionuclides. The purpose of these experiments is to identify thermodynamic properties of the radionuclides in the primary circuit during a severe accident and advance the chemisorption models. Boric acid is known to react with both cesium hydroxide and with cesium iodide to form the less volatile species cesium borate. The chemistry of boric acid is complex because it decomposes to a variety of different species, depending on the environmental conditions. The reactions under investigation will enhance the models already in the VICTORIA code.

HMS is a best-estimate, transient, three-dimensional code for analyzing the transport, mixing, and burning of hydrogen. The code can model geometrically complex structures with multiple compartments and can simulate the effects of condensation, heat transfer to walls and internal structures, chemical kinetics, and fluid turbulence. During FY 1993, the assessment work of HMS

against three test problems was completed, and a report was issued documenting the governing physical equations and computational model of HMS. During FY 1994, the HMS user's manual will be developed to provide the basic information for setting up and running problems with the code. Also, the HMS will be converted from a main frame computer code to a workstation environment code.

IFCI is an Integrated Fuel-Coolant Interaction code that was formulated as an independent stand-alone version in 1993 for workstation application. The capabilities of the code are described below in the section on **Fuel-Coolant Interactions**.

5.3 Severe Accident Phenomenology

5.3.1 Statement of Problem

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

5.3.2 Program Strategy

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

experiment at high fuel temperatures. Here the core geometry is changing and fission product chemistry and its effect on the retention of fission products within the RCS are significant. The FCI work is focused on the development and validation of appropriate phenomenological and analytical models addressing the fundamental aspects of FCI, namely, melt quenching and FCI energetics.

5.3.3 Research Accomplishments in FY 1993

The phenomenological research areas for improved understanding of accidents include the quantification of source terms, core melt progression, primary system failure from severe accidents, and fuel-coolant interactions.

5.3.3.1 Source Terms

"Source Terms" refers to the magnitudes of the radioactive materials released from a nuclear reactor core to the containment atmosphere, taking into account the timing of the postulated releases and other information needed to calculate offsite consequences following a hypothetical severe accident. NRC research in this area helps update TID-14844, which has been in use for three decades, in connection with plant siting assessments. The update to TID-14844 was published in draft "Accident Source Terms for Light Water Nuclear Power Plants" (NUREG-1465).

The NRC has also entered into an agreement with the Commissariat L'Energie Atomique of France (CEA) to participate in the PHEBUS-FP program. The program, sponsored by the CEA and the Commission of the European Communities, is aimed at studying in an in-pile facility, under sufficiently prototypical conditions, those phenomena that govern the transport, retention, and chemistry of fission products during LWR severe accident conditions. Phenomena to be studied are those occurring in the core, the primary reactor coolant circuit, and the containment. 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 reactor coolant system 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. The experimental data from PHEBUS-FP is confirmatory in nature and will be used to assess the revised source term assumptions used in NUREG-1465.

The PHEBUS-FP facility has received a license to refuel and start up. Final preparation for the first test, FPT0, is near completion, and the test is scheduled for the first quarter of FY 1994. The test matrix consists of six tests, with testing at a rate of one per year. The test matrix was revised recently to include fission product release tests

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under shutdown conditions for a degrading core in an air environment and for late-phase core geometry.

5.3.3.2 Core Melt Progression

"In-vessel core melt progression" describes the state of an LWR reactor core from core uncover 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 assessing the loads that may threaten the integrity of the reactor containment. Significant results of melt progression are the melt mass, composition, temperature (superheat), and the rate of release of the melt from the core and later from the reactor vessel if vessel failure does occur. Melt progression research also provides information about the in-vessel hydrogen generation, the conditions that govern the in-vessel release of fission products and aerosols and their transport and retention in the primary system, and the core conditions for assessing accident management strategies.

Much has been learned about the processes involved in core degradation and in the early phase of melt progression from integral tests in the PBF, ACRR, NRU, PHEBUS, and NSRR reactors, from the LOFT-FP2 test, from tests in the German CORA ex-reactor fuel-damage test facility, and from separate-effects experiments on significant phenomena. Most of the currently available information on late-phase melt progression has come from the post-accident examination of the Three Mile Island Unit 2 (TMI-2) core and the MP-1 and MP-2 experiments at the Sandia National Laboratories. In the TMI-2 core, a debris-supporting metallic blockage formed across the lower core from the relocation and freezing of metallic melt (mostly cladding) during coolant boildown. A growing pool of mostly ceramic uranium dioxide fuel melt was formed later by decay heating in the particulate ceramic (fuel) debris bed above the metallic core blockage. Even though the core had been previously reflooded by the high-pressure injection system, the growing pool melted out the side of the core and partially drained into the water-filled lower plenum of the reactor vessel.

Current NRC research on melt progression is focused on two major issues. The first issue is determining if there are any accident conditions for BWRs (and possibly also for PWRs) in which a metallic core blockage similar to that at TMI-2 would not be formed so the metallic melt, and later the ceramic (fuel) melt, would drain when formed from the core into the lower-plenum water. The second issue concerns the conditions for meltthrough of the growing pool of ceramic melt above the metallic blockage. The meltthrough threshold and location determine the mass of the melt released from the core and later from the reactor vessel.

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. All but one of the previous experiments for both PWRs and BWRs were performed for these wet core conditions, and this one experiment did not address the blockage or drainage question. The emergency operating procedures for U.S. BWRs, however, call for reactor depressurization, which lowers the water level below the reactor core 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 block the core as at TMI-2. Drainage 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 1993, a series of new laboratory experiments were prepared to address whether metallic melt drainage, core blockage, or core plate blockage occurs under BWR dry core accident conditions, and the initial two tests in a simplified development system were performed successfully. The experiment test assemblies in the follow-on tests will be a mockup at full radial scale of a cross section of the lower quarter of a BWR core (and core plate region) where blockages might occur. The experiment will have prototypic reactor materials, heat capacities, geometries, and temperature distributions. In these experiments, melts of metallic Zircaloy that also contain control-blade materials are poured into a test assembly at prototypic rates (dribbles), and the melt relocation and blockage behavior are observed.

Late-phase melt progression experiment MP-2 was performed in the Annular Core Research Reactor (ACRR) in FY 1993 to complete the currently planned program of experiments on late-phase melt progression, which involves ceramic (fuel) melting and relocation. Along with the earlier MP-1 experiment, MP-2 has provided unique information on the governing processes in the growth and meltthrough of a ceramic melt pool in a particulate ceramic debris bed in blocked core accidents. These are the conditions of the TMI-2 accident. With the results and interpretation of MP-1 and MP-2 in hand, an expert peer review group will be convened in FY 1994 to review the status of the current knowledge of late-phase melt progression, the significance of the remaining late-phase melt progression uncertainties, and the need for, efficacy of, and nature of any further research in this area.

5.3.3.3 Reactor Vessel Integrity

In 1988, the NRC—in cooperation with 10 foreign countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency (NEA)—undertook a follow-on program to the TMI-2 core examination conducted by the U.S. Depart-

ment of Energy. Under this program, called the TMI-2 Vessel Investigation Project (VIP), test specimens from the lower head of the TMI-2 reactor vessel were removed in 1990, and examinations of the specimens were conducted to obtain information on the melt attack on the lower head during the accident. The United States and the foreign countries participating in the OECD/NEA project performed metallurgical and mechanical examinations of the TMI-2 test specimens. Results of metallurgical examinations of the vessel steel samples have provided estimates of temperature histories of the lower-head samples. These specimens indicated that some regions of the lower head reached temperatures during the accident that exceeded the transformation temperature of the steel (727°C). In fact, a hot spot was found in an elliptical region on the lower head (about 1 m by 0.8 m) in which the inner surface of the reactor vessel steel reached temperatures as high as 1100°C and remained at that temperature for approximately 30 minutes. Results of examinations of instrument tube nozzles extracted from the lower head showed that some nozzles were melted off by interaction with molten core debris, whereas other nozzles were not affected at all. In general, it was found that the nozzles with the greatest damage were located in the vicinity of the hot spot. Data from examinations of the reactor vessel specimens performed under the VIP, as well as other data from the TMI-2 accident, were used to perform calculations on potential reactor vessel failure modes. Results of these calculations showed that instrument penetration tube failures, such as tube ejection or tube heatup and ex-vessel failure, had large margins-to-failure during the TMI-2 accident. Results also showed that creep rupture of the vessel, either via global creep rupture or local creep rupture resulting from the hot spot, were the more likely failure modes for the TMI-2 reactor vessel. All the technical work for the TMI-2 VIP was completed in FY 1993, with the final reports for this project issued in October 1993.

A more general study of reactor lower-head failure for both PWRs and BWRs has also been completed in FY 1993. The mode and timing of lower-head failure resulting from in-vessel melt progression have controlling effects on the subsequent containment loads during a postulated severe accident. A final report, "Light Water Reactor Lower Head Failure Analysis" (NUREG/CR-5642, October 1993), documents the results of this research. The report presents the results of potential failure mode analyses for a range of debris conditions, lower-head designs, and accident scenarios. The failure modes include global creep-rupture of lower head, penetration tube melt-through, tube ejection, and ablation by jet impingement of molten core material. In addition, an analysis of a limited vessel wall area that may be heated to a high temperature, as occurred in the TMI-2 accident, was also developed.

Results of the lower-head failure analysis are presented in NUREG/CR-5642 in terms of key dimensionless parameters to provide "failure maps" that indicate the relative potential for failure of the lower head in various failure modes. Thus, the methodology in the report should provide the most likely mode for the initial quantitative failure analysis.

Creep-rupture data and high-temperature material property data were required that were not previously available. Material data in the literature apply to design conditions, whereas these failure analyses require data in the vicinity of failure conditions, for example, creep-rupture times of 1 to 100 hours. Thus, the required data were obtained as part of the lower-head failure program for both pressure vessel steel and penetration materials (inconel, stainless steel, and SA105 steel). The new high-temperature data agree well with extrapolation of previous lower-temperature measurements and should be adequate for failure analyses.

To demonstrate that heat generated in the debris inside the reactor lower head can be effectively dissipated by boiling on the external surface of the vessel, a contract was awarded to Penn State. This program involves heat transfer measurement on a downward-facing surface (i.e., hemispherical and toroidal surfaces) to obtain the data base for critical heat flux and to develop a comprehensive model to predict the heat transfer from this surface.

5.3.3.4 Fuel-Coolant Interactions

Since the quantification of a steam explosion-induced missile as a possible mode of containment failure (alpha mode) in the reactor safety study called WASH-1400, significant progress has been made in understanding the limitations of damage from such potential missiles. For example, in NUREG-1150, alpha-mode failure is not a dominant contributor to early containment failure. The emphasis prior to NUREG-1150 on fuel-coolant interactions (FCIs) was directed at the conditions for in-vessel molten fuel pouring into a coolant pool and its likelihood of causing containment failure by energetic interactions. The current emphasis of FCI research is to provide the appropriate phenomenological and analytical tools to address those aspects of FCI that are germane to three specific issues: (1) FCI energetics, (2) fuel melt quenching in water pools, and (3) water added to a degraded core.

Complementary to the experimental programs on FCI, an Integrated Fuel-Coolant Interaction (IFCI) code was developed by the Sandia National Laboratories. Work was completed in FY 1993 to modify the IFCI modules and produce a stand-alone code for workstation computers. An operational report with examples of runs using the stand-alone version and a code manual were completed in FY 1993. Validation and peer-review of the code will be the major efforts for the next two years. The code

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calculates the four phases of the FCI process: mixing, triggering, detonating and propagating, and hydrodynamic expansion.

The NRC and the Safety Technology Institute of the Joint Research Center of the Commission of the European Communities at Ispra, Italy, have entered into a technical exchange arrangement to perform a series of FCI experiments at the FARO facility located in Ispra. At the FARO facility, large masses of prototypic reactor core materials can be melted and can interact with different depths of coolant at different temperatures and pressures. At least five molten FCI experiments will be conducted to obtain data prototypical of reactor conditions in the United States.

5.3.3.5 In-House Severe Accident Analysis Capability

Growth in the capability of workstation level computers provides the opportunity to run severe accident codes on other than main frame computers. In FY 1993, RES purchased workstations to enhance the in-house analysis capability at the NRC. These workstations are scheduled to be upgraded by March 1994. The upgrades will increase processing speed and electronic storage space. Reactor plant descriptions, or decks, for analyses using the MELCOR, SCDAP/RELAP, CORCON, CONTAIN, and VICTORIA codes have been installed on the workstations. Typical use of this new in-house capability has been to review input decks developed by NRC contractors and to use these decks to extend previous analyses. In addition, in-house analyses have been used to check new models in the codes and to do bounding calculations to determine the appropriateness of the new models. The following describes examples of some of the work done in this area in 1993:

1. A CORCON ABWR deck was developed, and sensitivity analyses were performed to investigate the effects of melt temperature, melt mass, melt composition, concrete composition, and other code parameters that influence chemistry and heat transfer modeling. Results of this work were provided to NRR.
2. An AP600 MELCOR input deck, developed by the Brookhaven National Laboratory, was run and evaluated. Several problems regarding control function activation levels were discovered, and NRR users were notified of these problems.
3. The in-house effort has included an application of CONTAIN to the AP600. Sandia National Laboratories developed the input deck and studied heat transfer modeling uncertainties and the uncertainties in passive containment coolant system (PCCS) shell wetting. The in-house effort extended these studies to look at uncertainties in thermal radiation modeling in the PCCS and aerosol modeling in the containment. Work has begun on an in-house code analysis effort using MELCOR 1.8.2 to review the new bottom-head model developed at the Oak Ridge National Laboratory.
4. A CONTAIN input deck has been prepared to model the SURTSEY facility at the Sandia National Laboratories. This deck is being prepared to gain experience with creating a CONTAIN input deck before attempting to create a larger, more complex CONTAIN deck modeling the ISP-35 (NUPEC mixing experiments).
5. The IFCI code (fuel-coolant interactions) was received, installed, and run on the in-house workstation. Also, the HMS code (gas flow, hydrogen) was transferred via internet from Los Alamos National Laboratory to the in-house workstation. The HMS code is currently being installed.

5.4 Reactor Containment Structural Integrity

5.4.1 Statement of Problem

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

5.4.2 Program Strategy

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

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

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

5.4.3 Research Accomplishments in FY 1993

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

In FY 1993, agreement was reached on the concept and preliminary design of a two-phase test of a model of a steel BWR containment. The model will be fabricated in Japan and erected at the Sandia National Laboratories in Albuquerque, New Mexico. The model will be at a scale of 1:10 to a typical BWR steel containment. The first phase of the test will not have a representation of a shield building and will simulate response to loading scenarios for which there is no contact between the containment and the shield building. That phase is planned for 1996. The second phase will involve the construction of a structure to represent the effects of a shield building and a test to simulate the response to loading scenarios for which contact is expected. That phase is planned for 1997.

In FY 1993, agreement was also reached on the preliminary design for a model of a prestressed concrete containment, representative of those used in PWR designs. The model will be at a scale of 1:4 to a typical large, dry PWR containment. The model structure will be designed by Japanese firms and constructed at the Sandia National Laboratories. The test, currently scheduled for 1998, will consist of a single phase in which the model will be pressurized up to failure.

5.5 Severe Accident Policy Implementation

5.5.1 Statement of Problem

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

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

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

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accident management programs at operating plants is a logical result of the IPE process.

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

5.5.2 Program Strategy

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

5.5.3 Research Accomplishments in FY 1993

In the 14 years since the TMI-2 accident, the NRC has sponsored an active program in research on severe nuclear power plant accidents as part of a multifaceted approach to the assurance of safety in this context. Other elements of this approach include improved plant opera-

tions, human factor considerations, and probabilistic risk assessments. In August 1985, the Commission issued a Severe Accident Policy Statement, which concluded that existing plants posed no undue risk to public health and safety. However, the Commission recognized that systematic examinations of existing plants could identify plant-specific vulnerabilities to severe accidents for which further safety improvements could be justified.

In connection with the implementation of the Commission's Severe Accident Policy Statement, the staff has required IPEs of all existing plants to identify any plant-specific vulnerabilities to severe accidents. The IPE process involves two different efforts. The first is an examination of existing plants for vulnerabilities to severe accidents resulting from events occurring within the plant (e.g., equipment failures, pipe breaks). The second effort is to consider severe accident vulnerabilities from external hazards (e.g., earthquakes, floods, winds). This activity is referred to as the individual plant examination for external events (IPEEE).

Twenty-six new submittals for IPEs were received from licensees in FY 1993, making an overall total of 63 submittals received to date. Staff evaluations were issued for Turkey Point 3 and 4, Oconee 1, 2, and 3, Beaver Valley 2, and Diablo Canyon 1 and 2 and draft staff evaluations were completed for FitzPatrick, Surry 1 and 2, Millstone 1, and Monticello. It is expected that all IPE submittals will be received and reviewed by the end of calendar year 1995.

The approach for review of the IPEEE will follow closely that developed for review of the internal-event IPE submittals. The staff initiated the procurement process to obtain contractual assistance for the IPEEE reviews. Four IPEEE submittals have been received, with two currently in the review process.

6. SAFETY ISSUE RESOLUTION AND REGULATION IMPROVEMENTS

6.1 Earth Sciences

6.1.1 Statement of Problem

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

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

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

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

6.1.2 Program Strategy

The strategy to resolve the seismic problem involves research to develop the methods and data that will support the necessary seismic criteria development and provide the evaluation tools. The research is focused on (1) im-

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

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

The backbone of the NRC program in the Eastern United States has been the seismographic networks deployed throughout the Eastern and Central United States. The NRC is currently funding seismographic networks in the following regions: Northeastern United States; Virginia; Charleston, South Carolina; the southern Appalachian region; the New Madrid (Missouri) region; Ohio and Indiana; eastern Kansas; and Oklahoma. An agreement was reached in 1986 between the United States Geological Survey (USGS) and the NRC to jointly support the establishment of the eastern portion of a national seismographic network. The eastern portion of the national network is now fully in place.

6.1.3 Research Accomplishments in FY 1993

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

Seismic hazards contribute a sizable proportion to its overall plant hazards and, because of inherent difficulties in defining them, they form an even more significant

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portion of the overall uncertainty in estimating plant hazards. In order to reduce these uncertainties, research into the causes and distribution of seismicity is continuing. Research is also progressing on improved methods of applying earth science information to estimates of ground motion levels for use in plant design.

6.1.3.1 Seismographic Networks

For about a decade and a half, the NRC has supported regional seismographic networks, primarily in the Eastern and Central United States where most of the nuclear plants are located and where seismicity is less well defined than in the Western United States. These networks have provided essential earthquake data to better describe the seismicity in this region and to compare the seismicity with geologic and tectonic information in order to gain insight into structures in the earth's crust that may create a potential for earthquakes. The NRC discontinued most of these networks in September 1992. Three networks, those operated by Weston Observatory, Massachusetts Institute of Technology, and Virginia Polytechnic Institute and State University, were continued through this reporting period until September 1993. The function formerly served by the regional networks has been taken over by the new National Seismographic Network (NSN), established through a cooperative agreement with the USGS.

The NSN was officially dedicated in April 1991. At present, the network consists of 16 network stations and 13 cooperating stations operated by the IRIS consortium and various universities. A number of regional seismographic network stations have also been integrated into the network, forming a national seismic system and providing more detailed coverage for special regions, such as the New Madrid, Missouri area. The NSN operates with high-quality three-component stations and satellite telemetry, thus providing data on significant earthquakes within minutes.

Near the end of this reporting period, a broad agency announcement was made public with the purpose of establishing research contracts for analyzing NSN data and other available seismological, geological, and geophysical data. This research will continue the type of investigations previously carried out by the universities operating regional networks. It is anticipated that the high-quality, broadband, and three-component data of the NSN will lead to new insights into the causes and distribution of seismicity and on the ground motion propagation characteristics of the earth's crust, particularly in the Eastern and Central United States.

6.1.3.2 Northeastern Neotectonics

During FY 1993, investigations were conducted at Wonalancet/Ferncroft in central-east New Hampshire, near the epicenter of the 1940 Ossipee earthquake; and in the

areas of northeastern Massachusetts and southeastern New Hampshire that were affected by the 1727 Newburyport and the 1755 Cape Ann earthquakes. Both areas are sites of current seismic activity.

In the Wonalancet/Ferncroft area, the investigations performed consisted of trenching, a ground-penetrating radar survey, geotechnical engineering tests, landslide/rockfall reconnaissance, and geophysics. No evidence was found that indicated the occurrence of prehistoric earthquakes larger than the 1940 event. Samples were taken for radiocarbon age-dating to determine the length of time during which a large earthquake has not occurred in this area.

In the Newburyport and Cape Ann region, many thousands of linear feet of exposure were observed along marshes, estuaries, and rivers. Although liquefaction-susceptible soils were found, no seismically induced paleoliquefaction features were identified along these exposures. Samples for carbon-14 age-dating were obtained to constrain the time period during which a large earthquake has not occurred.

During FY 1993, an investigation was begun in the epicentral area of the 1944 Cornwall-Messena earthquake (magnitude 5.5) to determine whether there was evidence for prehistoric moderate-to-large earthquakes. The 1944 event induced numerous liquefaction features, and the strategy is to identify similar features that pre-date the 1944 features and suggest earlier occurrences of similar earthquakes in the late Holocene.

Another paleoseismic investigation that is under way along the Atlantic coast of North America is a study of tsunami deposits left by the 1927 Grand Banks earthquake (magnitude 7), the development of criteria to distinguish between these deposits and storm-generated sediment, and a preliminary search for earlier tsunami deposits that were the result of prehistoric Grand Banks-sized earthquakes. Like the previously described studies, this one is an attempt to extend the relatively short historic seismic record back into time.

6.1.3.3 Faulting in Giles County Seismic Zone in Virginia

In June 1992, two faults were discovered at a borrow pit site in Pembroke, Virginia, near the epicenter of the 1897 Giles County Modified Mercalli Intensity VIII (magnitude 5.8) earthquake. The faults displace high-level terrace soils of the New River, which consist of bedded silts and gravels estimated to be of Quaternary age (less than 2 million years old). One fault strikes north 64° east, dips 60° to the northwest, and displaces the terrace strata more than 3 meters. The second fault strikes north 70° east, dips 80° to the northwest, and offsets the soils about 1 meter. Minor tension cracks and slip surfaces are also present in the outcrop. Further examination revealed that

these faults formed the eastern margin of a small graben (a narrow depression bounded by faults on either side) within the eastern limb of a small northerly trending anticline. Three possible origins for the faults are being considered: landslide, karst, and tectonic.

Additional investigations during FY 1993 revealed that the faults bounding the graben (the two faults described above on the east boundary and another fault forming the west boundary) experienced approximately 11 meters of normal displacement. Other faults were mapped in the excavation, which eventually exposed up to 50 feet of vertical face. The other faults consisted of small normal faults with 30 centimeters or less of offset and reverse faults with apparent displacements up to 1 meter. Preliminary geophysical investigations south of the exposure have been inconclusive as to the origin of the faults, but some methods show promise of helping to resolve this issue. Pending the availability of funding, core borings, trenching, age-dating of soils, and geophysical profiling are planned for FY 1994.

6.1.3.4 Paleoseismicity of Southern Illinois and Indiana

Investigation began in FY 1991 to identify and analyze paleoseismic evidence along the Wabash River valley and valleys of its major tributaries. To date, hundreds of planar, nearly vertical, sand- and gravel-filled dikes—caused by earthquake-induced liquefaction—have been discovered in these valleys in Indiana and Illinois. The dikes range in width from a few centimeters to as much as 2.5 meters; the largest of them being found around Vincennes, Indiana, and decreasing in size and abundance to the north and south of this area. Studies indicate that most of these features were caused by a large earthquake (estimated magnitude of about 7.5) that occurred in the Vincennes area between 2,500 and 7,500 years ago.

Investigations during FY 1993 were carried out in southern Indiana and Illinois. The results so far confirm the occurrence of a very large earthquake (magnitude 7.5) about 6,000–7,000 years ago, centered near Vincennes, and indicate that there was a liquefaction-producing, moderate earthquake about 4,000 years ago and a strong earthquake centered in southcentral Indiana 4,000–6,000 years ago.

6.1.3.5 New Madrid Seismic Zone

Several sites are being investigated in the New Madrid seismic zone to define faults associated with the seismicity there. The sites are at the intersection of Crowley Ridge and the west side of the Reelfoot Rift, Marston, Missouri, where waterfalls formed in the Mississippi River during the 1811–1812 New Madrid earthquakes; at the Crittenden County fault zone, which is the local east margin of

the Reelfoot Rift; and the Bootheel fault zone, which is located in the middle of the rift zone. Trenching will be carried out at specific sites pending the results of the geophysical studies.

Based on limited evidence, it is hypothesized that a recurrence interval for the 1811–1812 New Madrid earthquakes (magnitude 8) ranges from 550 to 1000 years. However, this idea was brought into question following the more recent work of Wesnousky, who examined the banks of numerous Corps of Engineers drainage ditches that exposed Holocene soils, which were strongly affected by the 1811–1812 earthquakes but showed no indication of deformation by prehistoric events. The focus of another research project in this region is to determine whether geologic evidence supports a recurrence of earthquakes like the 1811–1812 magnitude 8 events and to attempt to determine the ages of those events and the regional extent, if they exist, and to develop criteria for identifying them. Preliminary results indicate that there is paleoliquefaction evidence for at least one such prehistoric event.

6.1.3.6 Pacific Northwest

The Pacific Northwest, from southwestern British Columbia to northern California, is underlain by the Cascadia subduction zone, into which three minor oceanic plates—the Explorer, Juan de Fuca, and Gorda plates—are being subducted beneath the North American plate. Although geological and geophysical evidence indicates active subduction, there have been no historic large-thrust earthquakes along the plate interface—the type of earthquake that characterizes other active subduction zones around the rim of the Pacific Ocean.

The USGS has completed a major 5-year study of the geology and tectonics of the Pacific Northwest and continues to sponsor more limited research in this area. The NRC is partially funding several projects under this program in western Washington and Oregon. These efforts are continuations of investigations that revealed geological evidence suggesting the occurrence of several large prehistoric earthquakes during the past several thousand years. This evidence consists of several cycles of normal stratigraphic deposition of shallow marine sediments overlain by marsh deposits, each of which has been abruptly terminated by a catastrophic subsidence event and a new cycle has begun. These events are interpreted to be related to the occurrence of large subduction zone earthquakes. Along the coast, geologic and radiocarbon data indicate that the most recent of these events occurred about 300 years ago, affecting lowland soils at the Copalis River and at Willapa Bay about 65 kilometers apart. A 300-year-old event is also represented in northern California about 610 kilometers to the south. Two of the research projects have been concentrating on determining whether these widespread deformations were caused by a single magnitude 9 earthquake or by several

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smaller events of magnitude 8 or less. The data available so far supports either hypothesis.

Investigations along the coast of north-central Oregon confirmed the regional subduction zone subsidence events, but also identified geological evidence for local prehistoric earthquakes and subsidence-like evidence that may have been related to nonseismic phenomena such as storm surges or flooding due to the damming of estuaries by sand barriers.

In conjunction with these studies, a study is under way to identify and define seismically induced paleoliquefaction features in the region to determine whether strong shaking occurred during these subsidence events. Reconnaissance investigations in the Chehalis River valley and other drainages in southwestern Washington did not identify such features, even though there are long stretches of exposures of liquefaction-susceptible soils along the river banks.

The first positive evidence for seismic shaking that can be attributed to a subduction zone earthquake in the Pacific Northwest was found in the Columbia River estuary. The evidence consisted of seismically induced paleoliquefaction features (sand dikes and sills) on islands within the estuary. The features range from up to 0.3 meter in size, which are very numerous in the vicinity of Astoria, Oregon, to fewer in number and smaller in size upstream ranging from 7.5 to 10 centimeters about 30 to 40 kilometers away, and 2.5 to 5 centimeters wide about 60 kilometers inland. The dikes and sills are estimated to be about 300 years old, based on the estimated age of soils cut by the dikes (specifically, a 1,482-year-old layer of pumice), younger undisturbed soils, and the age of the oldest living trees (240 years) unaffected by the event. The evidence for shaking is correlated with the 300-year-old subsidence event in southwestern Washington.

Field studies in FY 1993 showed that paleoliquefaction features extend at least an additional 30 kilometers upstream in the Columbia River for a total distance of 90 kilometers from the coast. Preliminary geotechnical investigations of the liquefaction susceptibility of soils on Wallace Island in the Columbia River estuary suggest that the shaking that accompanied the 300-year-ago event was probably less than that which would be expected from a great subduction zone earthquake.

Geological evidence from excavations at West Point, Washington, 10 kilometers northwest of downtown Seattle, is interpreted to indicate that tsunami-like surges of sandy water from Puget Sound covered a tidal marsh that subsided at least 1/2 meter about 1,100 years ago. Estuarine mud about 1/2 meter in thickness overlies the sand and marsh deposits. Radiocarbon age dates of plants buried beneath the mud range from 900 to 1,300 years.

These data—along with other geological evidence gathered by other researchers in the Puget Sound region (such as submarine slides in Lake Washington, uplift at Restoration Point on Bainbridge Island, geophysical and stratigraphic evidence for a large east-west striking fault in south Seattle—Seattle fault)—suggest the occurrence of a large (magnitude 7) earthquake on the Seattle fault about 1,100 years ago.

Field studies in FY 1993 found additional evidence of a tsunami generated by the 1,100-year-ago event in a cove on Whidby Island in Puget Sound in the form of a buried sheet of sand that underlies the cove and laps up on its flanks. The cove is considered to be favorably oriented to receive a seismically generated sea wave from an earthquake on the Seattle fault.

6.1.3.7 Fault Segmentation Studies

It is well known that faults do not usually rupture over their entire length during a single earthquake. Numerous structural and paleoseismic studies and investigations of historical earthquakes indicate that there are physical controls within a fault zone that define the extent of rupture and divide a fault into segments and that these segments can persist through many earthquake cycles. This project is being carried out to establish a basis for recognizing and identifying geometrical and structural features that constrain or control rupture propagation within a fault zone.

Evaluation of the segmentation for selected faults was begun in FY 1991 using paleoseismic recurrence data and information on slip-per-event and slip rate. Studies in FY 1992 continued on these faults, including the Rodgers Creek-Hayward fault zone, the segment of the San Andreas fault that ruptured during the 1989 Loma Prieta earthquake, the Wasatch fault zone, and the Calaveras, Superstition Hills, Imperial, White Wolf, Lost River, Red Canyon-Hebgen, Dixie Valley-Stillwater, Pleasant Valley, North Anatolia (Turkey), Pitagcachi (Mexico), Oued Fodda (Algeria), Marriot Creek, Tennant Creek (Australia), and Landers faults.

Work during FY 1992 on the Rodgers Creek fault provided the first estimates of the timing of individual paleoearthquakes with events at about 1000 A.D., between 1200 and 1400 A.D., and between 1650 and 1808. Additional evidence supporting a 6-kilometer-wide step between this fault and the Hayward fault was found. Studies at Grizzly Flat on the San Andreas fault revealed evidence for the last two large surface faulting events, one after 1800 A.D. (probably 1906), and the other before 1636–1660 A.D. Along with evidence gathered by other researchers farther north along the fault, these findings indicate a recurrence interval along this part of the San Andreas fault of about 250 years. Initial data on fault geometry, lithology, and rupture directivity collected for the Coyote Lake, Morgan Hill, and Alum Rock

earthquakes on the Calaveras fault indicate a south-to-north progression of events. However, a north-to-south rupture propagation during each event was indicated.

Studies were begun late in FY 1992 on the complex 85-kilometer-long surface rupture of the 1992 Landers earthquake (magnitude 7.5) to determine its implications for segmentation modeling. This rupture was characterized by strike-slip faulting containing at least three major geometric segments with echelon steps up to 2.5 kilometers across.

In FY 1993, six trenches were dug across the 1992 rupture along the Homestead Valley fault where it truncated an alluvial fan and farther to the south where the rupture cuts across a playa. Four events have been identified: the 1992 event, an event about 4,000 years ago, one 8,000 years ago but with a very large error band, and an event about 14,000 years ago with an even larger error band. The next step is to excavate trenches across a segment of the Emerson fault that did not rupture in 1992 and continue to try to correlate events from fault segment to fault segment to test the fault segmentation model and the characteristic earthquake model.

6.1.3.8 Strong Ground Motion Studies

In 1989, in cooperation with the French Commissariat à l'Energie Atomique, a seismic experiment was undertaken at Garner Valley, California, to measure in situ amplification and attenuation of seismic waves that propagate through a soil column from bedrock to ground surface. The original contract was for the design, construction, and deployment of five downhole accelerometers and a field operable data acquisition system. In 1990, EPRI funded the installation of an additional downhole accelerometer and four surface accelerometers, along with additional data acquisition capability for the extra accelerometers. As presently deployed, the system is comprised of five surface accelerometers in a linear array spanning 310 meters and five accelerometers at depths from 6 meters to 220 meters. The network is located 7 kilometers from the San Jacinto fault, at the northern end of the Anza seismic gap on this fault, where a magnitude 6.5 or greater earthquake can be expected, and 35 kilometers from the Indian segment of the San Andreas fault.

Since its operation began, the downhole seismic array has recorded numerous earthquakes ranging in magnitudes from 6.1 to approximately 1.0. Analyses of the data through FY 1991 indicated that the spectral amplitudes recorded at ground surface are amplified on average by a factor of 10 over the spectral amplitudes at 220 meters depth. Resonance peaks have spectral ratios (surface spectral amplitudes divided by those at 220 meters) of about 40 for frequencies near 1.7, 3.0, and 12.0 Hz.

In FY 1992, more than 250 earthquakes were recorded, the largest of which was the April 23 Joshua Tree earthquake at a distance of 45 kilometers from the array and a depth of 13 kilometers. Maximum acceleration recorded from this event was 89 cm/s^2 at ground surface. Recordings were also obtained from the foreshock and the aftershock. Unfortunately, the data acquisition system was not working on June 25 during the Landers earthquake. Amplification characteristics for ground motion of FY 1992 earthquakes are being analyzed.

Because of the relative lack of near-field recordings of large intraplate earthquakes, such as those in the Eastern and Central United States, the prediction of strong ground motions radiated by these types of earthquakes is severely hindered. To compensate for this lack of near-field recordings, an analytic method was developed by the USGS to correct teleseismic recordings of the Global Digital Seismic Network for focal mechanisms, interference of the depth phases, and the teleseismic attenuation in order to estimate the acceleration source spectrum of the earthquake in the frequency band from 50 seconds to 2 Hz. Many large intraplate earthquakes have been analyzed to estimate the acceleration spectral level expected for near-field strong ground motion in northeastern North America. In FY 1992 the extensive near-field and regional accelerograph recordings from the 1989 Loma Prieta earthquake were analyzed with a view to applying the results to predicting strong ground motions in eastern North America.

During FY 1993, by studying the S-wave trains from 97 earthquakes recorded by the Eastern Canada Network, including the Saguenay, Mt. Laurier, Miramichi, Goodnow, Gaza, and Painesville earthquakes, a model for attenuation of ground motions was developed, and information was obtained about propagation and source characteristics in the Eastern United States.

One of the objectives in the USGS strong ground motions program is to use the stochastic model to predict ground motions from earthquakes in eastern North America. In FY 1992, an extension of the Boore and Atkinson (1987) ground motion predictions to deep soil sites was completed, representing an initial step in generalizing the prediction methodology to account for local variations in site geology. During FY 1992, much of the initial development of a strong-motion data base, including selection of those earthquake records that meet established quality control criteria, was completed.

The updated work in FY 1993 resulted in a division of sites into four building code-like classes based on shear wave velocities: soil A - $> 750 \text{ m/s}$, soil B - 750 to 360 m/s , soil C - 360 to 180 m/s , and soil D - $< 180 \text{ m/s}$.

Another ground motion study that was made during FY 1993 was in regard to rupture histories of eastern North

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American earthquakes. Sophisticated methods were used in the Western United States (inverting teleseismic and strong motion recordings for space-time slip distributions) to analyze large Eastern United States earthquakes such as Miramichi, two of the Nahanni events, Ungava, and Saguenay. The Saguenay event exhibited a concentrated source rupture pattern with an initial high-stress drop that spread over a broader area. The slip concentrations of the two Nahanni earthquakes were spatially complementary. The Ungava rupture took place within the upper 3 kilometers of the crust.

At the Savannah River site, a seismic array has been installed in a borehole. Four events in South Carolina were recorded, the largest of which was a magnitude 4 at Summerville. The data are still being analyzed, but initial results indicate that stress drops increase strongly with increasing moment.

Digital aftershock data from the 1992 Petrolia, California, earthquake sequence were analyzed in an attempt to determine the reasons for the high accelerations recorded at several of the stations. The results indicated that the high ground motions at the Cape Mendocino Station and the Petrolia General Store sites were most likely caused by site responses. The results of the analytic technique applied to other sites with anomalous readings indicated that the causes were attributable to either poor instrument calibration or to wave propagation characteristics.

6.1.3.9 Crustal Strain Measurements

During FY 1993, the crustal strain network for the Eastern and Central United States was measured for the third time since 1987. After this strain network was established, it became the backbone of a new geodetic network for the United States based on Global Positioning System (GPS) measurements. In addition, high-precision GPS networks have been established for many states; within the next few years all of the United States will be covered with detailed high-precision GPS networks for surveying purposes. Because of this, many stations are now available for strain determinations, in addition to the original 45 stations of the crustal strain network. These additional stations will also be periodically resurveyed and, in many locations, permanent GPS stations have been established that will provide a continuous record of measurements.

Because the intraplate strain rates in this region are expected to be low, many years may be needed to arrive at meaningful strain determinations. However, with the large number of high-precision GPS stations now available, it should eventually be possible to get a very detailed picture of strain distribution. Detailed information on deformations in the crust and their temporal rates will then provide a basis for refinements in seismic hazard determinations.

6.1.3.10 Probabilistic Seismic Hazard Assessments

Probabilistic seismic hazard assessments (PSHAs) were begun about a decade ago, and they have become an increasingly important aspect of site evaluations for nuclear power plants and other facilities. The revision to Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100, "Reactor Site Criteria," still in progress, will put substantial emphasis on PSHAs as part of the investigation required for proposed nuclear power plant sites. PSHAs are of particular interest in the Eastern and Central United States where uncertainties created by a lack of detailed knowledge of the seismicity make it difficult, by a deterministic evaluation, to arrive at a dependable estimate of seismic hazards.

Two large-scale PSHA studies are available for the Eastern and Central United States. One was performed by the Lawrence Livermore National Laboratory (LLNL) and sponsored by the NRC (NUREG/CR-5250); the other was performed by EPRI and sponsored by utilities in the Seismicity Owners Group. The two studies used similar methodologies and produced hazard curves with similar characteristics; they also produced consistent relative hazard rankings for plant sites in this region. A serious problem arises, however, from the fact that, at certain sites, absolute hazard levels may differ significantly.

Results from both studies are used by NRC staff for regulatory decisions, but, for future nuclear plant design and licensing, more consistent hazard values will be needed. At the end of FY 1992, an effort was begun to analyze differences between the LLNL and EPRI seismic hazard methodologies and to arrive at a more unified methodology that will produce more reliable absolute hazard levels. From previous analyses, it was known that methods of eliciting expert opinions and certain other factors—such as seismic parameters and ground motion models—cause some of the observed differences. The computer programs used for the LLNL and EPRI methods, although different, are designed to solve the same basic equation and do not seem to be a cause of discrepancy.

A new study will be conducted cooperatively by the NRC and the Department of Energy (DOE), both because its cost will be relatively high and because the DOE also has an interest in PSHA methods for assessing the numerous critical facilities it operates. EPRI will also make a significant contribution to the research through the DOE. The NRC is sponsoring a peer review by a panel formed by the Committee on Seismology of the National Academy of Sciences/National Research Council. The peer review panel will provide an independent, scientific review of the project and thus ensure the impartiality and objectivity of the study. It is expected that the study will be completed in 1994.

6.2 Plant Response to Seismic and Other External Events

6.2.1 Statement of Problem

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 PRAs that have been made.

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

6.2.2 Program Strategy

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

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

6.2.3 Research Accomplishments in FY 1993

6.2.3.1 Revision of Appendix A to 10 CFR Part 100

On October 20, 1992, the NRC published for public comment (57 FR 47802) the proposed revision of Appendix A to 10 CFR Part 100. The public comment period was extended twice—the first time (58 FR 271) so that the expiration date would be consistent with the expiration date of the supporting regulatory guides (57 FR 55601);

the second time (58 FR 16377) in response to a public request. The comment period expired on June 1, 1993.

Responses were received from approximately 47 domestic and foreign commenters. The domestic organizations providing comments included State geological surveys, the USGS, the Association of Engineering Geologists, and industry representatives. Nine foreign countries either individually or as a group provided comments. The staff is reviewing all the comments and will revise the regulations and guidance documents as appropriate during FY 1994.

Revisions of the geologic, seismic, and earthquake engineering criteria are being performed in conjunction with the revision of the reactor site criteria, 10 CFR Part 100.

6.2.3.2 Seismic Testing of Relays

Seismic testing of relays to determine the influence of relay chatter on circuit breaker tripping, among other things, was completed in FY 1993. The research initially was intended to support the resolution of USI A-46, "Seismic Qualification of Equipment in Operating Plants," but will also serve the needs of IPEEEs and seismic PRAs for advanced light-water reactors (ALWRs). Results obtained in FY 1993 indicate that relay chatter may or may not be acceptable in specific circuits depending on the circuit parameters. Thus, the two millisecond chatter criterion in IEEE codes may not be appropriate in all cases. Further evaluations are needed to decide on the appropriate course of action.

6.2.3.3 Hurricane Andrew

On August 24, 1992, Hurricane Andrew, a Category 4 hurricane, struck the Turkey Point nuclear power plant with sustained winds of 145 mph (233 km/h). During FY 1993, a combined NRC and Institute of Nuclear Power Operations (INPO) team investigated the impact of the hurricane on the Turkey Point Units 3 and 4 nuclear power plants. The emphasis of the investigation was on identifying those areas, events, or conditions that were problematic for the facility and Florida Power and Light (FPL) staff as well as those special preparations or actions that had a positive effect on the course and consequences of events relevant to plant safety. A RES staff member interviewed FPL staff about the performance of structures associated with the nuclear units, the chimneys and Bunker C oil tank associated with the fossil units, and an earlier systematic evaluation they had performed—the individual plant examination to identify severe accident vulnerabilities because it included the results of the wind and fire external-event analyses. The team report, "Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20-30, 1992" (NUREG-1474), was published in March 1993.

6.2.3.4 Earthquake Response

Following 1992 earthquakes in the Cape Mendocino and Landers/Big Bear areas of California, investigations were

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conducted to document the impact that the seismic events had on several selected non-nuclear industrial facilities. The facilities selected included a fossil-fueled power plant, cogeneration power plant, lumber mill, and a cement plant because they had steel framing, piping, and some equipment similar to that used in the nuclear power plant facilities. The effects of an earthquake on facilities of this type is perceived to be more severe than at a nuclear power plant because, in general, they do not have the stringent design requirements associated with nuclear facilities. The findings supplemented the existing experience data base and provided additional insights into the performance of nuclear power plant structures and equipment during an earthquake. Proposals have been made to use an experienced-based approach for the seismic qualification of selected equipment in ALWR designs.

6.2.3.5 Shear Wall Ultimate Drift Limits

The ultimate drift limit is defined as the lateral displacement at the top of the wall relative to its base normalized by the height of the wall. When performing seismic PRAs and seismic margin assessments (SMAs), the ultimate drift limit is necessary to estimate the seismic capacity of concrete nuclear power plant structures. In many cases, loss of equipment function has been considered to occur when the ultimate drift limits are reached; hence, the ultimate drift limit is a failure parameter in these studies. Seismic PRAs and SMAs have been identified as acceptable methods for performing the seismic portion of the IPEEEs for severe accident vulnerabilities. A research program was started this fiscal year with the objectives of establishing appropriate values of ultimate drift limit and obtaining the statistics to define this parameter in a probabilistic sense. It is anticipated that the technical report will be published in the second quarter of FY 1994.

6.2.3.6 Seismic Analysis of Piping

The ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I provides rules for the design of piping systems in nuclear power plants. In general, the design rules have been proved over the years to result in a design that affords reasonably certain protection of life and property and provides a margin for deterioration in service so as to give a reasonably long safe period of usefulness.

Recent developments in the nuclear industry have resulted in proposed changes to the design philosophy of piping systems for the ALWRs. Both the industry and the NRC staff have gained knowledge from piping tests conducted in the mid-1980's under NRC and industry sponsorship and from the failure data base obtained from actual seismic events at various facilities. The inherent ability of welded piping systems to withstand extremely large seismic inertia loadings is now recognized, and there

is a clearer understanding of the likely failure modes of piping systems under earthquake loadings. Design criteria need to address actual seismic failure modes in piping and need to be revised to eliminate excessive conservatism that do not add to safety and may hinder plant operation in the long term.

Therefore, the NRC has initiated a program in this reporting period whose objectives are: (1) to assist the NRC staff in developing regulatory changes on the subject of seismic analysis of piping systems and perform supporting research activities as needed; and (2) to evaluate the cumulative impact of proposed changes on the overall safety margins of the piping systems.

This program will be completed in 1995, allowing the staff to develop its position on the piping design requirements.

6.2.3.7 Cooperative International Seismic Programs

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

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

The LSST program was initiated in January 1990 and is expected to continue for 5 years. The goal of this program is to collect real earthquake-induced SSI data in order to evaluate computer codes used in SSI analysis of nuclear power plant structures. In the program, observations will be made on the motions of the reactor building model and the surrounding ground during large-scale earthquakes. The expectation is that the test model will be shaken by numerous earthquakes in this seismically active area of Taiwan. Instrumentation located on the scale model and in the field along a three-dimensional strong ground motion array will record any observed data. The LSST program at Hualien, Taiwan, is a follow-on to the SSI experiments at Lotung, Taiwan.

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

Framatome, the Korea Power Engineering Co. (KOPEC), and Korea Electric Power Corp.

During this reporting period, a collaborative effort involving exchange of technical information was established with the Ministry of International Trade and Industry and Nuclear Power Engineering Corporation (NUPEC) of Japan. In this effort, NUPEC is carrying out a seismic proving test program for a main steamline typical of the PWR plants and a feedwater system typical of the BWR plants. These tests will be conducted at the shake table of Tadotsu Engineering Laboratory and will begin in late 1994 and continue in 1995. Tests will be conducted for several levels of seismic excitation and will use energy absorber supports for the piping systems. The NRC in this collaborative effort will carry out pre- and post-test analyses to assess the applicability of currently available analytical models. In addition, data will also be obtained from NUPEC for seismic proving tests of a computer system and a reactor shutdown cooling system.

6.3 Generic Safety Issue Resolution

6.3.1 Statement of Problem

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

6.3.2 Program Strategy

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

6.3.3 Research Accomplishments in FY 1993

6.3.3.1 Priorities of Generic Safety Issues

The NRC revised the general methodology set out in the *1982 NRC Annual Report* for determining the priority of GSIs. An updated methodology (SECY-93-108), Revised Guidelines for Prioritization of Generic Safety Issues, was approved by the Commission and incorporated in NUREG-0933, "A Prioritization of Generic Safety Issues." In December 1983, a comprehensive list of the issues was published in NUREG-0933, and this list has been updated semi-annually with 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 1993, the NRC identified 5 new generic issues, established priorities for 12 issues (Table 6.1), and resolved 10 GSIs (Table 6.2). Table 6.3 contains the schedules for resolution of all unresolved GSIs.

6.3.3.2 Resolution of Human Factor Generic Safety Issues

GSI HF4.4 on procedures other than emergency operating procedures was resolved while an effort continued to develop a report describing work that relates to the use of procedures for low-power and shutdown operations. GSI HF5.1, on local control stations, was resolved and a lessons learned report on existing industry practices will be produced. Activity continued on the development of an advanced control room design review guide. The interactive version of the guide was demonstrated to the NRC staff. The final version is being developed for an application through Windows software. GSI HF5.2 on annunciators was resolved, and guidelines for the review of annunciators and alarm systems are being prepared for incorporation in the advanced control room design review guidelines, NUREG/CR-5908.

6.4 Reactor Regulatory Standards

6.4.1 Statement of Problem

RES has the primary responsibility to manage, coordinate reviews of, and control all NRC reactor-related (materials-related for Section 7.1.1) 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.

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Table 6.1 Generic Safety Issues Prioritized in FY 1993

Number	Title	Priority
146	Support Flexibility of Equipment and Components	RESOLVED
149	Adequacy of Fire Barriers	LOW
152	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	LOW
155.3	Improve Design Requirements for Nuclear Facilities	DROP
156.3.6.2	Emergency DC Power	LOW
159	Qualification of Safety-Related Pumps While Running on Minimum Flow	DROP
160	Spurious Actions of Instrumentation Upon Restoration of Power	DROP
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	DROP
162	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	DROP
164	Neutron Fluence in Reactor Vessel	DROP
166	Adequacy of Fatigue Life of Metal Components	NEARLY RESOLVED
168	Environmental Qualification of Electrical Equipment	NEARLY RESOLVED

Table 6.2 Generic Safety Issues Resolved in FY 1993

Number	Title
105	Interfacing Systems LOCA at LWRs
120	On-Line Testability of Protection Systems
142	Leakage Through Electrical Isolators
143	Availability of Chilled Water Systems and Room Cooling
153	Loss of Essential Service Water in LWRs
B-56	Diesel Reliability
HF4.4	Guidelines for Upgrading Other Procedures
HF5.1	Local Control Stations
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation
I.D.3	Safety System Status Monitoring

Table 6.3 Generic Safety Issues Scheduled for Resolution

Issue Number	Title	Priority	Scheduled Resolution Date
15	Radiation Effects on Reactor Vessel Supports	HIGH	03/96
23	Reactor Coolant Pump Seal Failures	HIGH	12/94
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	TBD
24	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	08/94
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	12/93
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	MEDIUM	TBD
106	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	10/93
B-17	Criteria for Safety-Related Operator Actions	MEDIUM	09/94
B-55	Improve Reliability of Target Rock Safety Relief Valves	MEDIUM	TBD
B-61	Allowable ECCS Equipment Outage Periods	MEDIUM	12/94
83	Control Room Habitability	NEARLY RESOLVED	12/93
145	Improve Surveillance and Startup Testing Programs	NEARLY RESOLVED	01/94
155.1	More Realistic Source Term Assumptions	NEARLY RESOLVED	01/94
166	Adequacy of Fatigue Life of Metal Components	NEARLY RESOLVED	TBD
168	Environmental Qualification of Electrical Equipment	NEARLY RESOLVED	TBD
B-64	Decommissioning of Nuclear Reactors	NEARLY RESOLVED	10/93
I.D.5(3)	On-Line Reactor Surveillance Systems	NEARLY RESOLVED	10/93

Needed reactor-related (materials-related for Section 7.1.1) regulatory products, e.g., regulations and regulatory guides, are developed. Rulemaking is proposed or initiated, as appropriate, and complex rulemakings that span the technical or organizational responsibilities of several groups or that involve novel or complex questions of regulatory policy are managed. Petitions for rulemaking are investigated.

6.4.2 Program Strategy

The purpose of the NRC nuclear regulatory program is to ensure that nuclear reactor (materials for Section 7.1.2) facilities are designed, constructed, and operated in a safe manner. Therefore, a continuing need exists to revise rules and guides and to develop new ones. The strategies

of this program are to (1) review the effectiveness of LWR regulatory requirements and guidance and make recommendations for revisions; (2) develop screening methodology to systematically review requirements and guidance; (3) coordinate and review proposed changes to the IAEA safety standards; (4) develop or assist the development of rules and regulatory guides; and (5) continue to develop and maintain management information systems for rulemaking.

6.4.3 Research Accomplishments in FY 1993

6.4.3.1 Elimination of Requirements Marginal to Safety

The NRC has instituted a program to eliminate requirements that are marginal to safety. The basic objective is to avoid dilution of safety efforts by reducing resource

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application to marginal safety issues. This improvement in efficiency is expected to result in a net beneficial effect on safety.

As part of this program (57 FR 55156) to eliminate requirements that are marginal to safety and yet impose a regulatory burden that more directly enhances safety, the NRC conducted a public workshop on April 27 and 28, 1993, in Bethesda, Maryland. The purpose of the workshop was to provide information on the NRC program, solicit comments from the public and regulated industry on the program, and discuss a number of specific initiatives being considered. The NRC encouraged the public and the regulated industry to attend the workshop and provide input to the NRC in the early stages of the program. In order to facilitate discussions at the workshop, advanced material on a framework for a performance-based regulatory approach and applications to three specific rulemakings were published (58 FR 6196) prior to the workshop.

Over 320 people attended the 2-day workshop, including representatives from 44 utilities, 5 industry groups, 8 vendors, 34 engineering and consulting firms, 4 public interest groups, and 6 State, Federal, and international government agencies. Representatives from an international union, law firms, and academia were also present. The discussions at the workshop have been documented in "Proceedings of the Workshop on Program for Elimination of Requirements Marginal to Safety" (NUREG/CP-0129), dated September 1993.

To implement this program, the NRC is currently taking action on requirements related to containment testing, fire protection, and quality assurance programs. The requirements in these areas will be modified to be less prescriptive and more performance-based to allow cost-effective implementation of regulatory safety objectives with marginal impact on safety. The NRC plans to use probabilistic risk analysis technology and its safety goals in reformulating requirements in these areas.

6.4.3.2 Other Rulemaking

The Commission issued a final rulemaking on April 26, 1993 (58 FR 21904), 10 CFR Part 50, on training and qualification of nuclear power plant personnel. The final rule amends the Commission's regulations to require each applicant and holder of a license to operate a nuclear power plant to establish, implement, and maintain programs that consider all modes of operation for the training of nuclear power plant personnel. The rule requires that the training programs be derived from a systems approach to training, as defined in 10 CFR Part 55. The objectives of the 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 (Public Law 97-425).

The Commission issued a final rulemaking on June 23, 1993 (58 FR 33993), 10 CFR 50.65, on monitoring the effectiveness of maintenance at nuclear power plants. The proposed rule was published on March 22, 1993 (58 FR 26938). The rule requires that the licensee conduct maintenance activities once every refueling cycle but not exceeding a period of 24 months. Because of the quality and quantity of data, this will provide a greater assurance that the nuclear power plant will operate safely.

The Commission issued a proposed rule on May 20, 1993 (58 FR 20336), 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses. The proposed amendment would delete the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator's 6-year license as a prerequisite for license renewal. Forty-two comments were received, the majority of which supported the proposed amendments. It is expected that the final rule will be published in FY 1994.

6.4.3.3 Regulatory Analysis

The Commission issued the proposed regulatory analysis guidelines for public comment (58 FR 47159) on September 7, 1993 (NUREG/BR-0058, Revision 2). The proposed guidelines represent the NRC's policy-setting document with respect to RIAs. The document contains a number of policy decisions for the preparation of an RIA performed to support NRC actions affecting reactor and non-reactor licensees.

Along with the guidelines, the NRC issued a draft report, "Regulatory Analysis Technical Evaluation Handbook" (NUREG/BR-0184). The purpose of the handbook is to provide guidance to regulatory analysts, to promote preparation of high-quality RIAs, and to implement the policies of the guidelines. The handbook expands upon the policy concepts included in the guidelines and translates the six steps necessary to prepare an RIA into implementable methodologies for the analysts. The guidelines and handbook establish the guidance and structure of the existing operating procedures, the better to integrate backfit analysis requirements and safety goal policy considerations.

Also to aid NRC analysts in preparing RIAs, the NRC published "Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States; 1992-1996" (NUREG/CR-4012, Volume 3), which updates replacement energy costs associated with short-term outages. These estimates can be useful in quantifying the overall impact of proposed regulatory actions when these requirements would necessitate retrofitting and short-term outages at nuclear power reactors. The NRC will continue to develop these methodologies in an effort to facilitate NRC decisionmaking in evaluating the need and effectiveness of the regulatory actions.

During this report period, about 15 safety-related RIAs were completed or initiated to justify specific regulatory actions for reactor and non-reactor licensees.

6.4.3.4 Maintenance Rule and Regulatory Guide

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 industrywide operating experience. Where monitoring proves unnecessary, licensees would be permitted the option of relying upon an appropriate preventive maintenance program.

The following chronology outlines the completion of the process to issue regulatory guidance to implement the maintenance rule.

- In November 1992, the draft regulatory guide and regulatory analysis for endorsement of the industry guidance document NUMARC 93-01 for implementation of the maintenance rule was issued for public comment (FR 57 55286). Eleven responses to the request for public comments were received. By the end of January 1993, the NRC staff had reviewed and resolved all public comments.
- In February 1993, the NRC staff made revisions to the regulatory guide in response to public comments. In April 1993, the NRC staff made presentations to the Advisory Committee on Reactor Safeguards and the Committee to Review Generic Requirements on the final regulatory guide.
- On April 28, 1993, the NRC sent a letter to NUMARC to suggest improvements in their guidance document resulting from the staff review of the public comments on the draft regulatory guide. By letter dated May 13, 1993, NUMARC transmitted a revised version of NUMARC 93-01, dated May 1993 in response.
- In June 1993, Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," which endorses NUMARC 93-01, dated May 1993, was issued.

A petition for rulemaking was received requesting that the regulations be amended to require licensees to specif-

ically consider the effects of electrical transients, including lightning strikes, on all electrical systems of nuclear power plants (PRM-50-56). On February 9, 1993 (58 FR 7757), a notice of denial for rulemaking was issued.

6.4.3.5 Summary of Rulemaking Actions

During FY 1993, 94 rulemaking actions were processed, of which 22 rules were formally published, 10 were terminated/withdrawn, and 62 are ongoing (see Table 6.4). Besides the 62 ongoing rulemaking actions, there are 30 potential rulemaking actions, and it is estimated that in FY 1994 there will be approximately 15-to-20 new rulemaking requests requiring RES review and approval by the Executive Director of Operations.

Table 6.4
Rulemaking Actions Processed During FY 1993

Rulemaking Activities	Number
Final Rulemakings Published	22
Rulemakings Terminated/Withdrawn	10
Ongoing Final Rulemaking Actions	22
Ongoing Proposed Rulemaking Actions	37
Rulemakings on Hold	3
Total Rulemakings	94

6.5 Radiation Protection and Health Effects

6.5.1 Statement of Problem

The NRC must provide reactor-related (materials-related for Section 7.1.1) 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 reactor-related (materials-related for Section 7.1.1) 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

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analyses are used to provide bases for severe accident consequence analysis, probabilistic risk assessment (PRA), the development of safety goals and emergency plans, the identification of radiation protection problems, the allocation of priorities for regulatory action, and environmental impact assessments. Recommendations of such organizations as the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP), Presidential guidance to Federal agencies, consensus standards, licensee performance indicators, cost and feasibility data, and available technical information also provide bases for developing regulatory and technical documents related to radiation protection for workers and the public.

6.5.2 Program Strategy

The Commission's regulatory process requires that safety enhancements to reactor (materials for Section 7.1.2) rules and guidance be systematically screened to ensure that there is substantial increase in public protection and that based on analysis the costs are justified. Realistic values of the dollar-per-person-rem criterion are needed for analysis to justify changes, but gaps in knowledge associated with radiation health effects cause uncertainties in these analyses. The strategies of this program are to identify and compensate for uncertainties in radiation risk coefficients used for health effect estimates in PRAs and regulatory decisions.

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

Federal guidance was approved by the President on occupational radiation protection. Further, the ICRP has published new recommendations for radiological protection. As a result of this new guidance, NRC reactor (materials for Section 7.1.2) regulations and regulatory guides will have to be revised. The strategies of this program are to (1) modify radiation protection guidance and standards to be consistent with Presidential guidance on radiation protection requirements and (2) continue to monitor licensee performance indicators by using the Radiation Exposure Information Reporting System program.

6.5.3 Research Accomplishments in FY 1993

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

6.5.3.1 Revision of Part 20 Radiation Protection Standards

Staff efforts in support of the implementation of the new 10 CFR Part 20 rule continued in FY 1993. These efforts included development of training courses, publication of questions and answers on Part 20, and publication of regulatory guidance. Also, several minor corrective rulemakings were completed.

Three new regulatory guides needed to implement the revised 10 CFR Part 20 were published. These guides are:

1. Regulatory Guide 8.9, Revision 1, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," was published in July 1993. The guide describes practical methods acceptable to the NRC staff for estimating intake of radionuclides using bioassay measurement techniques.
2. Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants," was published in June 1993. This guide describes a framework of graded radiation protection procedures recommended to ensure that control for access to high and very high radiation areas are appropriate to the radiation hazard present in those areas.
3. Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities," was published in July 1993. This guide provides guidance for materials licensees only and is addressed later in this chapter.

6.5.3.2 Brookhaven National Laboratory ALARA Center

The Brookhaven National Laboratory (BNL) ALARA Center, funded by the NRC, continued its surveillance and dissemination of DOE and industry dose reduction and ALARA research. BNL continued publication of the series that abstracts national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination (NUREG/CR-3469, Volume 7, July 1993). BNL also

continued publication of the newsletter, "ALARA Notes," on about a quarterly schedule. In 1993, BNL focused on making the data base more easily accessible, adding information from overseas contacts, making final plans for an international conference on dose reduction, and continuing development of an ALARA handbook. The center provided information and advice on dose reduction to NRC staff and licensees.

6.5.3.3 New Skin Dose Computer Code

A revised computer code (VARSKIN II) for calculating dose to the skin from radioactive materials on the skin was published (NUREG/CR-5873, December 1992). The revised code is more flexible than earlier versions, allowing consideration of factors such as self-absorption, particle shape, and particles on clothing.

6.5.3.4 Occupational Exposure Data Systems

The NRC continued to collect and process data in the computerized data system called the Radiation Exposure Information Reporting System (REIRS). REIRS provides a permanent record of worker exposures for reactors and several other categories of licensees. A report on 1991 exposures, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1991" (NUREG-0713, Volume 13, July 1993), was issued. Compilation of the statistical reports indicated that approximately 200,000 individuals were monitored and half received a measurable dose. The average measurable dose dropped from 0.36 rem in 1990 to 0.31 in 1991. The collective dose obtained from summing all the individual doses dropped from the 1990 value by 20 percent to about 32,000 person-rem. The data base also includes exposure data on individuals who have terminated employment with certain licensees. Data on some 687,000 persons are in the system, most of whom worked in nuclear power plants. NRC continued to respond to requests for individual exposure data from the system. The data also assist in the examination of the doses incurred by transient workers as they move from plant to plant (about 2,900 in 1991).

6.5.3.5 Water Chemistry and Decontamination

Advanced Process Technology determined the effects of hydrogen water chemistry on radiation buildup in BWRs and identified the most promising mitigating techniques.

The Idaho National Engineering Laboratory obtained information on out-of-core PWR power stations that will be useful to NRR in evaluating system contamination (radionuclide surface concentrations and exposure rates) in advanced reactor designs.

OMNI Tech International has developed an on-line UV-Vis spectrometer that will be used to determine the con-

centration of the vanadous ion during chemical decontamination of nuclear power plants.

The NRC published "Enhanced Removal of Radioactive Particles by Fluorocarbon Surfactant Solutions" (NUREG/CR-6081, August 1993). The report provides test results for the radiation stability and the application of environmentally compatible liquids to the nondestructive decontamination of nuclear equipment using ultrasonics.

6.5.3.6 Performance Testing of Extremity Dosimeters—Pilot Test

The NRC published "Performance Testing of Extremity Dosimeters—Pilot Test" (NUREG/CR-5989, July 1993). This report is the third of a series of tests run against the draft performance standard for personnel extremity dosimeters, ANSI N13.32, in order to establish the appropriateness of the standard for use in dosimeter processing certification. The NRC presently requires licensees to become accredited or to use dosimeter processors accredited under the National Voluntary Laboratory Accreditation Program (NVLAP) operated at the National Institute of Standards and Technology (NIST). At present, the NVLAP accredits whole body dosimeter processors and will add accreditation of extremity dosimeter processors as soon as the standard is jointly approved for use by the NRC and NIST.

6.5.3.7 National Institute of Standards Technology

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

6.5.3.8 Electronic Personnel Dosimeters

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

6.5.3.9 Gamma Dose Spectrometer

Work is being carried out under a Small Business Innovative Research Phase II contract that involves the development of a gamma-ray dosimeter/spectrometer that will measure the gamma-ray spectrum over a wide range of

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energies. From this information and the electronic signal retrieved from the dosimeter, it will be possible to calculate, through the use of appropriate algorithms, the dose delivered to the skin, the eye, and the whole body. To date, an Active Differential Absorption Spectrometer has been designed, developed, and tested.

6.5.3.10 Spent Fuel Heat Removal

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

6.5.3.11 Radiation Exposure Monitoring and Information Transmittal (REMIT) System

A new software package, REMIT, for electronically reporting radiation exposure measurements to the NRC was made available (58 FR 41526; August 4, 1993). REMIT is designed to assist NRC licensees in meeting the reporting requirements of 10 CFR 20.1001 through 20.2401, as outlined in Revision 1 to Regulatory Guide 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data." REMIT is a menu-driven system for use on

a personal computer. It includes provisions for various dose calculations and can produce NRC Forms 4 and 5 in paper and electronic format. In addition, REMIT can import and export data from ASCII and data base files.

6.6 Small Business Innovation Research

Pursuant to the Small Business Research and Development Enhancement Act of 1992, Public Law 102-564, the NRC supports the Small Business Innovation Research (SBIR) program, which stimulates technological innovation by small businesses, strengthens the role of small business in meeting Federal research and development needs, increases the commercial application of NRC-supported research results, and improves the return on investment from Federally funded research for economic and social benefits to the nation. The NRC has participated in the program since its inception in FY 1982, promoting high quality, "cutting-edge" research of relevance and potential importance to the NRC mission. One goal of the program is to couple this research with follow-on private funding, pursuant to possible commercial application. As of FY 1993, the NRC was supporting 20 SBIR projects-in-progress.

PART 3--NUCLEAR MATERIALS LICENSING AND REGULATION SUPPORT

7. NUCLEAR MATERIALS

7.1 Materials Regulatory Research

7.1.1 Statement of Problem

(See Sections 6.4.1 and 6.5.1.)

7.1.2 Program Strategy

(See Sections 6.4.2 and 6.5.2.)

7.1.3 Research Accomplishments in FY 1993

7.1.3.1 Materials Licensee Performance

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

Function and task analyses of the systems involved in teletherapy and remote after-loading brachytherapy were completed as a first step in better understanding the root causes of human error associated with these systems. In-depth studies of procedures, training, human-system interface, and organizational policies and practices were also completed. Reports are being prepared on setting priorities of function and task performance problems related to human errors in terms of their safety significance and an evaluation of alternative approaches for resolving safety-significant problems. Specific work on the human factors evaluation of the industrial radiography system was discontinued and is being reconsidered in view of the rule change to 10 CFR Part 34 established in 1993.

7.1.3.2 Materials Regulatory Standards

The Commission issued a proposed rule (10 CFR Parts 30, 40, 50, 70, and 72) on January 11, 1993 (58 FR 3515) that would allow self-guarantee as an additional mechanism for financial assurance. This proposed rule is in response to a petition for rulemaking (PRM-30-59) submitted by the General Electric Company and Westinghouse Electric Corporation. The rule would allow certain financially strong, non-electric utility licensees to use self-guarantee as financial assurance for decommissioning funding. It would not apply to electric utility licensees. It is expected that the final rule will be completed early in FY 1994.

A final rule (10 CFR Parts 31 and 32) is under preparation on requirements for the possession of industrial devices containing byproduct material. This rule would require licensees to provide the NRC with specific information about radioactive material used under a general license.

The proposed action would improve the public health and safety by reducing the likelihood of unnecessary radiation exposures from radioactive materials by ensuring that generally licensed devices are accounted for and disposed of properly. It is expected that the final rulemaking will be completed in FY 1995.

A final rule (Appendix H to 10 CFR Part 73) on day-firing (of firearms) qualifications for security personnel at category I fuel cycle facilities was published on August 31, 1993 (58 FR 45781). This rule was needed to provide assurance that security force personnel maintain required weapon-handling and marksmanship skills by annual performance testing. The rule is applicable to the specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.

A final rule and a proposed rule (10 CFR 72.214) adding casks VSC-24 and TN-24 to the list of approved spent fuel storage casks were published on April 7, 1993 (58 FR 17948) and on April 16, 1993 (58 FR 19786) respectively. The rules would increase the number of spent fuel storage casks from which the holders of power reactor operating licenses can choose to store spent fuel under a general license.

A final rule (10 CFR Parts 26, 70, and 73) on fitness for duty for category I facilities and shipments was published in June 1993 (58 FR 31467). The rule amends the regulations for the possession, use, or transport of strategic special nuclear material (SSNM). This action was necessary to ensure that specific employees of licensees who possess, use, or transport SSNM do not have a drug or alcohol problem.

A final rule (10 CFR Part 73) on physical protection requirements at fixed sites was published in March 1993 (58 FR 13699). The rule clarifies the Commission's regulatory intent that protection against both radiological sabotage and theft of special nuclear material is not required at all facilities. This final rule also adds a requirement that protection be provided against radiological sabotage at nonpower reactor licensees who operate at or above 2 megawatts thermal, where deemed necessary.

A proposed rule (10 CFR Part 73) that would require a physical fitness program for security personnel at category I facilities was published for public comment on October 6, 1993 (58 FR 52035). The rule would add new requirements for a physical fitness program and annual performance testing or a quarterly site-specific content-based performance test. It is expected that the final rulemaking will be completed in FY 1994.

A proposed rule (10 CFR Part 72) on reporting events at Independent Spent Fuel Storage Installations (ISFSIs)

and the Monitored Retrievable Storage (MRS) installation was published on September 14, 1993 (58 FR 48004). This rule would ensure that significant events such as contamination events, personal injuries, fires, and explosions at these facilities were promptly reported so that the Commission could evaluate whether the licensee has taken appropriate actions and whether prompt NRC action is necessary. The proposed rule would improve public health and safety by reducing the likelihood of unnecessary radiation exposures from these events. It is expected that the final rulemaking will be completed in FY 1994.

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

A proposed rule (10 CFR Parts 30 and 35) extending the expiration date of the Interim Final Rule related to the preparation and therapeutic use of radiopharmaceuticals was published in May 1993 (58 FR 26938), and a final rule was subsequently published in July 1993 (58 FR 39130). This action allows licensees to continue to use byproduct material under the provisions of the Interim Final Rule until the NRC completes a related rulemaking to address broader issues for the medical use of byproduct material (including those issues addressed by the Interim Final Rule). It is expected that the broader rule will be issued as a final rule in FY 1994. This extension of the expiration date was necessary to maintain the relief provided by the Interim Final Rule until the broader rule is issued.

The petitioner, Amersham Corporation, requested that petition PRM-35-8 (add iridium-192 wire for the treatment of intestinal cancer) be withdrawn. The withdrawal notice was published on August 23, 1993 (58 FR 44466).

7.1.3.3 Materials Radiation Protection and Health Effects

Irradiator Rulemaking. On February 9, 1993, the NRC published a final rule on licenses and radiation safety requirements for irradiators (58 FR 7715). The rule established a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators. Irradiators use gamma radiation to irradiate products to change their characteristics in some way. The safety requirements apply to panoramic irradiators (those in which the material being irradiated is in air in a room that is accessible to personnel

when the source is shielded) and underwater irradiators in which the source always remains shielded under water and the product is irradiated under water. The rule does not cover self-contained dry-source-storage irradiator devices, medical uses of sealed sources (such as teletherapy), or nondestructive testing (such as industrial radiography). The effective date of the rule was July 1, 1993.

The NRC is now in the process of publishing for comment a draft regulatory guide, "Guide for the Preparation of Applications for Licenses for Non-Self-Contained Irradiators," which is related to the irradiator rulemaking. The guide describes the information that an applicant should submit for a new or renewed license application. Issuance is scheduled for November 1993.

Air Sampling. In September 1993, the NRC published "Air Sampling in the Workplace" (NUREG-1400). The report provides technical information on air sampling that will be useful for facilities following the recommendations in the NRC's Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace." This guide addresses air sampling to meet the requirements in NRC's regulations (10 CFR Part 20) on radiation protection. The report describes how to determine the need for air sampling based on the amount of material in process modified by the type of material, release potential, and confinement of the material. The purposes of air sampling and how the purposes affect the types of air sampling provided are discussed. The report discusses how to locate air samplers to accurately determine the concentrations of airborne radioactive materials to which workers will be exposed. The need for and the methods of performing airflow pattern studies to improve the accuracy of air sampling results are included. The report presents and gives examples of several techniques that can be used to evaluate whether the airborne concentrations of material are representative of the air inhaled by workers. Methods for adjusting derived air concentrations for particle size and methods for calibrating for volume of air sampled and estimating the uncertainty in the volume of air sampled are described. Statistical tests for determining minimum detectable concentrations are presented. How to perform an annual evaluation of the adequacy of the air sampling is also discussed.

In April 1993, the NRC published "DEPOSITION: Software to Calculate Particle Penetration through Aerosol Transport Systems" (NUREG/GR-0006). DEPOSITION is user friendly software to calculate particle losses in aerosol transport systems. Revision 1 to Regulatory Guide 8.25 states that use of the DEPOSITION software is an acceptable method to calculate particle loss in aerosol transport systems. This software was developed at Texas A & M University under an NRC grant. Research will continue there in FY 1994 on the design of sampling probes to minimize loss of particles at inlets and on where to place sampling probes in ducts relative to bends and contrac-

tions in duct diameter in order to obtain a representative sample.

Solubility of Particles in the Lung. In August 1993, a research contract was awarded to the Inhalation Toxicology Research Institute on "Methods for Determining the Solubility of Radioactive Materials in Order to Implement 10 CFR Part 20." The new 10 CFR Part 20 lists derived air concentrations (DACs) and annual limits on intakes (ALIs) for inhalation of radioactive materials according to their solubility in the lung. Compounds are classified as "D" (soluble), "W" (moderately insoluble), or "Y" (highly insoluble) based on their clearance half times. This roughly translates to the time needed to dissolve in lung fluid, i.e., days (clearance half-time less than 10 days) for D class compounds, weeks (clearance half-time between 10 and 100 days) for W class compounds, and years (clearance half-time greater than 100 days) for Y class compounds. The objective of the research is to identify methods that the NRC can accept for determining the solubility in the lung of radioactive materials that may be inhaled so that licensees can determine internal radiation doses to meet the requirements in 10 CFR Part 20.

ALARA Levels for Effluents from Materials Facilities. Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities," was issued in July 1993. Section 20.1302(b) of 10 CFR Part 20 requires licensees to demonstrate compliance with the annual dose limit for members of the public in 20.1301. In addition, 10 CFR 20.1101(b) requires that licensees use procedures and controls to achieve doses to members of the public that are as low as is reasonably achievable (ALARA). This document provides guidance for materials licensees, such as medical and academic institutions, on acceptable methods of demonstrating compliance with this new mandatory ALARA requirement.

Patient Release Criteria. A proposed rule to amend 10 CFR Parts 20 and 35 concerning the criteria for the release of patients administered radioactive material, as well as a regulatory guide and a comprehensive regulatory analysis to be published as a NUREG report, was drafted and is expected to be published for comment in December 1993. This rulemaking action addresses the requests of two petitions for rulemaking: PRM-20-20 from Dr. Carol S. Marcus and PRM-35-10 from the American College of Nuclear Medicine. The petitioners requested that the Commission adopt a dose limit of 0.5 rem for individuals exposed to patients who have been administered radioactive material. It is expected that the final rulemaking, as well as the regulatory guide and NUREG report, will be completed in FY 1994.

Improvement of Health Effects Models. "Health Effects Models for Nuclear Power Plant Accident Consequences Analysis" (Revision 1 to NUREG/CR-4214) 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, "Modification of Models Resulting from Recent Reports of Health Effects of Ionizing Radiation," was published in 1991. The reports that led to modification of the models presented in NUREG/CR-4214 are the reports of the United States Scientific Committee on the Effects of Atomic Radiation (UNSCEAR, 1988), the National Academy of Sciences/National Research Council BEIR V Committee (NAS/NRC, 1990), and the revised recommendations of ICRP-60 (ICRP, 1991). A second addendum, "Modification of Models Resulting from Addition of Effects of Exposure to Alpha Emitters," was published in FY 1993. Revision 2 of NUREG/CR-4214, Part 1, "Introduction, Integration and Summary," which incorporates the new information presented in the two addenda was also completed in FY 1993 and will be published in early FY 1994. This project is completed.

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 1993, with significant progress. Revision 1 to NUREG/CR-5631, "Contribution of Maternal Radionuclide Burdens to Prenatal Radiation Dose" (March 1992), provides a method for calculating internal doses to the embryo/fetus with an expanded data base of radionuclides. Research that will permit inclusion of additional radionuclides, such as technetium, molybdenum, and additional transuranic elements began in FY 1993 and will continue in FY 1994.

An addendum, "Relationships Between Annual Limits on Intake and Prenatal Doses," to NUREG/CR-5631 was prepared in FY 1993 and will be published early in FY 1994. The methods and data developed under this project have been used by the NRC in preparing Regulatory Guide 8.36, "Radiation Dose to Embryo/Fetus," which describes acceptable methods of compliance with 10 CFR 20.1208. This guide might be revised to incorporate the information presented in the addendum. The methods developed under this project are also useful in calculating doses in cases of accidental releases of radioactive materials.

Criticality and Fuel Cycle Safety. A draft regulatory guide for criticality safety was published for comment on January 25, 1993 (58 FR 6022). This draft guide was developed to provide guidance to licensees on an appropriate nuclear criticality safety training program for the use of special nuclear material, especially the prevention of criticality accidents.

The Los Alamos National Laboratory continued its examination and revision of TID-7016, "Nuclear Safety Guide," for simplification of use, evaluation against new

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experimental data, and use of current computational codes. The document is a standard guide and reference used by the industry and the NMSS staff for initial criticality safety evaluations.

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

description of the critical experiments modeled, calculational results, quantification of trends in calculated k -effective's for different types of experiments, and recommended calculational uncertainties to be applied.

The availability of a draft regulatory guide for the fire protection of fuel facilities was published on April 22, 1993 (58 FR 21606). This regulatory guide was developed to provide guidance to applicants and licensees with respect to the information needed for the preparation of the fire protection sections (or chapters) of an application for a new license or for renewal of or amendments to an existing license for a fuel cycle facility. The guide also presents a standard format for submitting this information.

7.1.3.4 Uranium Enrichment

The Commission is considering issuing a proposed rulemaking to amend 10 CFR Part 76, "Regulation Governing the Operation of Gaseous Diffusion Facilities." This rulemaking is required by the Energy Policy Act of 1992 and will establish both the procedural and technical requirements for certification of the operation of the gaseous diffusion facilities by U.S. Enrichment Corporation. It is expected that a proposed rulemaking will be published early in FY 1994.

8. LOW-LEVEL WASTE DISPOSAL

NRC research in support of licensing activities for low-level radioactive waste (LLW) disposal facilities centers on (1) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial for LLW disposal and (2) evaluation of the overall performance of disposal systems. The NRC LLW research program is described in NUREG-1380, published in 1989. NUREG-1380 identifies issues, regulatory needs, a strategy, and a schedule for resolving them. 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 ’93” and the Annual DOE LLW Management Conference. In addition, the NRC and the U.S. Geological Survey conducted a 3-day meeting at Reston, Virginia, with State participation, on hydrology and geochemistry research addressing LLW concerns.

8.1 Statement of Problem

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

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

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

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

8.2 Program Strategy

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

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

8.3 Research Accomplishments in FY 1993

8.3.1 Materials and Engineering

8.3.1.1 Engineered Enhancements and Alternatives to Shallow Land Burial

Many States and State compacts are considering alternatives to shallow land burial for the disposal of LLW. Several concepts have been proposed as alternatives, and, of these, the most popular is the use of concrete as the principal construction material for engineered barriers to contain LLW. The National Institute of Standards and Technology (NIST) has continued investigating, on behalf of the NRC, the durability of concrete as an engineered alternative to shallow land burial, while the Idaho National Engineering Laboratory (INEL) completed an evaluation of concrete barriers in limiting radionuclide transport (NUREG/CR-6070). The NIST studies include concrete sulfate resistance research, the determination of diffusion coefficients for sulfate and chloride ions, the modeling of stresses caused by sulfate attack in concrete, investigation of cracking in concrete, and the durability of superplasticizers that may be used in concrete to reduce its transport

properties and improve its strength. Three reports are being prepared as NUREG documents that will be available in FY 1994; they describe (1) a new method to determine chloride diffusion coefficients, (2) the determination of sulfate diffusion coefficients, and (3) the modeling of stresses caused by sulfate attack in concrete. A new effort was started at NIST to develop computer models on the degradation of concrete. It is expected that a peer review panel report assessing these models and their application to LLW performance assessment will be issued early in FY 1994.

8.3.1.2 Anion Retention in Soil

The University of California at Berkeley investigated the use of natural soil materials to retard migration of anions at radioactive waste disposal facilities. Most soils are much more effective in retarding the migration of cations than anions. Certain long-lived radionuclides such as I-129 and Tc-99 may be in an anionic form at LLW disposal facilities. The anticipated application of this work is to identify materials that could be used to condition the near field either in or around LLW disposal facilities to retard migration and attenuate activities of radionuclides in anionic form. A literature review (NUREG/CR-5464) indicated that a group of soils called andisols, which are derived from the weathering of volcanic parent material, have significant potential for retardation of anionic forms of I-129 and Tc-99. Field work is under way in the Western United States to determine if exploitable deposits of andisols with anion exchange capacity are available. Preliminary results were published (NUREG/CR-5974) and work is continuing.

8.3.1.3 LLW Waste Forms

The stability of decontamination waste obtained from nuclear reactors using commercial decontamination processes and solidified in cement is being studied. Decontaminated LLW (collected from the Peach Bottom nuclear power station) is being tested at INEL. The studies are aimed at ensuring that radionuclide and chelating agent leaching characteristics, as well as the compressive strength of the cement solidified waste, are consistent with NRC technical positions and the requirements of 10 CFR Part 61 for waste form stability. Field lysimeter studies containing radioactive ion-exchange resins solidified in cement and vinyl ester-styrene are being conducted at the Oak Ridge and Argonne National Laboratories to determine radionuclide release rates under certain environmental conditions. Studies continued at INEL to investigate biodegradation of LLW by micro-organisms to ensure stability requirements, as required by 10 CFR Part 61. Studies continued at the Pacific Northwest Laboratories (PNL) to investigate activated metal and radioactive waste streams for radionuclides not included in the listing of long-lived radionuclides in 10 CFR Part 61, to

determine scaling factors for assessing hard-to-measure radionuclides in LLW, and to obtain activated metals from operating reactors for leaching and field lysimeter research studies. Studies were started at PNL to determine the presence of radionuclide-chelating complexes in leachates and behavior in soils.

8.3.1.4 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 6.) 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. Since this is a surface cover, it lends itself to use as a remedial action cover for sites susceptible to subsidence. The New York State Energy Research and Development Administration finished construction in 1993 of a bioengineering water management cover over such a trench at the West Valley LLW disposal facility, and the monitoring of performance has just begun. A second cover consists of a resistant 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.

PNL has developed an "infiltration evaluation methodology" (NUREG/CR-5523) that is being tested on the Las Cruces Trench field data set (NUREG/CR-5998) in the INTRAVAL project. The incorporation of various one- and two-dimensional analyses has also been applied and tested for conducting infiltration analyses for performance assessment and engineering design analysis (NUREG/CR-6114).

8.3.2 Hydrology and Geochemistry

8.3.2.1 Radionuclide Migration in Soil

A significant area of uncertainty in predicting site performance is the degree to which soils can retard radionuclide migration. To reduce this uncertainty, the NRC is investigating mechanisms controlling radionuclide movement through soils. The Sandia National Laboratories (SNL) are working on characterizing retardation mechanisms. The University of California at Davis is investigating the mechanisms and rates of dissolution for a variety of silicate minerals. This will be useful for understanding and modeling processes occurring on mineral surfaces that affect both sorption and leaching. PNL is examining the role in radionuclide transport played by microparticulates and naturally produced organic complexants.

8.3.2.2 Hydrology and Contaminant Transport

PNL has evaluated and developed a data set from an earlier field study involving subsurface injection of

radioactive tracers in heterogeneous unsaturated porous media at the Hanford site. The data sets reported in NUREG/CR-5996 cover a period of 10 years and will allow confirmatory analyses of existing flow and transport models that are to be used in LLW performance assessment. Work is being completed by the Massachusetts Institute of Technology (MIT) and Princeton University on the application of stochastic methods for simulating flow and transport in heterogeneous soils. In particular, Princeton University has completed and is now testing ground-water ventilation models for simulating vapor phase transport associated with LLW facilities, and MIT is applying their stochastic approach to field data sets.

8.3.3 Compliance, Assessment, and Modeling

8.3.3.1 Performance Assessment

Research is continuing on a performance assessment methodology. Emphasis is being given to engineering enhancements to shallow land burial, specifically the incorporation of concrete degradation models and the performance of cover materials. SNL has evaluated the current status of models used in performance assessment and published the results (NUREG/CR-5927, Volume 1). SNL is currently making improvements to the performance assessment methodology and assessing the validation approaches for the performance assessment models. 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.

8.3.3.2 LLW Source Term Modeling

During FY 1993, the existing LLW source term code, BLT (breach, leach, and transport), was benchmarked against field data and verified. Extensions to incorporate additional geochemistry and gaseous release are currently being investigated and planned for inclusion in the code during the next fiscal year.

8.3.3.3 Modeling of Tritium Migration at Arid Sites

The University of California at Berkeley, working cooperatively with CSIRO, developed a three-dimensional deterministic model based on soil physics to predict tritium migration at arid disposal sites (published as NUREG/CR-5980). The model awaits confirmation through the use of a controlled release from a known source at an arid site. Planning for such an experiment is under way.

8.3.4 Low-Level Waste Regulatory Standards

A final rule to amend 10 CFR Part 61 to clarify that requirements related to the performance of land disposal

facilities for LLW are applicable to aboveground disposal (i.e., built on the ground without an earthen cover) was published in June 1993.

A petition for rulemaking (PRM-61-2) from the New England Coalition on Nuclear Pollution was published in the *Federal Register* on July 23, 1992. The petitioner requests that the Commission amend its regulations regarding waste classification of low-level radioactive waste to restrict the number and types of waste streams that can be disposed of in near-surface disposal facilities. Recommendations on the need for rulemaking will be determined by December 1993.

8.3.5 Environmental Policy and Decommissioning

8.3.5.1 Enhanced Participatory Rulemaking

An Enhanced Participatory Rulemaking on radiological criteria for the decommissioning process was initiated to actively solicit early input from a wide spectrum of interests. Seven public workshops were held across the United States (Chicago, San Francisco, Boston, Dallas, Philadelphia, Atlanta, and Washington, D.C.), and these were followed by eight generic environmental impact statement (GEIS) scoping meetings in four cities (Washington, D.C., San Francisco, Oklahoma City, and Cleveland). The Environmental Protection Agency participated in these workshops and meetings and is a cooperating agency in the development of the GEIS. The staff began evaluating the input received and worked on preparing a staff draft of decommissioning criteria. The staff plans to complete the draft early in FY 1994 and continue seeking early input by releasing it to Agreement States and other interested parties in January 1994. The proposed rule is scheduled for publication in mid-1994.

The staff will continue to build experience with the use of the methodology described in the draft "Manual for Conducting Radiological Surveys in Support of License Termination" (NUREG/CR-5849) in the context of the present and proposed criteria for unrestricted release.

The NRC issued "Residual Radioactive Contamination from Decommissioning: Technical Basis For Translating Contamination Levels to Annual Total Effective Dose Equivalent" (NUREG/CR-5512, Volume 1) in October 1992. The complete report will consist of three volumes and one supplement. This first volume is designed to provide screening models, mathematical formulations for the screening models, and referenced parameter values for estimating doses, above natural background, to individuals from residual radioactivity associated with lands and structures after decommissioning licensed facilities. The modeling structure permits the use of either generic or site-specific parameters to be used as screening estimates of radiation doses from multiple environmental

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pathways. It is expected that the software to implement the models, D&DSCREEN, will be released for testing late in FY 1994 and will be accompanied by the user manual, NUREG/CR-5512, Volume 2.

8.3.5.2 Regulation Development

A proposed rulemaking (10 CFR Parts 30, 40, 70, and 72) on timeliness in decommissioning a materials facility was published on January 13, 1993 (58 FR 4099). The proposed rule would amend the Commission's regulations to establish timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities. The principal effect of these amendments is to formalize and codify the NRC's requirements for timeliness in the decommissioning of materials facilities. Seventeen comment letters were received, and the final rulemaking should be completed in FY 1994.

A final rule (10 CFR Parts 30, 40, 70, and 72) was published on July 26, 1993 (58 FR 39628) to amend the NRC'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 with the necessary information to ensure complete decommissioning of licensed facilities. This action is consistent with similar requests made at the Synar Committee hearing on decommissioning and an earlier GAO report.

A final rule (10 CFR Part 20) on disposal of waste oil by incineration at nuclear power plants was published on December 7, 1992 (57 FR 57649). The rulemaking action, responding to a petition for rulemaking originally filed by Edison Electric Institute and the Utility Nuclear Waste Management Group (PRM-20-15), allows reactor licensees to pursue the option of incineration of waste oils contaminated with small amounts of radioactivity without the need for specific authorization.

An advance notice of proposed rulemaking (10 CFR Part 40) concerning updating requirements for licensing of source material was published on October 28, 1992 (57 FR 48749). The contemplated rulemaking would improve the control of source material through more specific regulation and update applicable requirements to conform with the revised standards for protection against radiation. A broad range of issues are being considered in the categories of exemptions, general licenses, specific licenses, and mills and mill tailings. One specific issue identified in the advance notice is being handled in a separate rulemaking. A proposed rule conforming NRC's regulations governing uranium mill tailings to a proposed EPA

rule of June 8, 1993, was prepared for Commission consideration and is expected to be published early in FY 1994. This rule would address the timeliness of completion of radon barriers on uranium mill tailings and verification of the effectiveness of those barriers.

Responding to the Energy Policy Act of 1992, the Commission's Below Regulatory Concern Policy Statement was withdrawn. The withdrawal notice was published on August 24, 1993 (58 FR 44620).

The petitioner, University of Utah, requested that its petition, PRM-20-14 (disposal of biometical waste containing small amounts of radioactivity) be withdrawn. The withdrawal notice was published on July 22, 1993 (58 FR 39173).

8.3.5.3 Other Decommissioning-Related Actions

In September 1993, the NRC awarded a Small Business Innovative Research Phase II contract to Shonka Research Associates for a "Contamination Monitor Using Visual Identification." The monitor, using a position-sensitive proportional detector to detect contamination, is mounted on wheels and can be pushed over the floor like a lawn mower. The output from the detector goes to a computer, which creates a visual display in goggles worn by the operator. As the operator pushes the monitor over the floor, a "virtual reality" display of contamination can be seen on the floor. The monitor is faster and less expensive than conventional contamination detectors.

Brookhaven National Laboratory continued its determination of technical and safety criteria that should remain as part of decommissioning regulations under 10 CFR Part 50 when a licensee initiates action to permanently shut down the nuclear reactor in preparation for decommissioning activities. This project will develop a comparison of the safety requirements for a shutdown versus an operating nuclear power reactor after the reactor has permanently shut down. It will also perform financial assurance analysis for offsite liability requirements for shutdown reactors. It will examine the environmental impact of the potential increase in the spent fuel transport and radiological exposure to the public in the event the licensees prefer to ship and store their spent fuel. It is expected that a draft report for public comment will be issued early in FY 1994.

PNL continued providing support for nuclear facility decommissioning issues. An update of waste burial costs, "Report on Waste Burial Charges," was published in May 1993 as Revision 3 to NUREG-1307. "Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station" (NUREG/CR-5884) was published for public comment in September 1993. This report re-evaluated decommissioning costs for an earlier decommissioning study of a reference PWR reactor. "Estimating Pressurized Water Reactor Decommissioning Costs"

(NUREG/CR-6054) was issued in October 1993. This report describes the computer program used for estimating reactor decommissioning costs in the NUREG/CR-5884 reevaluation study.

PNL continued developing an information base on the actual radioactive contamination expected to be encoun-

tered at LWRs at the time of decommissioning, using actual field sampling and theoretical analysis. PNL is also updating and extending an information base on the technology, safety, and costs for decommissioning fuel cycle and non-fuel cycle nuclear facilities, using actual decommissioning data and analysis.

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

9. HIGH-LEVEL WASTE RESEARCH

The NRC maintains active research programs in rock mechanics and engineering, hydrology, geology, waste package performance, materials science, geochemistry, and several other disciplines related to the management of high-level waste (HLW). The research combines theoretical study with laboratory and field experiments to improve understanding of the physical processes that control and determine repository performance in the unsaturated volcanic tuff at the Yucca Mountain (Nevada) site currently under consideration by the Department of Energy (DOE) as directed by the Congress in December 1987. The ultimate goal of the NRC's HLW management research is to provide the technical bases for the licensing staff to make independent judgments as to the appropriateness and adequacy of DOE's demonstration of compliance for the HLW repository with NRC requirements and with the Environmental Protection Agency's HLW standard. Key technical issues being addressed include unsaturated flow and transport mechanisms, assessment of the potential for volcanic and seismic events, geochemical processes, and the long-term performance of engineered waste isolation systems.

9.1 Statement of Problem

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

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

program objective is to provide the technical capability necessary to evaluate DOE's site characterization activities as required by the NWPAA and to assess DOE's license application when it is submitted.

9.2 Program Strategy

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

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

9.3 Research Accomplishments in FY 1993

9.3.1 Engineered Systems Research

9.3.1.1 Stability of Underground Opening

When specifying suitable site conditions for an HLW 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 or motion caused by underground nuclear explosions at

9. High-Level Waste Research

the Nevada Test Site. Ground motion from either source could cause rock displacement and pressure changes in ground-water levels that could violate repository performance objectives.

To investigate the effects of seismicity on the underground openings for an HLW repository, the NRC is sponsoring research at the Center for Nuclear Waste Regulatory Analyses (CNWRA). The research includes the laboratory characterization of jointed fractured rock expected at the HLW repository horizon, the assessment of computer codes to calculate rock response to earthquakes, and field studies at the Lucky Friday Mine, Idaho, to measure rock displacements and ground-water response to mine seismic events. Results from the study indicate that underground openings at high states of stress are more sensitive to seismic loads than previously thought and that repetitive earthquake loading causes the displacement of rock joints, making the rock mass less stable. Seismic events of even small magnitudes cause changes in ground-water pressures as a result of volume changes in the rock. Rock/ground-water response to seismic events greater than those at the Lucky Friday Mine will be studied at the Garner Valley site in California. The California site can be subject to seismicity up to magnitude 6.5.

9.3.1.2 Thermohydrological-Mechanical Coupled Interactions

One important component of the safety analyses for HLW disposal is the coupling of the interactions between the rock mass, the ground water, and the thermal stresses induced by the high-temperature wastes. Coupling of the processes implies that one process affects the initiation and progress of the other, and independent consideration of each process is bound to be flawed. The NRC is a participant in an international multidisciplinary and cooperative research effort to study the coupled thermohydrological-mechanical (THM) processes under the acronym DECOVALEX (DEvelopment of COupled Models and their VALidation against EXperiments). The objectives of the study are to increase the basic understanding of THM coupled processes, support the application of codes for THM modeling for jointed hard rocks, and design validation experiments by means of THM model studies. In FY 1993, two benchmark problems simulating near-field conditions in an HLW repository were modeled, analyzed, and compared to results obtained by researchers from various countries using different computer codes. There was good agreement in the results. It is expected that the DECOVALEX study will be published in the International Society of Rock Mechanics Journal.

9.3.1.3 Materials Science

An understanding of the materials science aspects of the engineered barriers in HLW disposal systems is necessary

to permit the NRC to judge whether test data and models offer reasonable assurance of compliance with regulatory requirements. During 1993, the CNWRA conducted research on localized corrosion rates for candidate HLW package materials in tuff ground waters and continued to evaluate potential regulatory problems arising from stress corrosion cracking and metallurgical instability. Research was also initiated on the effect of water refluxing (due to thermal gradients) on degradation of waste package materials.

The CNWRA continues investigations of contaminant transport and material corrosion on ancient Minoan copper, bronze, and lead artifacts that were buried under silicic tuff 3,600 years ago. New metallic artifacts were uncovered, and the tuff immediately adjacent to the artifact was sampled to determine the extent of elemental transport in the unsaturated tuff. Hydrologic tests were conducted at the site in an attempt to model the hydrologic conditions to which the metals have been exposed. This work is providing data related to ongoing modeling research in copper alloy corrosion on unsaturated environments.

9.3.2 Geologic Systems Research

9.3.2.1 Hydrogeology

Since ground water is considered to be the primary agent of radionuclide transport from an HLW facility to the accessible environment, the NRC is actively studying ground-water infiltration, recharge, flow, and transport processes in partially saturated fractured rock. An experimental site in unsaturated fractured tuff, similar to Yucca Mountain, called the Apache Leap Tuff site, has been instrumented and characterized for testing instrumentation, methods, and analyses similar to those being used or proposed by DOE. In particular, field data sets are being collected and interpreted for testing mathematical models of flow and transport in partially saturated fractured rock. This work has been incorporated into the international project called INTRAVALE for model validation of ground-water transport models using field experiments. Key technical uncertainties dealing with determination and confirmation of ground-water travel times, presence and influence of perched-water systems, and preferential flow due to persistent discontinuities are being addressed.

Scientists at the CNWRA in San Antonio, Texas, are testing approaches to large-scale unsaturated flow in heterogeneous, stratified, and fractured geologic media. The BIGFLOW code has been developed, tested, and documented for use in their stochastic methodology (NUREG/CR-6028) for simulating flow in variably saturated, heterogeneous geologic media. To quantitatively account for spatial heterogeneity, CNWRA has applied the real space renormalization group method for parameter estimation. Work has begun on an analysis of the

regional hydrogeologic processes in the vicinity of the Yucca Mountain focusing on the appropriate integration of hydrogeologic, geophysical, and geochemical information and methods for testing alternative mathematical models of the regional flow and transport system.

The validity of conceptual and numerical models used to describe ground-water flow and radionuclide transport for various hydrogeologic settings is being evaluated in the INTRAVALE project. The NRC staff and research contractors from the CNWRA, University of Arizona, Sandia National Laboratories, Massachusetts Institute of Technology, Princeton University, and Pacific Northwest Laboratories are participating in this international effort involving 13 scientific parties from 10 countries.

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.

Research was completed on a grant at Johns Hopkins University to investigate numerical coupled thermohydrogeochemical modeling of large-scale transport processes that led to the formation of unconformity-type uranium ore bodies such as those found in the Alligator Rivers region of Australia and the Athabasca Basin in Canada. The model was able to simulate the detailed thermal, hydrological, hydrogeochemical, and mineralogical conditions and processes and the detailed final shape, structure, and mineralogy of the Cigar Lake, Canada, uranium ore body. This research model has the potential for providing a basis for major improvements in radionuclide transport modeling by performance assessment models.

9.3.2.2 Geochemistry

Geochemistry is involved in the assessment of all the HLW performance objectives and many of the siting and design criteria. Because the chemistry of the proposed site is dominated by the chemical effects of an enormous mass of rock compared to a relatively small mass of waste and engineered components, knowledge of geochemical effects on the behavior of the repository system and its components is essential. Over the long term, the performance of engineered components will depend on their compatibility with the geochemistry of the site. Moreover, there are geochemical clues to the past history of the site, including former site temperatures, the histories and ages of ground-water samples, and the ages of prehistoric geologic events such as earthquakes and volcanic eruptions. The NRC has a geochemistry research program actively investigating key technical uncertainties related to HLW disposal safety.

In 1993 geochemistry research focused on improving the tools used for the analysis of HLW geochemical pro-

cesses. Assessments of thermodynamic data for key minerals (zeolites) that strongly affect the ground-water chemistry of Yucca Mountain were completed by the CNWRA. Precise thermodynamic data and modeling showed that dissolution of the mineral analcime controls Yucca Mountain water chemistry. Research on the exchange of ions in solution with accessible sites in zeolite minerals continued to show the success of an ion-exchange model applied by the CNWRA to model interaction between zeolites and a number of dissolved fission product radionuclides. CNWRA research showed that the behavior of dissolved actinide elements could not be characterized well by simple linear models, and research continues on improving assessment methods for these important radionuclides. An extensive evaluation of more complex adsorption models was completed, and experiments were started to measure key sorption data.

From 1988 to 1993, the United States, represented by NRC, was one of five countries participating in the International Alligator Rivers Analog Project (ARAP). This project investigated the Koongarra uranium ore body in Australia. This ore body is a research site relevant to nuclear waste disposal because it has been subjected to dissolution and transport over the past one million years by oxidizing ground water flowing through rock fractures. A zone of dispersed uranium and other radionuclides has formed along the ground-water flow path. Processes controlling the long-term release and transport of uranium and other radionuclides can be intensively investigated. A 2-day final project presentation was given in October 1993 to the OECD Natural Analogues Working Group in Toledo, Spain. Mineralogical and hydrogeochemical analysis, combined with numerical geochemical equilibrium and kinetic modeling, showed that many of the same minerals and geochemical processes as those found at Koongarra will likely control radionuclide release and transport at Yucca Mountain. The project showed that integration of multidisciplinary geophysical, geological, geochemical, and hydrological data is necessary to describe a complex site in which ground-water flow occurs by both matrix and fracture flow. Numerical geochemical modeling powerfully described and predicted long-term evolution of the Koongarra site, but the need for better geochemical thermodynamic data for uranium silicate minerals (which will also tend to form at Yucca Mountain) was identified. Key uncertainties in making long-term predictions at Koongarra involve characterizing long-term climatic and hydrogeologic conditions. Performance assessment models were the subject of considerable investigation and discussion. Some investigators concluded that available performance assessment models were more sensitive than the Koongarra site to hydrological processes and less sensitive to geochemical processes. It is clear that these models would benefit from further research and development.

The NRC is sponsoring work by the CNWRA to investigate contaminant transport in an unsaturated tuff at the

9. High-Level Waste Research

Nopal I site in the Pena Blanca uranium district in Chihuahua, Mexico. The site is a tuff-hosted brecciated uranium ore body, which is analogous in many respects to the proposed repository at Yucca Mountain. This site is being studied to better understand the nature of contaminant transport in a fractured, unsaturated tuff (i.e., the relative roles and interaction of the matrix and fractures in transport and the alteration of uraninite) and in an oxidizing environment. Detailed geologic, fracture, and gamma spectroscopy maps have been completed on the cleared, exposed surface of the ore body. Transport of uranium tends to be concentrated along iron-stained fractures supporting in general the findings from the ARAP project on the association of uranium sorption with iron hydroxide minerals. Migration is generally thought to be fracture controlled; however, samples collected both along fractures and in the tuff perpendicular to the fractures indicate the relative mobility of uranium in the fractures versus the matrix. Uranium series disequilibrium studies are being conducted to determine the extent and nature of uranium mobility. The site also is being studied as an analogue of spent fuel corrosion. Detailed mineralogic and petrologic studies of the primary uraninite indicate a reasonable approximation of an oxidized spent fuel, but there are distinct differences in trace element concentration and grain size compared to unoxidized irradiated spent fuel.

9.3.2.3 Geology

The NRC has an ongoing project in volcanism in the Basin and Range to evaluate the potential for disruption of the repository by igneous activity. This work at the CNWRA focuses on determining the extent and availability of volcanic, tectonic, and geophysical data from the region surrounding Yucca Mountain. The NRC began a project on Central Basin and Range tectonics at the CNWRA. An extensive literature survey was completed for both volcanism and tectonics and a computerized digital geologic data base for the Yucca Mountain region has been successfully initiated. A review of age determination techniques for young basaltic volcanic rocks was also completed this year.

The NRC began an additional volcanology project at the CNWRA to examine the disruptive scenarios of small-volume basaltic volcanism in the Basin and Range. Three active volcanic sites have been identified for analogue studies of eruption dynamics, extent of disruption to the hydrogeology, and host rock mechanical stability based on volcanic events. Ancient cinder cone fields have also been identified as areas for the study of magma emplacement

dynamics and comparison with the data collected from active sites.

Research on contemporaneous deformation rates in the Death Valley region, using global positioning satellite interferometry, was started in FY 1993. This investigation will provide the NRC with an understanding of the regional geologic forces that will affect volcanism, seismicity, and faulting in the Yucca Mountain area and will provide a basis for informed NRC assessment of the DOE's site characterization data concerning geologic stability. At present the possibility that measurable deformation was occurring prior to and following the Little Skull Mountain earthquake near Yucca Mountain is being investigated.

9.3.3 Performance Assessment Research

The NRC will assess the claims of compliance made by the HLW licensee, the DOE, with the NRC's quantitative requirements for HLW disposal given in 10 CFR Part 60. Included (by reference in 10 CFR 60.112) in these requirements is the overall HLW repository performance standard, 40 CFR Part 191, set by the Environmental Protection Agency. The development of a methodology to quantitatively evaluate repository performance and the evaluation of the conceptual models used in the methodology are necessary to assess compliance.

The NRC is sponsoring research at the CNWRA to evaluate conceptual models used in the performance assessment of a repository in unsaturated, fractured tuff. In FY 1993, the CNWRA investigators completed development of a computer program (PORFLOW, NUREG/CR-5991) that is capable of simulating two-phase fluid flow conditions and radionuclide transport in geologic media. This program will be used to examine the representation of ground-water flow conditions during the thermal phase of the repository and simplifications used to represent the source term or releases from the engineered barrier system.

9.3.4 Regulation Development

A petition for rulemaking from the States of Washington and Oregon (PRM-60-4) was denied on March 4, 1993 (58 FR 12342). The petition requested the Commission to change the definition of high-level waste in its regulations so that some of the radioactive waste materials being processed at the DOE Hanford site could be classified as high-level waste. The petition was denied because the existing NRC regulations on waste classification are well established and can be applied on a case-by-case basis without revising the definition of high-level waste.

APPENDIX

FY 1993 Regulatory Products from the Office of Nuclear Regulatory Research

Date	Regulatory Product	Description
Part 1—NUCLEAR SAFETY RESEARCH—REACTOR LICENSING SUPPORT		
Reactor Aging and License Renewal		
Pressure Vessel Safety and Piping Integrity		
July 1993	Regulatory Guide	Revision 29 to Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability—ASME Section III, Division 1."
July 1993	Regulatory Guide	Revision 29 to Regulatory Guide 1.85, "Materials Code Case Acceptability—ASME Section III, Division 1."
July 1993	Regulatory Guide	Revision 10 to Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1."
September 1993	Draft Regulatory Guide	DG-1023, "Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-lb."
September 1993	Draft Regulatory Guide	DG-1025, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."
Standard Reactor Designs		
Regulatory Application of New Source Terms		
October 1992	Proposed Rule	Revisions to Part 100, "Reactor Site Criteria."
November 1992	Proposed Regulatory Guide	Proposed Revision 2 to Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations."
May 1993	Proposed Rule	The NRC regulation on emergency planning licensing requirements for independent fuel storage facilities.
May 1993	SECY-93-138	"Recommendations on Large-Release Definition."
June 1993	Proposed Rule	The NRC regulation on emergency planning.
August 1993	SECY-93-232	"Staff Approach for Assessing the Effectiveness of the Present Regulations with Respect to the Commission's Safety Goals."
August 1993	SECY-93-226	Public comments on proposed rule on ALWR severe accident performance.

Date	Regulatory Product	Description
Part 2—NUCLEAR SAFETY RESEARCH—REACTOR REGULATION SUPPORT		
Reactor Accident Analysis		
Severe Accident Policy Implementation		
October 1992	SECY-92-363	"Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations, and Status of Severe Accident Research."
April 1993	SECY-93-110	"Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations, and Status of Severe Accident Research."
May 1993	SECY-93-118	"Status of the Individual Plant Examination of External Events (IPEEE)."
Safety Issue Resolution and Regulation Improvements		
Earth Sciences		
October 1992	Proposed Rule	Proposed Appendix B, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," to 10 CFR Part 100 and proposed Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50 issued for public comment.
November 1992	Draft Regulatory Guide	DG-1015, "Identification and Characterization of Seismic Sources, Deterministic Source Earthquakes, and Ground Motions."
November 1992	Draft Regulatory Guide	DG-1016, Second Revision to Regulatory Guide 1.12, "Nuclear Power Plant Instrumentation for Earthquakes."
November 1992	Draft Regulatory Guide	DG-1017, "Pre-Earthquake Planning and Immediate Nuclear Power Plant Post-Earthquake Actions."
November 1992	Draft Regulatory Guide	DG-1018, "Restart of a Nuclear Power Plant Shut Down by a Seismic Event."
November 1992	Standard Review Plan	Section 2.5.2, Proposed Revision 3, "Vibratory Ground Motion."
August 1993	Proposed Rule	A proposed rule to amend 10 CFR 50.55a, "Codes and Standards," to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water

Date	Regulatory Product	Description
		Cooled Power Plants,” and Subsection IWL, “Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,” of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code.
Generic Safety Issue Resolution		
October 1992— September 1992	Generic Safety Issues	For generic safety issues prioritized and resolved in FY 1993, see Tables 6.1 and 6.2.
February 1993	SECY-93-004	“Resolution of GI-B56, Diesel Generator Reliability.”
June 1993	Regulatory Guide	Regulatory Guide 1.9, Revision 3, “Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants.”
June 1993	NUREG-1364	“Regulatory Analysis for the Resolution of GI-106: Piping and the Use of Highly Combustible Gases in Vital Areas.”
July 1993	NUREG-1463	“Regulatory Analysis for the Resolution of GI-105: Interfacing System Loss-of-Coolant Accident in LWRs.”
August 1993	NUREG-1461	“Regulatory Analysis for the Resolution of GI-153: Loss of Essential Service Water in LWRs.”
Reactor Regulatory Standards		
February 1993	SECY-93-028	Staff plans for the elimination of requirements marginal to safety.
April 1993	Final Rule	The NRC regulations (10 CFR Part 50) on training and qualification of nuclear power plant personnel. The rule amends the Commission’s regulations to require each applicant and holder of a license to operate a nuclear power plant to establish, implement, and maintain programs for the training of nuclear power plant personnel that consider all modes of operation.
May 1993	Proposed Rule	The NRC regulation, 10 CFR Part 55, on requalification requirements for licensed operators for renewal of licenses. The proposed amendment would delete the requirement that each licensed operator pass a comprehensive requalification written examination and an operating test conducted by the NRC during the term of the operator’s 6-year license as a prerequisite for license renewal.

Date	Regulatory Product	Description
June 1993	Final Rule	The NRC regulation, 10 CFR 50.65, on monitoring the effectiveness of maintenance at nuclear power plants. The rule requires that the licensee conduct maintenance activities once every refueling cycle but not exceeding a period of 24 months.
June 1993	Regulatory Guide	Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
June 1993	Regulatory Guide	Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants." This guide describes a framework of graded radiation protection procedures recommended to ensure that control for access to high and very high radiation areas are appropriate to the radiation hazard present in those areas.
July 1993	Regulatory Guide	Revision 1 to Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program." The guide describes practical methods acceptable to the NRC staff for estimating intake of radionuclides using bioassay measurement techniques.
July 1993	Regulatory Guide	Regulatory Guide 8.37, "ALARA Levels for Effluents from Materials Facilities." This guide provides guidance on ALARA for materials licensees only.
July 1993	Volume 13 to NUREG-0713	A report on 1991 exposures, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1991." It provides a compilation of the statistical reports of individual exposures.
September 1993	Revision 2 to NUREG/BR-0058	A draft report on proposed regulatory analysis guidelines for public comment. The proposed guidelines represent the NRC's policy-setting document with respect to regulatory impact analyses.
September 1993	NUREG/BR-0184	A draft report, "Regulatory Analysis Technical Evaluation Handbook." The purpose of the handbook is to provide guidance to regulatory analysts, to promote preparation of high-quality regulatory impact analyses, and to implement the policies of the guidelines.

Date	Regulatory Product	Description
Part 3—NUCLEAR MATERIALS LICENSING AND REGULATION SUPPORT		
Nuclear Materials		
October 1992	Final Rule	In an action related to the President's initiative to relieve the regulatory burden on nuclear licensees, a final rule to amend the NRC regulations (10 CFR Part 35) on departures from manufacturer's instructions on recordkeeping requirements. This revision eliminates recordkeeping requirements related to the justification for and a precise description of the departure and the number of departures from the Food and Drug Administration's approved manufacturer's instructions.
January 1993	Proposed Rule	A proposed rule (10 CFR Parts 30, 40, 50, 70, and 72) to allow self-guarantee as an additional mechanism for financial assurance. This proposed rule is in response to a petition for rulemaking (PRM-30-59) submitted by the General Electric Company and Westinghouse Electric Corporation. The proposed rule would allow certain financially strong, non-electric utility licensees to use self-guarantee as financial assurance for decommissioning funding. It would not apply to electric utility licensees.
January 1993	Draft Regulatory Guide	DG-3008, "Nuclear Criticality Safety Training." This draft guide was developed to provide guidance to licensees on an appropriate nuclear criticality safety training program for the use of special nuclear material, especially the prevention of criticality accidents.
February 1993	Final Rule	A final rule on licenses and radiation safety requirements for irradiators (58 FR 7715). The rule established a new Part 36 to specify radiation safety requirements and licensing requirements for the use of licensed radioactive materials in irradiators.
March 1993	Final Rule	The rule to amend the NRC regulations (10 CFR 73.40(a) and 73.60) on physical protection requirements at fixed sites. The rule clarifies the Commission's regulatory intent that protection against both radiological sabotage and theft of special nuclear material is not required at all facilities.
April 1993	Proposed Rule	The proposed rule to amend the NRC regulation (10 CFR 72.214) adding one cask to the list of approved spent fuel storage casks. The proposed rule would increase the number of spent fuel

Date	Regulatory Product	Description
		storage casks from which holders of power reactor operating licenses can choose to store spent fuel under a general license.
April 1993	Draft Regulatory Guide	"Standard Format and Content for Fire Protection Sections of License Application for Fuel Cycle Facilities."
April 1993	Final Rule	A rule to amend the NRC regulations (10 CFR 72.214) to add a cask to the list of approved spent fuel storage casks. The rule increases the number of spent fuel storage casks from which the holders of power reactor operating licenses can choose to store spent fuel under a general license.
June 1993	Final Rule	The rule to amend NRC regulations (10 CFR Parts 26, 70, and 73) on fitness for duty for Category I facilities and shipments of strategic special nuclear material (SSNM). The rule ensures that specific employees of licensees who possess, use, or transport SSNM do not have a drug or alcohol problem.
July 1993	Final Rule	A rule (10 CFR Parts 30 and 35) to extend the expiration date of the Interim Final Rule related to the preparation and therapeutic use of radiopharmaceuticals. This action allows licensees to continue to use byproduct material under the provisions of the Interim Final Rule until the NRC completes a related rulemaking to address broader issues for the medical use of byproduct material (including those issues addressed by the Interim Final Rule). It is expected that the broader rule will be issued as a final rule in FY 1994.
July 1993	Proposed Rule	A proposed rule to amend NRC regulations (10 CFR Parts 30, 32, and 35) on the medical use of byproduct material. This action, taken in response to a petition for rulemaking (PRM-35-9), is intended to provide greater flexibility by allowing properly qualified nuclear pharmacists and authorized users who are physicians greater discretion to prepare radioactive drugs containing byproduct material for medical use.
August 1993	Final Rule	The final rule to amend NRC regulations (Appendix H to 10 CFR Part 73) on day-firing qualifications for security personnel at Category I fuel cycle facilities. The amendment provides assurance that security force personnel maintain required weapon handling and marksmanship skills by annual performance testing for specific

Date	Regulatory Product	Description
		security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.
September 1993	NUREG-1400	"Air Sampling in the Workplace." The report provides technical information on air sampling that will be useful for facilities following the recommendations in the NRC's Regulatory Guide 8.25, Revision 1, "Air Sampling in the Workplace."
September 1993	Proposed Rule	A proposed rule to amend NRC regulations (10 CFR Part 72) on reporting events at Independent Spent Fuel Storage Installations and the Monitored Retrievable Storage installation. This rule would ensure that significant events such as contamination events, personal injuries, fires, and explosions at these facilities were promptly reported so that the Commission could evaluate whether the licensee has taken appropriate actions and whether prompt NRC action is necessary.
September 1993	Proposed Rule	A proposed rule to amend NRC regulations (10 CFR Part 73) on physical fitness programs for security personnel at Category I fuel cycle facilities. The amendment would require physical fitness training programs as well as annual performance testing for specific security force personnel at facilities authorized to possess formula quantities of strategic special nuclear material.
Low-Level Waste Disposal		
October 1992	Advance Notice of Proposed Rulemaking	An advance notice of proposed rulemaking (10 CFR Part 40) concerning updating requirements for licensing of source material. The contemplated rulemaking would improve the control of source material through more specific regulation and update applicable requirements to conform with the revised standards for protection against radiation.
December 1992	Final Rule	The NRC regulations (10 CFR Part 20) on disposal of waste oil by incineration at nuclear power plants. The rulemaking action, responding to a petition for rulemaking originally filed by Edison Electric Institute and the Utility Nuclear Waste Management Group (PRM-20-15), allows reactor licensees to pursue the option of incineration of waste oils contaminated with small amounts of radioactivity without the need for specific authorization.

Date	Regulatory Product	Description
January 1993	Proposed Rule	A proposed rulemaking (10 CFR Parts 30, 40, 70, and 72) on timeliness in decommissioning a materials facility. The proposed rule would amend the Commission's regulations to establish timeliness criteria for decommissioning nuclear sites or separate buildings or areas following permanent cessation of licensed activities.
June 1993	Final Rule	A final rule to amend the NRC regulations (10 CFR Part 61) to clarify that the requirements related to the performance of land disposal facilities for low-level waste are applicable to aboveground disposal (i.e., built on the ground without an earthen cover).
July 1993	Final Rule	A final rule (10 CFR Parts 30, 40, 70, and 72) to amend the NRC'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.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report, the ninth 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 1993. A special emphasis on accomplishments in nuclear power plant aging research reflects recognition that a number of plants are entering the final portion of their original 40-year operating licenses and that, in addition to current aging effects a focus on safety considerations for license renewal becomes timely.

The primary purpose of performing regulatory research is to develop and provide the Commission and its staff with the technical bases for regulatory decisions on the safe operation of licensed nuclear reactors and facilities, to find unknown or unexpected safety problems, and to develop data and related information for the purpose of revising the Commission's rules, regulatory guides, or other guidance.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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