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Vol. 2

NRC Safety Research in Support of Regulation - 1987

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

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ABSTRACT

This report, the third 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 1987.

The goal of this office is to ensure that research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

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PRINCIPAL RESEARCH ACCOMPLISHMENTS DURING 1987

The principal accomplishments achieved by the Office of Nuclear Regulatory Research during 1987 are listed below. Page references to more complete descriptions of these accomplishments are also provided.

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1-3	Completed the second Pressurized Thermal Shock Experiment (PTSE-2), a test of a deliberately flawed intermediate-scale pressure vessel containing material that simulated low upper-shelf weld metal. The test, which was performed under simulated pressurized thermal shock (PTS) accident conditions, supported the positions regarding PTS that appear in § 50.61 of 10 CFR Part 50.
1-4	Completed the experimental portion of the Steam Generator Group Project. Results included validation of empirical methods for predicting failure pressure of flawed steam generator tubing and highlighted possible inadequacies of eddy current inspection methods.
1-14	Completed a seismic margins review of the Maine Yankee nuclear power plant, using techniques developed under NRC sponsorship. A safety evaluation report, based on the results of this review, was issued.
2-3	Thermal-hydraulic facilities--At the MIST facility, which simulates B&W reactors of the lowered-loop design, completed all testing in the third phase of a four-phase program; at the ROSA-IV facility, used three test results in understanding and assessing the phenomena occurring during the depressurization process; and at the Japanese facilities that are part of the 2D/3D program, completed all testing.
2-3	Proposed amendments to 10 CFR Part 50 and Appendix K were published to revise the ECCS rule.
2-4	Code improvement focused on defining the accuracy of the TRAC-PWR code for large-break LOCAs in connection with the proposed Appendix K revision allowing realistic LOCA analysis.
2-4	Completed a data base management system (NUCLARR) for processing, storing, and retrieving human error and hardware failure data to support reliability evaluations, and completed a Cognitive Environment Simulation for analyzing intention formation aspects of human behavior.
2-5	Supported the implementation of an initial set of plant performance indicators.
2-6	Completed the Integrated Reliability and Risk Analysis System (IRRAS) version 1.0.
2-8	Developed five sets of guidelines for use in individual plant examinations.

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3-8 Draft NUREG-1150, assessing the risks from possible severe core damage accidents in five plants, was issued.

3-8 Updated the SARA program to accommodate NUREG-1150 plant PRA models.

3-9 Developed and used MACCS, a new model for assessing the consequences of radioactive releases.

3-5 Completed a pressure test of a 1/6th-scale model reinforced concrete containment built to ASME Boiler and Pressure Vessel Code standards. Failure occurred at 3.15 times the design pressure when a long tear developed in the steel liner. Pretest predictions of the pressure at which failure would occur and the nature of the failure were made by a number of U.S. and foreign organizations. The test result will be used to evaluate and validate the analytical techniques used for predicting containment performance.

3-6 Completed tests in the NRU reactor to provide full-length data on fuel damage during coolant boildown.

3-10 The ALARA Center, funded by NRC, reported that dose reduction research has resulted in a clear reduction of occupational radiation exposures.

4-2 Completed research on the use of ground-water dating techniques, which showed that a combination of isotopic and geochemical techniques have the potential to provide an independent data base for ground-water flow model validation. This is an important contribution to the bases for evaluating the site characterization report for a high-level-waste disposal facility.

5-1 Eight generic safety issues and two unresolved safety issues were resolved; full reviews were completed on 11 rulemakings.

5-4 NUREG-1226 was issued to provide guidance on the staff's advanced reactor review plans.

5-6 Completed a commission paper on the applicability of de minimis and below regulatory concern concepts that is to be used in developing a policy statement on allowable residual contamination levels following decommissioning.

1. INTEGRITY OF REACTOR COMPONENTS

The integrity of reactor components program examines reactor plant systems and related components to see that they perform as designed and that their functional integrity and operability are maintained over the life of the plant. Reactor safety depends on maintaining the integrity of the reactor system pressure boundary, i.e., maintaining it free from damage and leak-tight. Failure to maintain pressure boundary integrity could compromise the ability to cool the reactor and could lead to a loss-of-coolant accident accompanied by release of hazardous fission products.

1.1 Reactor Vessel and Piping Integrity

1.1.1 Statement of Problem

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

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 that of embrittlement of the pressure vessel steel as a result of bombardment by neutrons escaping from the fuel core during normal service. Experiments are thus conducted to develop a base of information on all the factors that will cause this embrittlement to increase during service life. Because it is known that cracks in thick-section materials representing real pressure vessels will respond to stress somewhat differently from cracks in small laboratory test specimens, much work is done to establish valid correlations between small-specimen predictions and thick-section material performance. Thus, use is made of large-scale "models" that realistically represent the true components. Similarly, the ability to predict integrity in piping has required testing of full-sized sections of pipe having a variety of cracks that could develop in service to determine if such cracks could cause failure during either normal service or an accident. For both vessels and piping, knowledge of the rate at which cracks grow is very important to ensure that a component will not fail during its forthcoming operational period. Thus, many experiments are conducted on a wide variety of pertinent materials under a very wide range of typical and expected exposure conditions to determine the maximum bounding rates of crack growth. Detection and sizing of flaws and cracks in all primary system components are conducted by the industry through periodic inservice inspections at shutdowns. To ensure that the inspections reliably detect and accurately size the flaws, extensive tests are conducted with inspection teams drawn from the industry using typical equipment and techniques on samples whose flaw conditions are known. From the results, it is possible to determine which techniques are effective and the magnitude of the error bands for both detection and sizing. Improvements in methods are proposed and qualification procedures developed that can provide better assurance for not missing flaws in future inspections and for sizing flaws more accurately. Use is made of materials and components removed from actual service to measure the real condition of material properties resulting from years of service, to establish the real corrosion state, and to validate the existence of flaws that have been "called" and estimated in size through nondestructive examination procedures.

1.1.3 Research Accomplishments in 1987

1.1.3.1 Pressure Vessel Safety

Under certain postulated accident conditions--such as small-break loss-of-coolant accidents, main steam line breaks, steam generator overfilling conditions, and associated instrument and component failures--a pressurized water reactor (PWR) pressure vessel could be subjected to severe cooling rates, which results in high thermal stresses, coupled with a continuing high pressure. This combination of thermal stresses and internal pressure, called pressurized thermal shock (PTS), could pose a serious challenge to the integrity of some older pressure vessels that have developed a significant degree of embrittlement due to neutron irradiation.

The resolution of the PTS problem, based on preliminary research results, took the form of a 1985 amendment of 10 CFR Part 50 (§ 50.61) that established an embrittlement screening criterion. In 1987, Regulatory Guide 1.154 was issued to furnish to the nuclear power plant licensees methods of analysis that may be performed to justify continued operation of the reactor beyond the screening criterion. Such justification may be based on corrective measures for reducing the rate of radiation-induced degradation of the vessel or for mitigating the effects of the degradation.

Although the rule amendment and the associated regulatory guide provide reasonable assurance that potential PTS accidents will not lead to failure of PWR pressure vessels, the actual margin against failure is clouded by the uncertainty introduced by the assumptions and engineering judgments needed to resolve the problem. Consequently, research has continued on several fronts to validate the rule and regulatory guide analysis and to quantify the inherent margin against failure. Topics that continue to be investigated include the effects of different materials (particularly the low upper-shelf welds), the effects of warm prestressing, the extension of the American Society of Mechanical Engineers (ASME) crack arrest toughness curves to higher values, and the effects of stainless steel cladding.

One way the NRC assesses the fracture behavior of pressure vessels subjected to PTS loadings is by intermediate-scale pressure vessel experiments. These experiments are performed by subjecting an intentionally flawed pressure vessel, whose wall thickness is nearly that of an actual PWR pressure vessel, to combined pressure and temperature histories that simulate a postulated PTS accident. Pressure, temperature, and flaw depth are carefully monitored so that the crack driving force and crack behavior (propagation or arrest) can be evaluated as a function of time. The second experiment in this series was performed in 1987 and evaluated a material that simulated a low upper-shelf weld material. The principal conclusions from the experiment are that (1) low upper-shelf material can exhibit very high crack arrest toughness--an important concept in evaluating crack stability during a PTS accident scenario; (2) warm prestressing inhibits brittle fracture to some degree even when crack driving forces are increasing with time although the benefits of warm prestressing are diminished by ductile tearing; and (3) a simple theoretical analysis of warm prestressing represented fracture conditions reasonably well, but calculations of ductile tearing based on state-of-the-art fracture analysis concepts did not consistently predict the observed fracture behavior.

The idea that a crack, extending rapidly through the pressure vessel wall with an increasing crack driving force, might slow and eventually arrest (stop) seems contrary to common sense. However, as a hypothetical crack would propagate from the inner surface to the outer surface of the reactor pressure vessel, the materials show an increasing resistance to crack propagation due to the increasing material temperature and to less severe radiation embrittlement. Recognizing these facts has led to including crack arrest concepts in the PTS methodology.

To make practical use of the crack arrest concept requires crack arrest toughness values for the material that are well in excess of values in the ASME Code. Therefore, the NRC began a study to validate the existing ASME Code curves, to extend the range of those curves, and to provide data to develop improved analytical models for a better understanding of the fracture process and of the margins against failure provided by current analysis criteria. The test specimens are 1 meter wide, 10 meters long, and either 0.1 or 0.15 meter thick. These large specimens are needed to prevent premature crack arrest and to preclude artificially low values of crack arrest resulting from stress waves reflected from the ends of the specimen. The work was started in 1984 and the 11th experiment was performed in September 1987. The results have shown that the existing ASME curves are a lower bound over their range. The higher crack arrest values obtained from the experiments agree with crack arrest values from the PTS experiments and from other work from Japan. These results and results from Europe and Asia were discussed during the third NRC-sponsored Crack Arrest Workshop held in May 1987.

Normal operation of reactors produces excess neutrons that strike the reactor pressure vessel walls, causing the steel of these walls to become embrittled and consequently lose its fracture toughness. The degree of toughness loss depends on several factors, including the chemical composition of the steel. This problem has been studied for many years, and the research has identified certain alloying and residual chemical elements that contribute to the radiation embrittlement problem. Based on this work, chemical composition standards for reactor pressure vessel steels have been developed to effectively minimize this problem. All the newer reactor pressure vessels were fabricated using materials made to these standards.

To evaluate the effects of radiation embrittlement, the NRC has sponsored several irradiation and testing efforts where test specimens of specific materials are irradiated to fluences corresponding to the projected end-of-life fluence for a typical reactor pressure vessel and then the specimens are tested. The results are contrasted to results from unirradiated samples to determine the degree of irradiation damage. Testing of the fifth irradiation series specimens was completed in 1987. This series is designed to validate the ASME Code's trend properties for the irradiation-induced changes in fracture properties used to evaluate pressure vessel integrity under both normal and accident conditions. The sixth irradiation series used the same material as the fifth series. The sixth series examines the effects of irradiation on crack arrest properties, again in order to confirm the ASME Code curves. The testing of the sixth irradiation series specimens began during 1987 and will be completed in 1988.

The results of this work will help to quantitatively define the margins of safety incorporated in the PTS rule. One of the questions yet to be resolved with the rule is the margins that exist for some of the low upper-shelf toughness energy welds found in some of the older plants. Still to be determined is the question concerning the similarity between the irradiated performance of this material undergoing a PTS accident as compared to the present-practice welds found in most United States reactors and studied in the fifth and sixth irradiation series.

1.1.3.2 Steam Generator Integrity

The Steam Generator Group Project at Battelle-Pacific Northwest Laboratories (PNL) has used a retired-from-service steam generator from an actual PWR facility as a test bed for measuring the effectiveness of eddy current inspection techniques to detect and size flaws in steam generator tubing. In addition, tube segments removed from the generator were burst tested to validate empirical [illegible] of [illegible] tube integrity developed earlier.

[illegible] examination of the removed tube segments and correlation [illegible] data were completed. Most defects were observed in the [illegible] near the top of the tube sheet. Pitting/wastage defects were [illegible] types observed in this region, and the most severely degraded [illegible] were from the hot leg side. Comparison of metallographic results and eddy current estimates of maximum defect depth showed a relatively large degree of scatter, and eddy current generally tended to underestimate the defect depth. The complex defect morphology coupled with the analyst's interpretation of the resulting complex eddy current signals were the primary causes of the observed variability in defect sizing.

Specimens removed from the generator with pitting/wastage defects along with tubes containing laboratory-produced stress corrosion cracks were burst tested in 1987 to validate empirical models of remaining tube integrity. Results indicated that these models adequately predict the failure pressure of inservice flawed tubing. Nearly all the specimens tested failed at levels several times the maximum pressure attainable during a main steam line break accident. This was because of the short axial extent of these defects and underscores the importance of knowing the length as well as depth of defects to arrive at a proper flaw evaluation.

1.1.3.3 Piping Integrity

A very significant problem encountered in BWRs has been the the intergranular stress corrosion cracking of austenitic stainless steel piping at weldments. This condition has been responsible for over 400 pipe-cracking incidents throughout the world over the last 10 years. Because these problems have resulted in extended and unscheduled outages, with extensive inspections, repairs, replacements, and significant occupational exposures, the NRC and the electric utility industry have devoted much research to their resolution.

NRC research in this area has been directed toward developing the capability to predict stress corrosion cracking in BWRs and to verify the acceptability of proposed fixes. In 1987, the research has included (1) the evaluation of stress relief and redistribution techniques, (2) studies of the stress corrosion cracking susceptibility of alternative piping materials, (3) the development of crack growth data for alternative materials and for pipe with weld-overlay repairs, and (4) studies on the effects of environment and temperature on stress corrosion cracking.

Analyses have been performed on welds treated by the mechanical stress improvement process developed to eliminate tensile stresses at the inside surface of piping. The welds were evaluated nondestructively and metallographically, and measurements of residual stresses were made. Experimental work found that the residual stresses due to this process generated the desired compressive stresses on the inside surface of piping in both axial and hoop directions at all locations and persisted through a substantial portion of the pipe wall. This indicates that the process can provide significant benefit even in the presence of small flaws that cannot be detected by nondestructive testing. This research has concluded that the basic concept of the mechanical stress improvement process is valid and sound and that the process is an effective means of improving the residual stress state of piping at welds.

The stress corrosion cracking susceptibility of Types 316NG and 347 stainless steel was investigated over a wide range of environmental and mechanical loading conditions using both constant extension rate and fracture-mechanics crack-growth-rate tests. A phenomenological model was developed to aid in the understanding and interpretation of the data. These tests have shown that Type

316NG stainless steel is extremely resistant to intergranular stress corrosion cracking but becomes susceptible to transgranular stress corrosion cracking at 289°C in water with dissolved-oxygen levels characteristic of conventional BWR water chemistries.

The process of crack growth in weld-overlay repairs of cracked pipe has been studied in simulated BWR environments and at low strain rates. The test specimens were fabricated, using standard industrial practice, in such a manner that the crack would propagate through the original sensitized pipe material into the weld overlay. The results of the experiment indicate that cracks do not extend into the weld overlay, confirming the suitability of this type of repair.

The pipe fracture experiment results from the Degraded Piping Program (phase II) have provided a means to validate the flaw evaluation procedures being developed by the ASME Code. For example, during 1987, work on welds in stainless steel pipe showed that the flaw evaluation procedures used for submerged arc welds are conservative and typically achieve actual margins of safety greatly in excess of the Code's intended margin. However, the research also showed that the fracture toughness of the actual weld fusion line is on the order of one-half that of the weld metal or the heat affected zone material. This discovery is important in defining material properties to be used in evaluating leak-before-break analyses as well as in making certain that the ASME Code is adequately conservative.

1.1.3.4 Inspection Procedures and Technologies

Research has been under way at PNL to develop the use of acoustic emission for the continuous online monitoring of reactors to detect and locate crack growth and to estimate the severity of the cracking from the acoustic emission signals. In 1987, activities focused on technology transfer by developing an ASTM standard for continuous acoustic emission monitoring of pressure boundaries, which has been approved, and by preparing a nonmandatory appendix to ASME Section XI Code for continuous monitoring of reactor pressure boundaries during operation, which is now in the approval process. Also, two successful applications of technology developed under this program were accomplished in the nuclear area. One was to monitor the High Flux Isotope Reactor vessel at Oak Ridge National Laboratory during a critical hydrostatic pressure test to verify that cracking did not occur in irradiation-embrittled sections of the vessel. The other was to monitor vitrified high-level waste during cooling to determine if and when cracking of the glass matrix occurred. Efforts continue to identify a circumstance where acoustic emission monitoring can be applied to reactor piping over a short period (1 year) to demonstrate acoustic emission detection of crack growth under actual reactor operating conditions. The availability and proper use of this technology will mean that reactors can be continuously monitored and that any cracks that develop can be detected and evaluated. In this way, proper and timely action can be taken to avoid extensive crack growth or component failure.

1.2 Aging of Reactor Components

1.2.1 Statement of Problem

Aging affects all reactor components, systems, and structures 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 components, systems, structures, 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 the initial license period of 40 years is contemplated, aging becomes a critical concern and will clearly be crucial to any assessment of the safety implications of license renewal.

Recently, the nuclear industry has initiated a significant effort aimed at extending the life of existing plants beyond their original license term of 40 years. According to a Department of Energy study, the projected net benefit to the United States economy can be on the order of $230 billion through the year 2030, assuming a 20-year life extension for current plants. If a 40-year life extension is judged feasible, the benefit is even larger. The benefit reflects both the lower fuel cost of nuclear plants compared to fossil-fueled plants and reduced outlays for replacement of generating capacity.

Utilities are currently planning to apply for license renewals and have outlined a tentative schedule for several steps in the process. The first submittal to the NRC is expected in 1993, with a large number of additional submittals to follow shortly thereafter. To keep pace with these industry plans, the NRC will need to devote effort over the next several years to license renewal. A firm NRC policy on the terms and conditions of license renewal applications can then be completed by early 1993. Review of these applications at an early stage will provide an indication to the industry of the viability of the life extension option in sufficient time to elect an alternative option if necessary.

1.2.2 Program Strategy

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

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

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

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

The program covers the electrical and mechanical components and systems important to the safe operation of the plant and the containment structure.

1.2.3 Research Accomplishments in 1987

1.2.3.1 Aging Research

Research studies were completed in 1987 on specific safety-related equipment in order to (1) identify failure mechanisms resulting from aging and service wear; (2) recommend maintenance, inspection, surveillance, testing, and condition monitoring to ensure operational readiness; and (3) establish degradation patterns for use in detecting incipient failures.

A survey and evaluation of power-operated relief valve (PORV) and block valve operating experience in nuclear plants (NUREG/CR-4692) was completed. This study provides data to support the resolution of Generic Issue 70 on PORV and block valve reliability.

An assessment of aging and service wear of auxiliary feedwater pumps (NUREG/CR-4597) was also completed. Potential damage to auxiliary feedwater pumps from low-flow testing operation was identified as a possible contributor to aging degradation.

The service water system was chosen as one of the reactor systems for study because it is the final link in the heat transfer chain between the reactor core and the ultimate heat sink. The focus of the investigation was on documenting from existing operational records the principal mechanisms of aging degradation of the system and on determining the adequacy of the current inservice surveillance and maintenance methods. The investigation revealed that the most prevalent degradation was related to corrosion of the piping, pumps, and valves forming the water passage from a natural source through the system.

Based on information derived from operating experience records, nuclear industry reports, and manufacturers' supplied information, reports were issued on electric motors (NUREG/CR-4156) and battery chargers and inverters (NUREG/CR-4564). Predominant electric motor failure modes are associated with the stator insulation system and the bearings. The failure mechanisms for stator insulation included loose laminations, shorted windings, overheating, and corrosion of electrical connections. The battery charger and inverter capacitors, transformers, inductors, and silicon-controlled rectifiers are the components most susceptible to aging. The research report concluded that overheating and electrical transients are two major causes of charger and inverter failures. Based on these research results, a national standard is being revised to reflect the research findings.

Emergency diesel generators used in nuclear power plants are exposed to aging stressors from the environment and from operational practices. A review of over 2,000 failures associated with emergency diesel generator systems revealed that roughly half the failures appear to be caused by some form of aging degradation. Examination of the various failure causes revealed that fasteners vibrating loose and metal fatigue are major contributors to emergency diesel generator system failures. Components of the instrumentation and control system (including governor, alarm system, and control air system), the cooling system pumps and piping, the fuel system injector pumps and engine piping, and the turbocharger are particularly sensitive to vibration-induced failure.

1.2.3.2 Chemical Effects

Effective amendments to 10 CFR Parts 30, 40, 50, 61, 70, and 72 were published in 1987 setting forth requirements that licensees notify the NRC in the event they are involved in bankruptcy proceedings. The purpose of this rulemaking is to require prompt notification to the NRC of licensee bankruptcy so that necessary action may be taken to deal with potential hazards to public health and safety posed by a licensee without the financial resources to properly handle licensed radioactive material or to clean up possible contamination.

The NRC continued to develop an information base for assessing the safety and effectiveness of decontamination alternatives for reducing occupational dose in nuclear power plants and for assessing the impact of decontaminations on nuclear plant solidification systems. Measurements were made of recontamination rates following chemical decontaminations at the Hatch Unit 2 (Ga.), Pilgrim (Mass.), Peach Bottom Unit 2 (Pa.), and Limerick Unit 1 (Pa.) nuclear power plants. A report analyzing these results and similar earlier measurements conducted at other nuclear power plants was being prepared at the end of the year. NUREG/CR-3444, published in 1987, describes the impact of LWR decontaminations on solidification and waste disposal.

1.3 Reactor Equipment Qualification

1.3.1 Statement of Problem

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

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

1.3.2 Program Strategy

NRC staff effort in the equipment qualification program is currently involved with developing the basis for resolving a high-priority generic safety issue (GSI 87) related to the operability of motor-operated valves (MOVs). Important information will be provided to eliminate the question as to whether gate valve thrust requirements should be based on opening or on closing motion. These results will be incorporated in the valve qualification standard and will clarify one of the areas the staff believes may be contributing to MOV problems. Another research effort is being devoted to understanding the effects of large earthquake loads on the operability of an aged gate valve. The effects on piping, support, snubbers, and anchors will also be determined. This cooperative program with the Germans uses one of their test facilities.

Future effort in the equipment qualification program will address new problems and safety issues consistent with safety and licensing needs. Since industry is expected to become more involved in solving some of the pressing valve safety problems, some NRC effort will be devoted to following this work and evaluating the results with regard to licensing applications.

The results from current and future efforts will be incorporated in appropriate qualification standards to provide the basis for ensuring safer components.

1.3.3 Research Accomplishments in 1987

1.3.3.1 Survival of Electric Equipment

Research on the survival of safety-related electric equipment when exposed to a hydrogen burn environment resulting from hydrogen deflagration following a LOCA core melt accident in PWR dry containment buildings (NUREG/CR-4763) was completed. The results of this research support resolution of Generic Issue 121 on a regulatory position on hydrogen control for large, dry and subatmospheric PWR containments.

Equipment temperatures were calculated for typical PWR large, dry and subatmospheric containments using multicompartment models of the TMI and the Surry nuclear plants and the HECTR hydrogen distribution and combustion code. A 75 percent metal-water reaction was assumed for the core melt accident as postulated in the Hydrogen Control Rule, § 50.44 of 10 CFR Part 50. Hydrogen concentrations in the detonable range were calculated to occur throughout containment for the smaller PWR subatmospheric containments in the absence of igniters. Detonable concentrations were calculated to be formed in all large, dry PWR containment subcompartments containing the LOCA system break in the absence of hydrogen igniters. Multiple hydrogen burns are possible in these subcompartments when igniters that can lead to equipment temperatures threatening their operation are present. The equipment temperatures from the analyses were verified by tests in the SCETCH facility at Sandia National Laboratories where the thermal and steam environment of a LOCA followed by a hydrogen burn was simulated.

1.3.3.2 Environmental Qualification of Mechanical Equipment

Experimental research conducted in 1987 demonstrated that typical dual-valve piping systems that penetrate the containment building will not experience failures when these piping systems are subjected to very large forces due to containment wall deflection. If a LOCA occurs inside containment, the internal environment will cause the containment wall to deflect outward and large forces can be transmitted to internal and external valves and piping supports through the attached piping. These forces can cause the piping and supports to plastically deform and create stresses in the valves that may cause binding of movable parts and affect the ability of these safety components to operate. The penetration structure welds may crack under these loads and result in a leakage path to the outside atmosphere. Although all these forces and stresses were simulated for three different, but typical, dual-valve piping systems, there was no evidence of system failures due to loads resulting from severe accident environments. These tests have provided the NRC staff with baseline data for assessing containment leak integrity and for demonstrating the ability of dual valves to function under severe accident conditions.

1.4 Seismic Safety

1.4.1 Statement of Problem

Earthquakes are among the most severe of the natural hazards faced by nuclear power plants. Very large earthquakes can seriously challenge the safety philosophies of redundancy and defense in depth that have been carefully designed into the plant by creating a common mode of failure. To ensure their safety, nuclear power plants are designed to withstand the stresses imposed by such extreme external events as earthquakes, floods, tornadoes, and other violent storms. The safety assumptions used in the plant designs are based on events that occur so rarely that the probability of the plant being faced with stresses that exceed these design bases is very difficult to quantify. As a consequence, they have presented a major source of uncertainty in reactor design and risk assessment in the past. Gradually the research program has resolved the main issues with respect to meteorological phenomena as they affect reactor design and the margin of safety. Seismic hazard in Central and Eastern United States remains an issue that is not likely to be easily resolved. Historically, the largest earthquakes in the United States have occurred at New Madrid, Mo., and at Charleston, S.C. The greater number of nuclear power plants in the Central and Eastern United States makes the seismic hazard issue one of major interest to the NRC. The problem is that the geology of the central and eastern regions makes it difficult to establish earthquake magnitudes or seismic parameters for specific locations or to ensure a proper design basis for individual power plants.

Generally, uncertainties such as those involved in seismic hazard analysis are not resolved quickly or with a single effort. For example, recent information from the United States Geological Survey (USGS) suggests that many of the currently operating nuclear power plants, particularly in the Central and Eastern United States, may be subjected to higher seismic loads than were specified when these plants were designed. While it is possible for a technical breakthrough to provide a definitive conclusion, it is more likely that the continuing accumulation of data and understanding will result in a concomitant gradual increase in the level of confidence held in the design basis of nuclear power plants for the Central and Eastern United States. This will require a continued level of effort in field investigation, data collection, and analysis. Faced with seismic hazard problems, the NRC needs validated analytical methods and an adequate data base to assess the capability of operating plants to sustain increased seismic loads. The research required to fill these gaps will support the development of simplified regulatory criteria and provide the NRC with an independent basis for licensing evaluations of seismic margins in operating plants.

1.4.2 Program Strategy

The strategy to resolve the 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) improvement of estimates of earthquake hazards by identifying potential earthquake sources and determining the propagation of seismic energy with distance, (2) assessment of the effect of earthquakes larger than the design basis on nuclear power plant structures, systems, and components, and (3) validation of the current seismic risk analysis methods. The result will be integrated assessments of seismic safety margins at much higher levels of confidence.

The work addresses short-term, high-priority research to resolve immediate licensing issues and long-term research to resolve concerns related to seismic design margins. The short-term research improves requirements by removing conservatisms where they are unnecessary and adding conservatisms where weaknesses in the regulations exist. Long-term research, on the other hand, aims at more strategic questions involving the overall view as to the importance of earthquakes in the regulatory process.

The program covers earth science (efforts to understand ground motion), seismic margins (capability to survive earthquakes larger than the design basis), seismic fragilities (seismically induced failure levels of structures or components), and validation (comparison of predictive methods with experimental data).

One very significant uncertainty in the seismic area is that associated with the "fragility," or failure mode and failure level, of structures, systems, and components. The seismic fragilities used in current licensing criteria, probabilistic risk assessments, and seismic margin evaluations rely heavily on subjective judgment, expert opinion, or military data. Thus, the research for resolving the uncertainties involves obtaining better estimates of the fragility of structures, components, and piping subjected to loads within and beyond their design level. In addition to providing failure data, structure research will identify how the parameters used in the design of equipment and components are affected by increased earthquake motion; component research seeks to test the hypothesis that electrical and mechanical components fail at higher levels than presently assumed and, as a consequence, that current licensing requirements are overly conservative; and piping research will provide the basis for improved balance between operating conditions in terms of overall safety.

Seismic probabilistic risk assessment methods have been developed to clarify safety issues for nuclear power plants since seismic events can simultaneously affect many plant systems and therefore be a significant or even dominant contributor to overall risk. However, these assessment methods have not been validated to eliminate uncertainties and as such cannot yet be used with the confidence desired to make sound regulatory decisions. Research will develop sufficient experimental data to allow the best possible validations of the complex probabilistic predictive methods. Research will also evaluate the adequacy of assumptions and subjective information used in seismic risk analyses. It also supports the seismic margins issue by providing experimental data to improve and reduce uncertainties in the current seismic design criteria.

1.4.3 Research Accomplishments in 1987

1.4.3.1 Earth Sciences

In early 1987, the NRC and the USGS entered into an agreement to establish the Eastern and Central United States portion of a National Seismographic Network.

Significant progress has been made in the design and specification preparation. A prototype station has been installed near Golden, Col., and is functioning quite well. It recorded the magnitude 6.1 Whittier, Calif., earthquake of October 1, 1987, at an epicentral distance of about 1350 kilometers.

Columbia University has completed its studies of surface structures in the epicentral area of the 1983 Goodnow earthquake in the Adirondack Mountains. No relationship between surface faults and the earthquake was found.

A study performed by the Pennsylvania State University focused on the Lancaster, Pa., and Moodus, Conn., seismic zones. The Lancaster area reveals a N-S trending structural and seismic zone that cuts across the strike of the major Appalachian structures. The zone is favorably oriented to be activated by the prevailing ENE compressional stress. The N-S trend is outlined by epicenters, aftershocks, and focal plane of the 1984 earthquake, and geologic and geomorphic trends, which include diabase dikes, faults, springs, drainage, and lineaments.

At Moodus, no specific seismogenic structure has been found, but there are some characteristics that may be related to the seismicity. Lineaments at Moodus are more numerous and shorter than in adjacent areas, possibly because the crust is more broken up and hence more easily activated by local stresses. Stream-water samples from the seismic zone have higher than normal pH, which is consistent with influx of water from the subsurface along fractures. A recent deep borehole has found a water-filled zone at 3,200 feet, below the Honey Hill fault. Above this zone, stresses are higher than below. This may explain the shallowness of the hypocenters, which seem to be concentrated in the more highly stressed surface layer.

A study of the Charleston seismicity has employed 2- and 3-dimensional stress models to clarify causes of the seismicity and to complement the sparse stress data in this region. The stress models take topography, density, and plate boundary stresses into account to derive the stress distribution over the area. There are two major structures that should influence local stresses, namely, the Appalachians and the continental shelf edge. The models show that these features indeed generate large stresses. However, when ridge-push forces are taken into account, the stress near the Blue Ridge of the Appalachians is enhanced whereas the shelf edge stress is largely cancelled. This corresponds to the observed seismicity, which is high in the Appalachians and low near the shelf edge. The Charleston area also emerges as a region of higher stress, while areas of minimal seismicity such as eastern North Carolina and southwestern Georgia are characterized by low stress. It appears, therefore, that stress computations can be a valuable tool for analyzing the seismic potential of certain areas.

Purdue University has analyzed seismicity, geologic and geophysical data, and borehole information of the midcontinent region between New Madrid, Mo., and Anna, Ohio. The improved data base that has been developed over the past 10 years has led to the conclusion that two hypotheses can explain most of the midcontinent seismicity. The dominant mechanism is reactivation of existing zones of crustal weakness that are favorably oriented with respect to the NE-SW direction of the maximum compressive stress in this region. Local basement inhomogeneities are a second mechanism that may explain seismic activity of low magnitude.

The initial NRC-funded investigations of the historically aseismic Meers Fault in Oklahoma have been completed. These investigations have shown, with several lines of evidence, that about 26 kilometers of the fault have undergone recent displacement, the latest of which probably occurred 1,100 to 1,200 years ago. Cumulative displacements of up to 5 meters of reverse offset and a much larger left lateral strike-slip offset were recorded.

1.4.3.2 Component Response to Earthquakes

The static testing of two large reinforced concrete models representing a portion of a nuclear power plant building (i.e., shear wall and floor segment) was performed in 1987. Based on a preliminary evaluation, it appears that the 1987 static test data contradict previous dynamic test observations. That is, an excellent comparison of analytically derived stiffness was obtained from the recent tests. The differences obtained from the static and dynamic tests are under investigation.

The majority of the experimental work in a cooperative EPRI/NRC piping and fitting dynamic reliability program was completed by the end of 1987. A milestone was reached with the completion of a series of design-level and high-level seismic input tests of a pressurized 6-inch carbon steel piping system. The piping system was well instrumented, and the recorded response data will provide valuable benchmarks for future evaluation of linear and nonlinear piping analysis methods. Of immediate interest is that for the first time a failure of a pressurized prototypical piping system was achieved under very high seismiclike loads. An input scaled roughly 25 times higher than normal safe shutdown earthquake (SSE) design limits produced hanger and valve operator failures and ratcheting in elbows, but not leakage. The input was then scaled even higher and excessive ratcheting at an elbow resulted in rupture.

Fragility data were developed for motor control centers, switchgears, panelboards, and dc power supplies. This information will be used in future seismic probabilistic risk assessments (PRAs) and margin studies to identify weaknesses and strengths in nuclear power plant seismic design and to assist in seismically related licensing decisionmaking.

1.4.3.3 Seismic Design Margin Methods

The successful completion of the Maine Yankee seismic margins review (NUREG/CR-4826) in March 1987 and the issuance of a safety evaluation report based on this review are major milestones in the seismic evaluation of nuclear power plants.

The Maine Yankee margins review followed the eight-step process outlined in the guidance of NUREG/CR-4482. The review involved the Maine Yankee Power Corporation, Yankee Atomic Electric, the NRC, Lawrence Livermore National Laboratory as project manager, and fragility and system analysis teams. The review demonstrated that the seismic design margins program methodology can be successfully implemented and be used to solve seismic licensing issues. Plant seismic vulnerabilities were found and upgraded as a result of the review. The importance of both peer review and utility cooperation was clearly shown.

2. PREVENTING DAMAGE TO REACTOR CORES

The program for preventing damage to reactor cores encompasses the operations of the reactor as a system, including control of power level, maintaining water in the reactor system, core cooling and heat removal, and maintaining proper coolant temperatures and pressures. It also includes consideration of operator actions as an integral part of the system. The research covers both normal and abnormal conditions, including accidents, such as a pipe break and loss-of-coolant accident, in which emergency systems are called upon to provide cooling water. A complete knowledge of the reactor operating as a system makes it possible to define the conditions of operation that prevent core damage and hence maintain safety. The emphasis of this research is on prevention of severe core damage through understanding of both plant and human behavior during accidents. This information is used to ensure that plant equipment, operational procedures, and training are adequate to deal with operating events and to prevent serious accidents. This program also includes accident management, which is a new undertaking and part of the implementation of the Commission's Severe Accident Policy Statement. Individual plant examinations (IPEs) of all existing plants to identify vulnerabilities for severe accidents and research to investigate strategies for managing severe accidents are included in this program.

2.1 Plant Performance

2.1.1 Statement of Problem

A wide range of reactor plant design variations exists in the United States, and the safety of these plants for a wide range of normal and abnormal operations must be ensured. The NRC is required to independently assess each licensee's assertions and performance of his responsibility to design, construct, and operate a reactor with respect to the safety of the plant 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 fluid flows would take too long to compute because of the complexity of heat exchange between reactor components, water, and steam, as well as because of the moving mechanical interfaces in pumps and the extensive baffle surfaces in the primary loops.

As a result, the NRC must combine, in a complex undertaking, limited experimental data, much of which is less than full scale, and limited calculational capability into a firm technical basis for evaluating design basis accidents, the safety implications of actual events in operating reactors, and hypothetical transient scenarios determined to be major contributors to risk as a result of probabilistic risk assessment studies and these operating events. Specific gaps in information available to or applied by NRC exist in the following areas:

1. Small-break loss-of-coolant accidents (LOCAs) in Babcock and Wilcox (B&W) plants.

2. Non-LOCA transients in B&W plants, including those for which full-power steam generator operation is a major factor, e.g., feedwater line breaks.

3. Evaluation of the transient, multidimensional response of the reactor vessel and core.

4. Assessment of the scalability, applicability, and uncertainty in code predictions of transient response of operating reactors.

5. Assessment of the NRC computer code predictions against a wide range of available experimental data.

2.1.2 Program Strategy

The NRC has a dual, complementary approach toward achieving a firm technical understanding of the thermal-hydraulic behavior of the reactor. The dual approach is analytical and experimental with feedback to the analytical models. The NRC starts by simulating the actual reactor's continuous flow of heat and fluids with a computerized model consisting of many small discrete cells exchanging heat, fluid, vapor, kinetic energy, and momentum at each small but finite time step. Physical laws are used when possible to calculate all these exchanges. Empirically derived formulas, obtained from experiments, are used as necessary to account for such complex effects as friction between vapor and liquid. The calculations are made for each time step and for each cell, in a manner familiar to animated computer games, except that reactor models have many more objects (cells) and these objects interact in a tightly coupled manner at every time step, requiring many more calculations. Hence, improved efficiency in the calculations is always an objective. There are two avenues for improvement. First, improved numerical methods, calculating across the cells and time steps, are sought and employed. Secondly, techniques are sought to better use the hardware possibilities of vector, parallel, and distributed computing.

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. Next are separate-effect experiments to test the code's predictions for a single, complex component such as a steam generator. Third are integral system tests that are used to evaluate the code predictions of a complete reactor. The results of these tests provide feedback to correct the code and our understanding of the transients. This feedback process is formalized in an International Code Assessment Program (ICAP) coordinating tests and code predictions in many countries as well as a code scaling, applicability, and uncertainty (CSAU) program.

2.1.3 Research Accomplishments in 1987

2.1.3.1 Multiloop Integral System Test (MIST) Program

The MIST facility is designed to simulate thermal-hydraulic aspects of transients in B&W reactors of the lowered-loop design. The research program conducted at this facility is jointly sponsored by the NRC, B&W, B&W Owners Group, and the Electric Power Research Institute.

In September 1987, all testing was completed in the third phase of a four-phase program. The experimental data obtained are being used to assess the capability of the NRC and industry thermal-hydraulic best-estimate codes to predict B&W transients and to provide the needed small-break LOCA data base.

2.1.3.2 Emergency Core Cooling System (ECCS) Rule Revision

It is now known that the methods specified in the ECCS evaluation models (Appendix K to 10 CFR Part 50), combined with other analysis methods currently in use, are highly conservative and that the actual cladding temperature that would occur during a LOCA would be much lower than those calculated using Appendix K methods. Therefore, on March 3, 1987, the NRC published proposed amendments to 10 CFR Part 50 and Appendix K. These amendments were proposed because, during the period of time since the issuance of § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," of 10 CFR Part 50 and the acceptable and required features and models specified in Appendix K to 10 CFR Part 50, considerable research has been performed that has greatly increased the understanding of ECCS performance during a LOCA. The revision to the ECCS rule would permit licensees to perform realistic evaluations of their ECCS based on the body of research currently available but accompanied by an evaluation of modeling uncertainties. The revised rule would also provide guidance with respect to the reporting of errors or changes appearing in evaluation models.

2.1.3.3 ROSA-IV

Based on the experiments run to date in the ROSA-IV facility in Japan, it appears that a 5 percent break produces the maximum core uncovery with a duration sufficient to cause some core heatup. The differential pressure in the upflow portion of the hot leg and steam generator will be small at normal reactor decay power levels so that the core uncovery will not be below the loop seal elevation. The magnitude of core heatup due to early core uncovery in all the experiments with realistic core decay power has been of such small magnitude so as not to pose any threat to core integrity. Thus, previous regulatory concerns on the magnitude and duration of potential unexpected core heatup during a small-break LOCA have been resolved.

Three ROSA-IV test results have been used in understanding and assessing the phenomena occurring during the depressurization process. Of concern is the need to depressurize the reactor vessel to avoid any potential for direct containment heating after a possible vessel meltthrough. These tests show that with existing power-operated relief valves (PORVs) and accumulator injection pressure setpoints the core could start to heat up before the pressure can be lowered below the accumulator setpoint in a station blackout transient without auxiliary feedwater. At the time of cladding temperature rise, the pressure is still above the accumulator setpoint. Since the ROSA-IV facility is one of the largest test facilities in the world and scaled well to simulate transients, one must be alert to the possibility of the occurrence of similar phenomena in reactor transients.

Tests show that higher accumulator injection pressure setpoints will help the quenching process since the accumulator water will be available for cooling of the core at the beginning of the core temperature rise. Tests also show that, as the number of PORVs is increased, the pressure at the onset of the core heatup should drop further. It should be possible to lower the pressure during the front end of the transient by adding PORVs.

2.1.3.4 Code Development and Assessment

The current versions of the computer codes used to simulate plant response to various transients and to assess procedures and operator training are TRAC-PF1/MOD1, RELAP5/MOD2, and TRAC-BF1. During 1987, code improvement was carried out to provide improved simulation capability in these codes and focused on defining the accuracy of the TRAC-PWR code for large-break LOCAs in connection with the forthcoming revision to Appendix K to 10 CFR Part 50 allowing realistic analysis of LOCAs.

A significant element of the NRC code improvement program is the International Code Assessment Program (ICAP), which was organized by the NRC to provide information for determining the scalability, applicability, and uncertainty of the transient codes. Information on code assessment also results from the 2D/3D, ROSA-IV, and MIST experimental programs. ICAP is a multiyear program that will continue through 1990. With the shutdown of all domestic experimental facilities except MIST, ICAP provides the NRC with a good part of its experimental data needs.

2.2 Human Performance

2.2.1 Statement of Problem

Probabilistic risk assessments (PRAs) conducted to date indicate that approximately 65 percent of all safety-related events at nuclear power plants involve human error. Methods are needed to identify, systematically set priorities for, and suggest solutions to human performance issues in the operation and maintenance of nuclear power plants during normal, transient, and emergency situations.

2.2.2 Program Strategy

The human reliability research program has two objectives: (1) to develop a technical support system (data and quantification tools) for doing human reliability analysis (HRA) segments of reliability evaluations, especially those employing PRA techniques and (2) to develop tools for systematically applying HRA and PRA quantitative and qualitative results to address outstanding generic issues and to identify immediate and long-term research needs of the NRC.

2.2.3 Research Accomplishments in 1987

Research for implementing the following systems has been completed: (1) a data base management system known as NUCLARR for processing, storing, and retrieving human error and hardware failure data to support reliability evaluations,

(2) a Cognitive Environment Simulation (CES) for analyzing the intention formation aspects of human behavior, and (3) a Cognitive Reliability Analysis Technique (CREATE) for applying CES outputs in reliability evaluations.

2.3 Reliability of Reactor Systems

2.3.1 Statement of Problem

Nuclear power plant engineered safety systems are designed with a high degree of reliability. This is so that they are available when needed to prevent reactor core damage. In order to maintain safe operations of nuclear power plants, licensees' programs for maintenance, training, and quality assurance are intended to minimize the frequency of transients severe enough to challenge safety systems and to ensure that safety systems function properly when challenged. Reliability methods are useful to achieve these objectives, to monitor plant safety performance, to help recognize declining trends, and to evaluate potential changes intended to preserve or improve safety performance. Potential multiple equipment failures also are a major concern in redundant safety systems and therefore identification, evaluation, and prevention of dependent (common cause) failures is important. External initiators (such as seismic events, internal flood, fire, tornado, and high winds) also challenge the reliability of the safety systems.

2.3.2 Program Strategy

Many areas contribute to the reliability of the critical safety systems and related components. Therefore, research is being conducted in these areas to ensure that the appropriate level of reliability is maintained. The first area focuses on the identification of the principal causes of equipment and safety system malfunctions. A second area to the strategy is to evaluate and document various programs used in industry and elsewhere to improve reliability (including test and maintenance requirements). The third area focuses on development of performance indicators that will allow the NRC to objectively monitor trends in licensee performance. Another phase of the strategy translates reliability methods into effective tools that make the regulatory decision process more efficient. These tools integrate dependent failure analysis, systems reliability, operational safety, and operator reliability into the regulatory process. This program will help NRC evaluate the licensee's ability to maintain acceptable levels of plant safety so that the NRC will have a sound basis to take rapid and effective enforcement action against the licensee whenever needed.

2.3.3 Research Accomplishments in 1987

2.3.3.1 Performance Indicators

Research has been undertaken to support NRC's development of plant performance indicators. In 1987, the Office for Analysis and Evaluation of Operational Data implemented an initial set of performance indicators for use by NRC. These performance indicators are logically related to safety in a qualitative way; but a reliability/risk-based method for quantitatively evaluating and integrating indicators was not yet available. RES is supporting this program by developing a method for reliability/risk evaluation of performance indicators. The results are intended to strengthen NRC's use of performance indicators (1) to improve Systematic Assessment of Licensee Performance (SALP) and (2) to identify trends of declining or improving performance between SALP reviews. During 1987, a research project conducted by the Brookhaven National Laboratory and SAIC developed an improved method for monitoring trends in the availability of important safety systems. The staff is evaluating this improved indicator as a possible addition to NRC's set of performance indicators.

2.3.3.2 Operational Safety Reliability

Operational safety reliability research is evaluating the effectiveness of a program that can help to maintain the reliability of important safety systems. The elements of a reliability program identified by this research are being adapted and used as part of the technical basis for the resolution of Generic Safety Issue B-56, "Diesel-Generator Reliability."

2.3.3.3 Plant Technical Specifications

Technical specifications are design and procedural limits that entail explicit restrictions on the operation of nuclear power plants and the maintenance of safety systems in a pre-accident-ready condition. In response to findings by an EDO task group on enhancing the safety impact of technical specifications (NUREG-1024), RES established a broad-based research program (Procedures for Evaluating Technical Specifications) in 1984 to examine approaches for developing methods for setting limiting conditions for operations (LCOs) and surveillance requirements based on reliability and risk analysis principles. During 1987 this program achieved progress in several areas.

For the first time, a rigorous basis was established to examine the influence of different parameters affecting diesel generator availability and the criteria in Regulatory Guide 1.108. NUREG/CR-4810 (dated May 1987) details this effort and discusses other related issues, such as the separation of demand and standby time-related failures, testing strategies (e.g., sequential, staggered, and adaptive), and the impact of maintenance activities on diesel test intervals in developing risk-effective surveillance test intervals. Revising Regulatory Guide 1.108 based on the study results will allow diesel generator accident unavailability to be more effectively monitored and controlled. Also investigated were risk control and regulatory considerations related to allowed cumulative outage times (ACOTs). If, based on ACOTs, LCOs were established for components, licensees could be permitted greater flexibility in being able to accommodate widely varying component downtimes. Another area of research focused on the importance of uncertainties of risk analyses related to modifications of technical specifications. This analysis provided an understanding of the range of uncertainty and insights for reviewing risk-based submittals.

2.3.3.4 IRRAS

The Integrated Reliability and Risk Analysis System (IRRAS) version 1.0 was completed. It is a fault tree construction and analysis tool that will run on a PC. Over 150 copies of IRRAS have been distributed in the United States. The recipients include staff at the NRC, national laboratories, contractors, vendors, utilities, other government agencies such as DOD and NASA, and universities. The reception of IRRAS has been very positive. National laboratories and their subcontractors have been using IRRAS on NRC risk problems. Nuclear power plant vendors are using IRRAS in their PRA work of standard plants. Utilities have shown interest in using IRRAS as a tool in the IPEs. IRRAS 2.0, being developed in 1988, will contain the capabilities of defining accident sequences and determining their minimal cutsets from the system cutsets.

2.3.3.5 Dependent Failure Analysis

During the last 2 years, EPRI and the NRC have been developing a common cause failure analysis procedures guide. The work on the main report is completed and documented in NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Reliability and Safety Studies." This work presents a framework for performing common cause failure analysis. This framework integrates qualitative insights from engineering assessments with historical evidence from recorded multiple failure events. This allows the analyst to approach common cause failure analysis in a practical, systematic, and clearly documented fashion.

2.4 Accident Management and Individual Plant Examinations

2.4.1 Statement of Problem

A severe accident in a nuclear power plant can be defined as an event in which the core is damaged and there is a potential for release of large amounts of radioactive fission products. In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded that existing plants pose no undue risk to the public health and safety and that there is no immediate need for generic rulemaking related to severe accidents. However, based on NRC and industry experience with plant-specific PRAs, the Commission is convinced of the need for a systematic examination of each existing plant to identify any plant-specific vulnerabilities to severe accidents. The policy statement indicated the intent of the Commission to take all reasonable steps to reduce the probability of a severe accident and, should a severe accident occur, to mitigate its consequences to the extent possible. As part of the implementation of the Commission's Severe Accident Policy Statement, the staff will require 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 (see Chapter 3). This work has shown that the causes and consequences of severe accidents can be greatly influenced by nuclear power plant operators and that many vulnerabilities to severe accidents can potentially be eliminated by proper operator actions. The TMI-2 accident and other abnormal occurrences in nuclear power plants have shown that operators do not stand idle but actively intervene in attempts to control the event. If operators are provided with proper guidance and training to take beneficial actions when needed and, most importantly, refrain from actions that can have adverse effects, the consequences of a severe accident can potentially be significantly reduced. Since many accident management strategies do not involve significant plant design changes, substantial safety benefits can be quickly achieved by ensuring proper operator actions. Thus, the initiation of accident management programs at operating plants is a logical result of the IPE process.

2.4.2 Program Strategy

RES has been given the responsibility for the implementation of the IPE. This implementation involves development of guidance for performance of the IPE, preparing a generic letter to plant operators requesting the IPE, and developing review plans and eventually reviewing the results of the IPE submittals in cooperation with NRR. The requirement to correct any identified plant-specific vulnerabilities not voluntarily corrected will be determined by the backfit rule. Accident management will not be required as part of the IPE process but will be highlighted in the IPE generic letter as a future requirement that will make use of the results of the IPE process. The consideration of severe accident vulnerabilities due to external hazards (earthquakes, flood, wind, etc.) will also be deferred until the staff considers how best to consider these hazards. Consideration of seismic hazards by the IPE process must be coordinated with a number of other seismic regulatory activities.

In support of the Severe Accident Policy Statement implementation and the IPEs, RES is initiating an accident management research program. This program has the objective of providing an independent evaluation of selected accident management strategies to provide a technical basis for staff review of licensee accident management programs. This program will also transfer the results of severe accident research to the industry in a form that can be practically applied and attempt to demonstrate the potential benefits of an accident management program.

2.4.3 Research Accomplishments in 1987

2.4.3.1 Individual Plant Examinations

During 1987 the accident management research program developed five sets of guidelines for use in IPEs, an integrated systematic approach (using either a method developed by the Industry Degraded Core Rulemaking (IDCOR) group or a more complete PRA method) to examine each nuclear power plant now operating and under construction for possible significant risk contributors (outliers) that might otherwise be overlooked. These guidelines apply to plant features and operator actions that were found to be important to either the prevention or mitigation of severe accidents. They incorporate the insights gained from industry-sponsored PRAs, NRC source term studies, and IDCOR reference plant analysis.

2.4.3.2 External Events

During 1987, existing PRAs were reviewed to determine the influence of various external events on plant risk. The objective of this review was to determine if external events are important contributors to plant risk so as to warrant consideration in the IPE process. This review confirmed that some external events (e.g., earthquake, fire) can be important to risk and should be considered by an IPE. This work will continue in 1988 to determine how best to include these external events in the IPE.

2.4.3.3 Accident Management

Accident management is a new research program in 1987 and its overall scope is still being planned. In 1987, preliminary investigations of containment venting as a strategy to prevent containment failure and vessel depressurization to preclude direct containment heating were completed. These investigations provided insights on these potentially important strategies for use in the IPE guidance but also highlighted the need for a more complete investigation to fully understand the implications of these strategies. This research is continuing in 1988.

3. REACTOR CONTAINMENT PERFORMANCE AND PUBLIC PROTECTION FROM RADIATION

To ensure that existing regulations relating to severe accidents (i.e., siting, general criteria, emergency planning) adequately protect the public, research is needed to confirm the technical bases upon which the existing regulations are founded. These bases include the behavior of fission products released from melting fuel, the temperatures and pressures produced during a core melt event, and the capabilities of containment buildings to retain radioactive materials during such events. The behavior of radioactive materials released to the environment (movement in air and water, uptake by plants and animals) is also an important consideration in protecting the public from radiation. Based on such information, the Commission will be better able to confirm the adequacy of its requirements for the siting, design, construction, operation, and reliability of those safety systems installed to mitigate the effects of severe accidents and to identify where improvements in the regulations are necessary.

3.1 Reactor Containment Performance

3.1.1 Statement of Problem

The potential for new or revised regulatory requirements and policy in reactor safety arises primarily from two sources, operational experience and research. In the early days of reactor licensing, very little information was available from either of these sources. This gave rise to a deliberate approach to regulation that stressed defense in depth and multiple layers of conservatism in the setting of design criteria to ensure that nuclear power plants could be operated safely and without undue risk to the public.

The physical possibility of serious accidents resulting in the release of fission products into the environment was recognized from the outset and embodied in public protection principles, both individual and societal, in the reactor siting criteria of 10 CFR Part 100 published in 1962. With the subsequent publication of the General Design Criteria, Appendix A to 10 CFR Part 50, and the codification of requirements and criteria for emergency core cooling systems (ECCS), an adequate regulatory basis was judged to exist to implement the defense-in-depth policy. Concurrently, and following the passage of the National Environmental Policy Act (NEPA), the Commission issued a proposed rule change dealing with the environmental impacts of reactor accidents. This defined a set of classes of accidents of increasing magnitude of impact, designated 1 through 9, in which all but the last were to be treated in environmental impact statements in accordance with NEPA. Class 9 accidents, however, were not to be considered in reactor design on the grounds of remote likelihood of occurrence. This system of classification was structured to be consistent with the safety philosophy and practice of 10 CFR Parts 50 and 100 and as discussed in Appendix I to Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations."

It is noteworthy that the staff concluded soon after Three Mile Island-2 (TMI-2) that the accident was a Class 9 event. Thus, the TMI-2 accident, with its offsite radiological consequences well within the licensing basis, demonstrated that recovery from a severe core damage accident is possible. Nevertheless, it gave rise anew to questions concerning the potential course of still more severe accidents that would represent a substantially greater challenge to the function of reactor containment systems.

The Reactor Safety Study (WASH-1400) dealt with such events but had very little in the way of experimental data with which to consider the complex physical and chemical processes involved in the structural degradation of a reactor core, the progress of melting, the attack on the reactor pressure vessel, subsequent meltthrough and ejection of molten core material into the containment, and the impacts of these phenomena on containment system performance and the release of fission products to the environment. The investigation of these phenomena has been intensively pursued through the severe accident source term research program since 1981. A major milestone in this program was reached in July 1986 with the publication of NUREG-0956, "Reassessment of the Technical Bases for Estimating Source Terms."

The aim of and justification for this research program has been to develop a scientifically sound basis for making decisions on severe accident matters that have arisen in the regulatory process. The need for such decisions flows primarily from the Commission's Severe Accident and Safety Goal Policy Statements and involves qualitative judgments and quantitative assessments of risk significance. Toward this end, the research findings and development of analytical methods described in NUREG-0956 were essential to the preparation of the "Reactor Risk Reference Document," draft NUREG-1150, in early 1987. Both of these documents reflect not only very considerable improvement in our state of knowledge of the potential outcomes of severe accidents, over and above that which existed at the time of publication of WASH-1400, but also the fact that there remain considerable uncertainties in our ability to predict these outcomes.

3.1.2 Program Strategy

The resolution of two key public health and safety issues provide the main focus of the current research program; namely, containment performance and accident management.

The containment performance issue is related to the capability of existing and future containments to mitigate the consequences of accidents. The bases for the issue are recent assessments of the ability of various containments (particularly Mark I, ice condenser, Mark II, Mark III, and small and large, dry containments) to successfully mitigate the consequences of severe accidents. For example, draft NUREG-1150 indicates that the failure likelihood of a Mark I containment of the Peach Bottom design in the event of a core melt accident could be as high as about 90 percent (a 10 percent chance of mitigating such accidents). The severe accident research program will aid in the resolution of issues relating to the acceptability of existing and future containment designs beginning in 1988 and ending in 1992 and in the confirmation of regulatory decisions thereafter.

The accident management issue relates to the improvement of existing accident procedures and operator training. This is to ensure that operators are able to reasonably minimize the likelihood and consequences of core degradation accidents. This conclusion is based on recent risk studies (e.g., draft NUREG-1150) and reviews of licensee procedures and training. Should such core damaging accidents be initiated, assurance is needed that proper actions are taken to minimize the likelihood of pressure vessel failure and containment bypass or failure. Accelerated and reprogrammed research on melt progression and the primary system, pressure vessel, and containment responses to such accidents is vital to ensuring adequate operator procedures and training.

There are a number of other related issues whose resolution would benefit from this research:

- Individual Plant Examinations - Resolution of issues related to the plant-specific searches for risk outliers as part of the implementation of the Commission's Severe Accident Policy Statement.

- Emergency Planning - Issues relating to the guidance for reducing early and latent health effects risks arising from accidental releases.

- Future Plants - Resolutions of issues and development of criteria and standards for future advanced light water and other advanced reactor designs.

- Adequacy of Existing Designs - The design of certain plant features (e.g., filter and heating, ventilation, and air-conditioning systems for operator and public protection; sprays; suppression pools; depressurization systems) under severe accident conditions has generally been recognized as providing some margin against severe accident conditions. The level of margin, however, and whether additional regulatory requirements would be cost beneficial needs further consideration.

- Safety Issue Resolution and Prioritization - Resolution of generic issues, unresolved safety issues, and backfit evaluations in large measure depend on evaluations of severe accident risks. The research is expected to help accelerate resolution by reducing the very large uncertainties in the present understanding of containment performance and accident management.

Research is needed in the following general areas to address the above issues:

- Core Melt Progression

    1. Support for continued examination of the TMI-2 core.

    2. Experiments to improve our understanding of melting cores, i.e., the temperature, pressure, and composition of core melt debris at and during vessel failure. This understanding, in turn, is vital to considerations of subsequent containment failure by overpressure from direct containment heating, core-concrete interactions, combustible gas ignition, ex-vessel steam explosions, and containment bypass.

- Primary System Response

    1. Experiments and calculations to better understand postaccident natural circulation and to assess why TMI-2 showed little or no heating of the upper internals that is usually associated with natural circulation effects.

    2. Improved understanding of vessel failure modes and analysis of strategies proposed to prevent or mitigate such failures.

- Containment Loading

    1. Experiments and analyses of the likelihood of, and loading phenomena resulting from, direct containment heating for the different existing containment designs.

    2. Experiments and analyses of pressure and temperature loading by core-concrete interactions and releases of fission products and combustible gases and how they may be mitigated by operator actions.

    3. Experiments and analyses of fission product chemical forms and their changes and interactions within the containment.

- Containment Response

    1. Analyses and experiments of core debris attack on containment liners and how such effects may be mitigated by operator actions.

    2. Analyses of containment bypass, including the attenuation of fission products, and how such situations may be mitigated.

    3. Analyses of containment structural and penetration responses to early and late temperature and pressure loadings.

- Probabilistic Risk Assessment (PRA)

    1. Development of improved models of severe accident phenomena in risk codes.

    2. Development of improved methods of analyzing and displaying uncertainties.

3.1.3 Research Accomplishments in 1987

3.1.3.1 Fission Product Behavior

The NRC is participating in an internationally sponsored project called LWR Aerosol Containment Experiments (LACE) to study the aerosol behavior of fission products within a containment and immediately after leaving a containment. The six experiments (now all completed) were performed to investigate inherent aerosol retention behavior in the containment or auxiliary buildings for postulated high-consequence accident conditions. These experiments will also provide a data base for validating containment aerosol and related thermal-hydraulic computer codes.

In 1987, as part of the LACE vessel blowdown experiments (LA-5 and LA-6), measurements were made of the localized deposition of particulates in the nearby ex-vessel area to determine if aerosols were plated out on the ground. A heavy condensed vapor atmospheric plume was observed under low wind speed and high humidity atmospheric conditions in LA-5. In LA-6, winds were much stronger, and only a thin vapor cloud was observed. Tentative results indicate that as much as 20 percent of the discharged aerosol was deposited locally on the ground by the vapor cloud, with the remainder apparently transported aloft as the hot, wet plume evaporated. This indicates that localized deposition could be important in contaminating the area near a site during a severe accident, but current offsite accident consequence models, which assume dry aerosol plumes, are not seriously in error in ignoring localized deposition.

3.1.3.2 Natural Circulation

Natural circulation in severe accidents is the buoyancy-driven steam circulation between the reactor core and upper-plenum region of a vessel (in-vessel circulation) with or without countercurrent flows in the hot legs and steam generators (ex-vessel circulation). This kind of multidimensional flow may exist during the core uncovery and core melt period of certain high-pressure severe accidents in a PWR. The flow serves as a means to transfer the decay heat from the core to the upper-plenum structures, hot leg piping, and steam generator tubes. As a result, the reactor coolant system pressure boundaries may be heated to high temperatures to challenge structural integrity.

Based on the EPRI-sponsored experiments at a 1/7-scale Westinghouse test facility, it has been concluded that multidimensional natural circulation indeed exists under certain simulated accident conditions. The COMMIX code (valid for intact-core geometry and single-phase flow) was used to compare with the Westinghouse data and showed good agreement. To assess whether the natural circulation would also exist in a full-sized reactor, COMMIX calculations for intact-core geometry were performed, and the results indicate that the natural circulation flow would also exist in a PWR. Since severe accidents involve core damage and core melt that is beyond the scope of COMMIX, MELPROG/TRAC calculations were performed for analyzing the in-vessel circulation in the Surry plant during a station blackout accident with the loss of auxiliary feedwater (the TMLB' accident). A comparative SCDAP/RELAP5 calculation was also performed for Surry using the countercurrent flow information calculated with COMMIX. These preliminary calculations suggest that either the surge line or the hot leg connection at the vessel may fail by creep rupture at high temperature and pressure before the vessel lower head failure. As a result, high-pressure melt ejection may not occur during the TMLB' accident in a Westinghouse PWR.

3.1.3.3 Structural Tests

A 1/6-scale model of a reinforced concrete containment was tested to failure in July 1987. The containment was designed and built in conformance with the ASME Boiler and Pressure Vessel Code, just as are actual containments. The model was 22 feet in diameter and 37 feet in height and included representative features such as four major penetrations (two airlocks and two equipment hatches) and several smaller penetrations that passed both separately and in clusters through the containment wall.

The containment had a design pressure of 46 psig. Pressure was increased in steps until failure occurred at 14[illegible] psig. At that point a major tear, 20 inches long, developed in the liner. Leakage through that tear overwhelmed the ability of the pressurization system. Additional minor tears were also present in the liner but the concrete outer structure, although visibly cracked, did not show great distress. Posttest analyses will focus on the measurements of strain and displacement taken at each discrete pressure step to evaluate the accuracy of pretest predictions made using different analytical techniques.

A full-sized personnel airlock, which was obtained from a cancelled nuclear power plant, was tested as part of NRC-sponsored containment integrity research. The objective of the tests was to obtain structural data on the behavior of an airlock, especially the sealing surfaces, under severe accident conditions. It was anticipated that leakage would not occur unless relative deformations between the sealing surfaces were developed and performance of the seal material was compromised. The sealing surfaces could separate because of a mismatch in the out-of-plane displacements of the door and bulkhead, which resist internal pressure through bending action. The performance of the seal material may be compromised in two ways: (1) a loss of resiliency associated with thermal or radiation aging and (2) degradation associated with exposure to very high temperatures.

Two of the four tests to be performed were conducted on June 30 and July 2, 1987, with satisfactory results. In the first test, the inner and outer doors were without gaskets and pressurized to 69 psig (1.15 times the design) at room temperature. Leakage was measured at 45 scfm and 35 scfm on the inner and outer doors, respectively. In the second test, the inner and outer doors had aged gaskets installed and were pressurized to 69 psig (this pressure was held for 1 hour), and no leakage was measured.

3.1.3.4 Core Melt Progression and Hydrogen Generation

In-vessel core melt progression is concerned with the state of the reactor core from core uncovery up to reactor vessel meltthrough in unrecovered accidents and up to the stabilization of core temperatures in accidents that are recovered by core reflooding, as at TMI-2. Sensitivity studies have shown that the uncertainties in the state of the core debris at vessel failure produce the greatest uncertainties in the ex-vessel containment loads, including core-concrete interactions and direct containment heating. The state of the core during core melt progression is also the primary determinant of in-vessel hydrogen generation, fission product and aerosol generation and attenuation, explosive and non-explosive rapid steam generation, and the potential for successful recovery actions in accident management.

Tests in the National Research Universal (NRU) reactor in Canada provided full-length data on fuel damage during coolant boildown. FLHT-4 provided information on fission product release and deposition for PWR high-burnup fuel rods. FLHT-5 was also conducted with power compensation for the bundle heat losses at high temperature.

The BWR-DF4 test was performed in the Annular Core Research Reactor (ACRR) to investigate the effects of the BWR channel boxes and the B_4C control blades upon fuel damage, early core melt progression, hydrogen generation, and system chemistry.

3.1.3.5 Core-Concrete Interactions

In those accident scenarios in which the reactor vessel fails, high-temperature core debris may fall into the reactor cavity where it interacts with structural concrete. The consequences of these thermal and chemical core-concrete interactions may significantly impact containment loading, the modes of containment failure, and the radiological source terms. To characterize the threat to containment integrity and the nature of the ex-vessel releases, experiments are being performed, and mathematical models are being developed and assessed.

The CORCON code was developed as a best-estimate computational tool to calculate the physical and thermodynamic variables needed to characterize the progression of high-temperature core debris as it erodes concrete in the reactor cavity. CORCON is incorporated into the NRC Source Term Code Package (available for licensing and regulatory applications) and has now been integrated into the CONTAIN and MELCOR codes. Improved models for the treatment of decay heat, time-dependent mass addition, and axial heat transfer to concrete have been developed. The code is being actively used in 17 research institutions throughout the world. Large-scale integral experiments with sustained induction heating were performed to study the effects of metallic zirconium present in molten stainless steel interacting with limestone and siliceous concrete. A summary review of available data on core debris-concrete interactions is being prepared in support of model validation.

3.1.3.6 Direct Containment Heating

In certain reactor accidents, degradation of the reactor core can take place while the reactor coolant system remains pressurized. Left unmitigated, core melt will slump and collect at the bottom of the reactor vessel. If molten core material attacks the bottom head of the reactor and a breach occurs, the core melt will be ejected under pressure. If the material should be ejected from the reactor cavity into surrounding containment volumes as small particles, thermal energy would be quickly transferred to the containment atmosphere. The metallic components of the sprayed core debris can further oxidize in air or in steam to generate a large quantity of chemical energy and further pressurize the containment. Simple analyses indicate that even a large, dry PWR containment can be pressurized beyond its ultimate strength if a significant fraction of the core material participates in direct containment heating (DCH).

During 1987, work on two experimental programs was continued: the Surtsey direct containment heating test program at Sandia and the separate-effect test program at Brookhaven National Laboratory. Preliminary observations of the Surtsey facility test results have been used to benchmark DCH models. Because of the complexity of the DCH problem coupled with the high cost of running large-scale tests in the Surtsey facility, separate-effect tests are being performed at Brookhaven to address core debris dispersal. 1/42-scale transparent plexiglass models of Zion, Surry, and Watts Bar reactor cavities were constructed.

3.1.3.7 Hydrogen Combustion

The hydrogen combustion program assesses both the consequences and methods used to control or mitigate deflagrations, diffusion flames, accelerated flames, and transition from deflagration to detonations (DDT) and detonations that might be caused by hydrogen burns in LWR plants. The HECTR computer code was developed at Sandia and is used in the analysis of nuclear reactor accidents involving the transport and combustion of hydrogen. The assessment of HECTR is continuing, and it includes the use of the data from large-scale hydrogen transport experiments performed at the HDR facility in the Federal Republic of Germany. Further, the three-dimensional finite-difference code, HMS-BURN, has been used to help design a hydrogen mixing experiment in the HDR facility as part of a cooperative program.

A theoretical model, based on chemical-kinetic equations, has been developed to assess the high-temperature behavior of hydrogen-air and hydrogen-air-steam mixtures. This model predicts behavior under accident conditions that might be found in DCH or core-concrete interactions that are different from what is observed (and predicted) for room temperature conditions. Because of facility limitations, these high-temperature predictions cannot be experimentally verified above 100°C.

3.1.3.8 Risk-Based Accident Methodology

The System Analysis and Risk Assessment (SARA) program provides a capability for computing and analyzing information on nuclear power plant risk characteristics for different levels of users requiring risk and reliability information for decisionmaking and regulatory analysis. In 1987, SARA was updated to accommodate the NUREG-1150 plant PRA models (Surry, Zion, Sequoyah, Peach Bottom, and Grand Gulf). The calculational capabilities were expanded to include the grouping of sequences; frequency calculations and importance measures can now be performed for a single sequence, group of sequences (including plant damage states), a single plant, or a group of plants. The graphics display capabilities were changed to support the NUREG-1150 plant models. The user interface was modified to be consistent throughout the various modules and to be much easier to use. The user manual was updated, and a data preparation and input reference manual was prepared as well as a SARA data management plan.

The Risk Methods Integrations and Evaluation Program (RMIEP) provides better assessment methods to support nuclear power plant PRAs. In 1987, the internal events analysis and the locations analysis for fire and internal floods were completed, thereby completing the technical work for the project.

3.1.3.9 Draft NUREG-1150

In February 1987, the NRC issued the draft version of NUREG-1150, "Reactor Risk Reference Document," as well as a series of supporting contractor reports, for public comment. This report assesses the risks from possible severe core damage accidents in five United States nuclear power plants. The five plants studied are Surry (Va); Zion (Ill.); Sequoyah (Tenn.); Peach Bottom (Pa); and Grand Gulf (Miss.). In addition, draft NUREG-1150 discusses the implications of the five risk assessments on regulatory issues such as the technical basis for present emergency planning regulations and implementation of the Commission's Safety Goal and Severe Accident Policy Statements.

3.1.3.10 Radioactive Release Consequence Model

In coordination with the NRC staff work on draft NUREG-1150, a new model for assessing the consequences of radioactive releases has been developed and used. This model--MACCS--has the capability to deal with radionuclide releases lasting for a short time or a prolonged period, including the effect of change in the wind direction at the reactor during the release, and to sample the variability of precipitation intensity from the reactor site's meteorological data.

It incorporates newer or more realistic (relative to WASH-1400) models for:

- Health effects projections developed for NRC after publication of WASH-1400 (1975) and BEIR-III (1980);
- Long-term (chronic) radiation exposure from continued use of contaminated environment;
- Emergency response and radiation protection measures; and
- Economic impact estimates.

3.1.3.11 Emergency Preparedness

On March 6, 1987, a proposed rule dealing with nonparticipation of State or local governments in emergency planning for reactor accidents was issued. An unprecedented number of public comment letters were received (approximately 38,000). All comment letters were evaluated by the staff and used in the development of a staff-proposed final rulemaking that was presented to the Commission on October 22, 1987. The rule was subsequently adopted.

On April 20, 1987, the NRC issued a proposed rule on emergency preparedness for fuel cycle and other radioactive material licensees. The rule would apply to about 30 large facilities. The facilities that will be required to comply with this regulation are those for which a release large enough to require the support of offsite response organizations to protect the public was considered credible. The rule would require, among other things, prompt notification of offsite response organizations in case of a serious accident, procedures and equipment for coping with the emergency, and training and exercises for response personnel. The rule is expected to be considered for final action in 1988.

3.2 Radiation Protection and Health Effects

3.2.1 Statement of Problem

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

3.2.2 Program Strategy

The Commission's regulatory process requires that changes to rules and guidance be systematically screened to ensure that there is a substantial increase in public protection and that based on analysis the costs are justified. Realistic values of the dollar-per-person-rem criterion are needed for analysis to justify changes, but technology gaps in knowledge associated with radiation health effects cause uncertainties in these analyses. The strategies of this program are to (1) identify and compensate for uncertainties in radiation risk coefficients used for health effect estimates in PRAs and regulatory decisions; (2) determine whether coefficient values used to establish dollars per person-rem for backfitting decisions are appropriate; and (3) evaluate and identify other areas where more realistic standards will reduce unnecessary costs.

The Commission approved the whole body dosimetry accreditation rule. At the same time, 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. As a result of this new guidance, NRC regulations and regulatory guides will have to be revised. The strategies of this program are to (1) modify radiation protection regulations, 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 (REIRS) program.

3.2.3 Research Accomplishments in 1987

3.2.3.1 ALARA Center

The BNL ALARA Center, funded by NRC, continued its work on surveillance of DOE and industry dose reduction and ALARA research. In 1987 BNL published Volume 3 of NUREG/CR-3469, which abstracts 252 national and international publications discussing dose reduction in areas such as plant chemistry, stress corrosion cracking, steam generator repair and replacement, robotics, and decontamination.

BNL reported in 1987 that dose reduction research has resulted in a reduction in occupational radiation exposure. A clear reduction in exposures is observable in countries with dose reduction research programs such as Japan, FRG, Canada, Sweden, France, and the United States.

3.2.3.2 Robotics

The Small Business Innovative Research contract for development of a surveillance robot was completed. NUREG/CR-4815, published in March 1987, analyzes the costs and benefits of the demonstration testing of a surveillance robot at the Browns Ferry (Ala.) nuclear plant. The NUREG/CR is the final report on the contract to design, construct, and demonstrate a remote-controlled vehicle for inspection and surveillance in radiologically controlled areas of nuclear power plants.

The robotic system was operated for 5 months at the Browns Ferry nuclear power plant by trained personnel in real job situations. Data collected by the evaluators indicate that the device can save over 100 person-rems per year at a nuclear power plant and permit more frequent and better inspections of safety-related components. Other benefits include labor and protective clothing cost savings, more complete surveillance data on components during power operation, improved worker safety, and decreased liability of plant operators for worker injury claims. The cost-benefit analysis concluded that the initial cost of approximately $160,000 would be recovered in 2 years of operation, and the system has an estimated useful life of 10 years.

3.2.3.3 Health Physics Technology Improvement

A report (NUREG/CR-4884) that provides a practical and consistent method for estimating intakes from both in vivo and in vitro bioassay measurements was published. To date, interpretations of bioassay measurements have been inconsistent, particularly in the early phases after an accidental intake. This report, in manual form, is aimed at completely describing a consistent approach and instructions for the computation of intakes and committed organ dose equivalents. Tables for the interpretation of bioassay results are compiled for several hundred radionuclides. The values in the tables were determined by using retention models published by the International Commission on Radiological Protection (ICRP-79).

A report (NUREG/CR-4915) of a 1-year study to examine the severity and duration of renal injury produced in the rat from exposure to low levels of uranyl fluoride was published in September 1987. Injury was apparent early in the dosing phase of the study at a time when renal uranium levels were between 0.7-1.4 microgram uranium per gram kidney and was most severe when the renal uranium burden was between 3.4-5.6 micrograms uranium per gram kidney. These levels are much lower than the nephrotoxic limit of 3 micrograms uranium per gram kidney used by the NRC in setting standards for exposure to soluble uranium compounds. Repair of the injury was rapid, with complete restoration within 35 days after the exposure.

The final report (NUREG/CR-4986) for a multiyear study of the metabolism of inhaled mixed (U, Pu) oxides was published in September 1987. Industrially collected aerosol materials were re-aerosolized in the laboratory to determine patterns of deposition, retention, and translocation in laboratory animals. Multiple species were used for inhalation exposures. A biokinetic model that used the measured physical/chemical characteristics of the particulates to describe the rate of dissolution of material deposited in the lung was developed. Results of the studies showed that, for a given elemental (U or Pu) component of the particulates, slight differences in the retention, distribution, and excretion of that element (U or Pu) were accounted for by slight differences in the physical/chemical characteristics of the aerosol. A dose/response study in rats exposed to (U, Pu) O_2 or PuO_2 resulted in development of pulmonary cancers, with no discernible difference due to the composition of the aerosol. The incidence of pulmonary cancers in rats exposed to these industrial materials containing plutonium were not different from rats exposed to laboratory-produced aerosols of plutonium oxide.

The NRC recently published NUREG/CR-4418, which describes the dose calculation for contamination of the skin using the computer code VARSKIN. The calculation method allows computation of the radiation dose rate at any desired depth beneath the skin from surface contamination and can be performed on an IBM PC or a compatible machine. The methods described in this report are considered acceptable for calculating skin dose from small, individual particles as well as from distributed sources. This work contributes significantly to resolving the current issue regarding appropriate management of hot particle skin contamination events at nuclear power plants. Work is continuing to prepare an addendum to VARSKIN that will address additional radionuclides and source/skin geometries.

3.2.3.4 Personnel Dosimetry Processing

In 1987 a final rule to improve personnel dosimetry processing has been completed. This rule requires licensees to use personnel dosimetry processors who have been accredited under the National Voluntary Laboratory Accreditation Program, which is operated by the National Bureau of Standards. The rule became effective on February 12, 1988, and is expected to improve the quality of dosimetry processing by requiring all processors to meet the guidelines of a national standard.

4. CONFIRMING SAFETY OF NUCLEAR WASTE DISPOSAL

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

4.1 High-Level Waste

4.1.1 Statement of Problem

The waste management issue involves both high-level waste and low-level waste. The high-level waste (HLW) disposal policy for the United States is defined by the Atomic Energy Act, the Energy Reorganization Act, the Nuclear Waste Policy Act, and the Nuclear Waste Policy Amendment Act (NWPAA). The latter, 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 and assigns responsibility for repository development to the DOE. 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 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, 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. NRC must have an independent capability to evaluate DOE safety analyses and confirm whether long-term releases predicted by DOE will be within established limits. The NRC research program objective is to provide the technical capability necessary to evaluate DOE's site characterization activities as required by the NWPAA and to assess DOE's license application when it is submitted.

4.1.2 Program Strategy

The research program has been guided by the need to provide the technical foundation for NRC development of a set of regulations 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 the management of high-level waste. The research combines theoretical study with laboratory and field experiments to identify and quantify the physical processes that determine repository performance and quantify the uncertainties associated with characterization and measurement of these processes. The ultimate goal of the NRC's waste management research is to provide the technical basis to support the licensing staff's independent judgment as to the appropriateness and adequacy of DOE's demonstration of compliance with 10 CFR Part 60 and the EPA's HLW standard. In addition, NRC's waste management research seeks to provide technical support to the licensing staff in their interactions with DOE, the State of Nevada, and other participants and interested parties and to develop regulatory standards to support the licensing of the disposal and management of high-level radioactive wastes.

4.1.3 Research Accomplishments in 1987

4.1.3.1 Geochemistry

Work on the use of ground-water dating techniques to help understand and model geohydraulic systems was completed. The results indicated that a combination of isotopic and geochemical techniques have the potential to provide an independent data base for ground-water flow model validation.

4.1.3.2 Waste Package Performance

Investigating the performance that can be expected from the waste form and waste package is essential if the NRC is to be able to independently evaluate DOE's demonstration that the waste form and waste package comply with the containment and controlled release requirements of 10 CFR Part 60. During 1987, NRC-sponsored research into the mechanisms of local corrosion of carbon steel was completed. As a result of this work, a significant new understanding of localized corrosion in carbon steel, of particular importance to geologic disposal of HLW, was realized.

4.1.3.3 Rulemaking

In February 1987, the NRC published an advanced notice of proposed rulemaking (ANPR) on the definition of high-level radioactive wastes. The purpose of the ANPR was to solicit comments on the classification and management of waste above class "C," which is not now considered HLW.

4.2 Low-level Waste

4.2.1 Statement of Problem

Disposal of low-level waste (LLW) involves issues at the forefront of technology, e.g., waste form and waste package integrity, and long-term retention of radionuclides in the disposal facility environment. Research is required to establish regulatory criteria 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. Establishing these criteria 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) set a very tight time schedule for establishing facilities within the various 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.

The direction of the LLW research program has responded to legislative mandates that resulted from an earlier 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., were used. 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 quickly complete the development of sound technical bases for regulatory decisionmaking regarding engineered LLW disposal methods, so-called alternative LLW disposal. This change has broadened the scope of NRC LLW research.

4.2.2 Program Strategy

At present NRC research in support of licensing activities for LLW disposal facilities is focused on addressing (1) water entry into disposal units, (2) performance of waste forms and waste packages, (3) characterization of the LLW source term, (4) mechanisms for transport of radionuclides from the disposal units, and (5) the safety and performance of engineered enhancements and alternatives to conventional shallow land burial for LLW disposal. This research is intended to support not only the NRC licensing but also those States that regulate LLW disposal under the Agreement State programs. This diverse user community makes the coordination and definition of LLW research and the dissemination of products a much more complicated undertaking than that for the HLW program. Further, the fact that many States will be the licensors and are looking to the NRC for technical support in their licensing and regulatory programs drives the LLW research to be more prescriptive and developmental than is the HLW research program.

4.2.3 Research Accomplishments in 1987

An NRC-sponsored cooperative project between Atomic Energy of Canada Ltd. (AECL) and the Battelle-Pacific Northwest Laboratories (PNL) used data collected from 40 years of LLW waste disposal at AECL's Chalk River facility to assess techniques for modeling LLW site performance. PNL approached the problem as though dealing with a pristine site, prior to waste disposal. This exercise confirmed the practicality and utility of modeling a site using a well-chosen data set collected during site characterization. This project is providing important insights into the design of data evaluation programs for future LLW disposal and the reliability of predictions based on site characterization data.

RES funded a project at the Idaho National Engineering Laboratory (INEL) to perform a safety assessment of engineering alternatives or enhancements to shallow land burial of LLW. The safety assessment is comprised primarily of a failure analysis and a reliability analysis. The failure analysis identified the cover as one of the most important design components for each of the alternatives considered. Also, components that improved the performance of the cover (i.e., fill) were found to significantly enhance system performance. This project has provided support to the selection of preferred alternative designs for LLW disposal by the licensing office.

5. RESOLVING SAFETY ISSUES AND DEVELOPING REGULATIONS

The program to resolve safety issues and develop regulations is directed toward developing the technical basis and related regulatory requirements needed to protect the health and safety of the public in the generation of electricity, and the manufacture, processing, transporting, and storing of nuclear fuel. This program also supports efforts to ensure that proposed Commission regulations are adequate and that their development is carried out in an efficient and timely manner.

5.1 Generic and Unresolved Safety Issues

5.1.1 Statement of Problem

The Commission directed the NRC staff to prepare a priority list of all generic safety issues, including TMI-related issues, based on the potential safety significance and cost of implementation of each issue. In December 1983, the listing was approved by the Commission. This guidance is reflected in the NRC Policy and Planning Guidance, the NRC Strategic Plan, and the RES Five-Year Plan.

5.1.2 Program Strategy

A generic safety issue is an issue 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-cost-effective and duplicate issues from further consideration, and for keeping the Commission and the public informed of the resolution of these issues. Currently, a backlog of approximately 50 proposed generic issues is awaiting prioritization. Strategies for this program are to provide timely prioritization of proposed new generic issues, eliminate the backlog of proposed issues (as resources permit), and issue periodic updates on the status and progress toward resolution of generic safety issues.

5.1.3 Research Accomplishments in 1987

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

During 1987, the NRC identified six new GSIs, established priorities for 19 issues, and resolved eight GSIs (see Table 5.1) and two USIs (described below).

Table 5.1 Generic Safety Issues Resolved in 1987

Issue Number	Title	Resolution Product	Date Resolved
91	Main Crankshaft Failures in Transamerica Delaval Diesel Generators	NUREG-1216	09/87
I.A.2.6(1)	Long-Term Upgrading of Training and Qualifications - Revise Reg. Guide 1.8	Reg. Guide 1.8	05/87
I.A.3.3	Requirement for Operator Fitness	No Req.	01/87
I.A.4.2(1)	Research on Training Simulators	Reg. Guide 1.149, Rev. 1	05/87
I.B.1.1	Organization and Management Long-Term Improvements	-	-
I.B.1.1(1)	Prepare Draft Criteria	No Req.	01/87
I.B.1.1(2)	Prepare Commission Paper	No Req.	01/87
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	No Req.	01/87
I.B.1.1(4)	Review Responses to Determine Acceptability	No Req.	01/87

In addition, 13 GSIs scheduled for resolution were integrated into the action plans for resolution as part of other GSIs.

5.1.3.1 USI A-46--Seismic Qualification of Equipment in Operating Plants

The design criteria and methods employed for the seismic qualification of mechanical and electrical equipment in nuclear power plants have changed significantly during the history of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and to perform intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition following a seismic event. USI A-46 entails investigation of alternative procedures for ensuring seismic adequacy of equipment in lieu of requiring qualification to current licensing requirements. The staff evaluated the various methods available for verifying seismic adequacy of equipment in operating nuclear power plants and decided that the use of seismic experience data and of test experience data would prove the most viable and cost-effective way of doing so. Based on its investigation of the issue, the staff concluded that there are three principal areas of concern: (1) the adequacy of equipment anchorages and supports, (2) the functional capability of electrical relays, and (3) equipment unique to nuclear power plants and outside the limits of the experience data base.

The NRC staff issued the final technical resolution of USI A-46 on February 19, 1987, as Generic Letter 87-02. Included as attachments to the generic letter were NUREG-1211, "Regulatory Analysis for Resolution of USI A-46," and NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants (USI A-46)." Sixty-nine operating plants that have not been reviewed to current licensing criteria for equipment seismic qualification are required to perform equipment seismic adequacy reviews. Implementation procedures are under development by the Seismic Qualification Utility Group (SQUG) and the Electric Power Research Institute (EPRI). Plant-specific implementation is scheduled to start in 1988.

5.1.3.2 USI A-49--Pressurized Thermal Shock

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

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

The final rule also required licensees to submit extensive plant-specific safety analyses regarding PTS risk 3 years before exceeding the screening criterion. Publication of guidance for performing those analyses in Regulatory Guide 1.154 in January 1987 resulted in completion of USI A-49.

5.2 Standardized and Advanced Reactors

5.2.1 Statement of Problem

The Commission has issued a policy statement on the regulation of advanced nuclear power plants (51 FR 24643), which states that the NRC will review and comment on new design concepts. 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 the Department of Energy's advanced reactor concepts. In-depth independent analysis will be performed as necessary to verify that advanced reactor designs have enhanced margins of safety, and appropriate means will be used to accomplish their safety function.

The Commission has issued the policy statement on standardization (52 FR 34884). The purpose of the policy statement is to encourage standardization and to provide for certification of plant designs that are essentially complete in both scope and level of detail. The policy statement also reflects the applicable provisions of the Severe Accident Policy Statement (50 FR 32138) and the proposed Nuclear Power Plant Standardization and Licensing Act of 1987. As part of this policy, the NRC will develop, review, and resolve reactor standardization policy issues and oversee their implementation.

5.2.2 Program Strategy

The Department of Energy has submitted three advanced design concepts for NRC review. These are the Sodium Advanced Fast Reactor (SAFR), Power Reactor Inherently Safe Module (PRISM), and Modular High-Temperature Gas-Cooled Reactor (MHTGR). The strategies for this program are: (1) conceptual design review of these three plants, (2) identification of major issues that need to be addressed prior to licensing, (3) identification of design features that should be verified, and (4) issuance of safety evaluation reports (SERs).

With regard to standardization, as part of the implementation of the Commission's licensing reform proposals, the legal staff is presently working on rule changes to clarify agency policy on design certification by rulemaking permitted under Appendix O. RES support has been requested in connection with this rulemaking.

Parallel with the policy development effort, NRR is reviewing three standard designs (ABWR, the Westinghouse SP-90, and the CE CESSAR) and design requirements prepared for standard plants by EPRI. Generic issues pertaining to standardization requiring further attention are expected to emerge from these reviews.

The strategies of this program are: (1) support of the legal staff's rulemaking activities on standardization and (2) resolution of generic issues pertaining to standardization arising from the ongoing standard plant reviews.

5.2.3 Research Accomplishments in 1987

5.2.3.1 Advanced Reactor Concepts

The staff continued to review three advanced reactor concepts submitted by the DOE. The purpose of these reviews is to determine the licensability of these unique designs. The conceptual designs consist of two advanced Liquid Metal Reactors and one advanced Modular High-Temperature Gas-Cooled Reactor. NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," was prepared to provide further guidance on the staff's advanced reactor review plans.

5.2.3.2 Plant Standardizations

The NRC believes that standardization of nuclear power plant designs is an important initiative that can significantly enhance the safety, reliability, and availability of nuclear plants. The Commission intends to improve the licensing process for standardized nuclear power plants and to reduce complexity and uncertainty in the regulatory process. In this regard, the Commission issued a revised Standardization Policy Statement (52 FR 34884) on September 15, 1987.

The purpose of the revised policy is to provide the regulatory framework for reference system design certification of nuclear power plant designs that are essentially complete in both scope and level of detail; cover plant design, construction, and quality assurance programs; satisfy regulatory requirements before construction begins; and can be referenced in individual plant applications. Use of certified reference designs in future license applications should enhance plant safety, increase the efficiency of the NRC review process, and reduce complexity and uncertainty in the regulatory process.

The Commission is also developing proposed regulations (10 CFR Part 52) to implement the revised standardization policy. The proposed Part 52 will provide a regulatory framework for certification of reference designs by means of rulemaking to alleviate the need to reconsider design issues in individual licensing proceedings on future license applications that reference the certified designs.

5.3 Fuel Cycle, Safeguards, Transportation, and Material Safety Research and Standards Development

5.3.1 Statement of Problem

Effective regulation of fuel cycle, safeguards, transportation, and material safety activities involves the the task of planning, developing, and issuing appropriate regulatory positions. Using information generated internally or through narrowly directed research, new positions are developed or existing positions are modified. These positions can take the form of regulatory requirements, policy statements, guidance, or criteria for activities within this element. Specific activities include: decontaminating and decommissioning licensed nuclear facilities; transporting radioactive materials; disposing of low radioactivity waste streams; and safeguarding facilities and special nuclear materials. Setting of priorities for regulatory needs or deficiencies should be established to ensure that the problems of greatest significance to the public health and safety or the common defense and security are addressed in an expeditious manner through properly defined regulatory and supporting safety programs.

5.3.2 Program Strategy

The program strategy for each of the activities in this element are as follows:

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

In the area of transportation of nuclear material, the U.S. Trade Agreements Act of 1979 directs Federal agencies to develop standards that are internationally consistent, whenever appropriate. A proposed rule will revise the transportation regulations for low specific activity material and ensure that the proprieties of the radioactive materials being shipped and the packages used in shipment adequately protect the public and the occupational health and safety of workers. The final rule will achieve maximum compatibility between NRC regulations and the transportation regulations of the International Atomic Energy Agency (IAEA).

The purpose of the safeguards program is to issue changes or additions to safeguards policy, regulations, or guidance that meet the needs of the Office of Nuclear Material Safety and Safeguards. The strategies are to (1) determine that physical security and accountability of strategic special nuclear materials (SSNM) remain adequate; (2) ensure that the value of security, physical protection, and material control and accountability are balanced against implementation costs; and (3) develop or modify safeguards regulatory requirements and guidance to be internally consistent.

In the area of material safety, the Low-Level Waste Policy Amendments Act of 1985 requires NRC to establish standards and procedures for expedited action on "below regulatory concern" (BRC) waste disposal petitions. Federal agencies are currently in the process of establishing BRC levels for radioactive waste. The strategy of this program is to carry out Commission policy to deregulate low-level waste streams by implementing standards and procedures that limit the control on disposal of these waste streams.

5.3.3 Research Accomplishments in 1987

5.3.3.1 Fuel Cycle

A commission paper discussing the current and potential applicability of de minimis and below regulatory concern concepts was completed. The concepts described are intended for use in developing a policy statement on allowable residual contamination levels following decommissioning.

5.3.3.2 Transportation

A report (NUREG/CR-4829) was issued in February 1987 to document the level of protection being provided by licensed spent fuel casks against transportation accident forces. The following month, the NRC issued a 30-page brochure (NUREG/BR-0111) to make the study's results more accessible and easily comprehended by people both within and outside the NRC. The brochure has been widely distributed to Federal and State authorities who have responsibilities to ensure that radioactive material shipments are conducted in a manner that protects public health and safety. A proposed rule modifying NRC's transportation regulations has also been issued. The rule maximizes compatibility between NRC and IAEA regulations and proposes limitations on the shipment of low specific activity (LSA) materials.

5.3.3.3 Material Safety

A report (NUREG/CR-4938), issued in July 1987, evaluated alternative low-level waste disposal methods (i.e., shallow land burial, belowground and aboveground vaults, earth-mounded concrete bunkers, augered holes, and mined cavities) from the standpoint of how occupational exposures are influenced by site design, operation, and closure. The results indicate that occupational doses do not vary greatly with disposal method but that slight changes in disposal site designs or operations could significantly affect the resulting occupational doses. A proposed rule allowing onsite incineration of waste oil at nuclear power plants has also been issued.

5.3.3.4 Safeguards

During 1987 efforts were completed on developing a proposed rule that would improve physical security at fuel facilities possessing weapons-grade nuclear material. This regulation, as proposed, would ensure that the safeguards requirements at licensed facilities are not only adequate but are comparable with requirements at similar facilities operated by the DOE. A proposed policy statement defining an access authorization program at nuclear power plants was published in the Federal Register, and a final rule modifying medical and physical requirements for armed security personnel was issued.

5.4 Developing and Improving Regulations

5.4.1 Statement of Problem

Based on the April 1987 reorganization, the Commission assigned to RES the primary responsibility to manage, coordinate independent reviews, and control all NRC rulemaking activities and to monitor scheduling of such rulemakings to ensure that rules are developed in a timely manner. In addition, RES provides support for preparation of regulatory impact analyses (RIAs) that accompany all rulemaking through development of generic methodology and guidance. Technical reviews of all RIAs are performed, and upon request technical oversight is provided or RIAs for the initiating offices are developed. The NRC Regulatory Agenda Report and other management information systems are maintained.

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

5.4.2 Program Strategy

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

5.4.3 Research Accomplishments in 1987

5.4.3.1 Develop or Modify Regulations

Continued in 1987 is a program to investigate existing regulatory requirements in terms of their risk effectiveness and to eliminate or modify those requirements that have a marginal safety importance. A two-volume research report was published in 1986, and in 1987 Volume 3 was added to NUREG/CR-4330. This report provided detailed technical evaluation of requirements in four regulatory areas: (1) postaccident sampling system, (2) impregnated charcoal filters, (3) recombiners in BWR Mark I and Mark II, and (4) turbine missiles. The NRC staff is currently considering these research results and will recommend whether to eliminate or modify related requirements that have marginal safety importance.

Some nuclear power plant licensees have requested that the NRC amend their operating licenses to permit them to keep their fuel in the reactor for a longer period than is their current practice (extended fuel burnup). In order to evaluate the environmental consequences if this method of fuel use receives a widespread following, the NRC commissioned a study documented in NUREG/CR-5009 entitled "The Environmental Consequences of Higher Fuel Burn-up." The study considered the various aspects of fuel production, transportation, power generation, and waste management in determining the effect of extended fuel burnup. The overall finding of the study was that there would be no significant increase in the environmental impact associated with the widespread use of extended fuel burnup. The NRC staff is currently considering this result and will recommend whether to modify the existing regulations on fuel burnup.

5.4.3.2 Independent Review and Control of Rulemaking

During 1987, RES was given lead responsibility in the NRC for rulemaking. Therefore, the control of rulemaking responsibilities was broadened to include coordination of rulemaking with the requesting offices and recommendations for assignment of requested rulewriting within RES.

During 1987, 57 rulemakings were processed for potential independent review by RES for initiation or continuation. Of these, full reviews were completed on 11 rulemakings, 30 were exempted from RES review because of final publication or EDO action, and 16 were in progress at the end of 1987 or had been deferred until 1988.

GLOSSARY

Acronyms and Initialisms

ABWR	Advanced boiling water reactor
ACOT	Allowed cumulative outage time
ACRR	Annular Core Research Reactor
AECL	Atomic Energy of Canada Ltd.
AEOD	(Office for) Analysis and Evaluation of Operational Data
ALARA	As low as is reasonably achievable
ANPR	Advanced notice of proposed rulemaking
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BEIR	(Committee on the) Biological Effects of Atomic Radiation
BNL	Brookhaven National Laboratory
BRC	Below regulatory concern
B&W	Babcock and Wilcox
BWR	Boiling water reactor
CE	Combustion Engineering
CEC	Continuous experimental capability
CES	Cognitive Environmental Simulation (to analyze intention aspects of human behavior)
CFR	Code of Federal Regulations
COMMIX	Code to analyze natural circulation flow patterns in the reactor coolant system
CONTAIN	Containment analysis code
CORCON	Code to model interaction between molten core materials and concrete during core melt accidents
CREATE	Cognitive reliability analysis technique
CSAU	Code scaling, applicability, and uncertainty (program)

DCH	Direct containment heating
DDT	Deflagration to detonation transition
DOD	Department of Defense
DOE	Department of Energy
ECCS	Emergency core cooling system
EDO	Executive Director of Operations
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
FRG	Federal Republic of Germany
GSI	Generic safety issue
HDR	Heissdampfreaktor (a decommissioned steam reactor in the Federal Republic of Germany where reactor safety experiments are conducted)
HECTR	Code to calculate hydrogen behavior
HLW	High-level waste
HMS-BURN	Code to provide detailed hydrogen transport and mixing calculations
HRA	Human reliability analysis
IAEA	International Atomic Energy Agency
ICAP	International Code Assessment Program
ICRP	International Commission on Radiological Protection
IDCOR	Industry degraded core rulemaking (program)
INEL	Idaho National Engineering Laboratory
IPE	Individual plant examination
IRRAS	Integrated Reliability and Risk Analysis System
LACE	LWR aerosal containment experiments
LCO	Limiting condition of operation
LLW	Low-level waste
LOCA	Loss-of-coolant accident
LSA	Low specific activity
LWR	Light-water reactor

MACCS	Code to assess the consequences of radioactive releases
MELCOR	Code to model meltdown accident assessment
MELPROG	Melt progression code
MHTGR	Modular high temperature gas-cooled reactor
MIST	Multiloop integral system test
MOV	Motor-operated valve
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection and Measurements
NEPA	National Environmental Policy Act
NRR	(Office of) Nuclear Reactor Regulation
NRU	National Research Universal (test reactor at Chalk River, Ontario, Canada)
NUCLARR	Data base management system that processes and stores human error and hardware failure data for supporting reliability evaluations
NWPAA	Nuclear Waste Policy Amendment Act
PNL	Battelle-Pacific Northwest Laboratories
PORV	Power-operated relief valve
PRA	Probabilistic risk assessment
PRISM	Power reactor inherently safe module
PTS	Pressurized thermal shock
PWR	Pressurized water reactor
REIRS	Radiation exposure information reporting system
RELAP	Code to model thermal-hydraulic behavior in reactor coolant system during transient and loss-of-coolant accident
RES	(Office of Nuclear Regulatory) Research
RIA	Regulatory impact analysis
RMIEP	Risk Methodology Integration and Evaluation Program
SAFR	Sodium advanced fast reactor
SAIC	Science Applications International Corporation
SALP	Systematic assessment of license performance (program)
SARA	Systematic analysis and risk assessment

SCDAP	Severe core damage analysis package
SER	Safety evaluation report
SSE	Safe shutdown earthquake
SSNM	Strategic special nuclear material
TMI	Three Mile Island
TMLB'	Station blackout accident with loss of auxiliary feedwater
TRAC	Code to model core reflood and quenching
USGS	United States Geological Survey
USI	Unresolved safety issue
VARSKIN	Code to calculate radiation dose of skin contamination

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This report, the third 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 1987.

The goal of this office is to ensure that research provides the technical bases for rulemaking and for related decisions in support of NRC licensing and inspection activities. This report describes both the direct contributions to scientific and technical knowledge with regard to nuclear safety and their regulatory applications.

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